IAEA-TECDOC-1536

Status of Small Reactor Designs Without On-Site Refuelling

IAEA
International Atomic Energy Agency

January 2007

IAEA-TECDOC-1536

Status of Small Reactor Designs Without On-Site Refuelling

IAEA
International Atomic Energy Agency

January 2007

The originating Section of this publication in the IAEA was:

Radiation and Transport Safety Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

STATUS OF SMALL REACTOR DESIGNS WITHOUT ON-SITE REFUELLING
IAEA, VIENNA, 2007
IAEA-TECDOC-1536
ISBN 92–0–115606–5
ISSN 1011–4289

Printed by the IAEA in Austria
January 2007

FOREWORD

There is an ongoing interest in member states in the development and application of small and medium sized reactors (SMRs). In the near term, most new NPPs are likely to be evolutionary designs building on proven systems while incorporating technological advances and often the economics of scale, resulting from the reactor outputs of up to 1600 MW(e). For the longer term, the focus is on innovative designs aiming to provide increased benefits in the areas of safety and security, non-proliferation, waste management, resource utilization and economy, as well as to offer a variety of energy products and flexibility in design, siting and fuel cycle options. Many innovative designs are reactors within the small-to-medium size range, having an equivalent electric power less than 700 MW(e) or even less than 300 MW(e). A distinct trend in design and technology development, accounting for about half of the SMR concepts developed worldwide, is represented by small reactors without on-site refuelling. Such reactors, also known as battery-type reactors, could operate without reloading and shuffling of fuel in the core over long periods, from 5 to 25 years and beyond.

Upon the advice and with the support of IAEA member states, within its Programme 1 "Nuclear Power, Fuel Cycle, and Nuclear Science", the IAEA provides a forum for the exchange of information by experts and policy makers from industrialized and developing countries on the technical, economic, environmental, and social aspects of SMRs development and implementation in the 21st century, and makes this information available to all interested Member States by producing status reports and other publications dedicated to advances in SMR technology.

The objective of this report is to provide Member States, including those just considering the initiation of nuclear power programmes and those already having practical experience in nuclear power, with a balanced and objective information on important development trends and objectives of small reactors without on-site refuelling for a variety of uses, on the state of the art in technology development for such reactors, and on their design status.

The report is intended for many categories of stakeholders, including electricity producers, non-electrical producers, policy makers, designers, and regulators.

The main sections of this report, addressed to all above mentioned groups of stakeholders, survey emerging energy market characteristics and draw a rationale for small reactors without on-site refuelling; provide a summary and an assessment of major design specifications, applications and user-related special features for the surveyed reactor concepts; review the design status and targeted deployment dates; and outline the possible fuel cycle approaches.

The annexes, intended mostly for designers and technical managers, provide detailed design descriptions of small reactors without on-site refuelling under development worldwide and are patterned along a common format, which makes it possible to identify the design philosophy, objectives and approaches, as well as technical features and non-technical factors and arrangements with a potential to provide solutions in the specific areas of concern associated with future nuclear energy systems.

The scope of this report is limited to reactors without on-site refuelling, i.e. small reactors of less than 300 MW(e) effective output that are designed for infrequent replacement of well-contained fuel cassettes in a manner that impedes clandestine diversion of nuclear fuel material. SMRs with conventional refuelling schemes have been addressed in previous IAEA publications.

The IAEA officer responsible for this publication was V. Kuznetsov of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION 1

1.1. Background 1
1.1.1. Developments in Member States 1
1.1.2. Previous IAEA publications 5
1.2. Objectives 6
1.3. Scope 7
1.4. Structure 8

2. DEFINITIONS, TERMS AND FORMATS FOR SMR DESIGN DESCRIPTION ... 9

2.1. Definitions and terms 9
2.1.1. Small and medium sized reactors (SMRs) 9
2.1.2. Small reactors without on-site refuelling 9
2.1.3. Innovative design 10
2.1.4. Safety related terms 10
2.1.5. Proliferation resistance related terms 13
2.1.6. Design and regulatory status 13
2.1.7. Enabling technologies 15
2.2. Formats for reactor design description 15
References for Chapters 1 and 2 15

3. SURVEY OF EMERGING ENERGY MARKET CHARACTERISTICS — A RATIONALE FOR SMALL REACTORS WITHOUT ON-SITE REFUELLING … 17

3.1. Introduction 17
3.2. The energy supply challenge of sustainable economic development 17
3.2.1. The link from energy supply to economic development 17
3.2.2. Projections of global primary energy demand growth 19
3.2.3. The challenge of sustainable development — a potentially expanded role for nuclear power 19
3.3. Characteristics of emerging markets for nuclear energy 20
3.3.1. Projected nuclear growth in developing countries 20
3.3.2. Demand for a broadened scope of energy products 22
3.4. Re-engineering nuclear energy to serve emerging energy markets — a rationale for small reactors without on-site refuelling 25
3.4.1. Possible energy architecture with small reactors without on-site refuelling 26
3.4.2. Energy security versus non-proliferation dilemma 27
3.5. Survey of customer needs 27
3.5.1. Villages and towns in off-grid locations 28
3.5.2. Energy intensive industrial sites in off-grid locations 30
3.5.3. Cities in developing countries 32
3.5.4. Future merchant plants for non-electric energy products 35
3.5.5. Other possible markets 37
3.5.6. Institutional prerequisites and requested support from the vendor 38
3.6. Business strategies for small reactors without on-site refuelling — an option of nuclear energy configuration tailored to the needs of emerging customers 40
3.6.1. Summary of customer needs 40
3.6.2. Common features of small reactors without on-site refuelling 43
3.6.3. Strategies to overcome the loss of economies of scale 44

3.6.4. Strategies to balance energy security with non-proliferation assurances 46
3.6.5. Strategies for transition to a sustainable fuel cycle 46
3.6.6. Decommissioning strategies 46
3.7. Conclusion to Chapter 3 47
References to Chapter 3 48

4. SURVEY OF POWER PLANT CONCEPTS, DESIGN APPROACHES, AND POSSIBLE VENDOR SUPPLIED OPERATIONAL SUPPORT SERVICES 50

4.1. Introduction 50
4.2. Structure of the chapter 51
4.3. Organizational structure for the concepts 52
4.4. Survey of nearer-term concepts 64
4.4.1. Very small reactors for unattended autonomous operation 64
4.4.2. Floating power plants with cogeneration capability 65
4.4.3. Liquid metal cooled concepts 67
4.4.4. Vendor supplied operational support services 68
4.5. Survey of longer-term power plant concepts 69
4.5.1. Motivation for innovative designs 69
4.5.2. LWR derivative power plants 69
4.5.3. Small LWRs with TRISO fuel 71
4.5.4. Concepts of liquid metal cooled reactors 73
4.5.5. High temperature concepts including hydrogen production 74
4.5.6. Remaining R&D and time to first deployment 74
4.6. Survey of power plant design approaches 75
4.6.1. Influence of targeted commercialization date on design choices 76
4.6.2. Approaches to achieve long refuelling interval 77
4.6.3. Approaches to reduce operating crew size and required skill level 79
4.6.4. Approaches to improve heat transport and energy conversion efficiency 80
4.6.5. Approaches to facilitate near-urban siting 81
4.6.6. Approaches for protection against external hazards 82
4.6.7. Approaches to achieve plant transportability and rapid site assembly 83
4.6.8. Approaches for multi-plant clustering 85
4.6.9. Approaches for high temperature process heat production 85
4.7. Strategies to facilitate plant commercialization 87
4.8. Conclusion to Chapter 4 88
References to Chapter 4 90

5. SURVEY OF FUEL CYCLES AND INSTITUTIONAL ARRANGEMENTS 91

5.1. Introduction 91
5.1.1. Near-term versus longer term fuel cycle options 91
5.1.2. Transition to a sustainable fuel cycle 92
5.1.3. Structure of the chapter 92
5.2. Parameters that characterize fuel cycle performance 93
5.3. Survey of fuel cycle performance characteristics and support services offered by the concepts 93
5.3.1. Nearer-term open cycle performance characteristics and fuel cycle support services 93

5.3.2. Longer-term closed cycle performance characteristics and fuel cycle support services 97
5.4. Efficiencies of fuel and ore usage in small reactors without on-site refuelling 98
5.4.1. Efficiency of fuel and ore utilization for open cycle small reactor concepts 98
5.4.2. Efficiency of fuel and ore utilization for closed cycle small reactor concepts 99
5.4.3. The fuel leasing option 100
5.5. Possible role of small reactors without on-site refuelling in the transition from an open to a global closed nuclear fuel cycle 101
5.5.1. Issues for a transition from open cycle to closed cycle 101
5.5.2. Effectiveness of closed cycle small reactor initial loadings as a waste sequestration strategy 102
5.5.3. Possible impact of small reactors on the dynamic response capability of the global fuel cycle 103
5.6. Fuel cycle institutional issues 105
5.6.1. Approaches to resolve energy security versus non-proliferation dilemma 105
5.6.2. Features of small reactors without on-site refuelling that could facilitate their deployment with outsourced fuel cycle services 106
5.7. Conclusion to Chapter 5 106
References to Chapter 5 108

6. PROGRAMMES IN MEMBER STATES FOR DEVELOPMENT OF SMALL REACTORS WITHOUT ON-SITE REFUELLING 110

6.1. Brazil 110
6.2. India 110
6.3. Indonesia 110
6.4. Japan 111
6.5. The Russian Federation 113
6.6. The United States of America 117
References to Chapter 6 119

APPENDIX I: OUTLINE FOR SMR DESIGN DESCRIPTION (FULL DESCRIPTION) 121

APPENDIX II: OUTLINE FOR SMR DESIGN DESCRIPTION (SHORT DESCRIPTION) 125

CONTRIBUTIONS FROM MEMBER STATES — DESIGN DESCRIPTIONS OF SMALL REACTORS WITHOUT ON-SITE REFUELLING: WATER COOLED SMALL REACTORS

ANNEX I: MULTI-APPLICATION SMALL LIGHT WATER REACTOR (USA) 129

ANNEX II: TRANSPORTABLE REACTOR FACILITY FOR ELECTRICITY SUPPLY IN REMOTE AREAS (UNITHERM) (Russian Federation) 157

ANNEX III: UNATTENDED SELF-CONTROLLED NUCLEAR THERMOELECTRIC PLANT (ELENA NTEP) (Russian Federation) 183

ANNEX IV: FLOATING NPP WITH VBER-150 REACTOR INSTALLATION (Russian Federation) 205

ANNEX V: WATER COOLED MODULAR NUCLEAR POWER REACTOR ABV (Russian Federation) 235

ANNEX VI: KLT-20 REACTOR FOR A FLOATING POWER UNIT (Russian Federation) 269

ANNEX VII: PASSIVELY SAFE SMALL REACTOR FOR DISTRIBUTED ENERGY SUPPLY SYSTEM (PSRD) (Japan) 299

ANNEX VIII: PACKAGE-REACTOR (Japan) 303

ANNEX IX: PARTICLE FUEL PRESSURIZED WATER REACTOR (PFPWR50) (Japan) 319

ANNEX X: BOILING WATER REACTOR WITH MICRO FUEL ELEMENTS (VKR-MT) (Russian Federation) 333

ANNEX XI: LONG-LIFE CORE SMALL BUYCLEAR REACTOR WITH CLOSED-VESSEL REFUELLING — ATOMS FOR PEACE REACTOR (AFPR-100) (USA) 367

ANNEX XII: FIXED BED NUCLEAR REACTOR (FBNR) (Brazil). 373

GAS COOLED SMALL REACTORS

ANNEX XIII: FAST NEUTRON BATTERY-TYPE GAS COOLED POWER REACTOR OF 300 MW THERMAL (BGR-300) (Russian Federation) 389

SODIUM COOLED SMALL REACTORS

ANNEX XIV: SUPER-SAFE, SMALL AND SIMPLE REACTOR (4S, TOSHIBA DESIGN) (Japan) 395

ANNEX XV: SUPER-SAFE, SMALL AND SIMPLE LIQUID METAL COOLED REACTOR (4S-LMR, CRIEPI DESIGN) (Japan) 425

ANNEX XVI: MODULAR PLANT WITH SODIUM COOLED FAST REACTOR (MBRU-12) (Russian Federation) 445

ANNEX XVII: WHOLE-CORE REFUELLED INTEGRAL DESIGN SMALL SODIUM COOLED FAST REACTOR (RAPID) (Japan) 469

ANNEX XVIII: TRANSPORTABLE MODULAR SODIUM COOLED REACTOR WITH GAS-TURBINE GENERATOR (BN GT-300) (Russian Federation) 491

LEAD AND LEAD-BISMUTH COOLED SMALL REACTORS

ANNEX XIX: LEAD BISMUTH COOLED FAST REACTOR SVBR-75/100 (Russian Federation) 511

ANNEX XX: ENCAPSULATED NUCLEAR HEAT SOURCE (ENHS) (USA) 551

ANNEX XXI: SMALL LEAD-BISMUTH COOLED REACTOR (Japan) 587

ANNEX XXII: SMALL SECURE TRANSPORTABLE REACTOR (SSTAR) (USA) 591

ANNEX XXIII: SECURE TRANSPORTABLE AUTONOMOUS REACTOR — LIQUID METAL (STAR-LM)(USA) 625

ANNEX XXIV: SECURE TRANSPORTABLE AUTONOMOUS (STAR-H2) (USA) 655

ANNEX XXV: LEAD-BISMUTH EUTECTICS COOLED LONG-LIFE SAFE SIMPLE SMALL PORTABLE PROLIFERATION RESISTANT REACTOR (LSPR) (Japan) 715

ANNEX XXVI: SMALL AND VERY SMALL LEAD-BISMUTH COOLED NUCLEAR POWER REACTORS WITHOUT ON-SITE REFUELLING (SPINNOR/VSPINNOR) (Indonesia) 739

ANNEX XXVII: LEAD-BISMUTH COOLED DIRECT CONTACT BOILING WATER SMALL REACTOR (PBWFR) (Japan) 761

NON-CONVENTIONAL SMALL REACTORS

ANNEX XXVIII: MICRO-PARTICLE FUEL AUTONOMOUS MOLTEN SALT COOLED REACTOR (MARS) (Russian Federation) 769

ANNEX XXIX: COMPACT HIGH TEMPERATURE REACTOR (CHTR) (India) 793

ANNEX XXX: MOLTEN SALT REACTOR FOR SUSTAINABLE NUCLEAR POWER — MSR FUJI (Japan) 821

CONTRIBUTORS TO DRAFTING AND REVIEW 857

1. INTRODUCTION

1.1. Background

1.1.1. Developments in Member States

According to the classification currently adopted by IAEA, small reactors are the reactors with an equivalent electric power less than 300 MW, medium sized reactors are the reactors with an equivalent electric power between 300 and 700 MW [1]. Small reactors came first historically, as power sources for nuclear submarines. In the early decades, civil nuclear power essentially borrowed from the experience of such reactors, and many nuclear power plants (NPPs) with medium-sized reactors have been deployed in 1960s and 1970s. Since the 1970s, however, the major focus for nuclear power has been on the design of nuclear plants of increasing size, with current new construction underway for several plants in the 1000–600 MW(e) range. This scaling up is generally appropriate for many industrialized countries, which can add generation capability to their electrical grids in larger increments and benefit from the reduced construction costs associated with scale factor. However, for many developing countries that have small electricity grids, limited capacity for investment and less developed infrastructure this approach may be less conducive. It may also be less vital if nuclear energy is considered for non-electric energy markets, such as district heating or advanced process heat applications [1, 2].

All analyses and forecasts of global energy needs project large increases in the century ahead. For example, a projection from the International Panel on Climate Change [3] indicates that primary energy demand in the world may double by 2050, see Fig. 1.

The IAEA's Nuclear Technology Review (2006 update) projects a 13% (low) to 72% (high) increase of nuclear generation from the current 368 GW(e) by the year 2030 [4]. The difference between high and low is 222 GW(e), with 66 GW(e) of the difference, or 30%, corresponding to Western Europe, and 52 GW(e), or 23%, corresponding to the Far East.

The trends in the world at large accompanying rising expectations of the future role of nuclear power, as identified in the Medium Term Strategy of the IAEA [5], are as follows:

- Unprecedented expansion of energy demand that the world faces in the next fifty years. This will be driven by continuing population growth, economic development and the aspiration to provide access to modern energy systems to the 1.6 billion people now without such access;
- Greater emphasis on technological advances that strengthen proliferation resistance whilst, at the same time, continue to facilitate the spread of nuclear power benefits to interested member states.

The principal drivers behind projected large increase in global energy needs are population growth and economic development in today's developing countries [3]; therefore, in defining pathways for nuclear power it is important to address the specific needs of such countries. These needs may vary between different groups of developing countries, but are often defined by limited capacity for investment and weak electricity grids [1]. Certain areas in some developing countries suffer from the deficiency of potable water [6]. Legal, institutional and human resource provisions for nuclear power are in many cases not emplaced or insufficient [7]. Many developing countries suffer from corruption and poverty, which fosters political instability and could make them an attractive domain for international terrorism. A transfer of traditional nuclear power and, especially, nuclear fuel cycle technologies to such countries might pose a proliferation and security risk.

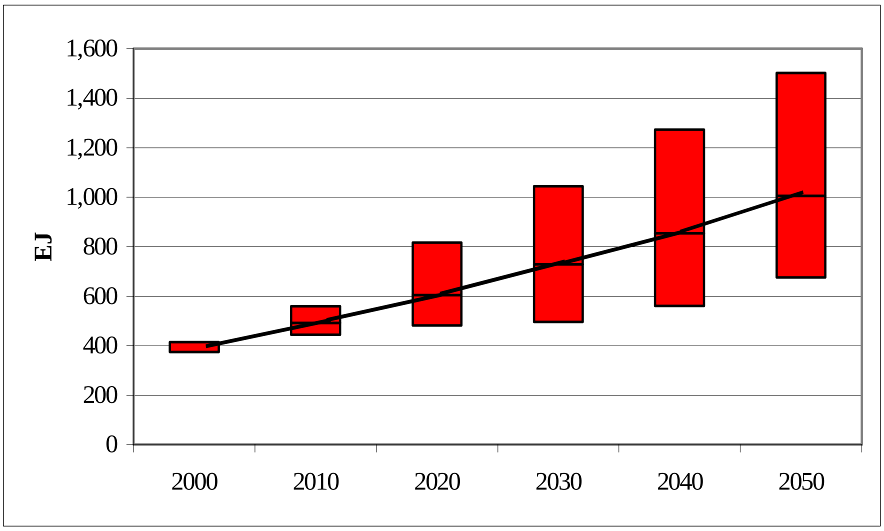


FIG. 1. Range of future primary energy demand in SRES scenarios, 2000–2050. Solid line represents median [3].

A general consideration for having SMRs in nuclear power's portfolio is that 'one size' is unlikely to fit for all, like it doesn't in other major industries, including car and aircraft industry, as well as energy production from fossil fuel [2]. However, a stagnation period that followed the Chernobyl accident, the ongoing liberalization of energy markets, and the progress in certain technologies of energy production from fossil fuel coupled, until recently, with relatively low prices for natural gas and oil altogether forced nuclear industries to fight for survival, making the construction of any new NPP a self-standing priority. Such situation, predominant over the past two decades, was generally unfavourable to many SMRs of older design, especially those that just reproduced the features of higher-capacity reactors at a reduced scale, e.g., some previous generation water cooled SMRs [8] that failed to compete with larger plants on an economy of scale basis, resulting in a loss of interest from many major vendors and utilities[1].

On the other hand, the above mentioned developments boosted SMR proposers and designers all over the world to pursue new design approaches making use of certain advantages provided by smaller reactor size to achieve reduced design complexity and, perhaps, simplified operation and maintenance requirements [1]. In many cases, the design approaches used for these innovative SMRs are unique, i.e., cannot be reproduced in the reactors of larger capacity and, therefore, represent alternative strategies to overcome loss of economies of scale [1, 2].

The attractive features of innovative SMRs that might facilitate their progress in certain energy markets are as follows:

- Fitness for small electricity grids, including an option of autonomous operation;
- Lower absolute overnight capital costs, as compared to large capacity plants;
- An option of incremental capacity increase that could perfectly meet the incremental increase of demand and minimize financial risk to the investor; and

---

[1] A notable exception is provided by national experience of India, a developing country that is successfully ongoing with operation and construction of new nuclear power plants with the domestically produced pressurized heavy water reactors of 209 and 490 MW(e) net capacity.

- Reduced design complexity, reduced impact of human factors, and, perhaps, reduced operation and maintenance requirements, which altogether could make SMRs a perfect vehicle to support local manufacturing and constructing industries as well as local and regional academic institutions in developing countries.

In addition to this, SMRs are a preferred option for non-electric applications that require a proximity to the customer (such as seawater desalination, district heating and other process heat applications).

As of 2005–2006, about 60 concepts and designs of innovative SMRs were analyzed or developed within national or international programmes in 13 IAEA member states, including both industrialized and developing countries [1, 2]. Innovative SMRs are under development for all principal reactor lines and some unusual combinations thereof. The focus is on innovative[2] design approaches aiming to provide increased benefits in the areas of economics and maintainability, safety and reliability, and proliferation resistance and security (physical protection), as indicated in Table 1.

About a half of the innovative SMR concepts represent small reactors without on-site refuelling [2], also known as battery-type or long-life core reactors. Small reactors without on-site refuelling are the reactors designed for infrequent replacement of well-contained fuel cassette(s) in a manner that impedes clandestine diversion of nuclear fuel material [1].

Small reactors without on-site refuelling incorporate increased refuelling interval (from 5 to 30 years and more), consistent with plant economy and considerations of energy security. Both front-end and back-end fuel cycle services are assumed to be completely outsourced for such reactors, i.e., they are either factory fabricated and fuelled or undergo a once-at-a-time core reloading performed at the site by a dedicated service team provided by the vendor of the fuel or the reactor itself; such team is assumed to bring in and take away the fresh and spent fuel load and the refuelling equipment.

About thirty concepts of small reactors without on-site refuelling are being analyzed or developed in the Russian Federation, Japan, India, the U.S.A., Brazil, and Indonesia. They cover different reactor lines: water cooled, sodium cooled, lead or lead bismuth cooled and molten salt cooled reactors.

The targeted dates for prototype or commercial deployment vary between early 2010s and 2030. Most of the concepts are at an early design stage, and only a few have reached the basic or detailed design stages.

The potential benefits of small reactors without on-site refuelling include:

- Possibly lower construction costs in a dedicated facility in the supplier country;
- Lower investment costs and risks for the purchaser, especially if the reactor is leased rather than bought and if the plant capacity increase is incremental;
- Reduced obligations of the user for spent fuel and waste management; and
- Possibly greater or easier non-proliferation assurances to the international community.

The proponents of small reactors without on-site refuelling reasonably expect that such reactors could add a certain degree of independence on fuel supplier and, in this way, support decisions of the user-countries to forego the development of indigenous fuel cycles.

---

[2] The IAEA-TECDOC-936 [1–9] defines an innovative design as the design “that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice” and would, therefore, “require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented”.

TABLE 1. SUBJECT AREAS FOR INNOVATIVE SMRs [2]

SUBJECT AREAS CONSIDERED BY DESIGNERS OF INNOVATIVE SMRS	SOLUTIONS PROPOSED (EXAMPLES)
Economics and maintainability	Incremental capacity increase through modular approach; design standardization and mass production; reduced design complexity and simplified operational requirements achieved through strong reliance on passive safety design options
Safety and reliability	Strong reliance on passive safety design options: inherent and passive safety features, and reliable passive systems; finding an effective combination between passive and active systems, etc.
Proliferation resistance and security	Broader reliance on intrinsic proliferation resistance features, specifically, to facilitate implementation of the extrinsic measures such as safeguards
Resources, waste management and environmental impacts	Improved fuel utilization, e.g., achieved through higher fuel burn-up or recycling; clearly defined strategies of spent nuclear fuel and waste management; design features to reduce off-site emergency planning
Applications	Generation of electricity, district heating, production of potable water, hydrogen etc; various cogeneration options, including purposeful use of the reject heat
User-related special features	Modular approach to reactor design and/or fabrication and construction; transportability (e.g., floating NPP options); infrequent refuelling; flexible design, siting and applications
Fuel cycle options	Flexible fuel cycle options, e.g., once-through use or recycling of U, Pu and Th fuel; centralized (e.g., regional) fuel cycle services
Enabling technologies	An enabling technology is the technology that needs to be developed and demonstrated to make a certain reactor concept viable. Upon a diversity of SMR designs, it may be useful to identify enabling technologies that are common to several SMR designs and, therefore, could benefit from being developed on a common or shared basis
Marketing strategy and deployment scenarios	Tailoring designs to specific market needs; considering SMRs as a part of innovative nuclear energy systems; figuring out deployment strategies on the basis of dynamic system simulations
Non-technical factors and arrangements to facilitate deployment	International cooperation; infrastructure developments to support centralized fuel cycle services or reactor leasing; guarantees of sovereignty to countries that would prefer to lease fuel; reciprocity of licensing/ design certification arrangements between countries; simplified licensing procedures, e.g., “License-by-Test” and reduced or eliminated off-site emergency planning

Not less important, there appears to be a market for such reactors in remote areas with no electricity grids and high current cost of fossil energy.

Small reactors without on-site refuelling could also offer a cost-effective decommissioning strategy in which the disassembling and all subsequent operations with a complete reactor module or even a complete NPP (e.g., barge-mounted) are outsourced to a centralized factory, and which would benefit from the absence of fresh and spent fuel storages at the site.

Reflecting on the developments in Member States, the IAEA's Medium Term Strategy [5] defines one of IAEA's objectives as the following:

"Expand partnerships and information exchange and facilitate collaborative research and development for beneficial uses of nuclear energy — including evolutionary and innovative technological developments for improved competitiveness, safety, proliferation resistance and waste reduction — particularly for developing countries in the areas of small and medium size reactors and non-electricity applications such as desalination, heat production and hydrogen production."

1.1.2. Previous IAEA publications

A direct predecessor of this report, sharing the same approach in information collection, assessment and presentation, is IAEA-TECDOC-1485 [2] titled "Status of innovative small and medium sized reactor designs 2005: reactors with conventional refuelling schemes." Different from it, this report describes design status for another category of innovative SMRs, not addressed in [2]. Because small reactors without on-site refuelling offer some special features that might be of benefit to certain categories of customers and/or support the initiatives for centralized or regional fuel cycle support services, the present report incorporates dedicated chapters with the analysis and assessment of a potential of small reactors with respect to certain customer groups and innovative fuel cycle architectures. As comes to other aspects, both reports are similar in sharing the same objective and being intended for several categories of stakeholders, including electricity producers, non-electrical producers, designers, regulators, and policy makers. Detailed design descriptions of SMRs in the annexes of both reports were prepared by the designers according to a common design description outline.

To support the preparation of this and the previous report [2], an IAEA technical meeting "Innovative small and medium sized reactors: design features, safety approaches and R&D trends" was held on 7–11 June 2004 in Vienna, and its final report was published as IAEA-TECDOC-1451 in May 2005 [1]. That TECDOC presented a variety of innovative water cooled, gas cooled, liquid metal cooled and non-conventional SMR designs developed worldwide and examined the technology and infrastructure development needs that are common to several concepts or lines of such reactors. It also introduced the definition of small reactors without on-site refuelling, which is referred to in this report. Both, the technical meeting and the report [1] provided recommendations on the objectives, structure, scope and content of this report and the report [2].

Before the publications mentioned above, the last status report published for SMRs was IAEA-TECDOC-881 [10], issued in 1995. That report included design descriptions of 29 innovative small and medium sized reactors, including some small reactors without on-site refuelling. Since that time many developments took place; for some designs the development activities have resulted in a significant progress towards detailed design and licensing, while for the others development activities for whatever reasons have been stopped. Many new developments for innovative SMRs have originated and progressed just within the past decade. The focus of IAEA-TECDOC-881 was primarily on safety and economics, while

IAEA-TECDOC-1485 [2] and this report incorporate the descriptions of features and approaches related to proliferation resistance and plant security (physical protection), fuel cycle and non-electrical applications, and outline non-technical factors and arrangements that could facilitate effective development and deployment of the presented SMRs.

In 2001–2002, the International Energy Agency (IEA), the OECD Nuclear Energy Agency (OECD/NEA) and the IAEA have been conducting a joint project to examine R&D needs on the innovative nuclear fission reactor technologies and to explore the potential for enhanced international collaboration in developing these technologies. This project, called "Three Agency Study" [11], also intended to highlight how new reactor designs are addressing the issues currently rated as critical for further deployment of nuclear power and, to this end, it has defined an approach partially similar to that used in the present report as well as in IAEA-TECDOC-1485 [2]. At the same time, the "Three Agency Study" was neither a dedicated publication on SMRs nor presented their technical descriptions as detailed as provided for by a status report.

It was with these, previous developments in mind that IAEA recommended to prepare a new report on design status of the small reactors without on-site refuelling.

1.2. Objectives

The general objective of this report is to provide member states, including those just considering the initiation of nuclear power programmes, and those already having practical experience in nuclear power, with a balanced and objective information on the definitions, possible applications and development trends of small reactors without on-site refuelling, and on the achieved state-of-the-art in design and technology development for such reactors.

Specific objectives of this report are the following:

(1) To introduce a rationale for small reactors without on-site refuelling and to identify certain categories of customers that might benefit from special features offered by such reactors;

(2) Through direct cooperation with the designers in member states, to define, collate and present the state-of-the art in the definitions, design objectives, design approaches and technical features of small reactors without on-site refuelling making a focus on their potential to provide solutions in the following areas of concern, rated important for future nuclear energy systems:

- Economics and maintainability;
- Safety and reliability;
- Proliferation resistance and security (physical protection);
- Resource utilization, waste management and environmental impacts;
- Fuel cycle options;
- Applications;
- User-related special features;
- Enabling technologies; and, to the extent possible;
- Marketing strategy and deployment scenarios.

(3) To provide a technical and information background to assist the designers of such reactors in defining consistent design strategies regarding the selected subject areas;

(4) To provide various categories of stakeholders in member states, including electricity producers, non-electrical producers, possible vendors, regulators, and policy makers, with a balanced and objective assessment and summary information on the application potential, development status and prospects of such reactors;

(5) To provide an information support to high-level technical managers and policy makers in member states who are planning to assess innovative SMR projects with a potential of deployment between 2010 and 2030.

1.3. Scope

The structure and scope of this report were defined through several technical [1] and research coordination meetings, with the support from IAEA Technical Working Groups (TWGs) on advanced water cooled, gas cooled, and fast reactors and accelerator driven systems, as well as from the International Coordinating Group (ICG) of the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) [12] and SMR designers in member states. No limits were set regarding the types of small reactors without on-site refuelling; therefore, the report includes design descriptions and summaries of water cooled, gas cooled, liquid metal cooled, and molten salt cooled reactors, as well as some non-conventional combinations thereof. The upper limit for targeted deployment dates was kept open, and it was generally accepted that many small reactors without on-site refuelling are at a conceptual or even pre-conceptual design stage. Regarding the objectives of this report, which addresses small reactors targeted for a prototype or commercial deployment roughly between 2010 and 2030, bringing out as many design approaches as possible was rated useful to foster further design adjustment, modification, merging and transformation and, perhaps, the origination of new concepts and designs that might better fit the requirements to future nuclear energy systems [12, 13].

To collect information on small reactors without on-site refuelling, a new common outline for design descriptions already applied in [2] was used, which provides for a structured description of the features and anticipated performance of innovative SMRs in all considered subject areas, see Table 1 and Chapter 2.2. Reflecting on the fact that some small reactors without on-site refuelling may be at a design stage too early to provide all data requested, a shorter version of the outline was used also, see Chapter 2.2.

The designers in Member States were then contacted with an offer of participation in this report and informed about its objectives and the approach to be used in its preparation. Specifically, they were informed about the adopted definition of small reactors without on-site refuelling (see Chapter 2.1.2) and the design description outlines to be applied.

In response to the above mentioned activity, twenty six full and four short design descriptions of small reactors without on-site refuelling were collected from Member States, roughly by the end of 2005.

All design descriptions, included as Annexes I through XXX, were edited by the IAEA secretariat and then reviewed and approved for publication by their respective designers (representatives of vendors, research and design organizations and academic institutions in Member States).The review was conducted throughout 2005 and into the first half of 2006.

Based on the inputs provided by member states, a group of international experts from developed and developing countries has prepared several cross-cutting and summary chapters (Chapters 3, 4, and 5), introducing a rationale for small reactors without on-site refuelling and identifying a potential customer base, providing an assessment of the presented power plant concepts, and outlining possible fuel cycle architectures and institutional arrangements to support deployments of such reactors in the near and longer term.

Prepared in the above mentioned way, the report covers all or nearly all efforts for development of dedicated small reactors without on-site refuelling that were ongoing in 2005 and early in 2006.

1.4. Structure

The report includes an introduction, 5 chapters, 2 Appendices and 30 Annexes.

The introduction (Chapter 1) describes the background and identifies the objectives, the scope and the structure of this report.

Chapter 2 collates the definitions and terms used. The outlines (formats) used in the preparation of design descriptions for this report are enclosed as Appendices I and II.

Chapter 3 gives a survey of emerging energy market characteristics in different regions of the world, introduces a rationale for small reactors without on-site refuelling and discusses possible strategies to introduce such reactors to different categories of potential customers.

Chapter 4 provides summary tables of reactor concepts and designs addressed in this report, including:

- Major specifications and applications;
- Achieved design and regulatory status; and
- Targeted deployment dates.

This chapter also summarizes the design approaches and technical features for the nearer and longer term concepts, outlines the scope of further necessary R&D, provides an assessment of timelines of readiness for deployment for certain groups of the concepts, and reflects on the strategies to facilitate plant commercialization.

For the addressed concepts of small reactors without on-site refuelling, Chapter 5 reviews the fuel cycle options and associated institutional issues, provides an assessment of material balance characteristics in once-through and closed fuel cycles, and outlines the possible role of small reactors without on-site refuelling in making a transition from open to closed nuclear fuel cycle. This chapter also summarizes the features of small reactors that could facilitate their deployment with outsourced fuel cycle services.

As a conclusion, Chapter 6 provides a review of the ongoing development programmes for small reactors without on-site refuelling in member states.

Annexes I–XXX present the contributions from Member States — structured design descriptions of water cooled, gas cooled, liquid metal cooled, and non-conventional (molten salt cooled, etc.) small reactors without on-site refuelling.

2. DEFINITIONS, TERMS AND FORMATS FOR SMR DESIGN DESCRIPTION

2.1. Definitions and terms

2.1.1. Small and medium sized reactors (SMRs)

According to the classification currently used by the IAEA, small reactors are the reactors with an equivalent electric power less than 300 MW, medium sized reactors are the reactors with an equivalent electric power between 300 and 700 MW [1, 14].

2.1.2. Small reactors without on-site refuelling

According to the definition given in [1], small reactors without on-site refuelling are the reactors designed for infrequent replacement of well-contained fuel cassette(s) in a manner that prohibits clandestine diversion of nuclear fuel material.

Small reactors without on-site refuelling should have the following essential features [1]:

- Capability to operate without refuelling for a reasonably long period consistent with the plant economics and energy security;
- Minimum inventory of fresh and spent fuel being stored at the site outside the reactor during its service life;
- Enhanced level of safety, consistent with the scale of global deployment of such reactors, through wider implementation of inherent and passive safety features and systems;
- Economic competitiveness for anticipated market conditions and applications;
- Difficult unauthorized access to fuel during the whole period of its presence at the site and during transportation, and design provisions to facilitate the implementation of safeguards;
- The capability to achieve higher manufacturing quality through factory mass production, design standardization and common basis for design certification.

Small reactors without on-site refuelling may have the following additional desirable features [1]:

- Factory fabrication and fuelling to facilitate delivery of a sealed core to the plant site;
- Capability to survive all postulated accident scenarios, including those caused by natural or human-induced external events, without requiring emergency response actions arising out of unacceptable radiological consequences in the public domain and without compromising the transportability of reactor back to the manufacturers;
- An overall reactor and fuel cycle enterprise that is highly unattractive for weapons purposes, e.g. offering limited overall amount of material, high degree of contamination providing noticeable radiation barriers, incorporating fuel forms that are difficult to reprocess and/or types of fuel that make it difficult to extract weapons-grade fissile material;
- A variety of applications, including generation or co-generation of electricity, production of heat, potable water, or hydrogen;
- A variety of options for siting, including those close to population centres, as well as remote and hardly accessible areas or dispersed islands, etc;
- Simplified operation procedures and robustness with respect to human errors;

- Minimum reliance on sophisticated local infrastructure;
- An overall reactor and fuel cycle enterprise that contributes to effective use of resources in a sustainable way.

2.1.3. Innovative design

IAEA-TECDOC-936, "Terms for Describing New, Advanced Nuclear Power Plants" [9], defines an *advanced design* as a "design of current interest for which improvement over its predecessors and/or existing designs is expected. Advanced designs consist of evolutionary designs and designs requiring substantial development efforts".

Evolutionary design is an advanced design that "achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining design proveness to minimize technological risks".

Innovative design is a design "that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice" and would, therefore, "require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented".

2.1.4. Safety related terms

Definitions from IAEA safety standards

The formats (outlines) for SMR design description, provided in Section 2.2, were developed keeping in mind the following consensus definitions given in the IAEA safety standard NS-R-1 [15]:

ACTIVE COMPONENT. A component whose functioning depends on an external input such as actuation, mechanical movement or supply of power.

PASSIVE COMPONENT. A component whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.

PLANT EQUIPMENT (see Fig. 2).

SAFETY SYSTEM. A system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents.

PROTECTION SYSTEM. System which monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

PLANT STATES (see Fig. 3).

NORMAL OPERATION. Operation within specified operational limits and conditions.

POSTULATED INITIATING EVENT. An event identified during design as capable of leading to anticipated operational occurrences or accident conditions.

ANTICIPATED OPERATIONAL OCCURRENCE. An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

ACCIDENT CONDITIONS. Deviations from normal operation more severe than anticipated operational occurrences, including design basis accidents and severe accidents.

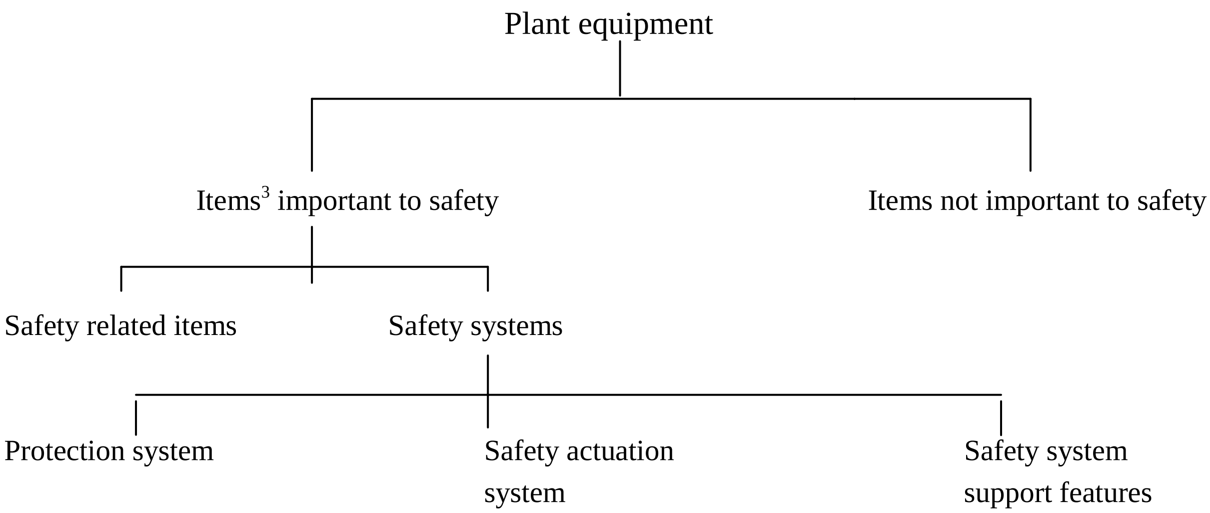
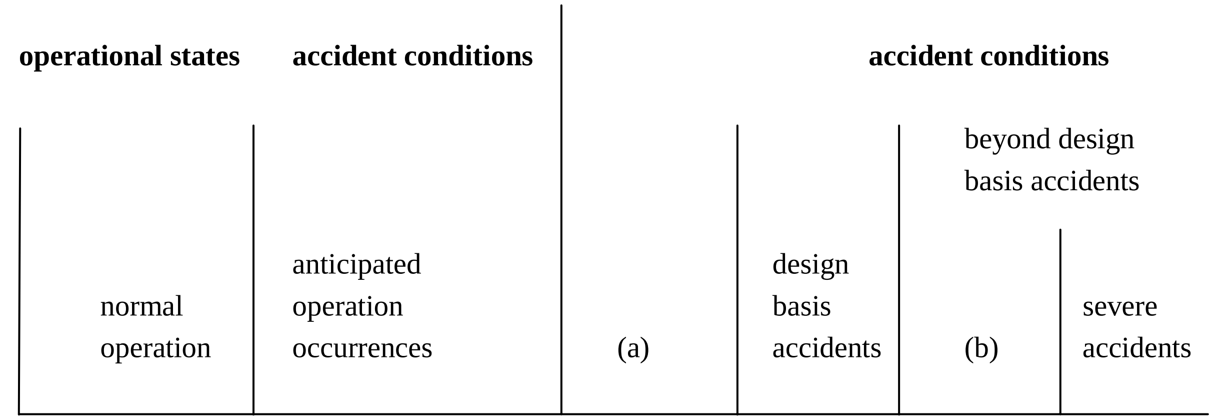


FIG. 2. Plant equipment [15].



(a) Accident conditions which are not explicitly considered design basis accidents but which they encompass.
(b) Beyond design basis accidents without significant core degradation.

FIG. 3. Plant states [15].

DESIGN BASIS ACCIDENT. Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

SEVERE ACCIDENTS. Accident conditions more severe than a design basis accident and involving significant core degradation.

ULTIMATE HEAT SINK. A medium to which the residual heat can always be transferred, even if all other means of removing the heat have been lost or are insufficient.

SINGLE FAILURE. A failure which results in the loss of capability of a component to perform its intended safety function(s), and any consequential failure(s) which result from it.

COMMON CAUSE FAILURE. Failure of two or more structures, systems or components due to a single specific event or cause.

SAFETY FUNCTION. A specific purpose that must be accomplished for safety.

---

[3] In this context, an 'item' is a structure, system or component [15].

Non-consensus definitions from IAEA TECDOCs

At the moment, the IAEA safety standards do not provide a complete set of definitions necessary for the description of safety features of NPPs with innovative reactors. In view of this, some missing definitions related to passive safety features could be taken from IAEA-TECDOC-626 [16]:

INHERENT SAFETY CHARACTERISTIC. Safety achieved by the elimination of a specified hazard by means of the choice of material and design concept.

PASSIVE COMPONENT. A component, which does not need any external input to operate.

PASSIVE SYSTEM. Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation.

GRACE PERIOD. The grace period is the period of time during which a safety function is ensured without the necessity of personnel action in the event of an incident/accident.

Recommendations from International Nuclear Safety Advisory Group (INSAG)

Although the IAEA safety standard NS-R-1 [15] provides a consensus definition of the defence in depth levels, the definitions suggested in INSAG-10 [17] may better suit for NPPs with innovative reactors. For the future reactors, reference [17] envisages the following trends of the different levels of defence in depth:

— Level 1, for the prevention of abnormal operation and failures is to be extended by considering in the basic design a larger set of operating conditions based on general operating experience and the results of safety studies. The aims would be to reduce the expected frequencies of initiating failures and to deal with all operating conditions, including full power, low power and all relevant shutdown conditions.

— Level 2, for the control of abnormal operation and the detection of failures, is to be reinforced (for example by more systematic use of limitation systems, independent from control systems), with feedback of operating experience, an improved human-machine interface and extended diagnostic systems. This covers instrumentation and control capabilities over the necessary ranges and the use of digital technology of proven reliability.

— Level 3, for the control of accidents within the design basis, is to consider a larger set of incident and accident conditions including, as appropriate, some conditions initiated by multiple failures, for which best estimate assumptions and data are used. Probabilistic studies and other analytical means will contribute to the definition of the incidents and accidents to be dealt with; special care needs to be given to reducing the likelihood of containment bypass sequences.

— Level 4, for the prevention of accident progression, is to consider systematically the wide range of preventive strategies for accident management and to include means to control accidents resulting in severe core damage. This will include suitable devices to protect the containment function such as the capability of the containment building to withstand hydrogen deflagration, or improved protection of the basemat for the prevention of melt-through.

—Level 5, for the mitigation of the radiological consequences of significant releases, could be reduced, owing to improvements at previous levels, and especially owing to reductions in source terms. Although less called upon, Level 5 is nonetheless to be maintained.”

Terms to be avoided

The designers were not requested to adjust safety related terminology of their projects accordingly when preparing the design descriptions for this report; they had rather followed the definitions accepted in their respective member states. However, in line with the recommendations of [9] and upon the approval from designers, terms such as “revolutionary design”, “passive, simplified and forgiving design”, “inherently safe design”, “deterministically safe design”, “catastrophe free design” etc. were edited out from design descriptions, except for the cases when they appear in the names of certain reactor concepts.

2.1.5. Proliferation resistance related terms

The terms and definitions used in the design description outline correspond to reference [18]:

- Proliferation resistance is that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices.
- The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures.
- Intrinsic proliferation resistance features are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures.
- Extrinsic proliferation resistance measures are those measures that result from States’ decisions and undertakings related to nuclear energy systems.

2.1.6. Design and regulatory status

The following simple classification, defined in [10], illustrates the design stages of an NPP project. Bullets indicate the items to be defined, determined, completed, achieved or established (whatever is applicable) within each stage.

(1) Conceptual design stage:

- Initial concept and plant layout;
- Single line flow diagrams for reactor coolant and other main processes (power production), and safety-related systems;
- Essential core configuration and composition;
- Specific safety features, including accident management;
- Overall sizes for main components, long delivery items and buildings;
- Main quantified parameters: power, flow rates, temperatures, pressures, sizes, etc.;
- Computer code development, verification and validation;
- Fuel cycle characteristics, if not conventional;
- Identification of required R&D: materials, components, systems, tests, etc.;
- Economic evaluation; and
- Design quality assurance (QA) programme;

(2) Basic (in some member states — preliminary) design stage:

- Essential R&D completed (except non critical items);
- Plant technical description;
- Engineering tools;
- Component conformity and principle feasibility tests;
- Design criteria manual "Engineering manual";
- System descriptions for the main reactor and auxiliary systems, with piping and instrumentation diagrams developed;
- Functional specifications for main components;
- Plant general layout: plans, evaluations, building sizes, floor loading, and embodiments;
- Basic safety studies and accident evaluation, part of Preliminary Safety Analysis Report (PSAR);
- First cost estimates;
- Marketing file;
- Preliminary schedule for construction;
- QA program for detailed design and procurement;

(3) Detailed design stage:

- Complete design of the plant, except very minor items. It can be unified (for example, for an envelope of site conditions) or site-specific;
- Large scale integral system tests;
- Equipment qualification testing;
- Design/engineering for systems and components
- Detailed specifications for procurement of all materials, components, systems, package units, construction/erection services, etc.;
- Preliminary safety analysis report;
- Detailed design reviews;
- Detailed construction planning;
- Final cost estimate;
- Final tender document;
- Construction and commissioning QA programmes.

The above mentioned classification is given as a reasonable example. The designers were not requested to adjust names of the design stages of their projects accordingly when preparing design descriptions for this report; they had rather followed the patterns established in their respective member states.

Regarding the regulatory status, different approaches for licensing in the individual member states make it difficult to establish milestones with precise meaning [10]. The following phases may represent a common approach to the licensing process:

- Licensing pre-application submitted;
- Preliminary licenseability assessment by regulator;
- Formal licensing application submitted;

- Review process by regulator;
- Permit(s) or licenses issued.

Regarding innovative reactors, an important observation is that R&D on key enabling technologies is often started before a completion of the conceptual design stage, to obtain the proofs of technological feasibility at as early stage as possible. Licensing pre-application at early design stages is rated useful by many designers of innovative reactors [1, 2] to establish good working relations with the regulator and secure more time for the resolution of the regulatory problems associated with a certain innovative design.

2.1.7. Enabling technologies

The enabling technology is a technology that needs to be developed and demonstrated to make a certain reactor concept viable [1]. Within this report, the term 'enabling technology' is used in a broad sense; for example, it could denote the technology of a particular system, structure or component as well as a combination of design approaches used to ensure inherent or passive safety features or high economic competitiveness of a certain design. Calculation technologies and data sets necessary for validation of nuclear power plant performance also fall under this definition.

2.2. Formats for reactor design description

The formats (outlines) used in the preparation of full and short design descriptions of small reactors without on-site refuelling for this report are enclosed as Appendices I and II, respectively.

REFERENCES TO CHAPTERS 1 AND 2

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Innovative Small and Medium Sized Reactors: Design Features, Safety Approaches and R&D Trends, Final report of a technical meeting held in Vienna, 7–11 June 2004, IAEA-TECDOC-1451, Vienna (2005).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Innovative Small and Medium Sized Reactor Designs 2005: Reactors with Conventional Refuelling Schemes, IAEA-TECDOC-1485, Vienna (2006).

[3] INTERNATIONAL PANEL ON CLIMATE CHANGE, Special report on emission scenarios, A Special Report of Working Group III, Cambridge University Press, Cambridge (2000).

[4] IAEA Nuclear Technology Review 2006, Report by the Director General, GC(50)/INF/3, 3 July 2006, http://www.iaea.org/About/Policy/GC/GC50/GC50InfDocuments/English/gc50inf-3_en.pdf.

[5] IAEA, Medium Term Strategy 2006–2011, http://www.iaea.org/About/mts2006_2011.pdf.

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Market Potential for Non-electric Applications of Nuclear Energy, IAEA Technical Reports Series No. 410, Vienna (2003).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidance for Preparing User Requirements Documents for Small and Medium Reactors and their Application, IAEA-TECDOC-1167, Vienna (2000).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs 2004, IAEA-TECDOC-1391, Vienna (2004).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Terms for Describing New, Advanced Nuclear Power Plants, IAEA-TECDOC-936, Vienna (1997).
[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Design and Development Status of Small and Medium Reactor Systems 1995, IAEA-TECDOC-881, Vienna (1995).
[11] INTERNATIONAL ENERGY AGENCY, OECD NUCLEAR ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Three Agency Study: Innovative Nuclear Reactor Development: Opportunities for International Co-operation, IEA Publications, 9 rue de la Federation, 75739 PARIS Cedex 15, Printed in France by Louis Jean (2002).
[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the assessment of innovative nuclear reactors and fuel cycles – Report of Phase 1B (first part) of the International Project on Innovative Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1434, Vienna (2004).
[13] GENERATION-IV INTERNATIONAL FORUM, A technology roadmap for Generation-IV nuclear energy systems, issued by the U.S. DOE Nuclear Energy Research Advisory Committee and the Generation-IV International Forum, December 2002, USA.
[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Introduction to Small and Medium Reactors in Developing Countries, IAEA-TECDOC-999, Vienna (1997).
[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, Safety Standards Series, No. NS-R-1, IAEA, Vienna (2000).
[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Related Terms for Advanced Nuclear Plants, IAEA-TECDOC-626, Vienna (1991).
[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Defence in Depth in Nuclear Safety, INSAG-10, Vienna (1996).
[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Proliferation Resistance Fundamentals for Future Nuclear Energy Systems, STR-332, Department of Safeguards, Vienna (2002).

3. SURVEY OF EMERGING ENERGY MARKET CHARACTERISTICS — A RATIONALE FOR SMALL REACTORS WITHOUT ON-SITE REFUELLING

3.1. Introduction

It is observed that there is a direct link between per capita energy consumption and economic development [1]. It is also expected that the growth of developing countries would take place faster in the coming decades. Nuclear energy may be required to fill a growing market share of world's primary energy supply in the future because its large resource base and its avoidance of greenhouse gas emissions are favourable features for sustainable development. Whereas in the past, nuclear deployments have been predominantly in developed countries, all projections forecast that the dominant energy capacity additions will occur in the currently developing economies. Small reactors without on-site refuelling can be designed to meet the needs of these emerging energy markets where industrial/technical infrastructure is generally poor. They could also provide additional assurances of energy security without exacerbating the risk of proliferation. This chapter presents an analysis of the emerging energy market characteristics and introduces a rationale for small reactors without on-site refuelling.

3.2. The energy supply challenge of sustainable economic development

3.2.1. The link from energy supply to economic development

A country's economic activity, Gross Domestic Product per capita (GDP/capita), is driven by energy input per capita in a cause and effect linkage. An increased GDP/capita leads to increased standard of living. Figure 1 plots GDP/capita versus energy use per capita for a number of countries. In developed countries, energy input of 4 to 9 toe/capita/annum[1] leads to a GDP/capita of 15 000 to 30 000 US$ per annum. A "quality of life" index versus energy use per capita is shown in Fig. 2; the index is seen to start saturating at around 4 toe/capita/year.

Use of energy and its resulting economic activity and standard of living are extremely non-homogenous among the world's countries and populations. Figure 3 shows that currently 20% of the world's population consumes 55% of the world's annual energy usage; 80% of the world's 6 billion people have access to less than half the world's energy. Over 4 billion of the world's people lack access to energy and are abjectly poor — living on less than 2 US$/day.

Poverty is not confined to rural populations living in a barter economy. Urban areas have become a magnet for the poor. Populations in over 35% of the more than a thousand urban concentrations in Asia, Africa, and South America are comprised of "bottom of the pyramid" consumers [2]. Rural-to-urban migrations are expected to continue, exacerbating the challenge; the World Energy Council predicts that 80% of the world's population will live in cities by 2050 [3].

The global economic development challenge for the 21st century is to improve the lives of the world's poor, a majority of whom will be living in developing countries and a majority of those in urban slums of ~15 000 people per hectare.

[1] *toe* = tons of oil equivalent primary energy input; 2.37 toe = 100 GJ.

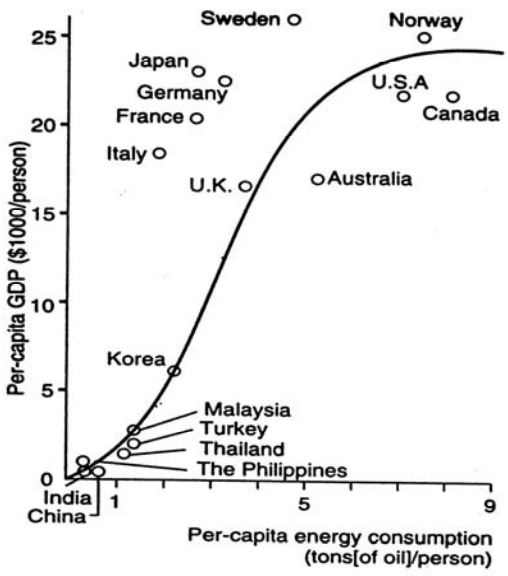


FIG. 1. Per capita GDP versus energy input per capita [4].

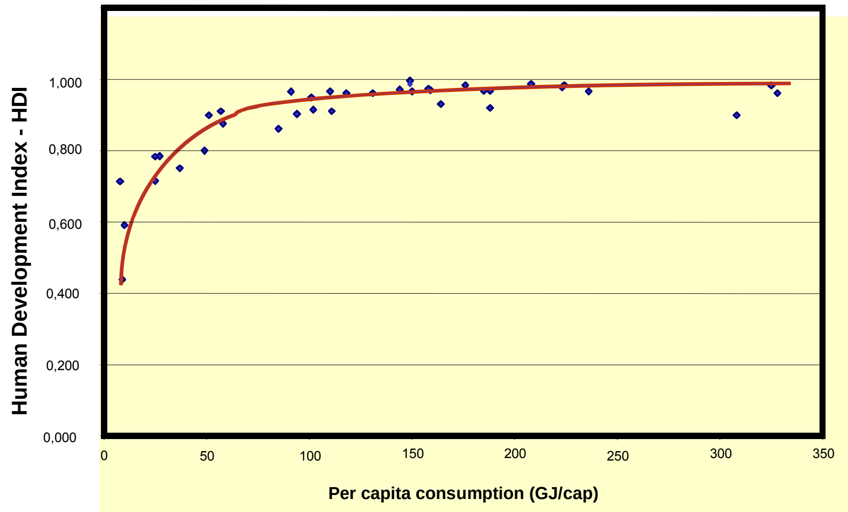


FIG. 2. Human development index (HDI) versus energy input per capita [5].

Economic development is the cure for poverty; energy input fuels economic development and institutional arrangements that favour investment enable it to happen; together they are the underlying drivers of economic development. A challenging goal for worldwide economic development could be to reach an energy input per capita of at least 4 toe/capita/year in all countries over the 21st century.

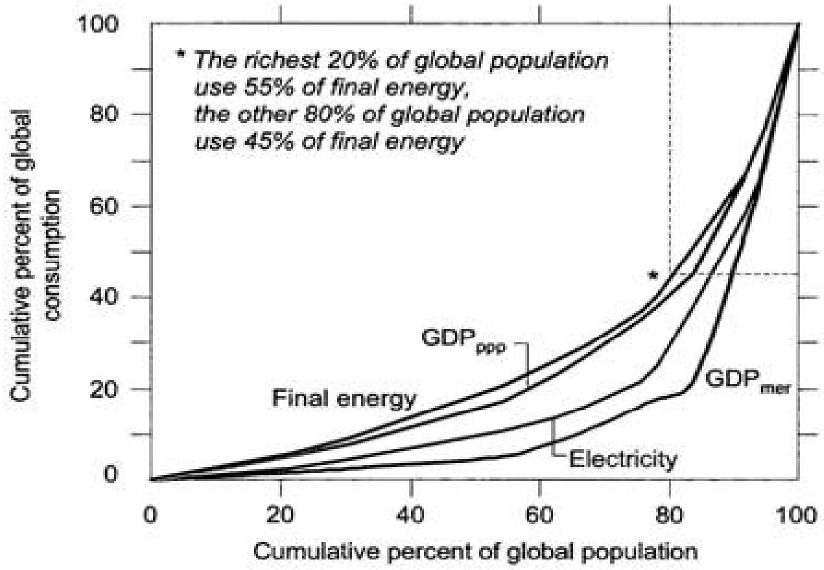


FIG. 3. Energy shares versus fraction of world population [6].

3.2.2. Projections of global primary energy demand growth

The 21st century will experience dramatic growth in energy demand as countries grow their economies and the quality of life for their populations. Because most developed countries have saturated in energy use per capita, whereas many developing countries are starting well below 4 toe/capita/year, the growth will likely be inhomogeneous — with faster growth taking place in developing than in developed countries.

A 21st century energy demand growth has been projected by numerous organizations, often in connection with the study of sustainable development or of global climate change. The demand projections are created by starting with projections of population growth and of economic development (annual growth rate of GDP) by region; they incorporate historical trends of energy use per capita versus GDP and energy intensity versus GDP to produce regional energy demand projections for a range of conceivable future scenarios. The regional outcomes are aggregated to produce global energy demand projections under a range of conceivable future conditions. Market shares of various energy resources (fossil, renewables, nuclear) are also projected under various assumptions concerning technological and institutional futures [6]. Although not predictions, these projections provide educated forecasts of what the future may hold under a range of postulated strategic approaches.

Most such projections, based as they are on hopeful projections of economic development, forecast massive growth in global demand for energy services over the next century. As an example, the Case B projections [6] show a 3 to 5 fold increase in world economic output by 2050 and a 10 to 15 fold increase by 2100, see Fig. 4 in section 3.3.1. When improvements in energy intensity, (GDP/toe energy input), are figured in, this corresponds with a 1.5 to 3 fold increase in primary energy use by 2050 and 2 to 5 fold increase by 2100.

3.2.3. The challenge of sustainable development — a potentially expanded role for nuclear power

A “business as usual approach” will rely on fossil fuels to support this growth. In fact, the Case B projections [6] mentioned above assumed that fossil will remain the dominant source of energy throughout the 21st century.

But increasingly in the past two decades, governmental attention has transcended simple economic development to seek sustainable economic development. According to [7], sustainable development is that "meeting today's needs without compromising the ability of future generations to meet their needs"; it rests on three pillars:

- Achieving mass flows of resources and wastes which are consistent with the ecosystem's ability to accommodate them;
- Ecological responsibility, and
- Social acceptability.

While fossil fuel has dominated energy supply for over 200 years, its continuing expansion faces a potential collision among four trends affecting the pillars of sustainable development:

- Continuing world population growth, primarily in developing countries where GDP/capita is currently low; coupled to
- Economic growth — especially in developing countries — thereby requiring increased energy input per capita there and, as a result, vastly increasing global energy usage overall.

The countervailing trends are:

- Increasing rates of consumption and competition for fossil energy resources, with shortfalls foreseen (except for coal) before mid century; and
- Increasing ecological assault from emissions attendant to fossil fuel use, both locally (soot, smog, Hg) and globally (CO_2, CH_4).

With the above mentioned factors taken into account, exclusive dependence on fossil based energy to fuel the 21st century global economic development in a sustainable fashion may turn to be not possible.

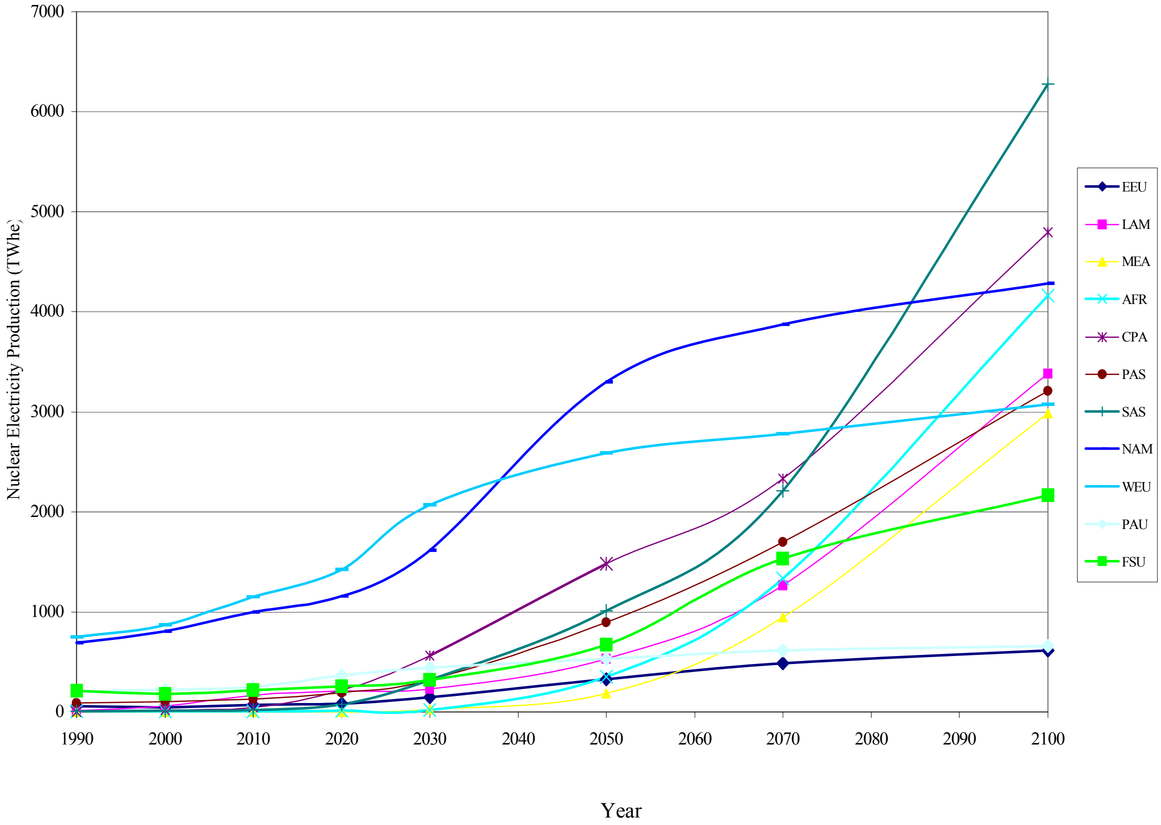
Nuclear's contribution to world primary energy supply is currently at 6%, even though it contributes nearly 20% of world electricity supply [1]. Nuclear energy may be required to fill a growing market share of world primary energy supply in the future because it's large resource base and its avoidance of greenhouse gas emissions are favourable features for sustainable development. At the same time, to increase the nuclear's share of primary energy supply, it must be tailored to meet the needs of the emerging market base. Thus, the characteristics and needs of the emerging markets need to be analyzed.

3.3. Characteristics of emerging markets for nuclear energy

3.3.1. Projected nuclear growth in developing countries

The Case B scenario from [6] shows that even for cases where fossil sources continue to dominate the energy market, and where nuclear is confined to electricity production only, potential exists for 2000 GW(e) nuclear by 2050 and nearly 6000 GW(e) by 2100, see Fig. 4. The range of nuclear deployments projected from alternate studies varies from 1200 GW(e) to as high as 5000 GW(e) in 2050 [8][2]. For perspective, starting at ~370 GW(e) nuclear capacity deployed currently [1], global nuclear deployments might grow by factors of three to ten over the next fifty years.

[2] IAEA Nuclear Technology Review 2006 gives a more moderate projection of 50 to 270 GW(e) increase by 2030, see Chapter 1.1.1.



EU = Central & Eastern Europe; LAM = Latin America; MEA = Middle East & North Africa; AFR = Sub-Saharan & Southern Africa; CPA = Centrally Planned Asia & China; PAS = Pacific OECD (Japan, Australia, New-Zealand); SAS = South-East Asia; NAM – North America; WEU = Western Europe; PAU = Other Pacific; FSU = Former Soviet Union

FIG. 4. Global energy perspectives by region [6].

Whereas until recently, nuclear deployments have been predominately in developed countries, (see Fig. 5), all projections forecast that by 2030 and thereafter the dominant energy capacity additions will occur in the currently developing economies due to two factors; greater population growth there than in developed countries and higher economic growth rate and energy use per capita there than in developed countries[3].

Figure 4 displays nuclear energy deployments by world regions from the Case B scenario [6]. The slopes of the curves represent new plant deployment rate per annum. It is seen that while the emplaced capacity of developing regions doesn't reach that of the developed regions until past mid century, their deployment rates (slope) surpass the developed regions as early as the 2030s.

[3] Figure 1 indicates that once the developed countries reached an annual energy use per capita in the range of 6–9 toe/capita, energy use per capita tends to saturate, and further growth depends mostly on population increases. But population growth rates also tend to saturate with increasing GDP/capita, and in many developed countries the birth rate has reduced to only a self-sustaining level or less.

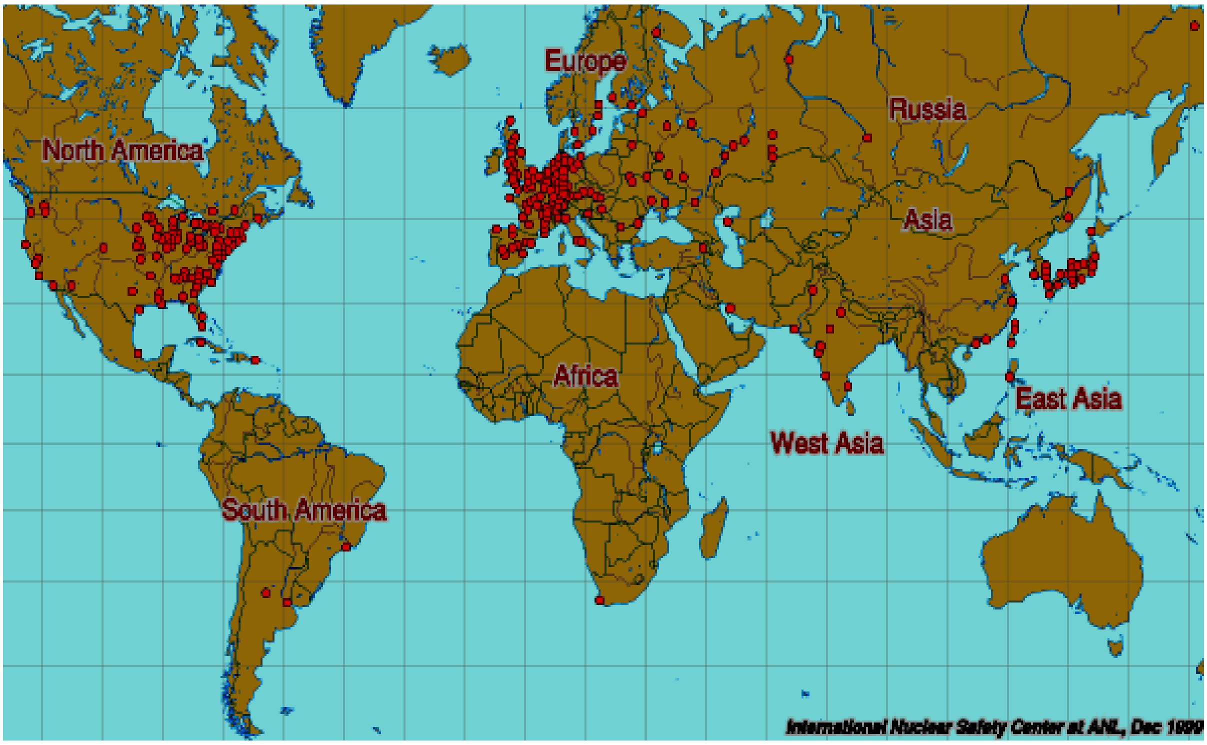


FIG. 5. Locations of nuclear power plants in 1999.

The Case B projections [6] forecast that:

- Between 2000 and 2025, roughly 460 GW(e) of new capacity might be added in North America and Europe, while about 475 might be added in the rest of the world; but
- Between 2030 and 2050, North America plus Europe add about 270 GW(e), whereas the rest of the world adds about 390 GW(e).

Post 2050, Fig. 4 shows the levelling off in North America and Europe to replacements primarily, whereas in the rest of the world the deployment rate accelerates even faster.

With several other deployment projections available (see, for example, [9]), and independent of the anticipated timeframes and scales, the conclusion is always the same — for nuclear power to play a significant role in world energy growth, it must be targeted to meet the needs of developing countries.

3.3.2. Demand for a broadened scope of energy products

In the 1950s, many and diverse applications for nuclear energy were envisioned and explored. However, over the ensuing decades only two applications came into widespread industrial use — electricity production and naval ship propulsion. Fossil fuel was abundant, cheap, and convenient, and the use of nuclear fission for diverse energy applications was simply not cost competitive with fossil-fuelled options.

Primary energy (in developed countries) is utilized in three roughly equal fractions [3]:

- A third is used to generate electricity;
- A third is used in the transportation sector;
- A third is used for domestic and industrial heating.

Therefore, for nuclear to significantly offset fossil use, it must enter the non-electric segments of the primary energy market. Off-peak electricity use, direct production of process heat and cogeneration bottoming cycles provide one avenue. Hydrogen production might provide another.

Potable water production

Perhaps the most prominent of the near-term non-electric applications is nuclear desalination. Because of population growth, surface water resources are increasingly stressed in many parts of the world, developed and developing regions alike. Water stress is counter to sustainable development; it engenders disease; diverts natural flows, endangering flora and fauna of rivers, lakes, wetlands, deltas and oceans; and it incites regional conflicts over water rights.

In the developing world, more than one billion people currently lack access to safe drinking water; nearly two and a half billion lack access to adequate sanitation services [10]. This would only get worse as populations grow. Water stress is severe in the developed world as well. As an example, Fig. 6 shows the 100-year history of Colorado River discharge into the Gulf of Cortez (U.S.A.). Human use has dried it up; meanwhile population density in the US southwest continues to grow. The numbers of water disputes are escalating in developed and developing countries alike [11].

As shown in Fig. 7, an important segment of population centres worldwide lies within 100 km of a seacoast, opening a vast new market for freshwater generated from desalination (fully 60% of the US population resides within 100 miles of an ocean coastline) [11]. In light of these trends, many opportunities in both developed and developing countries are foreseen for supply of potable water generated using nuclear process heat or off-peak electricity.

Low temperature process heat from cogeneration plants

When producing electricity, more than half of the heat is rejected at low temperature. If power plants are located near population centres to support a local electrical grid, opportunities open up for cogeneration bottoming cycles to supply industrial applications requiring low temperature (~100°C) heat.

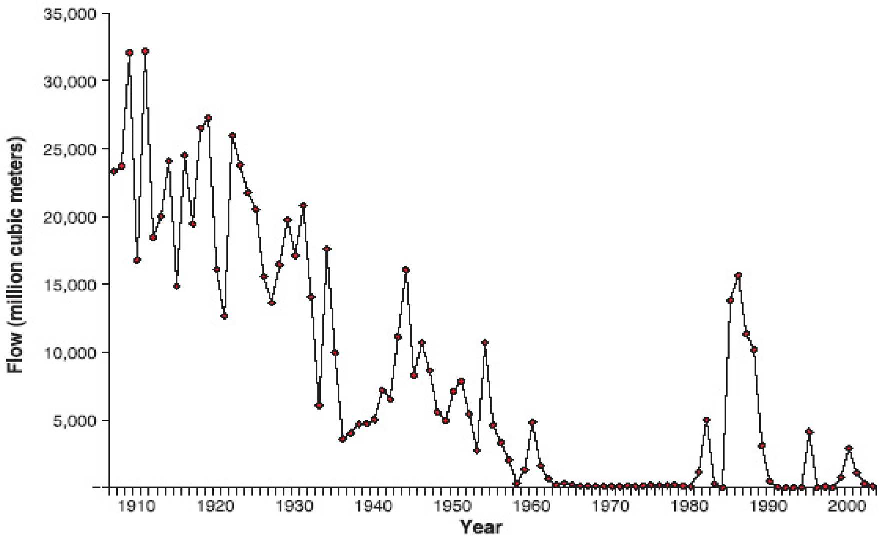
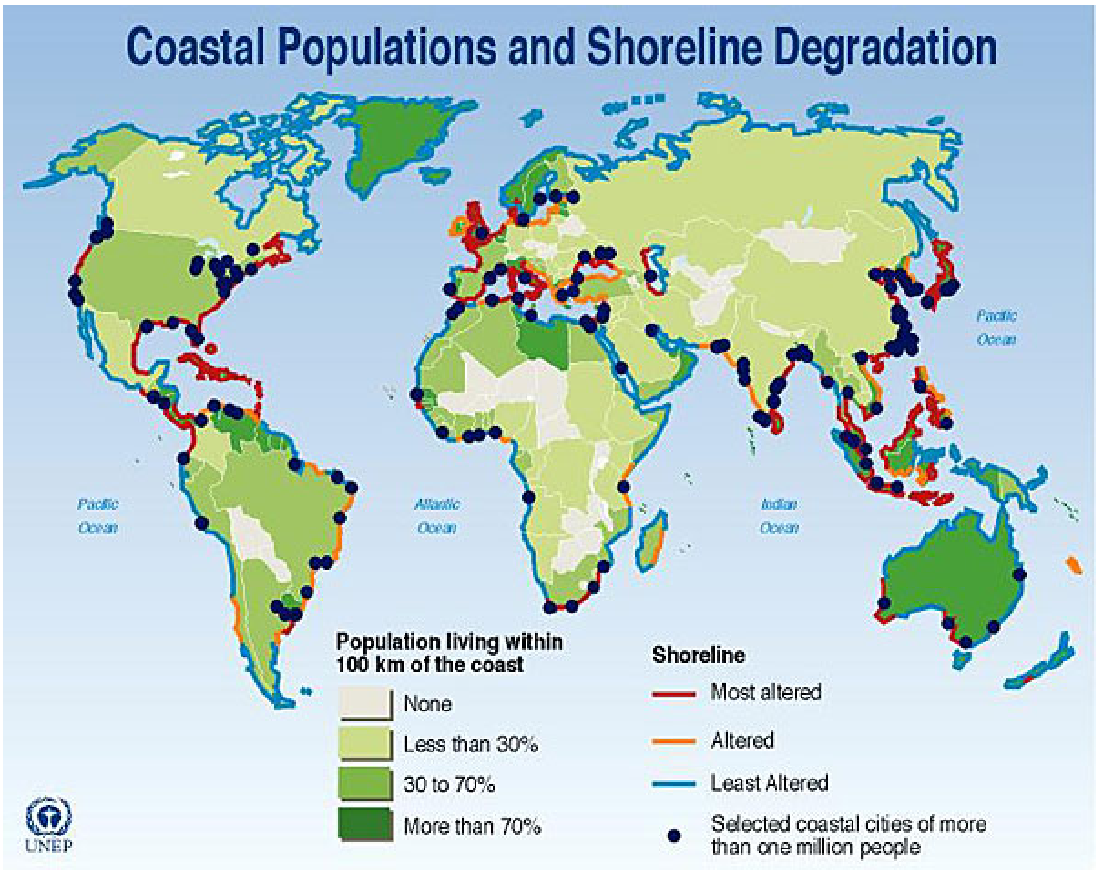


FIG. 6. Colorado River discharge by year [11].



Source: Burke et al., World Resources Institute, Washington DC, 2001; Paul Harrison and Fred Pearce, *AAAS Atlas of Population and Environment 2001*, American Association for the Advancement of Science, University of California Press, Berkeley.

FIG. 7. Coastal populations [11].

Likewise, when electricity consumption varies daily, monthly, or seasonally, off-peak electricity produced by a nuclear power plant could be used to power certain industrial applications.

Desalination missions are prominent as discussed above; they could be accomplished using either low temperature heat or off-peak electricity, depending on which option is more economical for a given case. District heating is an important possibility for low temperature process heat, specifically, for big cities as well as small off-grid villages in northern regions. Low temperature process heat applications to dry industrial and agricultural products are numerous, for example, the paper industry is a major user of process heat to boil water off the pulp.

High temperature process heat applications and hydrogen production

An emerging 21st century need for process heat conversion of water or hydrocarbon feedstocks to hydrogen is foreseen. Hydrogen is expected increasingly to upgrade and eventually displace liquid and gaseous hydrocarbon fuels in the decades ahead and to become, with electricity, one of the dominant energy currencies [12]. Hydrogen and electricity are convenient, clean and versatile energy carriers. They meet complementary needs; first they are convertible, each into the other; second, hydrogen is storable while electricity is not; third, electricity can be used to transmit and store information while hydrogen cannot.

Currently, fossil-derived processes dominate the hydrogen manufacture sector. But should the hydrogen economy develop, its favourable ecological features (no carbon release at point of use) will have little import unless a carbon-free hydrogen manufacture process is employed.

In the future, high temperature process heat for hydrogen production from water feedstock may become a major market for nuclear energy. Nuclear production of hydrogen was studied intensely during the 1970s to mid 1980s [13]. Interest waned once the effects of the 1970s oil shocks damped out. However, in light of growing concern regarding global climate change attendant to escalating fossil combustion, active research and development (R&D) programmes have recently been re-instituted worldwide.

Should the hydrogen economy come into being, the market potential for nuclear hydrogen production is larger by far than the market for nuclear generated electricity. By using nuclear heat for hydrogen manufacture, nuclear energy could enter the entire primary energy marketplace currently served by fossil fuel.

3.4. Re-engineering nuclear energy to serve emerging energy markets — a rationale for small reactors without on-site refuelling

The previous section has shown that emerging markets for energy deployments lie in two areas not previously served by nuclear energy — electricity to fuel the economic development of developing countries and non-electric primary energy products in both developed and developing countries.

Nuclear energy's vast resource base and carbon-free ecological features favour its gaining market share vis-à-vis fossil supplies to further the goals of sustainable global economic development. Such a market penetration of nuclear vis-à-vis fossil is not unprecedented. Starting from the 1970s, France and Japan, the countries that lack indigenous fossil resources, made major structural changes in their energy supply infrastructure, converting a significant fraction of electricity production to nuclear power.

The French and Japanese restructuring required more than deployment of nuclear power plants themselves; deployment of an entire indigenous fuel cycle infrastructure of front to back-end fuel cycle facilities was required as well.

To a greater or lesser degree many other developed countries made a similar shift from fossil to nuclear production of electricity during the past three decades. As a consequence of this first wave of market penetration, nuclear now supplies about 16% of the world's electricity [1], centred primarily in developed countries (see Fig. 4.).

These nuclear deployments in developed countries were made as incremental additions to already-existing, fossil based electric power systems:

- With extensive interconnected grids already in existence;
- With regulated utility markets wherein (owing to the regulatory compact) low risk premium on financing rates prevailed for new deployments;
- With institutional arrangements emplaced to govern the nuclear enterprise; and
- With skilled work force personnel available.

These conditions favoured an economy of scale approach to nuclear deployment; while first generation plants were ~200–400 MW(e), the industry rapidly advanced to 1000–1600 MW(e) plant size. With capital cost assignable to the overall utility rate base, large nuclear plant installations requiring 6 to 9 years of site construction employing thousands of workers were cost effective.

Additionally, front end (conversion, enrichment, and fuel fabrication) fuel cycle support know-how and infrastructure in developed countries benefited from technology that had been developed for military applications; it was deployed indigenously or was supplied by weapons states to closely politically-allied states.

In the future, a nuclear based strategy to achieve energy security and to fuel economic growth could offer similar advantages to an increasing number of the world's developing countries; with three things being different, however. First, an already existing fossil fuel based infrastructure is often not present, offering an opportunity to leap-frog to new energy supply architecture. Second, not all countries will wish to undertake the cost of (or have the capability to) deploy an entire indigenous fuel cycle if an acceptable alternative is provided for assuring their energy security.

Finally, emerging energy market conditions differ dramatically from the regulated electricity markets prevailing in developed countries during the previous half century. When an extensive pre-existing grid is absent, when financing is tight, and when energy use per capita is initially low but growth rate is high, the economy of scale power plant configuration supported by indigenous fuel cycle service facilities becomes ill-suited to the customer needs. For many potential new customers, low initial buy-in cost and outsourced fuel cycle services could offer a much better way to meet their energy supply needs.

3.4.1. Possible energy architecture with small reactors without on-site refuelling

In his plenary speech, "On the Nature of Nuclear Power and Its Future" at the Global'93 Conference [14], Wolf Häfele observed that the configuration of nuclear energy deployed in developed countries "was put into an existing technical and institutional (energy) infrastructure without much changing this infrastructure, still characterized by the use of oil in particular, but also of coal and gas", e.g., economy of scale nuclear power plants joined economy of scale fossil plants in driving a large pre-existing interconnected grid; financing arrangements benefited from adding capital costs into an existing rate structure and amortizing the new plant investment over long payback periods; and fuel cycle infrastructure benefited from military investments in technology development and/or facility emplacement. Nuclear reactors filled the role of coal furnaces raising steam for Rankine cycle turbines, and nuclear fuel, at less than 1% efficiency of use, supplemented fossil fuel supply. He concluded his speech by saying: "One must be prepared for evolution or even revolution when real nuclear power brings the factor of a million between nuclear and chemical bond energies to the surface; one cannot treat nuclear power like chemical power, uranium like yellow coal."

As it was already mentioned, the current configuration of nuclear energy is performing well in developed countries, and new deployments are ongoing or imminent. However, for a large segment of the new customer categories in developing countries, and new customers seeking to enter broader energy service sectors, the historical nuclear architecture comprised of economy of scale plants and an indigenous fuel cycle nuclear architecture might not fit well.

Small reactors without on-site refuelling are an approach to provide a new architecture for nuclear energy, specifically designed to meet the needs of these emerging markets.

This new configuration would have two defining features:

- Reactors of small power rating (a few MW(e) up to 300 MW(e)), which are delivered to the customer as a standardized, pre-licensed, turnkey plant requiring only a short site assembly period; and
- Very long refuelling interval (7 to 30 years) supported by outsourced fuel cycle and waste management services offered from centralized (economy of scale) fuel cycle support facilities operated under international safeguards oversight.

The re-engineering of nuclear energy architecture to meet needs of the emerging market could then involve technical, business, and institutional innovations, such as:

- Power plant design approaches and business strategies, which overcome the loss of economies of scale;
- Fuel cycle approaches and institutional strategies to mitigate the energy security/non-proliferation dilemma.

If this new architecture could be successfully deployed in the emerging energy markets, it holds a potential to ensure fuel-sustainable economic development with its economically harvestable resource base being good for a millennium of world energy supply. It also has a capacity to mitigate the energy security/non-proliferation dilemma by exploiting an incredible energy density of nuclear fuel to facilitate deployment of long refuelling interval reactors supported by a limited number of centralized fuel cycle centres. Finally, it has a potential capacity to eventually open itself to the entire primary energy market by manufacturing hydrogen from water.

3.4.2. Energy security versus non-proliferation dilemma

The essence of this dilemma is how to configure the nuclear energy enterprise in a way that provides each country with energy security while simultaneously providing the international community with non-proliferation assurances. The nuclear energy configuration with small reactors without on-site refuelling holds the promise of a practical approach to balancing energy security and non-proliferation assurances.

Like other nuclear reactors, small reactors without on-site refuelling exploit the incredible energy density of nuclear fuel. Whereas a kilogram of a chemical fuel can carry:

$$\text{1 kg of } H_2 \approx \text{1 gallon of gasoline} = 1.39\ 10^{-3}\ \text{(MW(th)-day) of heat, (LHV}^{4}\text{)} \qquad (1)$$

a kilogram of nuclear fuel that achieves 5 to 10 atom percent burn-up can carry a factor of ~35 000 to 75 000 greater:

$$\text{1 kg of nuclear fuel} \approx \text{50 to 100 (MW(th)-day) of heat.} \qquad (2)$$

Different from other nuclear reactors, small reactors without on-site refuelling attempt to use nuclear fuel as a long-life heat battery. Properly designed, a core loading of nuclear fuel in a small reactor may supply many years of energy without refuelling. When installed on a sovereign territory, such reactor could add a remarkable degree of energy security.

This key technical feature may also allow to change the world energy supply architecture from one optimized for fossil to the one optimized for nuclear. The new architecture could extensively distribute nuclear power plants but at the same time centralize fuel cycle support services to a small number of locations for conducting bulk fissile handling operations in the economy-of-scale facilities and under appropriate safeguards oversight.

Exploiting this new architecture for nuclear energy requires not only the technical innovation but institutional innovation as well; see the discussion in Chapter 5.6.

3.5. Survey of customer needs

This section takes a closer look at the targeted market segments for small reactors without on-site refuelling. Four main categories of customers are identified:

[4] LHV is lower heating value.

- Villages and towns in off-grid locations;
- Industrial installations in off-grid locations;
- Cities in developing countries;
- In a more distant future, perhaps, merchant plants[5] for non-electric energy services.

While these different categories each have unique needs, they share many common needs in terms of plant power rating, cogeneration opportunities, and requests for supporting fuel cycle services.

In the following sections, different categories of potential customers are surveyed and their needs are identified. Then, in the following sections, the common needs are shown to drive a number of the common design strategies shared by all small reactors without on-site refuelling.

3.5.1. Villages and towns in off-grid locations

Much of the world's land mass supports sparse populations and is unsupportable by an electric grid. The northern extremes of the North American and Eurasian continents are sparsely populated; the villages and towns are widely separated and are un-serviced by road, rail, or electrical grid. Examples include northern Alaska, Canada, and Siberia. For example, whereas only 3% of Europe's land mass is wilderness, fully 80% of northern Canada's land is wilderness having a population density less than $1/km^2$. Two-thirds of Russia's territory are off-grid and can be expected to remain so for decades if not forever. Such areas are often populated primarily by indigenous people; often they are in only the third generation after transition from a migratory lifestyle to village life. The villages are often small (≤1000 people). None-the-less, population centres, such as villages and small towns, and industrial sites in these off-grid areas require energy, both as electricity and as district heating.

In most of these vast regions, electricity and district heating supply currently come from diesel-generator sets, and often the logistics of diesel fuel delivery are extremely challenging owing not only to lack of roads but especially due to severe weather conditions over more than half the year. Even river access in unavailable during long months of winter freeze. Of the 620 individual generating plants in Alaska, about half employ diesel generators providing less than 1 MW(e) each. Many plants are in the several tens of MW(e); the largest is 335 MW(e) [15].

Island countries face a similar challenge for electricity delivery to widely dispersed population centres located on scattered islands separated by miles of ocean. For the island state of Hawaii with six separate power systems, the majority of generating units are less than 20 MW(e); the largest plant is 582 MW(e) [15]. Many island villages require more than electricity — desalinated water often comprises an additional necessary energy service. Indonesia, a country of 13 700 islands, is perhaps the most dramatic example [16].

The potential market for support of off-grid villages and towns is not confined to arctic and island regions. The government of India has identified 80 000 villages that are likely never to be connected to the grid [17]. The vast reaches of Brazil and Argentina contain hinterlands of low population density where grid emplacement is not cost effective.

[5] Merchant generation companies who operate outside the regulatory framework of regulated utilities and sell their product on a competitive market (i.e., they receive no guarantee of profitability in exchange for a guarantee of providing services to consumers).

Off-grid villages and towns require a standalone energy infrastructure, which includes electricity and perhaps desalination and/or district heating. Small power plant staffing levels are preferred and low staffing skill level may be desirable in some cases.

For nuclear to enter this market, the cost of energy must be competitive with the other options available locally, and the features must better meet the customer's needs than does the competitive option. Currently, diesel generators are the dominant supply sources for customers of this category. Capital cost of diesel generators is around US$ 1000/kW(e), but the difficulties attendant to fuel supply cause busbar energy cost to significantly exceed the rates experienced on well-developed urban grids.

For example, in northern Canada, busbar costs of 9–13 US$ cent/kW(e)-hour are typical; across Alaska the rates vary between 9.3 and 45 US$ cent/kW(e)-hour [15]. These exceed typical costs in the U.S. contiguous forty-eight states by factors of three to ten.

Figure 8 is a generic plot of electrical power requirements versus population, as a function of total primary energy use per capita[6]; this is a rough indicator only.

Figure 8 indicates that villages of 1000 require a plant in the range of 2 to 5 MW(e); towns of 50 000 require ~35 to 40 MW(e). Cogeneration missions for district heating, desalination or low temperature process heat are often required.

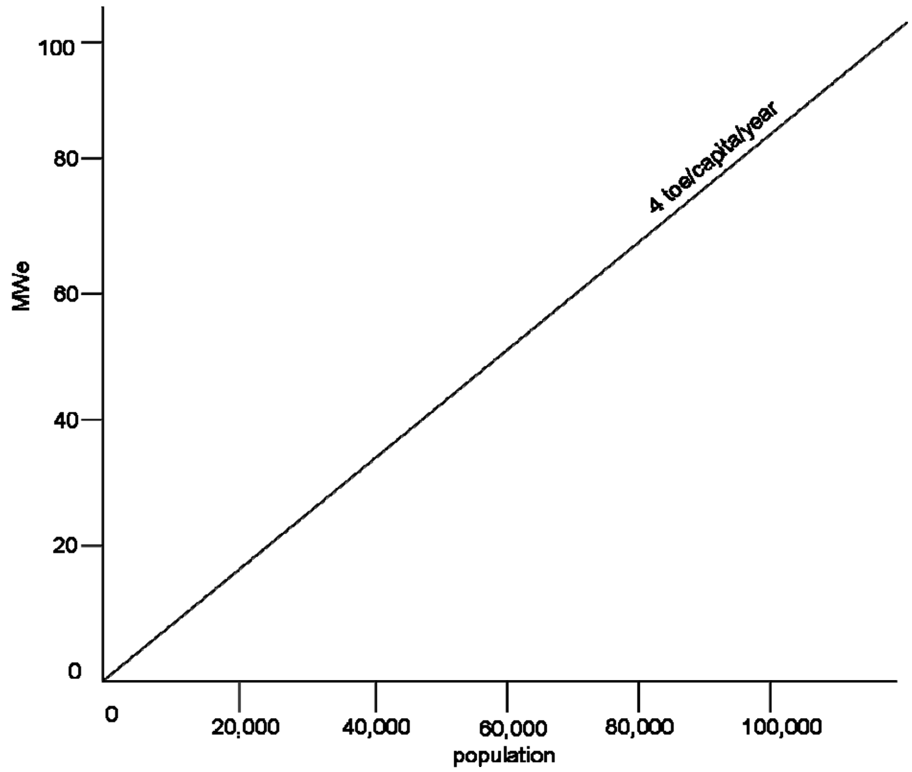


FIG. 8. Plant rating versus population.

[6] *toe* means "tons of oil equivalent"; the unit of primary per capita energy consumption was taken as 4 toe/year = 5 kW(th) year/year, corresponding to average energy consumption in Europe. In generating this plot, it was assumed that 1/3 of total primary energy use goes to electricity production and that primary energy is converted to electricity at a conversion efficiency of 0.4.

In this application, a small nuclear plant will be the sole source of power (and heating) to a local grid. It must be exceptionally safe to enable siting near the town, and it will be required to load follow the daily and seasonal variations in demand. Extremely high reliability and availability are essential. Load levelling energy storage products may be applicable in some cases. District heating, district cooling, desalination, hydrogen production by electrolysis, etc. could provide the town with additional energy products during periods of off-peak electricity demand. Long refuelling interval and the arrangements for replacement power during refuelling operations are essential.

Such mission could be best served by plants that are autonomous or near autonomous; which require only a small staffing level; and which are monitored at an off-site facility staffed to provide rapid-response specialized maintenance support to many such sites.

As indicated by the design descriptions in Annexes I through XXX, small reactors without on-site refuelling incorporate certain design features to meet the needs of this customer category. The designs attempt to minimize operating staff size and skill level and also incorporate many inherent safety features to achieve an exceptional level of safety for siting near population centres.

3.5.2. Energy intensive industrial sites in off-grid locations

Harvesting of natural resources is among the first steps for attracting foreign investment and initiating economic development in many developing countries and/or in sparsely settled regions. The economic activity of a majority of the remote villages discussed above is tied to harvesting of natural resources, such as mining, drilling, logging, fishing, etc. Along with permanent villages, dedicated work camps can be established temporarily to staff those harvesting activities.

Because of their remoteness and constrained transportation infrastructure, economic competitiveness of the harvested resource may require energy not only to harvest but also to add value to the raw product prior to shipping. For example, mines invest energy to mill the ore prior to shipping; fisheries use energy to process and pack the catch prior to shipping; loggers employ energy to produce paper, etc.

A market niche for nuclear power plants with small reactors without on-site refuelling could also be that of off-grid industrial applications, supporting the energy intensive processes which harvest and add value to natural resources. Two needs would arise in respect of this, which are supporting the work camp population and supplying power for the industrial operation.

First, for supporting the work camp population, the requirement is for only a few MW(e) of power and often may include cogeneration of district heat or seawater desalination (see Fig. 8).

Industrial needs are in the same range or somewhat larger. A survey of power requirements for mining operations in the Yukon Province of Canada (see Table 1) shows individual plants in support of the mining/milling industrial operations should be sized from 3 to 10 MW(e). Alternately, the electric power demand for milling can be as much as several hundred MW(e), and in those cases, the industrial demand vastly exceeds population support needs.

TABLE 1. POWER NEEDS FOR MINES IN THE CANADIAN YUKON [18]

MINE	DISTANCE FROM GRID (KM)	MINE LIFETIME (YEARS)	POWER REQUIRED, MW(e)	EMPLOYMENT (STAFF)
Operating:				
— Brewery Creek	133	8	3.0	100–220
— Mt. Nansen	60	2	6.2	65
Potential:				
— Cemerics Copper	45	8	7.2	90–136
— Casino	129	12	38	500
— Division Mt	25	15	2–4	100–200
— Dubin Gulch	25	10	2–5.3	179–205
— Grizzly & Grum	–	12–14	12 (on grid)	250–300
— Grum (open pt)	–	6	22 (on grid)	450
— Ketza River	50	2	3	75–100
— Kudz Ze Kayah	230	11	8.8	200
— Minto	88	12	2.5	76
— Skukum Creek	40	4	3	80–100
— SaDena Hes	58	4	6.2	81
— Tulsequah Chief	64	9	9	200
— Keno Hill	–	3	2–3.5 (on grid)	160
— Wellgren	200	12	35 mines + 261 smelter	400–500

The size, safety, and fuel cycle support requirements on plants for these off-grid industrial applications are similar to those for support of off-grid villages and towns; but several unique additional requirements apply. A striking feature of the mining application is the potentially short duration of deployment (2 to 15 years, as shown in Table 1). Because of the variability and inconsistency of the mining business, a value of 15 years is often used for gauging the financial viability of a proposed mine (the ore deposit may not last or the market price may fall, causing the mine to close). However, once the project is started, it may last for decades. This uncertainty in deployment duration sets an extreme requirement on transportability of the small power source, often over difficult terrain in extreme weather conditions.

Another unique feature of the mining application is the duty cycle, i.e., surge in power demand during the power shovel “bite” followed by a reduced demand during the turn and dump phase, accompanied by large minute-by-minute swings in demand, repeated continuously during the day, and followed by a massive scale-back to hotel load when mining operations stop such as to repair equipment (if milling is conducted on-site, the load is steady). Capability for rapid load following or load levelling via an energy storage arrangement may be a unique need for the off-grid industrial customer category.

3.5.3. *Cities in developing countries*

The truly massive future growth in energy demand would be for support of cities throughout the developing world; that is where energy infrastructure deployments could dominate throughout the 21st century.

The projection for massive energy demand growth in cities of the developing world can be understood as the product of population growth, rural-to-urban demographic migrations, and economic development.

Population growth in industrialized countries has essentially stopped; and world population growth has shifted almost entirely to the countries of Africa, Asia, and Latin America. Currently, of the 83 million people added to global population each year by the difference between births and deaths, only 1 million are in the industrialized countries. The developing world's population is projected to increase by 2.9 billion by 2050, compared with only 49 million in the more developed countries [19].

Demographic migrations from rural to urban setting are causing urban populations in developing countries to grow much faster than the rate for the country overall [2]. By 2015, there would be more than 903 cities in Asia; 225 cities in Africa, and 25 cities in Latin America. More than 368 of these cities will have more than 1 million people each. At least 23 cities will have more than 10 million residents. Collectively, these cities would account for about 1.5 to 2.0 billion people.

To foster economic development in these rapidly growing cities, "energy conservation" may not be an answer. Even though energy intensity (energy input/GDP) improves as a result of efficiency improvements and from shifts in economic activity as economies mature, such improvements can achieve only factors of two or three in intensity. This can play a significant role in already-developed countries where population growth and economic growth are both relatively low, but these "energy conservation" gains are apparently insufficient to compensate impending developing country population growth by factors of two multiplied by energy input/capita increases by a factor of ten. Therefore, conservation alone will be inadequate to cope with this demand growth, and increased energy input/capita and new energy deployments would then be needed.

The choices are fossil, renewables, and nuclear. Fossil is likely to remain dominant for decades, but in light of the trends of fossil resources and environmental emission, sustainable economic development might favour its displacement with non-fossil based energy supply if viable alternatives are available. The remaining options would then be renewables and nuclear.

When comparing renewables and nuclear energy infrastructure emplacements for rapidly growing cities, two characteristics are especially relevant: (i) energy density and (ii) energy payback period. A city would require high energy density and a fast-growing city would also require a short energy payback period.

The density of energy demand in cities is high because population density is high. It is higher still the higher is the energy use per capita. The peak power requirement in Manhattan (New York, USA) is ~1.5 kW/m^2; the corresponding annual average requirement is 1750 kW-hour/m^2 year. This exceeds by ten times the mean direct solar flow of 0.15 kW/m^2.

Figure 9 compares the societal power use per unit area footprint of Vienna to harvestable energy fluxes per unit area for several renewables [20]. As compared to demand per unit area, a factor of ten or more shortfall of harvestable renewable supply per unit area is observed.

Alternately, a 1000 MW(e) light water reactor power plant site occupies 500 to 1000 acres (about 2 to 4 km^2). This works out to a power per unit area of 750 to 1500 W/m^2 — two orders of magnitude larger that the energy demand per unit area typical of demand in a city (see Fig. 9). In other words, energy supply infrastructures deployed in support of urban centres should be high energy density sources, and nuclear has a benefit over renewables on this count.

The "harvest factor" of an energy asset is a measure of its energy payback period, i.e., the time it takes for the asset to deliver the amount of energy that it took to manufacture and emplace it in the first place. Table 2 shows that the harvest factor of renewables is highly unfavourable when growth rate is high, because the energy required to manufacture and emplace the next round of new assets to meet growing demand will divert a significant fraction of the energy from the first round that could otherwise have been directed to meeting society's energy needs. The reinvestment of energy for future deployments drags down the effectiveness of the deployment itself and forces the overall deployment rate up higher still. In fact, for growth rates having doubling time shorter than the harvest factor, the emplacement of assets can't keep up at all.

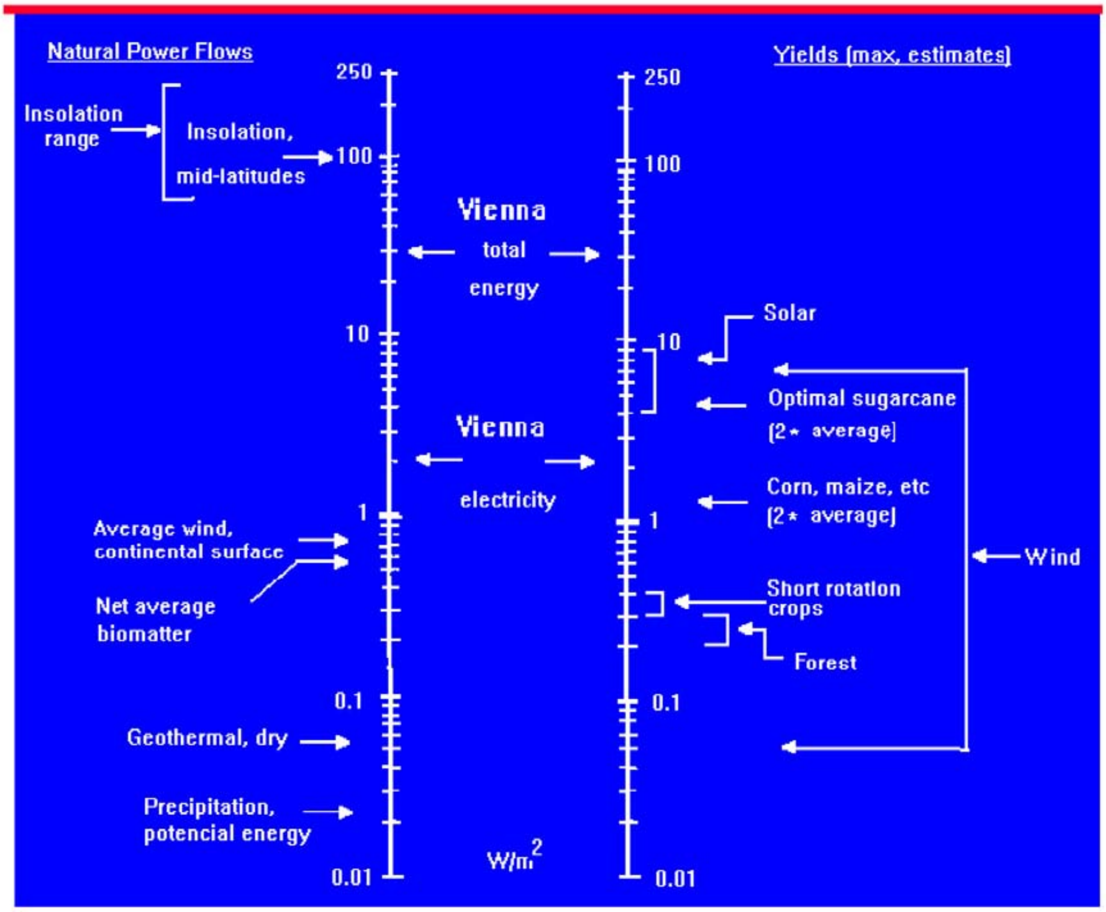


FIG. 9. Energy supply densities versus energy demand density [20].

Table 2 also shows that the harvest factor for nuclear asset emplacements is highly favourable in rapid growth situations; the energy payback period is reasonably short and emplaced reactors can be rapidly directed to meeting society's energy service needs. Again, nuclear option has advantages over renewables for rapidly growing cities. However, the features of nuclear deployments must be tailored to the customer's situation. Local grids can be small as city development starts. Economy of scale deployments may be inappropriate to the initially small needs. As shown in Fig. 8, for primary energy consumption at a rate of 4 toe/capita/year (about the usage in Europe), a town of 20 000 requires a plant of about 15 MW(e) rating; a city of a hundred thousand requires a plant of ~70 MW(e).

Initiating economic development of a small city may benefit from small-sized power plants. As one characteristic illustration of the market for downsized power plants in developing countries, Fig. 10 shows the situation for Mexico where out of an installed capacity of 42.3 GW(e), including fossil, hydro, nuclear, and geothermal, fully 85% of the plants are sized at less than 250 MW(e) [22].

The financial conditions faced by many developing cities may favour small initial capital outlay, with incremental additions deployed as population grows, as energy input per capita increases, and as the city becomes wealthier. To accommodate rapid growth but shortage of initial financing, a "just-in-time" capacity growth plan would be appropriate. Therefore, the small reactor plants must be designed to be easily expandable into clusters comprising ever-larger power installations.

TABLE 2. LIFECYCLE ENERGY RATIOS FOR VARIOUS ENERGY TECHNOLOGIES [21]*

ENERGY TECHNOLOGY	LIFETIME ENERGY RATIO (OUTPUT/INPUT)	LIFETIME ENERGY INPUT AS PERCENT OF OUTPUT	REFERENCE*
Solar PV (utility)	5	20	UCHIYAMA, 1996*1
LNG	6	17	UCHIYAMA, 1996*1
Wind	6	17	UCHIYAMA, 1996*1
Solar PV (roof top)	9	11	UCHIYAMA, 1996*1
Coal	17	7	UCHIYAMA, 1996*1
Nuclear (diffusion enrichment)	21	5	ERDA, 1976*2; PERRY, 1977*3
Natural gas-pipe	26	4	KIVISTO, 2000*4
Wind	34	3	KIVISTO, 2000*4
Hydro	50	2	UCHIYAMA, 1996*1
Nuclear (centrifuge enrichment)	59	2	ERDA, 1976*2; PERRY, 1977*3

**Source:* UIC Nuclear Issues Briefing Paper #57, May 2000, http://www.uic.com.au/nip57.htm

Note: Estimates of the energy ratios vary depending on the assumptions made in the analysis and on real operating conditions; for example, the significant difference between the two estimates of energy ratio for wind power represents, among other factors, significant differences in utilization factors related to site characteristics.

The outsourcing of fuel cycle services, including waste management, might be appreciated by many host countries of customers in this category; energy security could then be enhanced by the long refuelling interval plus the arrangements for guaranteed services, while, at the same time, additional costs for indigenous fuel cycle infrastructure emplacement will be avoided. And especially for small countries with few reactors, the outsourcing of waste management may be just a necessary condition to embark on any nuclear power programme.

Numerous concepts and designs of the small reactors without on-site refuelling described in this report could match the requirements of this emerging customer group, with power ratings from a few tens of MW(e) to 300 MW(e) being offered. Some of them specifically offer an option of incremental capacity increase to achieve as high as reasonable overall plant capacity. All offer long refuelling interval and outsourcing of the fuel cycle support services.

3.5.4. Future merchant plants for non-electric energy products

The non-electric markets for nuclear energy potentially include seawater desalination, district heating, low temperature process heat, and high temperature heat (including a potential for hydrogen manufacture by water splitting). These markets are likely to be served by commercial entities, which are separate from electric utilities, and for which financing relies on commercial bank loan rates or usual rates of return on investor equity.

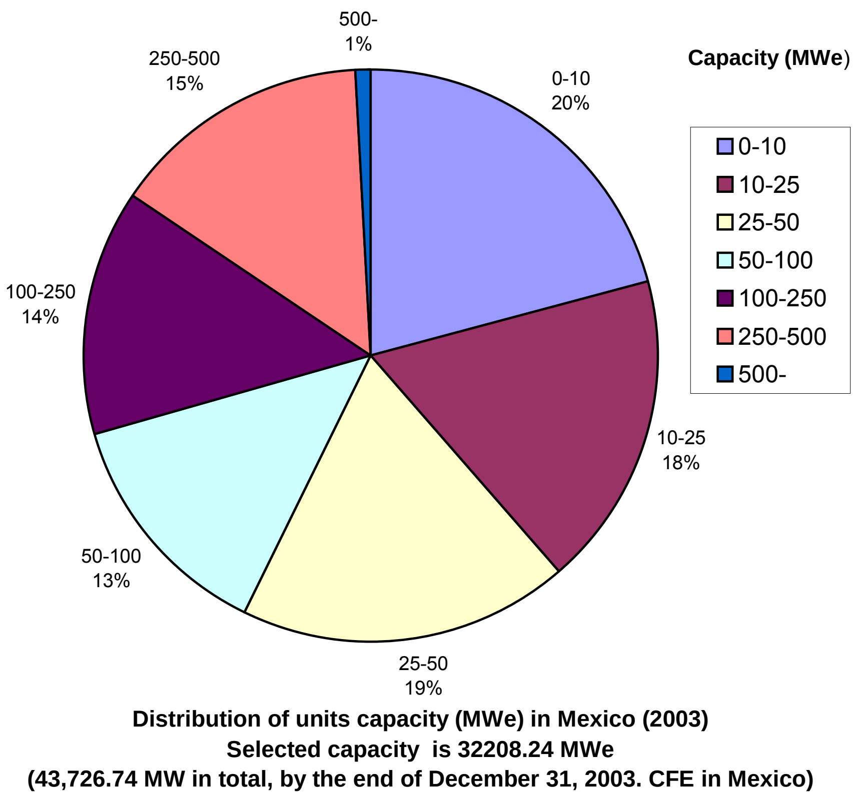


FIG. 10. Distribution of power plant sizes in Mexico [22].

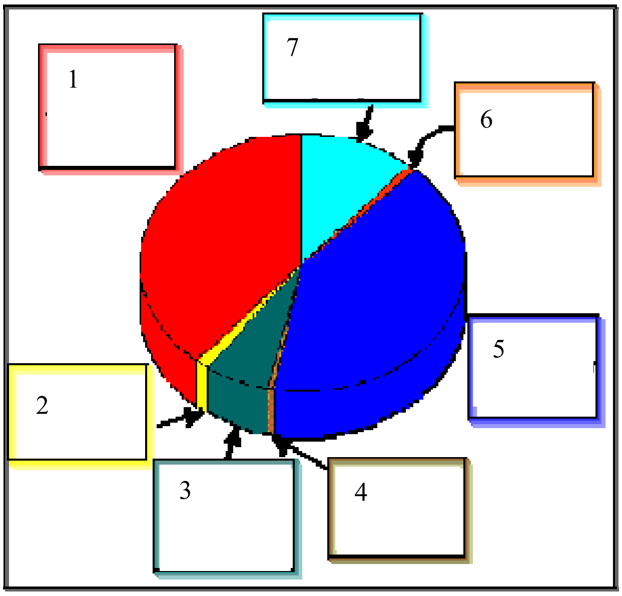
As it was discussed in previous sections, a commercial market for potable water production is expected to grow significantly in developed and developing countries alike. The need for water and for electricity to support population centres occur together. Among the small reactors without on-site refuelling, both dedicated and cogeneration nuclear desalination plant designs are being developed. Two technology options are considered. For dedicated desalination plants, the reverse osmosis process uses electricity directly for desalination. Cogeneration plants employ bottoming cycle distillation processes driven by heat from an extraction steam turbine or from a Brayton cycle cooler.

It is not a widely known fact that water withdrawals for the purpose of waste heat rejection from electricity-producing power plants can be equal to those for agricultural use (irrigation), see Fig. 11. The same may be true for hydrogen production plants. Power plant cooling water demands can be enormous [21]. For example, the Pacific Gas and Electric Diablo Canyon Station in the USA (two 1164 MW(e) pressurized water reactors) pumps 2 billion gallons (7.75 billion litres) of seawater per day for heat rejection and discharges the water at 20°F (~11°C) hotter than the ambient sea temperature.

(a)

Total freshwater withdrawals in the USA in 1995, by category of water use

Bgal/day = Billion of galons of water used per day



1 Power Generation 132 Bgal/day (39%)
2 Livestock 5.5 Bgal/day (1%)
3 Industry Mining 23.3 Bgal/day (7%)
4 Commercial 2.9 Bgal/day (1%)
5 Irrigation 134 Bgal/day (39%)
6 Domestic 3.4 Bgal/day (1%)
7 Public Supply 40.2 Bgal/day (12%)

(b)

Amount of water used to produce thermoelectric power in the USA, 1950-1990

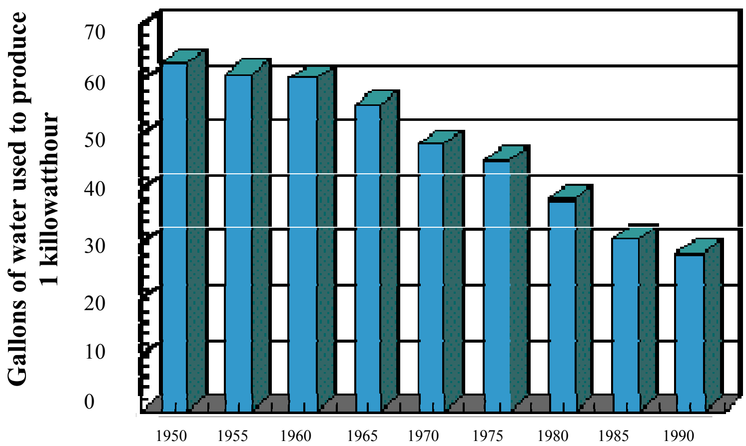


FIG. 11. Water use in the USA for all needs (a) and for power plants (a, b).

The need for water withdrawals for power plant heat rejection competes with water withdrawals for other purposes. Designs of power plants that achieve reduction in competition for water resources and reduction of the thermal plume ecological footprint are becoming important necessities for any energy architecture supporting sustainable development. This suggests the commercial efficacy of cogeneration plants producing electricity plus potable water using a thermal desalination bottoming cycle.

Some analysts believe that a hydrogen economy will eventually replace the fossil economy altogether [12]. Hydrogen can be combusted for heating applications or converted to electricity in a fuel cell; it may supplant fossil energy carriers serving all primary energy applications. Hydrogen can be manufactured from water by electrolysis or (potentially) using thermochemical water cracking cycles. Nuclear production of hydrogen in place of steam methane reforming would eliminate CO_2 emissions at both the supply and the end use links of an energy supply chain and would further the ecological pillar of sustainable development. Should the hydrogen economy come into being in the decades ahead, the market for nuclear production of hydrogen might be huge, becoming twice that of the nuclear electricity.

It is too early to predict the configuration of a hydrogen economy driven by nuclear energy. It might be highly centralized with huge, economy of scale reactors manufacturing hydrogen in regional centres for shipment to end users throughout the region (here the hydrogen itself would serve as a long distance energy carrier) [23]. Or alternately, it might take the form of a “hub-spoke” architecture with small reactors without on-site refuelling manufacturing hydrogen locally (here nuclear fuel serves as the long distance energy carrier) [24], see ANNEX XXIV.

Design requirements for extreme levels of reliability and safety apply to the non-electric applications because of the necessity to site process heat sources close to population (and industrial) centres.

The business opportunities for non-electric nuclear energy services do not necessarily fall under the governance of electric utilities. At least in the more distant future, deployments of such plants could be made under merchant plant financing arrangements, for which payback period must be short, internal rate of return on investment must be high, and financial risk minimization would be at a premium. This would make the financing needs very similar to those of fast growing cities in developing countries.

Many of the small reactors without on-site refuelling described in this report take the cogeneration route to supply non-electric energy products. A few are dedicated reactors for either district heating, or hydrogen production.

3.5.5. Other possible markets

Although it is explicitly mentioned in conjunction with only two designs of small reactors described in this report — the SVBR-75/100 (ANNEX XIX) and the BN GT-300/100 (ANNEX XVIII), both coming from the Institute of Physics and Power Engineering (IPPE) of the Russian Federation — an option to use such reactors for the so-called “renovation” of decommissioned older power plants, i.e., after necessary checks, to use the remaining premises and infrastructure and balance of plant of these older plants to accommodate and plug-in certain number of small reactor modules for another decades of operation, should be mentioned as another market opportunity for small reactors without on-site refuelling.

More exotic applications of such reactors may include underwater (e.g., to support underwater mining) or extraterrestrial (e.g., as a power source on the Moon, etc.) locations.

3.5.6. *Institutional prerequisites and requested support from the vendor*

The needs of the customer categories discussed above extend beyond technical needs to encompass needs for institutional and business support, as well as support of operations.

A country's economic development can be 'fuelled' by energy input supplied from small reactors without on-site refuelling. But this economic development must first be enabled by creating an institutional infrastructure favourable to investment and enterprise. The rule of law, enforceable property rights and an independent judiciary, a favourable tax structure, a stable currency, etc. are examples of institutional prerequisites for economic development.

Over and above such enterprise-fostering institutions, deployment of nuclear energy assets requires emplacement of additional institutional arrangements [25]. A nuclear licensing authority must be created and its authorities defined under law. Nuclear issues must be added to the various permitting processes for air and water quality, etc.; waste management legislation is required; and liability laws should be passed. Finally, trained staff is required for the operations and the oversight of nuclear facilities.

Upon request of a member state, the IAEA is able to provide guidance and assistance in emplacing these institutional prerequisites for the country's initial nuclear deployments [25]. Beyond that, the nuclear vendor may also offer assistance to potential customers as a part of the business arrangement. The degree to which such institutional and business prerequisites are currently in place versus what could be requested of the vendor varies dramatically.

The World Economic Forum has assessed 104 countries against a "business competitiveness index", which measures the status of a country's institutional prerequisites for economic development [26]. A summary is shown in Fig. 12, which plots the country's GDP/capita against its "Business Competitiveness Index". Broadly speaking, the countries listed in the lower left corner of the plot are "developing" countries, those clustered in the middle of the plot are "transitional" countries already having significant institutional infrastructure and growing GDP/capita (a subset of this group are countries of the former Soviet Union and Eastern Europe whose economies are transitioning out of the economic trauma of the break-up of the Soviet Union), and those in the upper right of the plot are "developed".

Vendors of small reactors without on-site refuelling would generally (initially) reside in developed or transitional countries[7]. When customers in these countries acquire nuclear assets, the service they are most likely to request from the vendor is waste management. For developing countries (and for small towns in off-grid locations as well as future merchant plants for non-electric energy products), the services requested of a nuclear vendor may extend all the way to a full-service energy supplier, wherein the vendor owns and operates the power plant and sells power to the customer under a commercial contract. Other customers may wish to own the plant but request assistance with manpower training, infrastructure development, or technology transfer.

However, the so-called developing and transitioning countries are very far from homogeneous in their situations, needs, and aspirations. The range of country characteristics can be appreciated from country presentations by energy ministers at the recent International Ministerial Conference on Nuclear Power for the 21st Century held in Paris on March 21–22, 2005 and sponsored by the IAEA and the OECD-NEA [27].

[7] This is evidenced by the countries of origin of the 30 concepts described in this report.

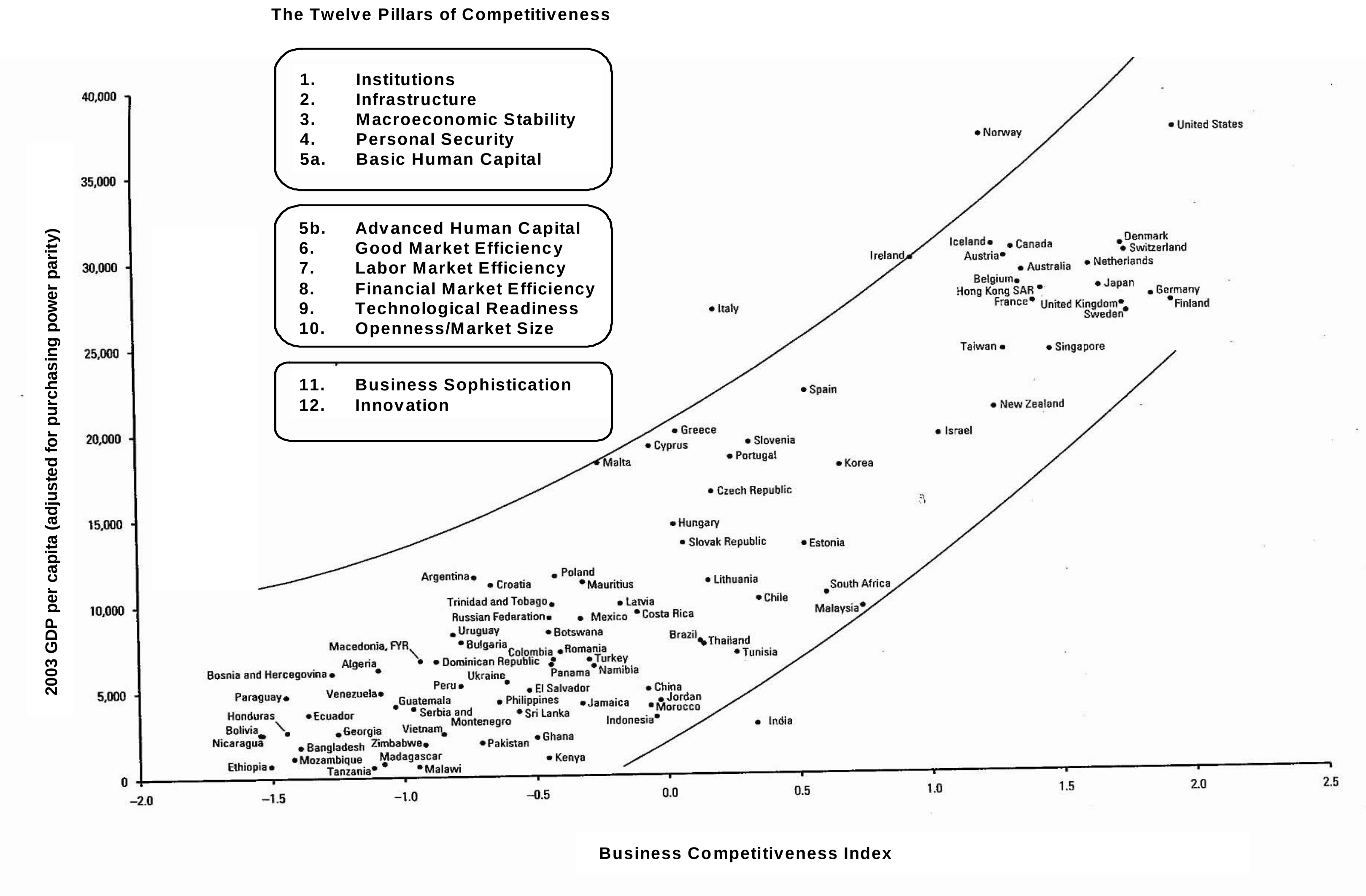


FIG. 12. World Economic Forum's assessment of the status of institutional prerequisites for economic development for 104 countries [26].

Several characteristics were widely expressed, such as:

- Desire to attain (or maintain) high economic growth rate;
- Trending to liberalization of the country's power sector.

Several aspirations were essentially universal, including:

- Desire to improve security of energy supply;
- Desire to reduce greenhouse gas emissions, even though not bound by the Kyoto commitments; and
- Desire to support a regime of non-proliferation.

Alternately, substantial diversity exists in numerous characteristics regarding readiness for nuclear deployments:

- Some developing countries have skilled technical work force already in place (e.g., Turkey, Hungary, Bulgaria, Romania), while many do not and would welcome assistance with training;
- Some developing and transitional countries have prior experience with nuclear energy, and enabling nuclear laws and legal infrastructure are already in place, including ratification of international norms, while some do not and are desirous for help in emplacing the enabling legal framework for nuclear energy (e.g., Indonesia).

Some developing countries that are rich in fossil reserves value fossil exports as comprising a main component of their export revenue stream (e.g., Indonesia, Iran) and desire to secure their own energy security using nuclear so as to sustain fossil exports.

Some transitioning countries aspire to achieve complete self-sufficiency in the nuclear power plant and fuel cycle enterprise (e.g., India, China, Iran); others want to exploit selective indigenous competitive advantage in natural resources or industry acumen for specific segments of the enterprise (e.g., Brazil, Argentina, Bulgaria); and, finally, some may want to purchase all or most capability from outside sources.

Not only are the characteristics, aspirations, and needs of a developing and transitional country customers heterogeneous now, they would remain heterogeneous in the future as each country's economy evolves and the country shifts up the curve in Fig. 12. These differing aspirations will always significantly affect the technology transfer, institution building, and training provisions which enter into the negotiations with suppliers.

Potential vendor support service requests are evaluated in Table 3, on a preliminary basis.

3.6. Business strategies for small reactors without on-site refuelling — an option of nuclear energy configuration tailored to the needs of emerging customers

3.6.1. Summary of customer needs

Certain categories of customers surveyed in the previous section (depending on the case, they may represent certain segments of a country as well as the country as a whole) are characterized by having:

- Small or non-existent electricity grid;
- Limited access to financing;
- Perhaps, immature institutional framework and limited skilled workforce.

TABLE 3. POTENTIAL RANGE OF CUSTOMER REQUESTS FOR VENDOR SERVICES

CUSTOMER TYPE		REQUIRED ENERGY PRODUCTS					SERVICES REQUESTED FROM THE VENDOR						
							POWER PLANT			FUEL CYCLE		INSTITUTIONS	
Characterization	Prototype examples	Electricity MW(e)	District heat	Potable water	Process heat	Other	Reactor heat source	Balance of plant	Operating crew	Front end fuel cycle services	Back end and waste	Institu-tional training	Technology transfer
1. Power purchase a. Small, isolated areas	North Siberia; Alaska; Indonesia Islands	1–10	Yes No	Option Yes	Option		Vendor owned	Vendor owned	Vendor	Outsourced	Outsourced	No Yes	No
b. Cities of countries at an early stage of institutional and economic development	Selected countries of South America, Africa	10–50	Option	Yes	Option	Power for remote mines, logging	Vendor leased or customer owned	Customer owned	Vendor or customer	Outsourced	Outsourced	Yes	Option
Indigenous infrastructure development a. Towns/urban regions in countries of economic growth and developing institutions	Turkey; Mexico; Egypt; etc.	100–300	Option	Yes	Option	H_2	Purchase or lease for purchase	Customer built and owned	Customer	Outsourced	Outsourced	Yes	Option
b. Cities/urban regions in countries of developed institutions and recovering economies	Eastern Europe	600–1600	Option	Option	Option		Purchase or lease for purchase	Customer built and owned	Customer	Indigenous or outsourced	Outsourced	No	Option

CUSTOMER TYPE		REQUIRED ENERGY PRODUCTS					SERVICES REQUESTED FROM THE VENDOR						
							POWER PLANT			FUEL CYCLE		INSTITUTIONS	
Characterization	Prototype examples	Electricity MW(e)	District heat	Potable water	Process heat	Other	Reactor heat source	Balance of plant	Operating crew	Front end fuel cycle services	Back end and waste	Institu-tional training	Technology transfer
<u>Aspiring suppliers of nuclear technology</u>													
a. Large developing countries having a nuclear history	China; India; Republic of Korea; Argentina Brazil	100–1600	Option	Yes	Yes	H_2	Customer built or purchased	Customer built or purchased	Customer	Indigenous	Indigenous	No	Yes/ option
b. Developing countries having U ore at an early stage of institutional and economic development	Kazakh-stan, etc.	100–600	Option	Option	Yes	Elec-tricity; fuel for mines	Vendor or customer owned	Vendor or customer owned	Vendor or customer	Indigenous (goal)	Outsourced	Yes	Yes/ option

Regarding a nuclear power option, these customers may require:

- Very high levels of plant safety, because of siting near population centres;
- Infrequent refuelling, as justified by:
  - The reactor being the sole source of energy on a small grid; or
  - Difficult fuel delivery owing to remoteness; or
  - Safeguards concerns; and
- Full front and back end fuel cycle service support, including waste management.

There could be additional unique needs of specific customer categories:

- Cities of developing countries experiencing rapid growth would benefit from scalability, i.e., potential for cost effective incremental clustering of power plants as their power demand grows;
- Merchant plants for non-electric energy services may require advanced cogeneration options;
- Off-grid industrial applications would require easy and potentially frequent transportability over difficult terrain and may present unique duty cycles.

3.6.2. Common features of small reactors without on-site refuelling

Thirty specific concepts and designs of small reactors without on-site refuelling are presented in Annexes I through XXX of this report. While each one is unique in its technical approach, there are certain common business strategies that are either explicitly described by the designers or just implicitly match the proposed concepts.

For small reactors without on-site refuelling, the observed common business strategy is to tailor the offering specifically to meet the needs of certain customers by providing a standardized turnkey plant that is:

- Easily transported and installed;
- Superbly safe on the basis of multiple inherent and passive safety features;
- Pre-licensed (standardized design certification) in the supplier country;
- With vendor-supplied front and back end fuel cycle services, including waste management;
- Has a long (many years) whole-core refuelling interval with a potential for refuelling equipment to be brought to the site by the refuelling team, or with entire reactor module change-out.

The mentioned above special features of small reactors without on-site refuelling make them compatible with proposed future institutional means to centralize fuel cycle facilities at only a few locations worldwide. Moreover, item accountancy could be performed on entire cores during shipment and operation deployment of such reactors.

Many concepts of small reactors without on-site refuelling indicate a potential for fuel load leasing (in some cases, internal breeding ratio of unity maintains fissile mass “principal” for the load owner). Several concepts indicate a potential for reactor module or complete nuclear power plant leasing.

3.6.3. Strategies to overcome the loss of economies of scale

The concepts of small reactors without on-site refuelling are sized under 300 MW(e); some are as small as a few MW(e). Then, all of them need to forego economies of scale of the power plant, i.e., employ alternative approaches to be competitive in targeted markets.

Alternately, the supporting fuel cycle and waste management infrastructure could be centralized and would benefit from economies of scale as compared to indigenous emplacements in every country. Importantly, safeguards costs might benefit from centralizing the bulk fuel handling operations, from precluding access to fuel at the distributed reactors, and from employing item accountancy on entire cores during shipment and operation deployment of such reactors.

Market penetration requires both the initial buy-in cost and the ongoing cost of energy to be competitive with the prices of the competition that is available to the customers. Then, for the targeted categories of customers, large economy of scale power plants may be just not available or affordable. Instead, the competition can comprise:

- For small villages and mines:
  - Diesel;
- For growing cities in developing countries:
  - Hydro power;
  - Gas;
  - Coal;
  - Oil;
- For future merchant plants for process heat production:
  - Bio;
  - Gas;
  - Coal.

The cost of the competition varies dramatically among countries and customer categories. In general, costs are higher (or even much higher) than found in developed countries serviced by interconnected grids and massive customer bases.

Beyond cost considerations, market penetration requires that small reactors without on-site refuelling meet the customer’s non-cost-related needs better than does the competition available to them. Here, the benefits of small reactors without on-site refuelling may include:

- Better energy security;
- Better reliability;
- Lower environmental pollution;
- Better local jobs;
- Less ancillary infrastructure investment required;
- Less waste legacy, etc.

Thirty specific designs described in Annexes I through XXX each use a unique mix of design and business approaches to cope with the loss of economies of scale on the plant. However, most of them imply several common strategies targeting:

- To transfer risk from the customer to the vendor:
  - Customer would receive a standard pre-licensed turnkey plant delivered and assembled by a skilled vendor team with only a short interval between securing financing and the start of a revenue stream;
  - The vendor has a start-up risk of building a factory for mass production and creating a logistics and installation capability, but could spread his cost of risk over many plants;
- To reduce site construction time and construction cost and achieve an early start of a revenue stream by:
  - Sizing the reactor for transportability (or transportability of modules);
  - Targeting a standardized pre-licensed design with no site-specific modifications provided for;
  - Providing for shipping from a factory and assemble of modules using an itinerant vendor assembly team who are focused on speed and efficiency;
  - In some cases, assigning no (or limited) nuclear safety function to the balance of plant, so that it could be built to local standards by local constructors using local labour with financing denominated in local currency;
- To benefit from factory mass production through:
  - A pre-licensed design certification for a standardized plant with no site specific modifications;
  - Serial manufacture of standardized plant modules;
  - Achieving reciprocity arrangements among licensing authorities in customer and vendor countries;
- In multi-module plants, to take a benefit of smaller module sizes to:
  - Achieve learning curve acceleration and discount rate savings per total capacity installed; and
  - To minimize capital-at-risk.

To reduce operation and maintenance (O&M) costs, the designers of small reactors without on-site refuelling target:

- To reduce operating staff number and required skill level through:
  - In some cases, the use passive load follow control (less demand on operation and maintenance and well qualified staff);
  - In some cases, providing for off-site monitoring and dispatch of specialized maintenance crews from a centralized facility supporting region deployments;
  - Refuelling operations outsourced to a specialized vendor team;
- Fuel leasing operation — some approaches employ whole core fuel cassettes with internal conversion ratio of unity that maintains fissile loading, ensuring that there no loss of fissile mass "principal" by end of life.

3.6.4. Strategies to balance energy security with non-proliferation assurances

All small reactors without on-site refuelling are being designed to offer an enhanced energy security in operation with the outsourced front-end and back-end fuel cycle services:

- High energy density of nuclear fuel enables designs such that once sited, the power plant with small reactor without on-site refuelling can deliver energy for many years with no need of any operations with fuel[8] during this whole period − relaxing the dependence on foreign suppliers, fuel cost changes, political and economic tensions and conflicts between countries, etc. − altogether, increasing energy security to the customer;
- Whole core refuelling is conducted infrequently by vendor crews using one of the two methods:
  - Total change-out of the reactor module; or
  - Whole core refuelling, e.g., using equipment brought to the site and removed with the used core;

Shipments of whole reactors or whole core fuel loads could then employ item accountancy procedures and Global Positioning System (GPS) monitoring. All bulk handling of fissile material (enrichment, fabrication, reprocessing, and refabrication) could be conducted in a limited number of centralized facilities that can serve many hundreds of small reactors. These centralized facilities could operate under international safeguards oversight[9].

Altogether, the above mentioned features may help better balance energy security with non-proliferation assurances.

3.6.5. Strategies for transition to a sustainable fuel cycle

The centralization of fuel cycle support operations offers a potential pathway to achieve sustainable world energy supply via symbiotic fuel cycles involving a time evolution of reactor types. Large-capacity light water reactors (LWRs) will maintain dominant market share in developed countries and some transitional countries for many decades. Small reactors may be favoured in numerous emerging markets. Some of the thirty small reactors described in this report are nearer-to-medium term designs and are based on uranium fuel loads; others are medium-to-longer term designs and use transuranics recovered from LWR spent fuel — these sources are sufficient to fuel deployments for the early decades of a transition toward a longer-term fissile self-sustaining symbiotic fuel cycle. Ultimately, fast breeder reactors could be sited at the fuel cycle centres to serve a fuel manufacture and waste transmutation function. With all bulk fuel handling conducted at centralized facilities, such symbiotic cross flows of fissile material have the potential to maintain fissile production in balance with demand, and could be conducted under an appropriate safeguards regime.

3.6.6. Decommissioning strategies

Reference [28] defines the following objective for minimizing decommissioning costs of advanced nuclear plants:

---

[8] Such as procurement, handling, transportation, storage, etc.

[9] The institutional approaches to prevent global dispersal of fuel cycle facilities and possible synergies with small reactors without on-site refuelling are summarized in Chapter 5.6.

"Decommissioning of nuclear plants is a significant cost factor for which the planning for decommissioning and the accumulation of an adequate reserve during plant operation to cover the costs requires greater consideration. Consideration of such preliminary decommissioning plans should be part of the design effort so as to optimize, where possible, the capability to decommission the plant and minimize the associated costs".

Small reactors without on-site refuelling incorporate potentially beneficial decommissioning strategies in their original design concepts. For example, floating (barge-mounted) NPPs with such reactors could benefit from full factory-performed decommissioning of the entire plant, borrowing from the experience of nuclear-propelled ships and submarines. Some concepts of small reactors suggest the use of factory fabricated and fuelled transportable reactor modules; all operations with the fuel and internals for such modules would then be outsourced to a centralized factory. Finally, nearly all concepts of small reactors without on-site refuelling provide for no fresh or spent fuel storages at the site and all concepts assume a long period of operation without reloading and shuffling of fuel in the core.

It could be recommended that, in line with the recommendations of reference [28], designers of small reactors without on-site refuelling pay more attention to considering preliminary decommissioning strategies at early design stages of their concepts, to take full advantage of the potential benefits outlined above.

3.7. Conclusion to Chapter 3

In reviewing the needs of different categories of potential customers for small reactors without on-site refuelling, one can find many needs which are shared across several customer categories in terms of:

- Power rating and the option for incremental capacity increase;
- Energy products, such as electricity, potable water, heat and, potentially, hydrogen;
- Enhanced safety and reliability requirements, e.g., to allow plant siting in immediate proximity to its customer;
- Outsourced front and back end fuel cycle support services, including waste management.

One can also find a very diverse range of non-technical and business needs involving:

- Power plant operational support;
- Assistance with emplacement of legal and institutional infrastructure;
- Technology transfer; and/or
- Participation in the construction project.

Common business strategies to meet these needs have been summarized in this chapter, including:

- Strategies to tailor the offering specifically to meet the needs of certain categories of customers;
- Strategies to overcome the loss of economies of scale;
- Strategies to balance energy security with non-proliferation assurances; and
- Strategies for transition to a fissile self-sustainable symbiotic fuel cycle.

Table 3 summarizes these characteristics of the several categories of customers at a coarse level. A more detailed survey of customer needs based on a dialogue with member states could be helpful to bring out more exactly the current situation in each specific case.

The performed preliminary consideration indicates a potentially huge and sustainable market for small reactors without on-site refuelling once they become ready to serve the needs of rapidly growing cities in developing countries and, later on, independent merchant type plants for non-electric energy products. A nearer-term opportunity for such reactors is provided by industrial sites and small towns in off-grid locations — the market that also has a large potential for expansion, once conveniently located deposits of natural resources become exhausted.

REFERENCES TO CHAPTER 3

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Technology Review Update 2005, GC(40)/INF/3, IAEA, Vienna (2005).

[2] PRAHALAD, C.K., The Fortune at the Bottom of the Pyramid: Enabling Dignity and Choice through Markets, Wharton School Publishing, ISBN 0-13-146750-6 (2005).

[3] World Energy Book 2005, World Energy Council: http://www.worldenergybook.com/.

[4] YODA, S., Ed., Trilemma — Three Major Problems Threatening World Survival, Central Research Institute of Electric Power Industry, Tokyo 100, Japan (1995)

[5] SPLATNIK, J., Sustainable development: Theory and practice, Global 2003 (Proc. Int. Meeting, New Orleans, USA, November 16–20, 2003), paper #1964, ANS/ENS.

[6] NAKICENOVIC, N., GRÜBLER, A., Mc DONALD, A., EDS. Global Energy Perspectives, Cambridge University Press, Cambridge, UK (1998).

[7] BRUNTLAND, G., Chairman, World Commission on Environment and Development (1987), Our common future, The Brundtland Report, Oxford University Press, Oxford, United Kingdom (1987).

[8] LANGLOIS, L., ROGNER, H., Mc DONALD, A., VERA, I., Energy system expectations for nuclear in the 21st Century, a plausible range, ICONE-10 (Proc. of 10th International Conference on Nuclear Engineering, Arlington, VA, April 14–18, 2002), paper ICONE10-22499.

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidance for the Evaluation of Innovative Nuclear Reactors and Fuel Cycles Report of Phase 1A of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1362, Vienna (2003).

[10] GLEICK, P.H., Global freshwater resources: Soft-path solutions for the 21st century, Science Magazine, Vol. 302, pp 1524–1528, (Nov 28, 2003 issue).

[11] WALKER, J.P., Dealing with the increased problems associated with trans-boundary water, Eos, Vol. 84, No. 46 (Nov 28, 2003 issue).

[12] AUSUBEL, J., Where is Energy Going?, the Industrial Physicist, American Institute of Physics, Melville, NY, USA (January 2000 issue).

[13] MARCHETTI, C., Hydrogen and energy, Chemical Economy and Engineering Review, CEER, Vol. 5 (1) (1973).

[14] HÄFELE, W., On the nature of nuclear power and its future, plenary speech, GLOBAL'93, Future Nuclear Systems: Emerging Fuel Cycles and Waste Disposal Options (Proc. Int. Meeting, Seattle, Washington, USA, Sept. 12–17, 1993) pp. 3–17.

[15] US-DOE-NE Report to Congress, Small modular nuclear reactors, available at DOE web site, http://www.doe.gov, (2002).

[16] SU'UD, Z., The role of nuclear energy in Indonesia in the 21st Century, GENES4/ANP2003, paper 1227 (Proc. Int. Conf., Kyoto, Japan, Sept. 15–19, 2003).

[17] WEBB, J., Daring to be different: India has bold plans for a nuclear future, New Scientist, www.newscientist.com, (Feb. 19, 2005 issue).

[18] Department of Energy, Mines and Resources, Yukon Government, http://www.emr.gov.yk.ca/energy/hydro.html.
[19] Population Reference Bureau website, http://www.prb.org and Bugliarelo, G., Megacities and the developing world, The Bridge Vol. 29, #4, Winter 1999, available at the National Academy of Engineering, website, http://www.nae.edu.
[20] ROGNER, H., Building sustainable energy systems: The role of nuclear derived hydrogen, Proceedings of the American Nuclear Society 2001 Winter Meeting, Reno, NV, (2001).
[21] Table 2, in The Greening of the Nuclear Age, a publication prepared by the American Nuclear Society for distribution at the World Summit on sustainable Development in August. 2002. www.ans.org.
[22] MINATO, A., (CRIEPI, Japan), private communication to D. Wade 28 Nov. 2005.
[23] MARCHETTI, C., Energy islands in the final configuration of the hydrogen economy, Plenary lecture at the World Hydrogen Conference (Proc. of Int. Conf. June 21–26, 1998), www.cearemarchetti.org.
[24] WADE, D., DOCTOR, R., PEDDICORD, K.L., STAR-H2: A 400 MW_{th} lead-cooled fast reactor for hydrogen manufacture in a sustainable hierarchical hub-spoke energy infrastructure, paper # 1189 GENES4/ANP2003, Kyoto, Japan (Sept. 15–19, 2003).
[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Basic Infrastructure for a Nuclear Power Project, IAEA-TECDOC-1513, IAEA, Vienna, Austria (2006).
[26] PORTER, M., SCHWAB, K., SALA-I-MARTIN, X., LOPEZ-CARLOS, A., The Global Competitiveness Report 2004–2005, World Economic Forum, published by Palgrave Macmillan, New York, NY (2004).
[27] International Ministerial Conference on Nuclear Power for the 21st Century, held in Paris on March 21–22, 2005 and sponsored by the IAEA and the OECD-NEA, www.parisnuclear2005.org.
[28] INTERNATIONAL ATOMIC ENERGY AGENCY, Objectives for the Development of Advanced Nuclear Plants, IAEA-TECDOC-682, Vienna (1993).

4. SURVEY OF POWER PLANT CONCEPTS, DESIGN APPROACHES, AND POSSIBLE VENDOR SUPPLIED OPERATIONAL SUPPORT SERVICES

4.1. Introduction

Thirty concepts of small reactors without on-site refuelling are presented in this report. All of them have a potential to meet the needs of one or more of the several groups of customers who have not previously been served by nuclear energy, see the discussion in Chapter 3. The list of these potential customers includes:

- Stand-alone power plants for small remote communities and/or industrial sites in off-grid locations facing difficult conventional fuel delivery challenges;
- Cities in developing countries having high population growth and economic growth rates, but having undeveloped grids, shortages of capital financing, and immature industrial infrastructure and trained workforces;
- In a longer-term, perhaps, independent power producers1 (IPP) and/or cogeneration process heat customers in developed countries who are entering markets for non-electric, energy intensive products such as hydrogen or potable water production, and who face "merchant" plant financial conditions requiring short payback period and high internal rate of return on investment.

Given such diverse groups of potential customers, the thirty concepts addressed span broad ranges of technical parameters and features, as shown in Table 1.

TABLE 1. SUMMARY OF ENERGY PRODUCT OFFERS AND SPECIAL FEATURES OF SMALL REACTORS ADDRESSED IN THIS REPORT

PARAMETER	RANGE
Energy products	Electricity; heat for district heating; potable water; process heat
Electrical power rating	From several tens of kW(e) to 300 MW(e)
Common fuel cycle support strategy: • Long refuelling interval and outsourced front and back end fuel cycle services	Implemented by one of the following options: • Return a floating plant to the factory; • Return a transportable land-based plant to the factory; • Whole core cassette change-out; • Once-at-a-time core sub-assembly change-out; • In-situ pebble bed recharging

Moreover, the various concept proponents plan to offer a progression of business arrangements to the customer — from providing vendor-owned and operated transportable power plants for customer purchase of power, absent all further responsibility on the part of the customer — all the way to providing the customer a purchased reactor delivered and installed by the vendor adjacent to a customer-constructed balance of plant and supported by a

[1] Independent power producers (IPPs) could be merchant generation companies who operate outside the regulatory framework of regulated utilities and sell their product on a competitive market, i.e., they receive no guarantee of profitability in exchange for a guarantee of providing service to consumers.

long-term contract for vendor supplied fuel cycle and waste management services. In some concepts, vendor and customer may both rely on “regional fuel cycle centres” owned and operated under multilateral arrangements and offering front end and back end fuel cycle services to regional power plant operators.

Despite the diversity of targeted customers and business strategies, most of the concepts employ several fundamental features:

- Power rating within a small reactor power range and factory fabrication as modularized components;
- Ease of transportability and rapid field assembly of the plant;
- Very high levels of safety to support siting near population centres;
- Long refuelling interval; and
- Avoidance of the necessity for pre-existence of indigenous fuel cycle and waste management support infrastructures in the customer’s country.

Annexes I through XXX contain design descriptions for each one of the concepts, presenting design details but also the background rationale (which underlies the design approaches taken), the enumeration of related technology experience base, and the surveys of research and development (R&D) activities still necessary to perform before commercial readiness.

4.2. Structure of the chapter

This chapter has two parts. In the beginning, the power plant concepts are surveyed from the point of view of a customer:

- What is offered in terms of power rating and energy products?
- Which types of vendor support services are offered?
- How soon could the offering be commercially available?

First, nearer-term concepts (with a potential to become available within 5–10 years) are discussed; then, longer-term concepts, which require more substantial further R&D, are addressed. This survey may be helpful to a potential customer for identifying specific concepts best meeting his needs; a more detailed examination could then be pursued in the detailed descriptions found in the annexes.

In the second part of the chapter, attention is turned to the viewpoint of the development community — by drawing from the detailed descriptions in the annexes to delineate, inter-compare and discuss the several classes of design approaches that have been employed by different design teams to meet anticipated customer needs. These discussions address, among others:

- Approaches to achieve long refuelling interval;
- Approaches to reduce staffing;
- Approaches to remove and transport heat from the fuel lattice and drive energy converters;
- Approaches to achieve plant transportability and rapid site assembly;

- Approaches to achieve safety consistent with siting near population centres;
- Approaches to minimize the cost impact attendant to loss of economies of scale and to maximize the cost savings of economies of mass production and rapid site assembly; and
- Strategies to facilitate plant commercialization.

4.3. Organizational structure for the concepts

Table 2 lists the concepts in the order provided by the annexes. For each of the entries, the table specifies concept name, principle designer and the country of origin, rated thermal and electric capacity and operation period between refuellings, current design stage (at the time when this report was prepared), timeline for detailed design development including licensing (a projection by the designer), and the annex number where the concept is described in more detail. For each concept, the table also indicates the availability of a multi-module plant option. In this way, the table lists water cooled reactor concepts first; then, gas cooled, sodium cooled, lead and lead-bismuth cooled and non-conventional designs, correspondingly.

For the purpose of categorizing the concepts and discussing the trends of common features and design approach, it has been found convenient to group the concepts in a hierarchical fashion, as follows:

- Very small plants for autonomous or unattended operation;
- Water cooled thermal-spectrum reactors for electricity production with optional desalination or heating bottoming cycles, including:
  - Adaptations from marine-reactor experience base;
  - Adaptation of commercial light water reactor (LWR) experience base;
  - Use of TRISO type fuel in LWRs;
- Liquid metal cooled fast reactors for electricity production with optional desalination or heating bottoming cycles, including:
  - Sodium (Na) cooled reactors with conventional fuel;
  - Lead (Pb) and lead-bismuth (Pb-Bi) cooled reactors with conventional and new types of fuel; and
  - A Pb-Bi cooled direct contact boiling water reactor;
- High temperature lead cooled, molten salt cooled and gas cooled reactor concepts for hydrogen production and other applications.

The concepts organized into these hierarchical groupings are presented in Tables 3 through 6.

TABLE 2. SUMMARY OF THE CONCEPTS PRESENTED IN THE ANNEXES

No.	Concept name	Principal designer, country	Power, MW(th)/MW(e); Operation period between refuellings*	Energy products	Multi-module plant option	Design stage	Timeline for detailed design development including licensing**	Annex No.
Water cooled reactors								
1	MASLWR	Oregon State University, USA	150/35; 5 years	Electricity, potable water	Yes	Conceptual design		I
2	UNITHERM	NIKIET, Russian Federation	30/(2.5 or 6); 20 MW(th) for process heat applications; 16.6 years	Electricity, district heating, potable water, process steam	Yes	Conceptual design	5 years	II
3	ELENA	RRC “Kurchatov Institute”, Russian Federation	3.3/0.068; 72 GCal/hour for district heating; 21.7 years	District heating		Conceptual design; detailed design of fuel element		III
4	VBER-150	OKBM, Russian Federation	350/110; 5.7 years	Electricity; district heating; potable water	Yes	Conceptual design	3 years	IV
5	ABV	OKBM, IPPE; Russian Federation	45/11; 8 years	Electricity; district heating; potable water		Conceptual design; detailed design of the previous version (ABV-6M)	3 years	V
6	KLT-20	OKBM; Russian Federation	70/20; 8 years	Electricity; district heating; potable water; emergency source of heat and power for natural disasters		Conceptual design	2 years	VI
7	PSRD	JAEA, Japan	100/31; More than 5 years	Electricity	Yes	Conceptual design		VII

No.	Concept name	Principal designer, country	Power, MW(th)/MW(e); Operation period between refuellings*	Energy products	Multi-module plant option	Design stage	Timeline for detailed design development including licensing**	Annex No.
8	Package-Reactor	Hitachi, Ltd. and Mitsubishi Heavy Industries, Japan	10–100 MW(th); the ratio between electric and thermal output is flexible; 5–10 years	Electricity and thermal power	Yes	Feasibility study		VIII
9	PFPWR50	Hokkaido University; Japan	50 MW(th); ~7.3	District heating and hot water supply		Feasibility study	~5 years	IX
10	VKR-MT	VNIIAM and RRC “Kurchatov Institute”; Russian Federation	890/300; ~10 years	Heat and power plant		Feasibility study	~5 years	X
11	AFPR	PNNL, USA	300/100; ~36 years	Electricity	Yes	Feasibility study		XI
12	FBNR	Federal University of Rio Grande do Sul, Brazil	134/40; more than 10 years	Electricity; potable water; process steam; district heating	Yes	Feasibility study	~10 years	XII
Gas cooled reactors								
13	BGR-300	RRC “Kurchatov Institute”; Russian Federation	300/130; 12 years	Electricity; hydrogen		Feasibility study		XIII
Sodium cooled reactors								
14	4S Toshiba Design	Toshiba Corporation and CRIEPI, Japan	30/10 or 135/50; 30 years	Electricity; potable water; hydrogen and oxygen		Conceptual design completed	5–6 years	XIV
15	4S-LMR, CRIEPI Design	CRIEPI and Toshiba Corp.; Japan	135/50; 10 years or more	Electricity; potable water		Conceptual design	~10 years	XV
16	MBRU-12	OKBM, SPb AEP, and IPPE, Russian Federation	48/12; 30 years; fuel is shuffled internally each year	Electricity; potable water; district heating; process heat		Conceptual design		XVI

No.	Concept name	Principal designer, country	Power, MW(th)/MW(e); Operation period between refuellings*	Energy products	Multi-module plant option	Design stage	Timeline for detailed design development including licensing**	Annex No.
17	RAPID	CRIEPI, Japan	10/1; 10 years	Electricity; potable water		Conceptual design	6 years	XVII
18	BN GT-300	IPPE, Russian Federation	730/300; 4.5–6 years	Electricity; district heating; potable water		Early conceptual design	3 years	XVIII
				Lead or lead-bismuth cooled reactors				
19	SVBR-75/100	IPPE and Gidropress; Russian Federation	280/101.5; 6–8.8 years	Electricity; heating; potable water; coal liquefaction; etc.	Yes	Basic design completed	6–8 years, including the construction of a prototype plant	XIX
20	ENHS	LLNL, ANL, LANL, and the University of California at Berkeley	(125–180)/(50–75); 20 years or more	Electricity; potable water; process heat; district heating	Yes	Feasibility study	~15 years	XX
21	Small Lead-bismuth Cooled Reactor	JAEA, Japan	132/50; 30 years	Electricity		Conceptual design		XXI
22	SSTAR	ANL, LLNL, LANL, and INL; USA	45/20; 20 years	Electricity; Potable water; district heating		Early conceptual design	10 years	XXII
23	STAR-LM	ANL, US Universities; USA	400/178; 15 years	Electricity; potable water	Yes	Feasibility study	10 years	XXIII
24	STAR-H2	ANL, US Universities; USA	400 MW(th); 20 years	Hydrogen and potable water	Yes	Feasibility study	15–25 years up to commercial deployment	XXIV
25	LSPR	RLNR TITech; Japan	150/53; 11.5 years	Electricity; district heating; potable water; hydrogen; process steam		Feasibility study	20 years	XXV

No.	Concept name	Principal designer, country	Power, MW(th)/MW(e); Operation period between refuellings*	Energy products	Multi-module plant option	Design stage	Timeline for detailed design development including licensing**	Annex No.
26	SPINNOR/ VSPINNOR	ITB and BATAN, Indonesia	55/20 or 27.5/10 or 17.5/6.25; 15 or 25 or 25 years	Electricity; potable water		Feasibility study	More than 10 years	XXVI
27	PBWFR	RLNR TITech; Japan	450/150; 15 years	Electricity		Conceptual design		XXVII
Non-conventional designs								
28	MARS	RRC "Kurchatov Institute", Russian Federation	16/6; 8.5 MW(th) in heat supply mode; 15-60 years	Electricity; high temperature process heat; district heating, potable water, very high temperature heat, etc.		Conceptual design	5–8 years	XXVIII
29	CHTR	BARC, India	0.1/0.023; 15.1 years	Electricity; hydrogen; Potable water	Yes	Conceptual design		
30	MSR-FUJI	ITHMSI, Japan	450/200; More than 30 years; but on-line fissile and fertile feeding from an internal reservoir is needed	Electricity; hydrogen; potable water	Yes	Early conceptual design	8 years	XXX

* Whenever clearly identified by designers, the operation period between refuellings is in effective full power years; the actual operation period will be that divided by a load factor (for example, for a load factor of 0.85, the given value should be multiplied by 1.18).
** Under favourable conditions, including adequate financing.

TABLE 3. VERY SMALL REACTORS WITH AN OPTION OF UNATTENDED OPERATION*

(a) Water cooled

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
1.	ELENA; Russian Federation	III	PWR	0.068/3.3	21.7	UO_2 pellets; 15.2 MOX fuel is an option	Natural circulation	H_2O	311/328	Heating reactor; direct; thermo-electric		~9000 US$/ kW(th)	0.06
2.	UNITHERM Russian Federation	II	PWR	6/~30	16.6	U-Zr metal ceramic (cermet); 19.75	Natural circulation	H_2O	258/330	Indirect; Rankine	5 years	3048	0.04

(b) Sodium cooled

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
3.	RAPID; Japan	XVII	Fast reactor	1/10	10	U-Pu-Zr metal fuel; 14–19	Forced circulation	Na	380/530	Thermo-electric	6 years	8000	

* In Tables 3 through 6, annex number also corresponds to the reactor concept position in Table 2.

TABLE 4. WATER COOLED REACTORS

(a) Commercial LWR experience base

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
4.	PSRD; Japan	VII	PWR; integral design	31/100	> 5	UO_2 pellets; <5	Natural circulation	H_2O	270.4/311	Indirect; Rankine			
5.	MASLWR; USA	I	PWR; integral design	35/150	5	UO_2 pellets; 8	Natural circulation	H_2O	218.8/271.4	Indirect; Rankine		1458	0.07
6.	Package-Reactor; Japan	VIII	PWR/ BWR/ CANDU	Variable/ (10–100)	5–10	UO_2 pellets; 5	Natural circulation	H_2O	340/345	Indirect; Rankine			

(b) Icebreaker reactor experience base

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
7.	VBER-150; Russian Federation	IV	PWR	110/350	5.7	UO_2 pellets; 4.75	Forced circulation	H_2O	291/322	Indirect; Rankine	3 years	1636	0.025
8.	KLT-20; Russian Federation	VI	PWR	20/70	8	UO_2 granules in inert matrix; 19.2	Forced circulation	H_2O	289/317	Indirect; Rankine	2 years	2500	0.04
9.	ABV; Russian Federation	V	PWR; integral design	11/45; (18–60) MW(th)	8	UO_2 in silumin matrix; 16.5	Natural circulation	H_2O	247/330	Indirect; Rankine	3 years	4300	0.03

TABLE 4 (continued).

(c)TRISO fuel based

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
10.	PFPWR50 Japan	IX	PWR with TRISO fuel	–/50	~7.3	UO_2 based TRISO within graphite column in fuel rods; <5	Forced circulation	H_2O	~240/260	Heating reactor	~5 years		
11.	VKR-MT; Russian Federation	X	BWR with TRISO fuel	300/890	9.6	UO_2 based micro fuel elements in direct contact with coolant; 10	Forced circulation	H_2O	280/290	Direct; Rankine	~5 years	2340	0.075
12.	AFPR; USA	XI	BWR or direct flow system with TRISO fuel	100/300	~36	UO_2 based micro fuel elements in direct contact with coolant; 8–13	Forced circulation	H_2O	270/291 (BWR version)	Direct; Rankine			
13.	FBNR; Brazil	XII	PWR with TRISO fuel	40/134	>10 (target)	Fixed bed of the spherical fuel elements made of TRISO particles based on UO_2	Forced circulation	H_2O	290/326	Indirect; Rankine	~10 years	1000	0.021

TABLE 5. LIQUID METAL COOLED REACTORS

(a) Sodium cooled

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
14.	4S Toshiba Design; Japan	XIV	Fast reactor	10/30 50/135	30	U-Zr metal fuel; 17/19 12/18	Forced circulation	Na	355/510	Indirect; Rankine	5–6 years		0.1–0.24 0.05–0.1
15.	4S-LMR CRIEPI Design; Japan	XV	Fast reactor	50/135	10 years or more	U-Pu-Zr metal fuel; 17.5–20	Forced circulation	Na	355/510	Indirect; Rankine	~10 years	<2000	
16.	MBRU-12; Russian Federation	XVI	Fast reactor	12/48	~30 1-year internal shuffling	UO_2-PuO_2; 28	Forced circulation	Na	330/480	Indirect; Rankine			
17.	BN GT-300; Russian Federation	XVIII	Fast reactor; no IHTS*	300/730	4.5–6	UO_2; 17 UO_2-PuO_2	Forced circulation	Na	~450–500	Indirect gas turbine cycle (Ar–N)	3 years	593	0.01

* IHTS is for intermediate heat transport system

TABLE 5 (continued 1)

(b) Lead-bismuth cooled

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
18.	SVBR-75/100; Russian Federation	XIX	Fast reactor	101.5/ 280	6–8.8	UO_2 pellets; 16.1 MOX; UN; UN-PuN	Forced circulation	Pb-Bi	320/482	Indirect; Rankine	6–8 years; incl. prototype plant construction	1000	0.0146
19.	ENHS; USA	XX	Fast reactor	(50–75)/ (125–180)	20 years or more	Pu-U-Zr metal fuel; 12.2 (U-Pu)^{15}N; 13.1	Natural circulation; lift pump as an option	Pb-Bi; Na is an option	400/503	Indirect; Rankine or super-critical CO_2 Brayton	15 years	2000	0.034
20.	LSPR; Japan	XXV	Fast reactor	53/150	11.5	(U-Pu)N; 10–12.5	Forced circulation	Pb-Bi	360/510	Indirect; Rankine	20 years		
21.	Small Lead-bismuth Cooled Reactor; Japan	XXI	Fast reactor	50/132	30	(U-Pu)^{15}N; 10.5–18.2	Natural circulation	Pb-Bi	335/505	Indirect; Rankine			
22.	SPINNOR/ VSPINNOR Indonesia	XXVI	Fast reactor	20/55; or 10/27.5; or 6.25/17.5	15; or 25; or 35 years	(U-Pu)N; 10–12.5	Forced circulation	Pb-Bi	(340–345)/ (505–515)	Indirect; Rankine	~10 years to establish a pilot project	1500; or 1750; or 2000	
23.	PBWFR; Japan	XXVII	Fast reactor; direct contact steam production	150/450	15	(U-Pu)^{15}N; 11.5–15.8	Natural circulation for Pb-Bi; forced circulation for water	Pb-Bi; H_2O is injected in hot Pb-Bi above the core	307/457 – Pb-Bi 220/296 – H_2O	Direct; Rankine		3300	

TABLE 5 (continued 2).

(c) Lead cooled

#	Concept name; country	Annex No.	Type	Power, MW(e)/ MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
24.	SSTAR; USA	XXII	Fast reactor	20/45	20	TRU nitride, 100% ^{15}N enriched; 1.7/3.3/16.6/ 19.9 (TRU)	Natural circulation	Pb	420/566	Indirect; super-critical CO_2 Brayton cycle	10 years		0.05–0.08
25.	STAR-LM; USA	XXIII	Fast reactor	178/400	15	TRU nitride, 100% ^{15}N enriched; 13.3/18.2/ 21.3 (TRU)	Natural circulation	Pb	438/588	Indirect; super-critical CO_2 Brayton cycle	10 years		

TABLE 6. HIGH TEMPERATURE GAS COOLED, LEAD COOLED, AND MOLTEN SALT COOLED REACTORS

#	Concept name; country	Annex No.	Type	Power, MW(e)/MW(th)	Core lifetime (EFPY)	Fuel type, enrichment; weight %	Primary circulation type	Coolant	Core inlet/ outlet temperature, °C	Cycle type	Time for detailed design development including licensing	Capital cost, US$/ kW(e)	Electricity cost, US$/ kW-hour
26.	CHTR; India	XXIX	Non-conven-tional	0.023/0.1	15.1	$^{233}UC_2$-ThC_2-Gd TRISO particles in compacts within graphite matrix; 33.75	Natural circulation	Pb-Bi	900/1000	Thermo-electric	Critical facility in 2012		
27.	MARS; Russian Federation	XXVIII	Non-conven-tional	6/16	15 or 60	UO_2 based TRISO in spherical graphite; 10	Natural circulation	Molten fluoride salts	550/750	Indirect; open cycle air turbine	5–8 years	3500–2500	
28.	BGR-300; Russian Federation	XIII	Fast reactor	130/300	12	(U-Pu)C, (U-Pu)N; 14–15.5 (fissile Pu isotopes)	Forced circulation	He; or He-Xe as an option	350/850	Indirect; air turbine			
29.	STAR-H2; USA	XXIV	Fast reactor	–/400	20	TRU nitride, ^{15}N enriched; 13.1–18.4	Natural circulation	Pb	663.7/793.4	Process heat; optional indirect supercri-tical CO_2 Brayton cycle	15–25 years (commercial deployment)		
30.	MSR-FUJI; Japan	XXX	Molten salt reactor	200/450	>30 (on-line fissile and fertile feeding)	Molten fluoride salt of LiF-BeF_2-ThF_4-$^{233}UF_4$	Forced circulation	Molten fluoride salt	570/710	Indirect; super-critical steam Rankine cycle	8 years	1584	0.012

Table 3 lists technical specifications for small autonomous reactors; Table 4 contains specifications for water cooled reactors; Table 5 gives data for liquid metal cooled reactors; and Table 6 presents the characteristics for high temperature lead, gas, or molten salt cooled reactors.

Core outlet temperature is inherent to this hierarchical grouping of concepts; water cooled reactors for district heating operate at ~100 to 300°C core outlet temperatures; water cooled power reactors at ~300°C; Na and Pb-Bi cooled reactors at ~500°C, — all at temperatures for which materials are well proven. Then, Pb cooled reactors operate at ~600°C and, finally, high and very high temperature gas cooled, molten salt cooled and lead cooled reactor concepts, developed for a variety of applications including hydrogen manufacture by water cracking, operate at ~750–800°C and ~1000°C, respectively. Such reactors would require substantial R&D programmes for development of advanced materials to be carried out or completed.

Within these four categories, a final level of hierarchy is plant power rating — from as small as 100 kW(th) to as large as 890 MW(th), for a single-module[2] plant.

Later on in this section, the concepts are identified by numbers (1 through 30); these designations correspond to reactor identifications in Tables 3 through 6. The time for detailed design development and cost estimates provided in Tables 2 and 3–6 are those indicated by the developers; harmonization of estimating methodology among the developers has not been undertaken.

4.4. Survey of nearer-term concepts

Some of the concepts are based on well proven technologies, or borrow from the experience of previously operated marine propulsion reactors, or require a limited scope of R&D to be completed to become viable; these are rated by the designers as capable of becoming available within a 5 to 10 year time frame, under favourable conditions.

4.4.1. Very small reactors for unattended autonomous operation

One of the concepts, the water cooled ELENA (1), is being designed for district heating as its primary function. Another two concepts, the water cooled UNITHERM (2) and the sodium cooled RAPID (3), are being designed for a variety of applications, including cogeneration options with potable water and/or district heat production. All three concepts are sized for remotely sited towns of several tens to one hundred thousand populations; two are water cooled thermal spectrum reactors, one is a sodium cooled fast spectrum reactor. Their characteristics are summarized in Table 3.

The ELENA at 3.3 MW(th) is based on well proven technology of light water reactors with uranium dioxide fuel; its thermoelectric conversion array for providing electricity for on-site use is supported by the operating space reactor prototype, GAMMA.

The UNITHERM at ~30 MW(th) is based on proven water cooled, cermet fuelled marine reactor experience in the Russian Federation. These two concepts could be offered for deployment in the very near term.

The RAPID (3) is an adaptation of a proposed autonomous reactor for a moon base; it uses proven metal alloy fuel and Na coolant at traditional temperature but incorporates innovative

---

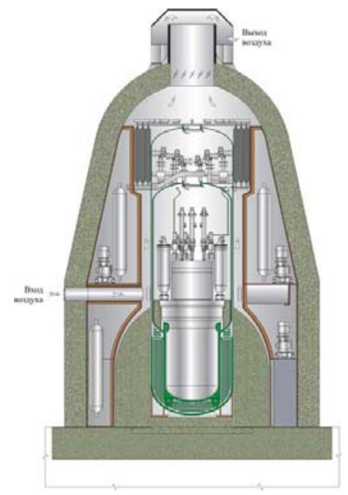
[2] Many of the concepts presented in this report incorporate modular approach to reactor design, which allows building multi-module plants of higher overall capacity with an option of incremental capacity increase. Such possibility is indicated in Table 2 and outlined in more detail in the corresponding annexes.

heat pipe and molten Li autonomous control concepts, see ANNEX XVII; therefore, it may require somewhat more time to be deployed. The fuel option specified as basic is U-Pu-Zr alloy, which assumes the operation within a closed nuclear fuel cycle. Elaborating and mastering a closed fuel cycle technology would require additional time; however, the U-Zr metal fuel option could be considered for a shorter-term deployment.

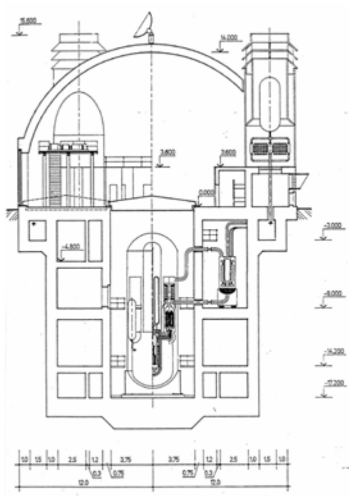
The ELENA and UNITHERM use natural circulation cooling and are intended for minimal operational staffing; an option of unattended operation is being considered, with monitoring from centralized regional support facilities.

The RAPID uses forced convection in normal operation mode; it is being developed as an operator-free reactor.

The infrequent refuelling is supposed to be handled by the vendor once in 10 to ~20 years, depending on concept, and in the cases of the ELENA and UNITHERM is performed by switch-out of the entire transportable reactor plant. The used plant is returned to the factory for refuelling and maintenance. In the RAPID case, a whole core once-at-a-time refuelling is performed at the site. Sketches of the UNITHERM and ELENA plants are shown in Fig. 1.



UNITHERM



ELENA

FIG. 1. Plant general view of UNITHERM (left) and ELENA (right), see ANNEXES II and III for details.

4.4.2. Floating power plants with cogeneration capability

Three of the concepts, VBER-150 (7), KLT-20 (8), and ABV (9), are being designed as barge-mounted, complete power plants which can be towed from the factory to a water-accessible site, moored in a pre-prepared lagoon, and connected to a localized grid[3]. Table 4 summarizes their characteristics. At ~10 to 150 MW(e), these plants could support electrical needs for off-grid towns of up to several hundred thousand populations. They are also properly sized for support of industrial operations at remote, water-accessible locations. Moreover, all of these plants offer potable water production or district heating.

The VBER-150, KLT-20, and ABV concepts are water cooled thermal reactors based on the Russian nuclear icebreaker experience. The ABV is a pressurized water reactor of an integral design; the VBER-150 and KLT-20 are lower core power density versions of the modular

[3] Land-based power plant option is also being elaborated for some of these concepts, to increase the flexibility of energy offer.

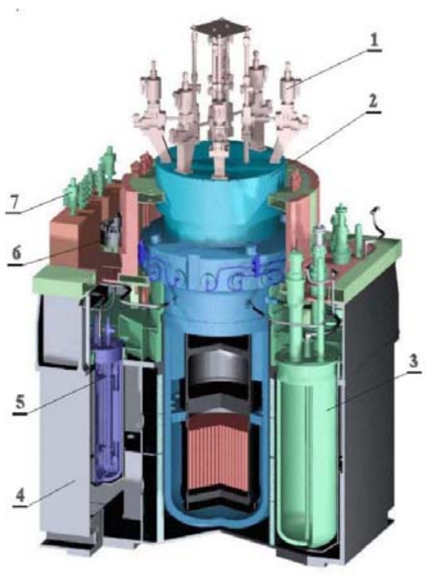
loop-type VBER-300 [1] and KLT-40S [2] reactors. The ABV, the KLT-20, and the UNITHERM (described in the previous section) use cermet fuel of enriched UO_2 dispersed in zirconium alloy or silumin matrix. The VBER-150 uses uranium dioxide fuel as currently used in the Russian VVER type reactors.

As an example, Fig. 2 shows the ABV nuclear steam supply system layout.

The reactors addressed in this section drive Rankine steam cycles; options are provided for turbine extraction-driven bottoming cycles for district heating and potable water production.

Refuelling interval is 6 to 8 years for the various reactors. The entire barge-mounted plant is exchanged for a fresh one in its moorings and is returned to the factory for refuelling and maintenance.

The VBER-150, ABV, and KLT-20 concepts have been developed by the Russian design and industrial consortia having a long history of support of the Russian marine reactor programmes; they have a potential to become commercially available in the near term. Specifically, the KLT-20 is a downsized version of the KLT-40S floating plant [2]; the latter has been started in construction in the Russian Federation in June 2006 and would be deployed 2010.



1 – CPS drive	5 – Purification and cooldown system cooler
2 – Reactor	6 – Purification and cooldown system pump
3 – Pressurizer	7 – Valves
4 – Metal and water shielding tank	

FIG. 2. General view of the steam-generating unit for the ABV barge-mounted plant, see ANNEX V for details.

4.4.3. *Liquid metal cooled concepts*

The SVBR-75/100 (18) lead-bismuth cooled reactor concept is based on the Russian submarine reactor technology; it has already completed basic design and is entering the detailed design stage[4]. The near-term deployments may employ a once-through fuel cycle with uranium dioxide fuel. However, the design is flexible in fuel and fuel cycle option and can easily be accommodated to those of them that are preferred at the moment. Design versions already exist for MOX and transuranic nitride fuelling in a closed fuel cycle, and a transition strategy to such long-term closed cycle operations has been elaborated. The plants could be deployed individually or within higher capacity multi-module plants. A multi-reactor cluster configuration of the SVBR-75/100 has been designed for a 1600 MW(e) power plant.

In addition to the Pb-Bi cooled SVBR-75/100, two sodium cooled reactor concepts with a potential for nearer-term deployment have adapted established fast reactor fuel, coolant, structures, and component technologies to small sized plants of long refuelling interval, see Table 7.

TABLE 7. LIQUID METAL COOLED REACTOR CONCEPTS WITH NEARER-TERM DEPLOYMENT POTENTIAL

NAME	TYPE	ORGANIZATION; COUNTRY
SVBR-75/100 (18)	Pb-Bi cooled/UO_2 fuel	IPPE — “Gidropress”; Russian Federation
4S Toshiba Design (14)	Na cooled/U-Zr metal alloy fuel	Toshiba — CRIEPI; Japan
MBRU-12 (16)	Na cooled/UO_2-PuO_2 fuel is mentioned as basic; but UO_2 option could be considered also	OKBM; Russian Federation

These designs remain in an established sodium reactor temperature range (core outlet temperature ~500°C). They use proven fuel/cladding combinations[5] and a pumped primary circuit. All reactors drive superheated Rankine steam cycles.

The MBRU-12 has maintained a conservative approach, providing for the shuffling of fuel under a closed guard vessel cover, which could help achieve early market availability. The 4S reactor, however, incorporates a small-diameter core of high neutron leakage rate with moving reflector control of burn-up reactivity loss, as a way to assure negative sodium void worth under all conditions. The reflector in the 4S is located outside the core and the power control is executed via the feedwater control from the steam-water power circuit. Some further related R&D is required on these features (ANNEX XIV).

These concepts of sodium cooled reactors are being developed by the design teams, which include industrial partners. The designs take credit for high degree of passive response in unprotected transients without scram and employ non-traditional, low pressure containments; these features may be non-routine in previous licensing interactions in some countries and may require a departure from traditional licensing norms used historically for LWRs.

[4] On 15 June 2006, the Scientific and Technical Council No. 1 of the Rosatom of Russia supported the continuation of works for the detailed design of the SVBR-75/100 plant with a link to a certain deployment site.

[5] The 4S Toshiba Design (14) uses U-Zr alloy fuel in a once-through cycle to achieve early deployment; the 4S-LMR CRIEPI Design (15) uses U/TRU/Zr alloy fuel based on reprocessed LWR spent fuel as a source.

4.4.4. *Vendor supplied operational support services*

Table 3 in Chapter 3 contains a survey of potential customer requests for services to be provided by the nuclear energy system supplier. The potential requests vary by customer category, from isolated village to large city, and pertain not only to the power plant hardware itself, but also to operational support, to supporting fuel cycle and waste management services, and to vendor or vendor-country assistance with institutional development and technology transfer. The requests range from "power for purchase" arrangements to plant purchase contingent on technology transfer and customer participation in the construction project.

All concepts discussed in this report have a potential to satisfy a majority of the enumerated requests; only those plants that are evaluated as having a nearer-term deployment potential are described in this section.

The reactors for autonomous operation could address the electricity, potable water and heating needs of villages and towns on a "contract for services" basis, see Table 3. In this, plant emplacement, plant maintenance and operation, the refuelling and fuel cycle services could all be vendor supplied.

The ELENA (1), UNITHERM (2), and RAPID (3) are designed for or provide for an option of unattended operation/passive load follow. The ELENA and RAPID use passive thermoelectric energy conversion to generate electricity for on-site power or consumer needs; the UNITHERM can use a steam Rankine cycle. In the cases of unattended operation, remote monitoring and rapid response maintenance teams are intended to be available from centralized regional facilities.

The three near term barge-mounted concepts from the Russian Federation, VBER-150 (7), KLT-20 (8), and ABV (9), could meet the electricity and cogeneration needs of towns having tens of thousands population, towns that are water-accessible and that desire full service electricity supply on a "contract for service" or lease basis, see Table 4b.

These plants are offered as barge-mounted, relocateable power sources operated by Russian crews under power purchase arrangements with all fuel cycle and waste management conducted by the supplier. Not only cities, but also dedicated industrial applications can benefit from relocateable power supplies in the indicated range of ~10 to 150 MW(e). Land-based versions of the same plants could in a similar way meet the needs of similar customers that are water-inaccessible.

Several land-based power plants with fast reactors evaluated as having a potential of being deployed in a relatively near-term, are listed in Table 7 above. These plants are sized for towns of several tens of thousands to a hundred thousand. If successful, they could be offered for purchase with customer operation, but with arrangements for fuel cycle and waste services made by the vendor under commercial contract at the time of sale. Fuel leasing may also be an option, as the conversion ratio of a fast reactor core could be made high.

Tables 3 through 6 show numerous other concepts that are under development but appear to be more far away from commercialization status; they are discussed in the following section.

4.5. Survey of longer-term power plant concepts

The remaining concepts not yet discussed often employ significant innovation and, therefore, they may require multiple years of R&D for technology development prior to a prototype integral demonstration. In general, if new fuel/cladding/coolant combinations are employed, the deployment times can be no sooner than 15–25 years.

4.5.1. Motivation for innovative designs

The innovations include:

- New fuel/cladding/coolant combinations, such as:
  - Nitride fuel in Pb or Pb-Bi cooled reactors;
  - TRISO type particle fuel in water cooled reactors;
- Innovative system configurations;
- Use of lift or jet pumps in place of mechanical pumps;
- New energy conversion systems, such as Brayton cycles;
- New passive safety design approaches with non-conventional containments;
- New autonomous load follow control approaches with reduced staffing.

Given that nearer-term concepts already exist, the motivation for further innovations and R&D expenditure is to enhance performance in one or more aspects, such as:

- More efficient use of fuel resources by closing the fuel cycle;
- Improved environmental performance by closing the fuel cycle;
- Reduced capital cost;
- Reduced staffing and operation and maintenance (O&M) cost;
- Higher levels of operational reliability and safety; and
- Improved proliferation resistance.

These enhanced features are intended to help nuclear energy secure a significant and growing role in long-term global energy supply.

4.5.2. LWR derivative power plants

Three of the water cooled reactor concepts, PSRD (4), MASLWR (5) and Package-Reactor (6), are downsized light water reactors (LWRs) using traditional uranium dioxide fuel and operating on natural circulation, see Table 4a.

The PSRD reactor at 31 MW(e) is a natural circulation pressurized water reactor (PWR) derivative with an integral design of the primary circuit, hosting steam generators and a pressurizer. The design uses conventional UO_2 pellet type fuel with less than 5 % enrichment by ^{235}U, allowing to achieve continuous core operation within 5 effective full power years. The design objective is to achieve system simplification resulting in a reduction of costs for the construction, operation, and maintenance. No chemical and volume control is used during power operation. The PSRD is being designed to produce electricity and incorporates a modular approach allowing for incremental capacity increase of a multi-module plant.

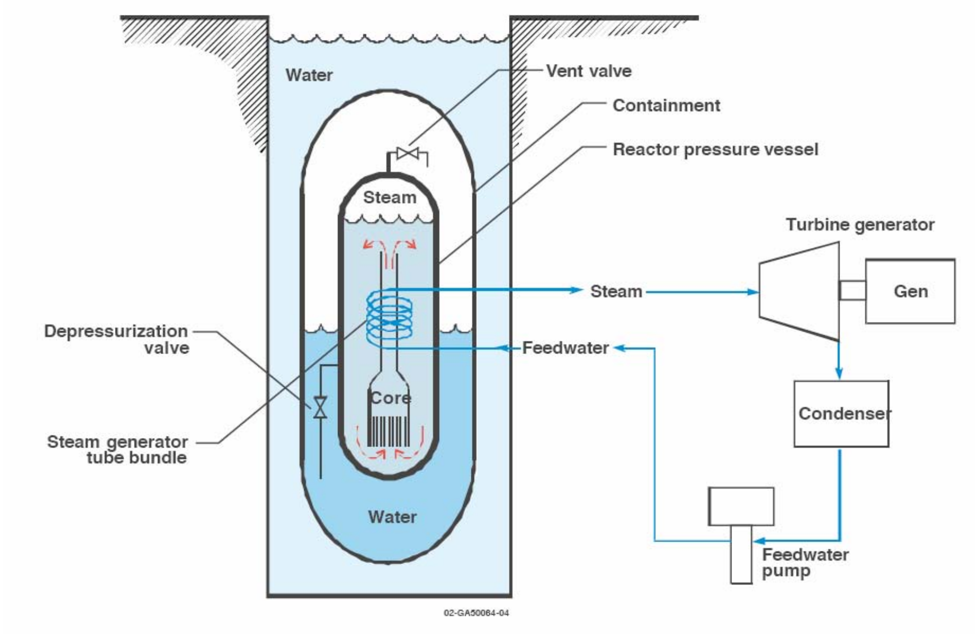


FIG. 3. Simplified diagram of the power cycle for MASLWR, see ANNEX I for details.

The MASLWR reactor at 35 MW(e) is a natural circulation PWR derivative with integral steam generators and cogeneration capability, see Fig. 3. Each 35 MW(e) plant is modularized for delivery in three shippable components: reactor, balance of plant, and condenser. It is intended that multiple MASLWR power plant modules of 35 MW(e) each will be clustered to form a multi-module power plant of up to 1050 MW(e) total output. For refuelling, the reactor components themselves are taken out and returned to the factory on a 5-year interval.

The Package-Reactor is a natural circulation BWR/PWR/CANDU derivative at 6–8 MW(e) per module power. Individual encapsulated "cassette" fuel assemblies (coupled neutronically) boil water under natural circulation; the steam is piped from each cassette to the in-vessel secondary loop steam generators, which drive the Rankine cycle energy conversion, see Fig. 4. The Package-Reactor incorporates a bottoming cycle cogeneration capability, using a chemical heat pipe concept for long-distance heat transport. Reactivity control is performed using control rods located between encapsulated fuel assemblies. Refuelling on a 5 to 10-year interval is accomplished by removal and replacements of the fuel "cassettes". The Package-Reactor concept provides for the option of incremental plant capacity increase.

All of these LWR-derivative concepts rely on traditional uranium dioxide fuel clad in Zr alloy. However, heat transport and refuelling are non-traditional. The designs are either at a feasibility study or conceptual design stage. Several years of further R&D will be required prior to licensing and commercialization of these concepts, but they have a potential to be available for deployment sooner than the other longer-term concepts.

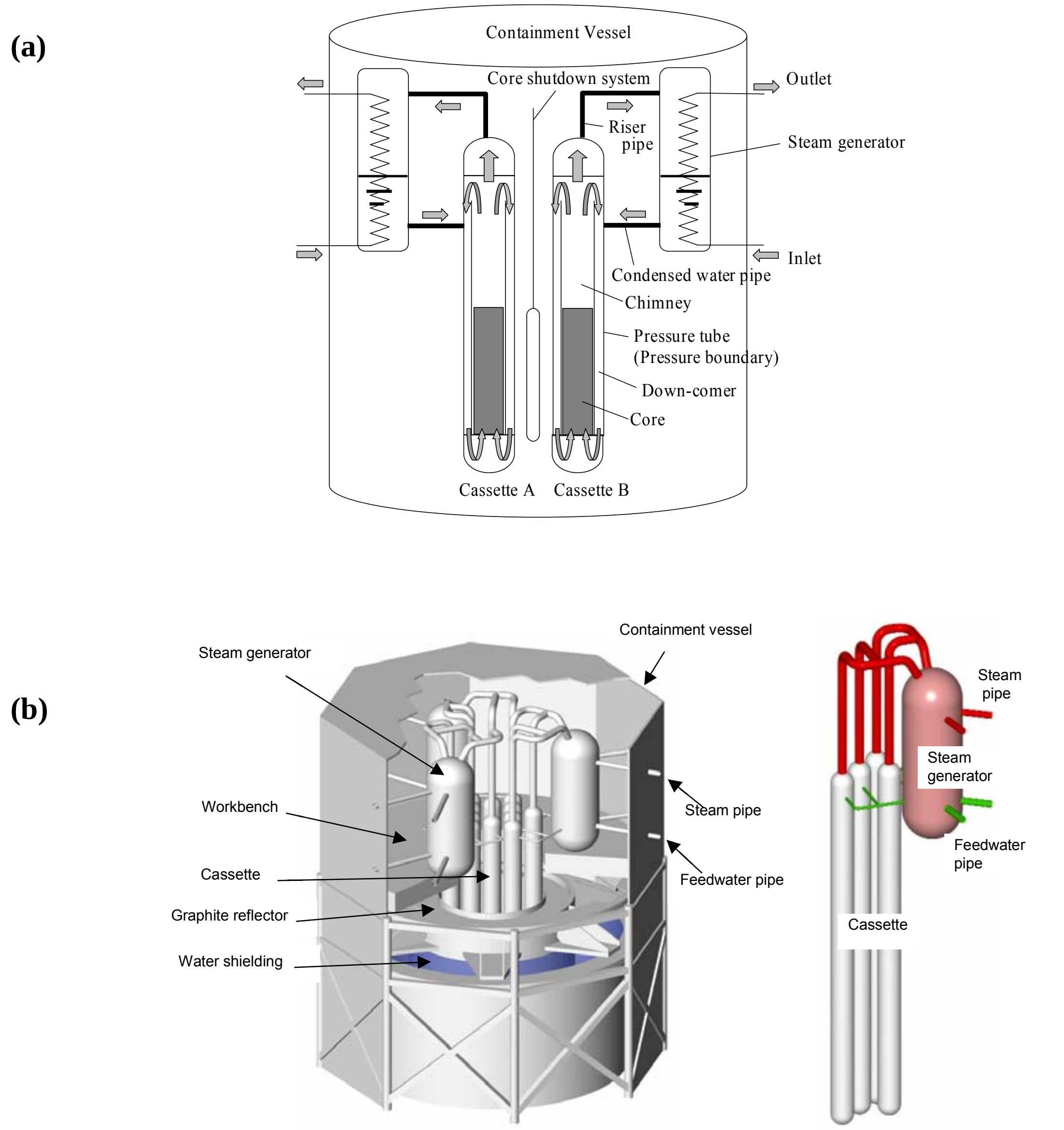


FIG. 4. Nuclear power generation unit (a) and configuration of multi-cassettes/single steam generator (b) in the Package-Reactor, see ANNEX VIII for details.

4.5.3. Small LWRs with TRISO fuel

Several of the water cooled concepts, PFPWR50 (10), VKR-MT (11), AFPR (12), and FBNR (13), adopt TRISO fuelling as a strategy to simplify their safety approach as compared to traditional LWR engineered safety system approaches. The goal is to exploit the robustness of the TRISO fuel to avoid loss of fuel integrity even in beyond design basis events. This may help simplify plant layout, reduce cost and strengthen passive safety performance for near-urban siting. All designs in this category are at an early stage of development; however, comprehensive corrosion tests were performed and in-pile irradiation testing is being performed for the Russian VKR-MT.

The PFPWR50 is a small PWR with proposed ‘mild’ variant of the use of HTGR type TRISO fuel within graphite columns packed in Zr-alloy claddings of conventional LWR type fuel elements. It uses a hexagonal lattice that is tighter than the square lattice of conventional

PWRs. The PFPWR50 is a dedicated reactor for district heating; its concept requires further R&D before commercialization.

The highest in power rating of all small reactors described in this report is the 890 MW(th) VKR-MT concept. It is a BWR based cogeneration plant providing up to 300 MW(e) of electricity and up to 600 MW(th) of heat. The VKR-MT draws on the Russian VK-300 vessel type BWR technology [1]; however, the fuelling is unique. Fuel assembly cages are filled with micro-fuel elements (MFE) — coated particles of 1.8 mm outer diameter with SiC outer coating layer. The MFE containing fuel cages are cooled by water cross flow and in-core boiling, see Fig. 5. The cages are refuelled in situ, without opening the reactor head, every ten years using hydraulic transfer of MFE into and from refuelling canisters brought to the site. With the use of comprehensive burnable absorber scheme, the design achieves a 10-year interval of operation without reloading or internal movement of fuel. Safety calculations confirm that the goal for no loss of fuel integrity for selected beyond design basis accidents is attainable (ANNEX X).

The AFPR concept of 100 MW(e) is generally similar to the VKR-MT in overall design approach, but suggests the use of continuous in-vessel transport of micro fuel elements to achieve a 36-year operation cycle without refuelling. The concept is in two versions, a BWR and a direct flow system with superheated steam at core outlet.

The FBNR uses TRISO particles within SiC-coated spherical fuel elements in an up-flow coolant stream, which, if interrupted, allows the particles to relocate into a subcritical, well-cooled configuration.

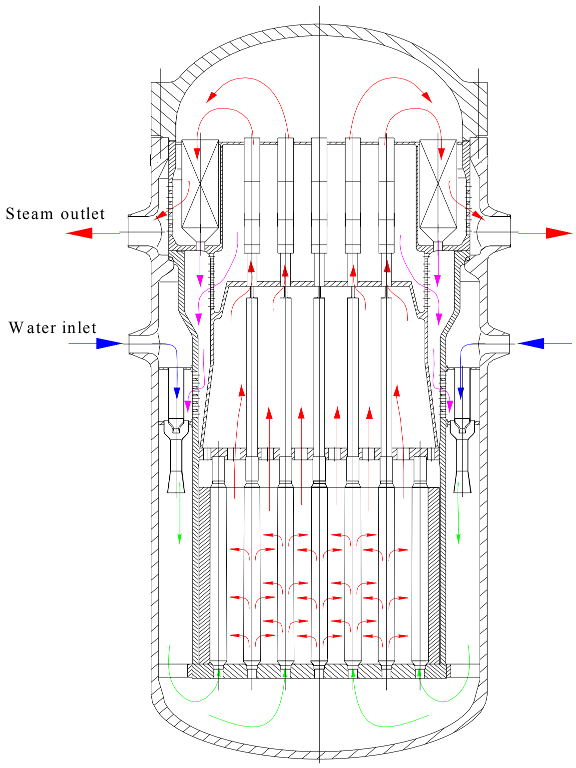


FIG. 5. Scheme of coolant circulation inside the VKR-MT reactor vessel, see ANNEX X for details.

4.5.4. Concepts of liquid metal cooled reactors

Of all small reactors without on-site refuelling in this report, about half are liquid metal cooled reactors with fast neutron spectrum; they include the following concepts (see Table 5):

- Sodium cooled;
- Lead-bismuth cooled, using either traditional fuel, i.e., MOX or metal alloy fuel, or nitride fuel in steel cladding at moderate (~550°C) temperatures; and
- Lead cooled, using nitride fuel at somewhat higher (~600°C) temperature.

The sodium cooled concepts are based on proven fuel/cladding/coolant combinations (dioxide or metal alloy fuel in steel cladding) in a traditional temperature range, and rely on the proven reactor, fuel cycle, and balance of plant technology[6]. However, these concepts introduce inherent and passive safety features, use space between the reactor and guard vessels to accommodate an intermediate sodium heat transport system, and incorporate non-traditional low volume containments in their safety strategy; therefore, they may require adaptations of prior LWR-based licensing norms. Most of the concepts foresee the operation in a closed nuclear fuel cycle. Under adequate financing, they could be ready for a prototype plant construction within 10–15 years.

All of the lead-bismuth cooled concepts are being designed to operate in the traditional temperature range of the Russian submarine experience, reflecting on the SVBR-75/100 (18) design. While the near-term option for the SVBR-75/100 is a uranium dioxide once-through fuel cycle concept, the closed cycle version of the SVBR-75/100 is designed to operate with MOX fuel with an option for nitride fuel (ANNEX XIX). All others rely on either metallic alloy fuel or nitride fuel, and in either latter case, as the fuel/cladding/coolant combination is new, fuel testing will be required. Refuelling is done either by entire reactor module change-out or by whole core cassette/sub-assembly change-out; the refuelling interval varies by concept from 4.5 to ~30 years. The lead-bismuth cooled designs use pool layouts; mechanical pumps drive the primary coolant; or, in the ENHS (19) and Small Lead-bismuth Cooled Reactor (21), natural circulation in the primary circuit at full power is used. These latter two designs employ passive load follow, i.e., semi-autonomous control.

Except for the BN GT 300 (17), all sodium and lead-bismuth cooled reactor concepts use a Rankine superheated steam cycle, and a number of them provide options for the extraction turbines to support seawater desalination, district heating, or process heat cogeneration. The BN GT 300 (17) uses a gas-turbine cycle.

The SSTAR (24) and STAR-LM (25) lead cooled reactor concepts are based on nitride fuel and use a higher core outlet temperature to drive a supercritical CO_2 Brayton cycle at 550 to 600°C, with a potential to gain energy conversion efficiencies of ~43% at these temperatures. Moreover, the outlet temperature on the cool side of the recuperator can lie in the range of 70 to 125°C with only weak influence on the efficiency. As the inlet to the compressor is just above 31°C, these conditions facilitate installation of bottoming cycles for district heating, seawater desalination, or process heat production, using the heat otherwise rejected in thermodynamic cycle (see Annexes XXII and XXIII). The supercritical CO_2 Brayton cycle lacks an industrial experience base; this non-conventional Brayton cycle will require R&D.

The PBWFR (23) is a non-conventional direct contact boiling water reactor with lead bismuth cooled core. Water is distributed into hot lead-bismuth above the core, and the generated bubbles act as a lift-pump to facilitate natural circulation of the lead-bismuth coolant. The

[6] An exception is the BN GT-300 (17) sodium cooled reactor concept of 300 MW(e), which couples a sodium-cooled reactor with gas-turbine Brayton cycle for electricity generation; it also eliminates intermediate heat transport system (ANNEX XVIII).

thermodynamic cycle is direct, and the feedwater system is based on forced convection. Such combination of a lift pump approach and a direct contact steam production is being considered primarily for reasons of plant economy; it is backed by a substantial amount of thermal-hydraulic testing (ANNEX XXVII).

All of the fast spectrum concepts provide for or do not exclude the operation in a closed fuel cycle, with their high conversion ratio providing a pathway for growing deployments of small reactors without on-site refuelling, even as the economically affordable uranium ore resource base is drawn down over future decades.

4.5.5. High temperature concepts including hydrogen production

The more future-oriented concepts of small reactors without on-site refuelling are those designed for core outlet temperatures of ~700 to 1000°C, see Table 6.

All of these concepts provide for at least partial use of generated high temperature heat for process applications, such as hydrogen production. They retain small size and long refuelling interval characteristics that may satisfy customer needs in developing countries, but also might conform to the future “merchant” plant financial conditions in developed countries, by offering small initial capital outlay and short payback period.

The various high temperature concepts propose three coolants - molten salt; Pb or Pb-Bi; and gas. The fuel is nitride for fast spectrum concepts (BGR-300 (28) suggests the use of advanced porous fuel) and TRISO derivative for thermal or intermediate spectrum concepts. Two of the 5 concepts are reactors intended for operation in ^{233}U-Th fuel cycle.

All of these concepts require substantial R&D on both the reactor and the balance of plant, including systems for high temperature process heat applications. Licensing norms matching the design specifics of such reactors will need to be developed and emplaced.

4.5.6. Remaining R&D and time to first deployment

Concepts based on downsized commercial PWR technology

The concepts based on well-established coolant/fuel/cladding technology packaged for small scale power plants include:

- PSRD (4), based on PWR technology (JAEA, Japan);
- MASLWR (5), based on PWR technology (Idaho National Laboratory, USA); and
- Package-Reactor (6), based on PWR/BWR/CANDU technology (Mitsubishi Heavy Industries/Hitachi, Japan)

While the reactor core technology is proven, the heat transport and plant layout selected for these concepts are quite innovative. The concepts have neither completed final design nor been submitted for licensing; they are up to a decade into the future before prototype construction and subsequent commercialization.

Concepts based on innovative combinations of fuel, cladding, and coolant

Most of the longer-term concepts incorporate new fuel/cladding/coolant combinations; for example, they suggest:

- The use of TRISO type particle fuel in LWR:
  - PFPWR50 (10), within the compacts substituting conventional pellets in conventional Zr-alloy claddings (PWR);
  - VKR-MT (11), as a pebble bed (BWR);
  - AFPR (12), as a movable pebble-bed (BWR or direct flow reactor);
  - FBNR (13), within relocateable bed of spherical fuel elements (PWR); or
- The use of nitride fuel (often enriched by ^{15}N) in Pb or Pb-Bi cooled reactors:
  - SVBR-75/100 (18), longer-term version;
  - Small Lead-bismuth Cooled Reactor (21);
  - ENHS (19);
  - LSPR (20);
  - SPINNOR/VSPINNOR (22);
  - STAR-LM (25); and
  - PBWFR (23).

It is clear that in light of the in-pile irradiation testing time lags, those concepts that rely on new fuel/cladding/coolant combinations will require 10–15 years of in-pile fuel testing before a prototype demonstration is feasible and before a fuel irradiation testing database is sufficient for licensing interactions to begin. When recycle and waste form production R&D is included, even additional time may be required before the entire fuel cycle for the concept, including fuelling with the recycled fuel, can be proven.

Very high temperature concepts

Very high temperature concepts, see Table 6, are motivated by a goal to broaden nuclear energy's role by mid century; moving from electricity and low temperature process heat applications into high temperature non-electric markets.

The high temperature concepts would require development and confirmation of non-traditional structural materials as well as new chemical processes, new types of fuel and new fuel/cladding/coolant combinations at challenging temperatures. Addressing safety issues of nuclear plant/chemical plant co-location would require extensions of safety approaches and licensing norms.

All the high temperature concepts will require at least a decade and a half or two before commercialization; thermal spectrum concepts using TRISO fuel and molten salt or gas coolant could reduce this time by drawing on a partial experience base from previous programmes conducted in the 1950s through the 1970s.

4.6. Survey of power plant design approaches

All concepts of small reactors without onsite refuelling are based on small power rating with factory fabrication of modularized components, rapid field assembly of the plant, long refuelling interval, and assume providing a very high level of safety to support siting near population centres. The design approaches that have been taken to achieve these features vary significantly according to:

- Neutron energy spectrum;
- Fuel and coolant choices;

- Energy converter choice;
- Targeted commercialization date; and
- Vendor operational support offerings.

Any given design attempts to achieve an optimal trade-off among numerous, often conflicting considerations based on unique priority ranking applied by each design team. A detailed description is presented for each of the concepts in the Annexes I through XXX, where the rationale for the choices is explained by each designer.

The purpose of the following sections is to identify and discuss common and contrasting approaches used to meet the fundamental features of small reactors without on-site refuelling.

4.6.1. Influence of targeted commercialization date on design choices

The hierarchy used for grouping of the concepts listed in Tables 3 through 6 derives from their projected time to market. These time-to-market goals have very often influenced design strategy decisions, sometimes in non-apparent ways. For example, the Russian icebreaker derivative designs presume an early deployment date because small reactor deployments in the Russian Far North are needed as soon as available, and they presume a trained Russian crew will be operating the plant because the Russian law requires all nuclear plants in the Russian Federation to be operated by Rosenergoatom personnel. Therefore, for these Russian marine-derivative early-deployment concepts:

- Proven design solutions are favoured; or
- The technological extensions from prior practice are limited; and
- These are the consortia of designer, scientific advisor and industrial firms with a history of cooperation that are offering the plant, using the existing infrastructure.

Therefore, forced circulation and proven fuel at conservative discharge burn-ups, conventional temperatures and pressures are preserved, and Rankine steam cycle energy conversion equipment is used. Moreover, as trained Russian crews will be operating the plants in any case, design goals to reduce staffing numbers and skill level have less weight on design decisions.

Similarly, the near-term sodium cooled fast reactor concepts stick with well proven fuel/coolant/cladding combinations and heat transport components operating in the traditional temperature range as a way to enter the market relatively quickly.

Alternatively, concepts having deployment target dates further out in time tend to "push the technology envelope" and employ more aggressive temperatures, new types of fuel and materials, new energy converters and less proven heat transport approaches, etc. The motivation is to reduce cost, and to take steps toward autonomous operation for reduced operating staff size and skill level. Many of these concepts assume the operation in a closed fuel cycle to ensure better fuel utilization and further minimization of environmental impacts. Further substantial R&D will be required to achieve these gains.

In light of the above, in the following discussions of common and contrasting features and design approaches, appropriate attention is given to distinctions between well established versus still "paper" design solutions.

4.6.2. Approaches to achieve long refuelling interval

Long refuelling interval is one of the defining features of small reactors without on-site refuelling; *inter alia*, it is an approach to enhance energy security for the customer while at the same time facilitating higher levels of non-proliferation assurances by reducing the motivation for developing an indigenous nuclear fuel cycle infrastructure.

The two issues, which dominate design approaches for long refuelling interval, are (i) fuel and structure endurance to the in-core environment, i.e., achievable discharge burn-up under the conditions of long-life core operation; and (ii) reactivity loss with fuel burn-up. The latter is mainly a concern for the designers of fast reactors targeting passive compensation of burn-up reactivity (the so-called 'zero' burn-up reactivity swing) or the designers of very small thermal spectrum reactors aimed at operator-free, autonomous operation[7]. One or the other of these two issues ultimately limits energy delivered per unit mass of fuel loading.

Design approaches for extending refuelling interval differ according to the choices for fuel and for the neutron spectrum. The reactor concepts described in this report employ four major fuel systems:

- Traditional LWR UO_2 pellet fuel in Zr-alloy cladding for thermal spectrum reactors;
- Fast reactor fuel of thorium, uranium, or uranium/transuranic (TRU) oxide and nitride or of uranium/TRU/Zr metal alloy; all in stainless steel claddings;
- Particulate UO_2 fuel kernels in a cermet configuration with Zr alloy cladding for thermal neutron spectrum reactors; and
- TRISO type UO_2 fuel as a bed of micro fuel elements in direct contact with coolant or in HTGR-type pebble or compact configuration for thermal spectrum reactors.

These different fuel systems are capable of very different discharge burn-ups, controlled by very different physical phenomena.

The last two fuel systems are capable of extremely high fission fraction of the initial fissile mass loaded in the fuel kernels, approaching 90%. However, particle volume fraction in the fuel form is below 50–70%, reducing fissile loadings below that needed in fast reactors. Therefore, these types of fuel are suggested for use in thermal reactors. For those thermal reactor designs with long refuelling interval that use these fuel types, the strategy is to load enough ^{235}U into the core, sufficient to produce the required number of fissions, i.e., energy in MW(th)-day, and to employ burnable absorbers for partial compensation of the resulting reactivity loss. Increased enrichment is used, ranging from 10 to ~20%, and little reliance is placed on production and in-situ burning of Pu from neutron capture on ^{238}U. However, for application of these types of fuel in commercial designs, the designers self-impose a limit on initial ^{235}U enrichment of ≤20% by weight[8]; this in effect sets an upper bound on available energy extraction per kilogram of the initial heavy metal (IHM) inventory. Even though this limit may be smaller than the fuel's innate endurance limit, in effect it places a bound on discharge burn-up (MW(th)-day/kg IHM), the one imposed by safeguards considerations rather than by fuel degradation phenomena.

---

[7] Conventional LWRs of medium-to-large capacity provide no option for full passive compensation of reactivity changes resulting from fuel burn-up, even if burnable poisons are used; the 'remaining' reactivity changes vary between ~10 and ~15%ΔK/K and are compensated by moving the mechanical control rods or changing the liquid boron concentration in the coolant, with necessary reactivity margin being provided at the beginning of operation cycle.

[8] The choice of 20% as a ^{235}U enrichment limit is guided by the IAEA supported recommendation of 20% as the lower enrichment boundary for direct use materials [3].

Thereafter, a relatively straightforward three way design trade-off exists among the reactor power level, the refuelling interval (given a capacity factor), and the mass of fuel initially loaded. In other words, given a specified core power rating, longer refuelling interval requires more fuel to be loaded. The net effect is a reduction in specific power (kW(th)/kg IHM) compared to that attainable absent a limit on enrichment.

Many water cooled reactor concepts use conventional uranium dioxide LWR type fuel. The fuel pin endurance limit for UO_2 pellet fuel in Zr-alloy cladding lies in the range below ~55−60 (MW(th)-day/kg IHM). This fuel type is not constrained by particle packing fraction as is cermet or TRISO type fuel; i.e., high fuel density is possible and, therefore, the enrichment is generally lower, ranging from 3 to 8%. Initial ^{235}U is essentially completely consumed, and plutonium production with in-situ fission contributes significantly to energy delivery prior to fuel discharge. This fuel type experiences several life limiting morphological degradation phenomena around 55−60 (MW(th)-day/kg IHM) burn-up, such as (i) "rim structures" of high porosity leading to intensive fission gas release under the cladding; (ii) fast-fluence induced cladding degradation; and (iii) formation of brittle zirconium hydride platelets in the cladding [4]. Therefore, again the discharge burn-up (MW(th)-day/kg IHM) is a constraint, and, given a reactor power rating, the extension of refuelling interval relative to standard LWRs is achieved by initially loading more fuel per MW(th), with concomitant reduction of the specific power (kW(th)/kg IHM). The discharge burn-up remains nearly unchanged, the enrichment is modestly increased, and the burn-up reactivity loss is compensated in a conventional way, by control rods and burnable poisons.

The situation is quite different for fast neutron spectrum reactors where burnable poisons do not exist but excellent neutron economy allows for plutonium breeding to be conducted internal to the core lattice itself to produce new fissile materials that can be bred in-situ to completely compensate reactivity loss with burn-up.

The traditional fast reactor types of fuel are constrained in their achievable discharge burn-up by fast neutron fluence damage to claddings. Whether oxide or metal alloy, they have a well demonstrated average burn-up capacity in the range of 100 to 120 (MW(th)-day/kg IHM), with a small database available for burn-ups as high as 200 (MW(th)-day/kg IHM). Nitride fuel irradiation testing data is relatively sparse but suggests that ~100 (MW(th)-day/kg IHM) average burn-up can be reached. Therefore, again for any given power level the design strategy to extend reload interval relative to standard fast reactors is to load more fuel, effectively derating the fuel specific power (kW(th)/kg IHM) and the core power density (kW(th)/litre). In order to attain a 20-year refuelling interval, power densities for fast reactors drop from their traditional range of several hundreds of kW(th)/litre down into the range of traditional LWRs, i.e., ~100 kW(th)/litre.

In designing extremely small fast reactors for remote villages, a neutronics-driven trade-off — the one absent for thermal reactors — may constrain the degree to which reactor power rating can be reduced. In order to compensate burn-up reactivity loss by internal breeding, two conditions are required: (a) the ^{238}U content in the core lattice itself must remain high (initial enrichment $\lesssim$ 15% by weight), and (b) the neutron economy must remain high. But as reactor power rating is reduced and the reactor becomes physically smaller at a given power density, neutron leakage increases thereby requiring an enrichment increase to sustain the chain reaction. Both trends — leakage and enrichment — diminish the fast reactor's internal breeding; one finds that ~50 MW(th) is the lower limit on reactor power rating, below which one can no longer compensate burn-up reactivity loss with internal breeding. In order to reduce power rating still further, active control rod compensation becomes a necessity.

In contrast, in thermal reactors within the power range above 10 MW(th) there is no possibility to compensate burn-up reactivity loss by passive means only, even with the use of burnable absorbers, because of a low conversion ratio inherent to thermal spectrum cores[9]. In thermal reactor designs below 10 MW(th), self-compensation of burn-up reactivity loss by temperature reactivity effects plays an increasing role, which explains why the designers of autonomous plants go to very small power rating in the <10 MW(th) range.

A second consideration, which is peculiar to the fast reactors, exists. Some licensing authorities may simply forbid fast reactor designs to exhibit a positive coolant void worth, even when the overall power coefficient of reactivity is negative. For deployments of fast reactors in countries where that would be the case, two options are available:

- To spoil the neutron economy to increase leakage; to give up on internal conversion ratio and go to active control systems to compensate burn-up reactivity loss; or
- To change the fuel from TRU/^{238}U to ^{235}U or ^{233}U in ^{238}U or ^{232}Th.

As an example, the first choice has been made for the 4S-LMR CRIEPI Design (15); while the second choice is made for the near-term configurations of the SVBR-75/100 (18).

In summary, whether for fast or for thermal neutron spectrum small reactors without on-site refuelling, the fuel discharge burn-up and the irradiation of core structures never exceed standard practice from the conventional designs. The refuelling interval is being extended by derating specific power (kW(th)/kg fissile). Power densities never significantly exceed ~100 kW(th)/litre and often are much lower. Burn-up reactivity loss is mitigated by using burnable poisons and active control rods in thermal systems and by designing for internal breeding in fast systems.

Alternative to what was discussed above is continuous reactor refuelling or uploading performed without opening the reactor vessel head, as suggested by some concepts of reactors with relocateable pebble bed of spherical fuel elements, e.g., the AFPR (12) or the FBNR (13). Generically, such an approach may result in higher unit power of a reactor without on-site refuelling — a trend to be explored further.

4.6.3. Approaches to reduce operating crew size and required skill level

In most remote villages and in some developing countries the targeted customer base for small reactors without on-site refuelling may initially be deficient in an indigenous skilled workforce, or some customers may aspire to reduce operating costs by means of reducing staff size. Most concepts discussed in this report have sought to reduce demands on indigenous work force at the reactor plant without compromising safety.

An extremely broad range of approach has been taken. The Russian icebreaker reactor derivatives — VBER-150 (7), KLT-20 (8) and ABV (9) — presume that the plant will be operated by skilled Russian crews who live on the barge-mounted plant or in the village where a land-based plant is emplaced. The crew is highly skilled, and is sized for operating a relatively complex plant. At the other extreme, certain of the very low-power autonomous plants are designed for unattended operation and for passive load follow. This is the case for the dedicated district heating reactor ELENA (1) and also for the smallest cogeneration plants, the marine derivative UNITHERM (2) of 2.5 MW(e) and the moon-base reactor derivative sodium cooled RAPID (3) of 1 MW(e). Water cooled small autonomous reactors are natural

[9] High conversion ratio can be reached in some thermal spectrum cores with ^{233}U-Th fuel, e.g., modified LWRs or HTGRs.

circulation cooled and the ELENA and the RAPID use passive thermoelectric energy conversion requiring little maintenance.

All other concepts lie in between these extremes, and they all employ one or more of the following approaches to simplify plant operation so that crew size and/or skill level could be reduced without compromising safety:

- Reducing equipment count and associated maintenance requirements, for example:
  - By simplification of energy conversion equipment, use of Brayton cycles, passive thermoelectric converters; or
  - By simplification of the primary circuit, e.g., via the use of natural circulation;
- Simplifying control requirements and required skill levels, for example, by allowing a passive load follow mode (semi-autonomous operation);
- Removing all safety functions from the balance of plant and reducing the number of engineered safety systems of the nuclear steam supply system, to reduce necessary skill levels of the maintenance work force;
- Increasing the role of inherent and passive safety features by design, as comes to both decay heat removal and innate feedbacks to keep power/flow in balance with heat removal, to reduce necessary number of the operating personnel.

In general, all concepts could benefit from remote monitoring of plant performance with rapid response specialty maintenance crew support provided from centralized multi-plant support centres.

4.6.4. Approaches to improve heat transport and energy conversion efficiency

All small reactors without on-site refuelling have given up economy of scale benefits at the outset and seek to benefit instead from mass production of standardized, modularized plants in a factory, and from rapid site assembly. But even more can be done on the plant itself to lower capital cost by other means. One such means is improved conversion efficiency per unit of capital cost in the balance of plant.

Almost all of the water cooled concepts use a Rankine steam cycle with saturated or slightly superheated steam for energy conversion. The energy conversion efficiency has a maximum of ~33% based on reactor core outlet temperatures from 270 to 345°C.

The nearer-term sodium cooled reactor concepts — 4S Toshiba Design (14) and MBRU-12 (16) — and the Pb-Bi cooled SVBR-75/100 (18) employ conventional core outlet temperatures in the range from 480 to 510°C and drive superheated Rankine steam cycles attaining conversion efficiencies near 39%. The number of loops transporting heat to the balance of plant never exceeds two.

For these nearer-term plants, high priority has been placed on being first to the market; therefore, traditional heat transport conditions and balance of plant designs are used. They rely on well-proven heat transport components (pumps, steam generators, pressurizes, etc.).

On the other hand, for the concepts having longer-term commercialization targets, alternative heat transport and energy conversion approaches are being considered. The incentives for their application are as follows:

- To simplify and reduce capital and/or operating cost of heat transport;
- To improve energy conversion efficiency per unit of capital cost;

- To simplify energy conversion equipment to reduce staffing and skill requirements;
- To generate or co-generate alternative energy products from nuclear heat.

In the area of simplification of heat transport, chemical compatibility among heat transport working fluids (e.g., Na and Ar-N; Pb-Bi and steam or CO_2) is introduced so that an intermediate circuit is eliminated and integral (in-vessel) steam generators or heat exchangers can be employed, e.g., like in the BN GT-300 (17), SSTAR (24), STAR-LM (25), and STAR-H2 (29). Natural circulation of the primary coolant at full power is used for many concepts. In the lead-bismuth cooled ENHS (19), a gas lift pump option is considered as an alternative to either natural circulation or mechanical or electromagnetic pumps. Similar approach, coupled with direct contact production of steam is employed in a lead-bismuth cooled PBWFR (23).

In the area of heat engines, Brayton cycles are under study. The MARS (27) high temperature concept proposes an open cycle air-turbine. The STAR concepts (24, 25, 29) consider a supercritical[10] CO_2 Brayton cycle, which can theoretically reach conversion efficiencies of about 43% at core outlet temperatures of ~500−530°C, "traditional" for sodium and lead-bismuth cooled reactors. Since the Brayton cycle rotating machinery is smaller and the component count is smaller than for the Rankine cycle, the targeted result would be higher conversion efficiency at lower capital cost as well as smaller required operating crew and skill level to achieve reduced operation and maintenance costs.

Low-efficiency passive thermoelectric energy conversion is considered for the two unattended very small plants for remote outposts, ELENA (1) and RAPID (3), and for the high temperature concept, CHTR (26). In all three cases the electricity generation is optional, just to serve on-site needs.

4.6.5. Approaches to facilitate near-urban siting

Small reactor concepts are attractive to be sited near population centres because:

- They could supply energy to cities in regions where only a local electrical grid exists,
- They could produce energy products such as potable water and district heat, which cannot be transported to significant distances; and
- In industrial cogeneration applications, they must be sited adjacent to the industrial site for delivery of process heat.

These siting considerations lead to a requirement for very high levels of safety and reliability.

Protection of population from consequences of accidents resulting from internal initiators relies on traditional defence in depth strategies. However, in addition to active safety systems, nearly all concepts reinforce the first and subsequent levels of the defence in depth by broad incorporation of inherent and passive safety features into design concept. The goal is to eliminate as much accident initiators as possible, with the remaining part then being dealt with by appropriate combinations of active and passive systems. The expected result is a highly-assured level of safety response through reliance on innate laws of physics to maintain reactor heat production and heat removal in balance (e.g., via optimum combinations of reactivity feedbacks combined with natural circulation) and on innate laws of physics to assure decay heat removal (e.g., via a natural circulation heat removal channel to the ultimate heat sink).

---

[10] CO_2 critical state is at T = 31°C and P=7.1 MPa.

In the case of many water cooled reactor concepts, the loss of coolant hazard continues to be addressed in the traditional ways of water injection and robust containment; however, compactness is maintained by trading off reduced containment volume with increased containment pressure rating (e.g., the rating of the KLT-20 (8) containment is 1 MPa). Other designs introduce innovative features; for example, the PSRD (4) immerses the vessel in a guard vessel filled with water; integral type PWRs place the steam generators inside the vessel to reduce water pipe rupture hazards; some arrange for passive low pressure injection of make-up water. Finally, several water cooled reactor concepts are seeking to exploit the robustness of TRISO fuel to dramatically ameliorate dependence on engineered safety systems and, perhaps, to improve public acceptance of near-urban plant siting via the concept of "each fuel element having its own containment".

In the case of fast neutron spectrum reactors, the efficacy of inherent and passive safety features to anticipated transients without scram (ATWS) has been well established by testing in the EBR-II, FFTF, and Phoenix reactors [5 and 6]. So too has passive decay heat removal. The severe accident hazard of reactivity addition upon core rearrangement in small fast reactors is addressed by a range of approaches including:

- Assured passive termination of unprotected accident sequences prior to fuel damage;
- Avoidance of a positive coolant void coefficient of reactivity; or
- Precluding prompt criticality events by achieving fuel dispersal at a low value of energy deposition in the fuel.

While LWR type thermal reactors using uranium dioxide fuel of low enrichment do not experience a reactivity insertion hazard due to compaction, it is noted that some small reactor concepts with thermal neutron spectrum that are based on cermet or TRISO fuel at enrichments approaching 20% might face fuel reconfiguration reactivity addition issues, which requires further examination.

4.6.6. Approaches for protection against external hazards

In a world of natural and human initiated vicissitudes, nuclear power plants are to be configured to protect against external events such as high winds/missiles, earthquakes, floods, fires, aircraft crash [7] and also purposeful attacks such as airplane attack or rocket or tank shells.

As on-site fuel storage and fuel cooling do not take place for small reactors without on-site refuelling, the need for protecting spent fuel pools or fresh fuel storages does not arise. Protection of fuel shipments could generally follow traditional practice; additionally, Pb and Pb-Bi cooled reactor concepts suggest to transport cores or cassettes of fuel encased in frozen Pb or Pb-Bi.

The design approaches adapted for all small reactors without on-site refuelling fit well with the imperative for siting near population centres; they intend to achieve an exceptionally robust safety posture with respect to initiating events both internal and external and combinations thereof.

For protecting a power plant itself, the plant hardening design approaches are intended to preclude an external event from becoming an accident initiator and, separately, to protect the fissile material from being easily stolen. Hardening of the reactor building or containment has been considered. For example, the Russian barge-mounted plants are designed to operate under harsh conditions and employ extremely rugged layouts and protection from external hazards, Fig. 6. Many of the land-based plants are on seismic isolation pads and most are emplaced in silos. As an example, Fig. 7 illustrates the silo emplacement/protection berm design strategy for the STAR (24, 25, and 29) concepts. As discussed above, all concepts incorporate inherent and passive safety features and/or increased safety margins targeted to achieve an extremely high level of safety response under both internal and external impacts and combinations thereof, even in the event of failure of active safety systems with no external power and water supply and operator intervention over a long period, reaching several days or more.

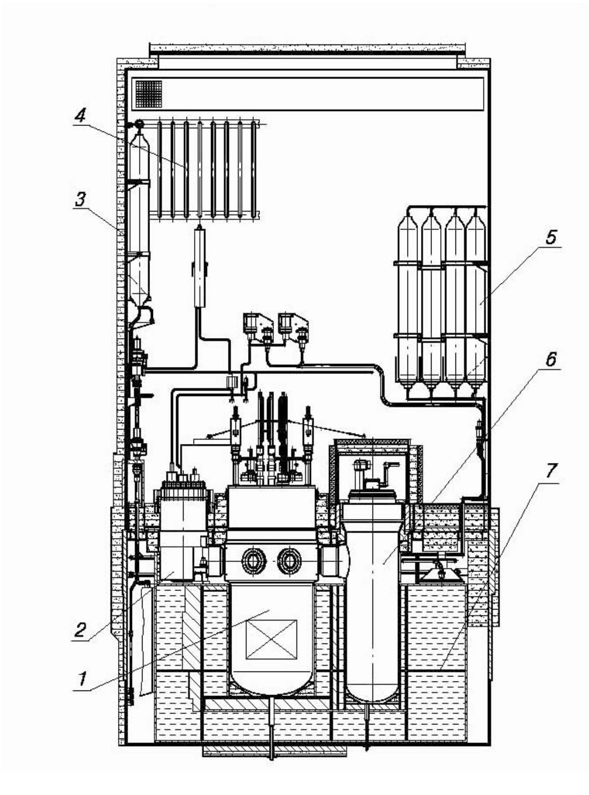
Some of the concepts adopt an autonomous or semi-autonomous passive load follow feature such that no conceivable combination of multiple equipment and human errors occurring outside the reactor vessel could lead to core damage. As an example, this feature is incorporated in both very small reactors for autonomous operation (Table 3), as well as in the higher-powered Na cooled 4S (14 and 15) and the Pb-Bi/Pb cooled ENHS and STAR (24, 25, and 29) reactor concepts.

The longer-term concepts of water cooled reactors — PFPWR50 (10), VKR-MT (11), AFPR (12), and FBNR (13) — attempt to exploit the high temperature durability of the TRISO fuel form, which offers a potential to preclude fuel disruption in nearly all conceivable off normal events.

4.6.7. Approaches to achieve plant transportability and rapid site assembly

Having eschewed an economy of scale cost reduction approach at the outset, two broad strategies to recover economic benefit are employed across the board by designers of small reactors without on-site refuelling:

- Factory assembly-line serial manufacture of standardized plant modules; and
- Plant modularization designed to promote both transportability and rapid site assembly.



1 Reactor
2 Main circulating pump
3 Protective shell
4 Pressure suppression condensation system
5 High pressure gas cylinders
6 Steam generator
7 Metal and water shielding tank

FIG. 6. Protective shell on the barge-mounted KLT-20 (8), see ANNEX VI for details.

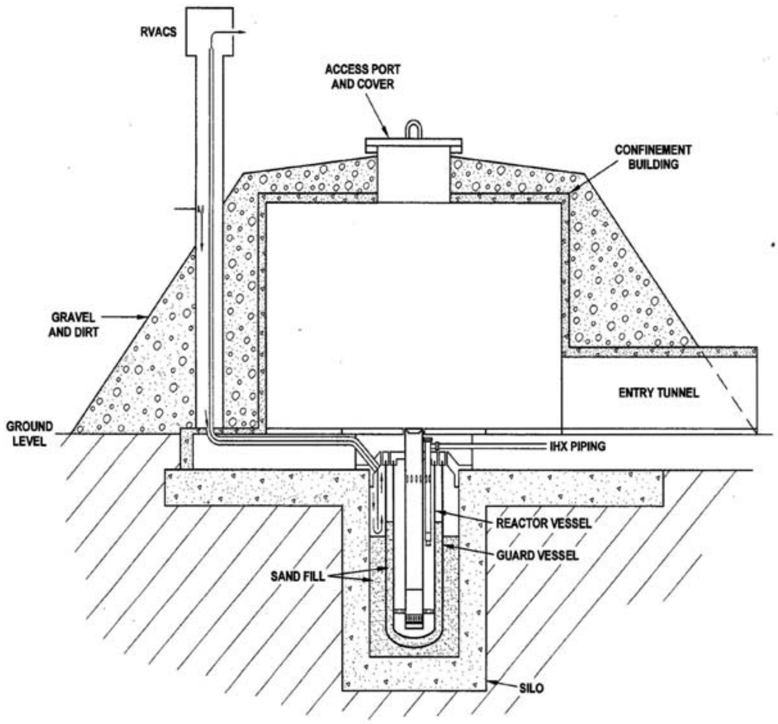


FIG. 7. STAR-H2 (29) heat source reactor, sited in a silo; see ANNEX XXIV for details.

The first approach is intended to benefit from learning curve and mass production efficiencies to achieve cost reductions for standardized modules in a factory environment. The second is intended to dramatically slash the customer's cost of interest on capital financing during site assembly and to hasten faster achievement of a revenue stream. Different designers approach these goals in several ways.

At one extreme (the icebreaker reactor derivatives), the entire power plant is factory built on a barge which is towed to the customer's mooring site, anchored in a pre-prepared lagoon, and connected to a pre-installed power grid switchyard. Trans-ocean shipping times[11] (6 days for the Atlantic, 10 days for the Pacific) are so short that the logistics time interval for this class of plants could be no more than several months. Crew training and physical start-up tests are minimized because factory start-up tests are run and a trained Russian crew is assumed to arrive with the plant.

All other concepts require various degrees of site assembly, at least, to connect the balance of plant to the nuclear steam supply system (NSSS). Most of the small modularized plants have been sized such that their largest components (e.g., the vessel or the NSSS mono-block) are rail shippable; many are even sized (~3 m × ~12 m) for highway delivery. Helicopter delivery of some components could be envisaged.

Some concepts intend to reduce the number of safety-related functions of the balance of plant, e.g., the SVBR-75/100 (18), or even to release the balance of plant from any nuclear safety function whatsoever, e.g., the STAR-H2 (29). Then, the balance of plant could be pre-constructed or constructed in parallel to reactor site assembly by local companies and local labour to local building standards, and can be financed in local currency.

[11] Trans-ocean times quoted for large container ships.

4.6.8. *Approaches for multi-plant clustering*

Based on the data presented in Chapter 3, one possible motivation for multi-plant clustering could be as follows. Many cities in developing countries may be expected to continue in a spiral of rapid population growth for decades. Their energy requirements could accelerate even faster than their population as economic development leads to an increased energy use per capita. This places a scalability requirement on the small reactor concepts, the one which can be met using a multi-plant clustering strategy.

In situations when the initial energy demand is small and financing availability is constrained but growth is expected, the economic benefits of "just in-time" incremental capacity additions with rapid revenue generation could be enhanced using a multi-plant clustering approach. Anticipating growth, a city having an initial need for only a single plant but anticipating that additional plants will be added incrementally in the future could position itself to recover some of the economy of scale benefits of a large installation. Such scale benefits would derive primarily from anticipating growth in designing the site infrastructure:

- Setting aside space for future incremental plants;
- Sizing the switchyard, water and district heat distribution pipelines, etc. for growth; and
- Sharing of railroad, road, and seaway access facilities among future increment plants.

There could be other motivations for multi-plant clustering; in general, it is often foreseen just to make an energy source offer more flexible and better tailored to the needs of potential customers, irrespective of who they might be. The designers of many small reactors without on-site refuelling incorporate the provisions for creation of multi-module plants with certain shared components and infrastructure, as indicated by the data of Table 2. On the total, thirteen out of the 30 presented concepts provide for such an option by relying on modular approach to reactor design.

A more detailed approach to achieve multi-plant clustering has already been elaborated for the MASLWR (5), SVBR-75/100 (18), and STAR-H2 (29) concepts. As an example, one version of the SVBR-75/100 concept comprises a cluster of sixteen plants altogether producing 1600 MW(e), see Fig. 8. Another version with smaller number of modules is being suggested for the renovation of older nuclear power plants with decommissioned larger-capacity reactors (ANNEX XIX).

4.6.9. *Approaches for high temperature process heat production*

Nuclear energy has a potential to carry out future missions for high temperature process heat production. In particular, hydrogen manufacture via thermo-chemical water splitting or steam electrolysis, when driven by high temperature nuclear heat, holds the promise for achieving a carbon-free, electricity/hydrogen energy supply architecture to fuel global sustainable development in the second half of the 21st century.

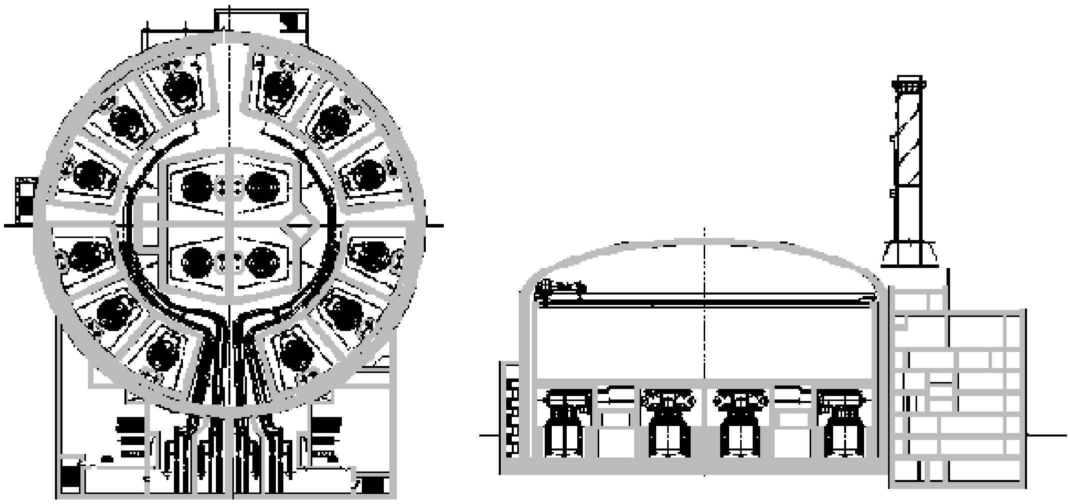


FIG. 8. A plan and a longitudinal section of the clustered modular NSSS SVBR-1600.

TABLE 8. ENERGY PRODUCT OFFERS OF HIGH TEMPERATURE SMALL REACTOR CONCEPTS

CONCEPT (NO. IN TABLE 6)	ENERGY PRODUCTS
CHTR (26)	H_2 with optional cogeneration of electricity using thermoelectric conversion and waste heat
MARS (27)	Electricity with an open cycle air-turbine; high or very high temperature process heat; district heating; potable water
BGR-300 (28)	Electricity with an open cycle air-turbine; H_2
STAR-H2 (29)	H_2 by thermo-chemical cycle
MSR-FUJI (30)	Electricity with Rankine cycle; H_2; potable water

Five of the small reactor concepts presented in this report incorporate an option of high temperature process heat production, see Tables 6 and 8.

Most of these concepts require development and demonstration of currently undeveloped high temperature structural materials, high temperature heat transport components, new combinations of coolant/fuel/cladding and, in some cases, the development and optimization of process heat application cycles, such as thermo-chemical water cracking.

Moreover, most of the concepts select a closed fuel cycle technology as basic, which would require developing and mastering such technology at a commercial level.

Two concepts originating from the Russian Research Centre “Kurchatov Institute” — the fast spectrum gas cooled BGR-300 with advanced porous fuel and the thermal spectrum MARS with TRISO based pebble bed of spherical fuel elements and molten salt coolant — employ the technologies for which a partial experience database already exists in the Russian Federation from previous nuclear R&D programmes.

The technologies for the high-temperature lead or lead-bismuth cooled reactor concepts (the thermal spectrum CHTR using ^{233}U-Th based TRISO type fuel and the fast spectrum STAR-H2 using nitride U-TRU fuel) are less well developed but have given rise to nascent development programmes for structural materials in the high temperature, high radiation, Pb environment.

4.7. Strategies to facilitate plant commercialization

Although all concepts of small reactors without on-site refuelling presented in this report assume 'moderate' values of fuel burn-up and irradiation on the structures, not exceeding those proven in operating practice or typically considered for advanced types of fuel, they all incorporate substantially increased refuelling interval, ranging from ~5 to 20–25 years and beyond. The operating experience for such elongated refuelling intervals is generally unavailable in civil nuclear power. The known experience of marine reactors confirms the possibility of a 7 to 8-year continuous operation of small reactors. Such experience might be relevant to those concepts of small reactors without on-site refuelling that are derived from the corresponding technological and engineering solutions, e.g., those given in Table 3 (2), in Table 4 (7, 8, and 9), and in Table 5 (1); or those that have a prototype successfully operated for many years, see Table 3 (1). In most of the cases, however, licensing and construction of a prototype would be a must for small reactors without on-site refuelling. Moreover, it is unlikely that a prototype of a plant with, say, 20 or 30-year operation cycle could be immediately licensed for uninterrupted operation within the whole cycle. A strategy to proceed could then be as follows.

A reactor prototype could be built and subjected to a pre-agreed set of anticipated transient without scram (ATWS) and other accident initiators. By demonstrating safety based on passive response, on the prototype, the licensing authority might be able to certify the design, permitting the manufacture of many tens (or hundreds) of replicate plants to the set of prints and design specifications used for the prototype. In order to assure that aging effects do not degrade the passive safety features of deployed plants, the licensing authority could prescribe the performance of periodic in-situ tests on the plant to confirm continued presence of reactivity feedbacks in the required range and of passive decay heat removal continuously operating at the required rate. Such an approach is often referred to as "licence-by-test". An example of regulatory provisions for such approach is given below.

There are two current sets of the U.S. Nuclear Regulatory Commission regulations for the licensing of nuclear power plants in the United States [8]:

(1) 10 CFR Part 50 — Domestic licensing of production and utilization facilities; and

(2) 10 CFR Part 52 — Early site permits; standard design certifications; and combined licenses for nuclear power plants.

The regulations contained in 10 CFR Part 50 have been used to license current generation light water reactors (LWRs) in the U.S.A. The regulations in 10 CFR Part 52 provide for many options. Subpart B — "Standard design certifications" discusses the requirements for testing. In particular, Part 52.47 "Contents of applications" states:

- "(b)...(2)(i) Certification of a standard design which differs significantly from the light water reactor designs described in paragraph (b)(i) of this section or utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions will be granted only if

- (A)(1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;
- (2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;
- (3) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; and
- (4) The scope of the design is complete except for site-specific elements such as the service water intake structure and the ultimate heat sink; or
- (B) There has been acceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If the criterion in paragraph (b)(2)(i)(A)(4) of this section is not met, the testing of the prototype must demonstrate that the non-certified portion of the plant cannot significantly affect the safe operation of the plant.
- (ii) The application for final design approval of a standard design of the type described in this subsection must propose the specific testing necessary to support certification of the design, whether the testing be prototype testing or the testing required in the alternative by paragraph (b)(2)(i)(A) of this section.
- The Appendix O final design approval of such a design must identify the specific testing required for certification of the design."

The items beginning with (B) above refer to "acceptable testing of an appropriately sited, full-size prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions". This approach is an example of what is referred to as "license-by-test." There is no experience with any applications proposing license-by-test under 10 CFR Part 52 [8].

4.8. Conclusion to Chapter 4

The thirty concepts of small reactors without on-site refuelling addressed in this report span a range from 0.1 to 900 MW(th) and from less than 1 MW(e) to 300 MW(e). The majority of them are designed for cogeneration of electricity and heat for district heating or seawater desalination, or incorporate other low temperature process heat bottoming cycles. Five concepts are being designed with an option of high temperature process heat applications.

A hallmark of most of the concepts is rapid site assembly of a modularized turnkey plant, a strategy well suited to meet customer needs in off-grid locations where fossil fuel delivery is difficult and, potentially, in developing countries with rapidly growing populations.

Refuelling intervals range from 4.5 years to 60 years, and all concepts could be offered with vendor supplied full-scope fuel cycle services.

Slightly over a third of the concepts represent thermal spectrum water cooled ^{235}U fuelled reactors designed to operate in a once-through fuel cycle. Slightly under half of the concepts are fast spectrum liquid metal cooled reactors, in most cases providing for the operation in a closed fuel cycle.

The surveys presented above indicate that a few small reactors without on-site refuelling might be ready for deployment within the next decade; only one concept has reached detailed design stage, and two others have reached basic design stage. Licensing pre-application in 2006 was considered only for one concept; vendor's discussions with tentative customers have been started in several cases. In most cases, however, new configurations, even if they employ proven fuel/cladding/coolant combinations, will likely require a prototype plant to be constructed prior to commercialization.

About half of the concepts introduce new types of fuel and/or higher temperatures, and/or advanced energy conversion cycles designed to improve plant performance. These new features require further R&D on both power plant and non-electric application technologies.

The once-through fuel cycle could be entirely appropriate for initiating market penetration of small reactors without on-site refuelling. On the other hand, successful market penetration may lead to a requirement to close the fuel cycle as the way to achieve efficient waste minimization and fuel utilization.

Technologies for closing the fuel cycle for fuel types identified in this report are at various stages of development, usually motivated by parallel nuclear research ongoing for reactors beyond the power range of small reactors.

A license-by-test approach providing for acceptable testing of an appropriately sited, full-size prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions, could be re-examined to facilitate commercial deployment of small reactors.

Table 9 summarizes estimated technology development timeframes for the concepts of small reactors without on-site refuelling considered in this report. This estimate is provided under 'optimistic' conditions, including adequate financing.

TABLE 9. ESTIMATES OF TIME TO DEPLOYMENT

DEVELOPMENT APPROACH	TIME SCALE TO FIRST DEPLOYMENT*	EXAMPLES
Adaptation of proven designs and technologies by historical industrial consortia	~7 years	• Derivatives of Russian icebreaker (water-cooled) and submarine (water cooled or Pb-Bi cooled) reactors
Size reductions using already commercialized components and types of fuel and coolant	5–10 years	• Small PWRs with uranium dioxide fuel • Small sodium cooled fast reactors
Designs in conventional temperature ranges using new types of fuel, coolant, and structural materials	10–20 years	• Nitride fuelled Pb-Bi cooled reactors • TRISO fuelled water cooled reactors
High temperature designs including hydrogen production	15–25 years	• Pb and molten salt cooled reactors at 700°C to 1000°C core outlet temperature using nitride or TRISO type fuel

* First deployment, except for the first row of this table, will generally mean deployment of a prototype.

REFERENCES TO CHAPTER 4

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Innovative Small and Medium Sized Reactor Designs 2005 Reactors with Conventional Refuelling Schemes, IAEA-TECDOC-1485, Vienna (May 2006).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs 2004, IAEA-TECDOC-1391, Vienna (July 2004).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safeguards Glossary, 2001 Edition, International Nuclear Verification Series No. 3, Vienna (2001), http://www-pub.iaea.org/MTCD/publications/PDF/nvs-3-cd/Start.pdf.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Technical and Economic Limits to Fuel Burn up Extension, IAEA-TECDOC-1299, Vienna (July 2002).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of National Programmes on Fast Reactors (Proceedings of the 26th Meeting of the International Working Group on Fast Reactors, Vienna, 4–7 May 1993, IAEA-TECDOC-741, Vienna (April 1994).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Liquid Metal Fast Reactor Development, IAEA-TECDOC-791, Vienna (April 1995).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Advanced Nuclear Power Plant Design Options to Cope with External Events, IAEA-TECDOC-1487, Vienna (March 2006).

[8] U.S. NUCLEAR REGULATORY COMMISSION, New Reactor Licensing — Licensing Process, http://www.nrc.gov/reactors/new-licensing/licensing-process.html#inspections.

5. SURVEY OF FUEL CYCLES AND INSTITUTIONAL ARRANGEMENTS

5.1. Introduction

As it was discussed in previous Chapters, small reactors without onsite refuelling have a potential to meet the emerging needs of the following main groups of customers:

- Off-grid customers who experience difficult fuel delivery challenges, such as remote settlements and industrial operations;
- Cities of developing countries at the early stages of economic development, and
- In a more distant future, probably, independent power producers in developed countries that wish to enter markets for new energy products such as process heat, seawater desalination and hydrogen production.

In other words, the targeted categories of customers for small reactors without on-site refuelling are those for which standard economy of scale plants might be ill suited. The designers of such reactors anticipate that the benefits of nuclear energy could be made available to this broader range of customers in markets expected to grow rapidly by a strategy in which vendors provide serially produced standardized small turnkey plants with outsourced front and back end fuel cycle support services, including waste management.

5.1.1. Near-term versus longer term fuel cycle options

The concepts of small reactors without on-site refuelling designed for the nearer term are based on enriched uranium fuel and a once-through fuel cycle. The vendors who are proffering these concepts are based in countries that already have front-end fuel cycle facilities (enrichment and uranium dioxide fuel fabrication) deployed and already have or intend to have approaches for spent fuel management.

On the other hand, many designers already foresee changes that might take place in the longer term, when increasing number of nuclear deployments would require:

- Closing the nuclear fuel cycle;
- Making a transition from ^{235}U fuelling to fuelling with transuranics and ^{233}U bred from ^{238}U and ^{232}Th, respectively; and
- Lifecycle waste management, including wastes from reprocessing.

Additionally, over time, the number of “vendor countries” can be expected to grow to include some currently developing countries as their economic development matures.

The fuel cycle strategies for supporting small reactors without on-site refuelling should, therefore, be designed with these imminent transitions in mind.

5.1.2. *Transition to a sustainable fuel cycle*

Among other features, the concepts of small reactors without on-site refuelling could be configured to share a common symbiotic fuel cycle with current deployments of large capacity light water reactors and nearer-term deployments of small reactors based on a once-through cycle. The integrated symbiotic fuel cycle could be one wherein spent fuel from once-through (open) cycle near-term concepts will be reprocessed to serve as a feedstock to fuel initial working inventories of longer-term closed fuel cycle concepts. The longer term members of the small reactor portfolio, including fast neutron spectrum reactors and, possibly, some thermal spectrum reactors with ^{233}U-Th fuel, can be designed to be fissile self-sufficient and could also self-consume their long lived transuranic "wastes" through recycling and fissioning them as fuel. The configuration of nuclear energy based on small reactors without on-site refuelling thereby offers a potential to manage the foreseen transitions (open → closed cycle; ^{235}U → bred fissile materials; developing to developed countries) in ways, which meet the tenants of sustainable development [1], including:

- Longevity of resource base;
- Ecological responsibility;
- Social acceptability, including:
  - Interregional and intergenerational equity;
  - Safety;
  - Affordability; and
  - Non-proliferation of nuclear weapons.

In addressing the social acceptability pillar of sustainable development, an architecture comprised of long refuelling interval power plants with the centralization of fuel cycle and waste support services has a potential to provide a pathway to contain the hazard of proliferation without constraining widespread global use of nuclear energy.

Managing the foreseen transition to a sustainable symbiotic fuel cycle cannot be accomplished by technology innovations alone. New institutional arrangements governing the global nuclear fuel cycle must be specifically designed to enable centralized fuel cycle support services for small reactors without on-site refuelling. Such enabling institutional changes can build on existing international norms and treaties, but would require substantial further effort to be emplaced.

5.1.3. *Structure of the Chapter*

The Chapter is organized as follows. First, some fuel cycle characteristics used in comparative analysis of energy systems with different reactors are introduced. Second, open fuel cycle features and support strategies are surveyed for the nearer-term concepts. Then, the several proposed fuel cycle features and support options for the closed cycle concepts are surveyed. After that, fuel and ore resource utilization efficiencies for small reactors with long refuelling interval are discussed and are compared to those of standard light water reactors (LWRs) and typically projected liquid metal cooled fast breeder reactors (LMFBRs). Implications on fuel cycle costs are discussed, and the notion of fuel leasing is presented.

Then, the dynamic response features of a transition from an open to a global closed fuel cycle based on symbiosis among near-term and longer-term nuclear power plant (NPP) concepts are outlined, and the issues for a smooth transition are highlighted.

Finally, institutional initiatives that hold a potential to facilitate the transition to a nuclear-based sustainable world energy supply, which meets the criteria for strengthening the proliferation resistance of a global nuclear fuel cycle, are presented in brief; and the common features of small reactors without on-site refuelling that could facilitate their deployment with outsourced fuel cycle services are summarized.

5.2. Parameters that characterize fuel cycle performance

Independent of design specifics, the dynamic characteristics of a fuel cycle of reactor concepts relate to specific inventories and mass flow rates.

Specific mass working inventories can be characterized by the following main parameters:

- In-core specific fissile inventory, *in-core kg fissile loaded/MW(th) deployed*;
- Cycle specific fissile inventory, *in-core + out-of-core kg fissile loaded/MW(th) deployed*;
- Ore specific draw down, *kg U ore required/MW(th) deployed*.

Specific mass flow rates can be characterized by the following main parameters:

- Efficiency of fissile material use, MW(th) delivered/in-core kg fissile loaded;
- Efficiency of ore use, *MW(th) delivered/kg U ore withdrawn*;
- Cycle conversion (or breeding) ratio, *net kg fissile generated in cycle/net kg fissile consumed in cycle*;
- Cycle turnaround time, *years*.

Tables 1 and 2 tabulate the above mentioned fuel cycle dynamic response characteristics for selected open cycle and closed cycle concepts of small reactors without on-site refuelling, based on the inputs provided by designers in the corresponding annexes. For comparison, Table 3 lists the corresponding values for typical LWRs [2] and projected high-breeding LMFBRs [3]. IHM is for initial heavy metals; and CR is for conversion (or breeding) ratio.

5.3. Survey of fuel cycle performance characteristics and support services offered by the concepts

5.3.1. Nearer-term open cycle performance characteristics and fuel cycle support services

All of the nearer-term concepts in this report provide for centralized fuel cycle support services based on a once-through fuel cycle with enriched uranium fuel. Some of the longer-term concepts, e.g., those that extend current water cooled reactor technology to incorporate TRISO type fuel, are also based on an open fuel cycle with ^{235}U fuel.

Depending on the concept, the ^{235}U enrichment varies from 4.7% to just under 20% by weight, see Table 1.

TABLE 1. FUEL CYCLE ATTRIBUTES OF SELECTED OPEN CYCLE REACTOR CONCEPTS

NAME (ORGANIZATION; COUNTRY, ANNEX NO.)	MW(e)	MW(th)	kW(th)/LITRE	FUEL TYPE	ENRICHMENT (WEIGHT %)	REFUELLING INTERVAL, YEARS*	AVERAGE BURN-UP; MW(th)DAY/ KG IHM	IN-CORE FISSILE/ MW(th)	MW(th)DAY/ KG FISSILE LOADED	MW(th)DAY/ KG ORE WITHDRAWN**	CR
Water cooled reactors: icebreaker experience base											
VBER-150 (OKBM; Russian Federation; IV)	110	350	39	UO_2 pellets	4.7	5.7	31.3	3.13	666	3.6	<< 1
KLT-20 (OKBM; Russian Federation; VI)	20	70	33.3	UO_2 granu-les in inert matrix	19.2	8	121.5	4.61	634	3.3	<< 1
ABV-6M (OKBM; Russian Federation; V)	11	45	33	UO_2 in silumin matrix	16.5	8	94.5	5.11	573	2.97	<< 1
UNITHERM (NIKIET, Russian Federation; II)	6	~30	27.2	U-Zr metal ceramic	19.75	16.6	114.9	10.4	582	2.4	<< 1
Water cooled reactors: TRISO fuel											
VKR-MT (VNIIAM — RRC KI; Russian Federation; X)	300	890	36.3	UO_2 TRISO fixed bed	10	9.6	53.3	5.79	533	2.8	<< 1

* Effective full power years
** ^{235}U fraction in the tails is assumed to be 0.2%

TABLE 2. FUEL CYCLE ATTRIBUTES OF SELECTED CLOSED CYCLE REACTOR CONCEPTS

NAME (ORGANIZATION; COUNTRY, ANNEX NO.)	MW(e)	MW(th)	KW(th)/ LITRE	FUEL TYPE	ENRICHMENT (WEIGHT %)	REFUELLING INTERVAL, YEARS*	AVERAGE BURN-UP; MW(th)DAY/ KG IHM	IN-CORE FISSILE/ MW(th)	MW(th)DAY/ KG FISSILE LOADED	MW(th)DAY/ KG ORE WITHDRAWN**	CR
Sodium cooled reactors											
4S-LMR (CRIEPI — Toshiba, Japan; XV)	50	135	89.8	U/TRU/ Zr metal	17.5/20	10	70	9.78	373		0.71
Lead-bismuth cooled reactors											
SVBR-75/100 (IPPE — “Gidropress”; Russian Federation; XIX)	101.5	280	140	U-TRU oxide	14	8.8	92.9	4.9	663	~1000	1.03
Lead cooled reactors											
STAR-LM (ANL; USA; XXIII)	178	400	42.1	U/TRU/ Nitride	16 (average)	15	85.7	10.2	536		Slightly less than 1

TABLE 3. IN-CORE FISSILE INVENTORY OF CONVENTIONAL LWRs AND TYPICALLY PROJECTED LMFBRs

REACTOR TYPE	IN-CORE KG FISSILE/MW(th)	MW(th) DAY/ KG FISSILE LOADED*	MW(th) DAY/ KG ORE WITHDRAWN*
Light water reactor; fuel burn-up 33 MW day/kg IHM	0.887	1100 at 3% initial enrichment	~6.1
Light water reactor; fuel burn-up 50 MW day/kg IHM	1.165	1250 at 4% initial enrichment	~6.8
Liquid metal fast breeder reactor; CR=1.6	2.04		~1000

* For this calculation, tails enrichment (ε_{tails}) was taken as 0.2%; the formulas used were:

$$\frac{MW(th)\ days}{kg\ of\ U\ ore\ withdrawn} = Discharge\ burn-up\left(\frac{\varepsilon_{ore}-\varepsilon_{tails}}{\varepsilon_{fuel}-\varepsilon_{tails}}\right)$$

$$\frac{MW(th)\ days}{kg\ fissile\ loaded} = \frac{Discharge\ burn-up}{Enrichment}$$

The refuelling interval varies from 5.7 to 16.6 effective full power years for water cooled concepts; the average discharge burn-up varies from ~30 (MW(th)-day/kg IHM) for uranium dioxide fuelled concepts and up to ~120 (MW(th)-day/kg IHM) for cermet fuelled concepts.

Nearer-term water cooled reactor concepts and their associated fuel cycle support services are proposed mainly by industrial firms in the Russian Federation and Japan, the countries that already have emplaced front-end and back-end fuel cycle infrastructures and have a prior history of reactor construction and operation. Full scope front end and back end fuel cycle services, including spent fuel management, are supposed to be centralized. The institutional instrument guaranteeing future fuel cycle services to the customer could be a commercial contract.

Extensive worldwide experience exists for providing fuel supply services under commercial contracts. Certain experience in spent fuel return to the originating country does exist; for example, the former Soviet Union received spent fuel returns from affiliated republics and some countries of the Eastern Europe, and the Russian Federation continues to do so currently. At the same time, none of the countries in the world has so far agreed to shoulder the disposal of final waste originating from back-end fuel cycle services being provided by companies operated at its territory [4].

Up to now, the amounts of spent fuel returns have been relatively small; therefore the institutional arrangements for a widespread expansion of such practices would require evaluation and, perhaps, may require enhancement.

5.3.2. *Longer-term closed cycle performance characteristics and fuel cycle support services*

Many (but not all) of the longer term concepts surveyed in Chapter 4.5 anticipate closing the fuel cycle, and they rely on numerous facets of a symbiotic, integrated fuel cycle shared by different reactor types; specifically:

- Numerous of the fast reactor concepts propose to reprocess discharged LWR fuel to capture the uranium and transuranics (or only the uranium and plutonium) and to use that recovered fissile-fertile mass to supply initial working inventories for small fast reactor deployments[1]; this could provide a "win-win" symbiosis in managing LWR "waste" while fuelling small reactors with the recovered transuranics and irradiated uranium;
- Additionally, many of the fast spectrum small reactors are fissile self-sufficient, having an internal conversion ratio of about 1.0, see Table 2; once their initial working inventory is committed, thereafter each fresh fuel reload is obtained by recycling; their spent fuel is reprocessed to recover the fissile and fertile mass needed for a fresh loading, to remove fission products, and to add fresh ^{238}U;
- Waste management benefits accrue to the multiple transuranic recycle of a feedstock symbiotically recovered from once-through cycle concepts, facilitating that only fission products are consigned to the waste destined for geologic disposal;
- The resource extension benefits of a fuel cycle based mostly on fertile ^{238}U feedstock with breeding of fissile mass can be attained with fast spectrum small reactors without on-site refuelling; similar resource extension benefits could, perhaps, be attained in a closed fuel cycle based mostly on fertile ^{232}Th, if appropriate high conversion thermal spectrum small reactors are employed, e.g., see ANNEX XXX.

The recycle technology to support the closed fuel cycle concepts is only partially in place at present. Technological experience in reprocessing and refabrication at an industrial level is mostly confined to uranium based fuel[2]. On the other hand, extensive research and development (R&D) programmes are in progress in many countries concerning reprocessing, refabrication, recycle, and high level (HLW) waste form production and durability characterization for all relevant fuel forms (oxide, carbide, nitride, metal alloy, and TRISO type fuel), as indicated in Sections 1–5 "Outline of fuel cycle options" in Annexes I through XXX. These R&D programmes are run in parallel to advanced reactor development programmes with the intent to arrive at "systems" of advanced nuclear power plants and associated advanced fuel cycles in due time, e.g., [5, 6].

Currently, the institutional experience base for transfers of transuranic-containing reload fuel across national borders is practically nonexistent except within the European Union and from Europe to Japan (for MOX fuelled LWRs). An experience for the transfers of factory assembled and fuelled reactor modules is not available at all. Therefore, new institutional arrangements will need to be developed and emplaced to support the progress of both, making a transition to a global symbiotic nuclear fuel cycle and deployment of the small reactors.

---

[1] The SVBR-75/100 (ANNEX XIX) proposes to use LWR spent fuel as it is, with fission products remaining; its fraction in the overall core loading would then be limited to not more than 12% by weight.

[2] Mixed oxide (MOX) fuel reprocessing for one or two MOX recycles of plutonium in LWRs in being performed in France by AREVA; Japan is expected to start the industrial operations within a few years.

5.4. Efficiencies of fuel and ore usage in small reactors without on-site refuelling

5.4.1. Efficiency of fuel and ore utilization for open cycle small reactor concepts

As it was already discussed in Chapter 4.6.2, the discharge burn-up attained in small reactors without on-site refuelling remains in the same range as for traditional designs as an upper bound. When the same discharge burn-up is attained, then the small reactors use fissile material per unit energy delivered as efficiently as do traditional reactors.

On the other hand, the design strategies for achieving long refuelling intervals impact fuel working inventory utilization and fuel cost; generally speaking, the resulting cost implications are unfavourable compared to traditional reactor designs. As discussed in Chapter 4.6.2, the design approaches employed to attain longer refuelling interval are as follows:

- Small thermal-spectrum reactors of long refuelling interval using uranium dioxide pellet type fuel derate their core power density (kW(th)/litre) and specific power (kW(th)/kg ^{235}U) compared to commercial LWRs, thereby increasing fissile specific inventory;
- Small thermal-spectrum reactors of long refuelling interval using cermet or TRISO type fuel increase fissile uranium enrichment per unit power rating compared to commercial LWRs, thereby increasing fissile specific inventory (kg ^{235}U/MW(th));
- Small fast reactors of long refuelling interval decrease core power density and specific power compared to projected commercial fast reactors, thereby increasing fissionable specific inventory.

Given a fuel with a physically-achievable discharge burn-up, the left hand side of the identity:

$$\frac{MW(th)\,day}{kg\ \ IHM} = \frac{MW(th)}{kg\ IHM} * days\ (core\ lifetime) \qquad (1)$$

is fixed. As it follows from identity (1), in order that the refuelling interval (days corresponding to core lifetime) can be increased from 1 or 2 years to as much as 20 or 30 years, the specific power (MW(th)/kg IHM) must be significantly decreased. The end result is that the fissile mass initially loaded into the reactor per unit power rating increases significantly. Therefore, even when the discharge burn-up is the same, the cost of the fuel feedstock and of the fuel fabrication that must be paid up front for small reactors without on-site refuelling will be recovered over longer time periods than in traditional designs. Given the size of the reduction in specific power of the fuel, the net present value of revenue to offset these initial costs will be significant.

First looking at thermal-spectrum open cycle small reactor concepts (Table 1) as compared to traditional LWR reactor designs (Table 3), the effect of increased fissile inventory requires more ^{235}U in the initial working inventory per unit of power deployed. Table 3 indicates that traditional LWRs require in the range of ~1 (kg fissile/MW(th) deployed). Different from it, as shown in Table 1, the water cooled small reactors have specific loads in the range of 3 to 6 (kg fissile/MW(th) deployed). Given a specified deployment rate of new nuclear capacity, (MW(th)/annum), small reactor deployments would draw down the virgin ^{235}U reserves 3 to 6 times faster than will the deployments of standard LWRs.

On the other hand, the small reactors perform relatively better on the basis of energy delivered per unit of initial fissile mass and of uranium ore withdrawals. As it can be seen from Table 3, standard LWRs with fuel enrichment between 3 and 4% can deliver around ~1100–1250 (MW(th) days/kg fissile loaded) or ~6.1–6.8 (MW(th) days/kg U ore withdrawn). The corresponding numbers for water cooled small reactors of long refuelling interval are about half that (Table 1).

Some of the thermal spectrum reactor designs, e.g., that using cermet fuel, employ high enrichment $\lesssim 20\%$ to raise power density and discharge burn-up above values attainable using the uranium dioxide fuel of <5% enrichment[3]. The higher burn-up compensates the higher enrichment, and their fuel utilization parameters remain in line with other small reactors, see Table 1. However, for a given capacity deployment rate, (MW(th)/annum), a larger deployment of enrichment capacity becomes necessary.

5.4.2. Efficiency of fuel and ore utilization for closed cycle small reactor concepts

The fissile loading per MW(th) deployed in small fast-spectrum reactors exceeds that for standard fast reactors, again because for a given allowable discharge burn-up the specific fuel inventory (MW(th)/kg fissile) has been derated to lengthen the refuelling interval, see the discussion in Chapter 4.6.2. Whereas the projected sodium cooled LMFBRs require ~2 (kg in-core fissile/MW(th) deployed), see Table 3, the corresponding value for small reactors is 5 to 10, a factor in the range of 2.5–5 times higher, see Table 2.

On the other hand, the small reactors perform relatively better on their *overall fuel cycle working fissile inventory.* A typical uranium dioxide fuelled LMFBR may operate a three-batch core on a one-year refuelling interval with a five year cooling period, a one year reprocessing/refabrication period, and one year of fresh fuel storage/shipping before refuelling [3 and 7]. Thus, the total number of core inventories tied up in the overall closed fuel cycle is:

$$\frac{1}{3} * \{3\ in-core + 5\ in\ the\ cooling\ \ pool + 1\ in\ recycle/refabrication + 1\ in\ storage\} =$$

$$= \frac{10}{3} \approx 3.33\ in-core\ inventories \quad (2)$$

Different from it, a small reactor with whole core cassette refuelling on a 20-year interval using the same recycle and refabrication technology gives:

$$1 * \{1 + \frac{7}{20}\} = \frac{27}{20} = 1.35\ in-core\ inventories \quad (3)$$

With a demerit factor of 5 in in-core fissile inventory per MW(th) deployed, the ratio of small fast reactor cycle specific working inventory to that of a typical projected large-capacity LMFBR becomes:

$$\sim 5 * \frac{1.35}{3.33} \approx 2 \quad (4)$$

The closed cycle achieves a very high efficiency of uranium ore use. Although the fuel achieves only 6 to 10 atom percent burn-up per reload cycle, multiple transuranic recycle in reactors with CR ≥ 1can achieve nearly total conversion of ^{238}U to transuranics with subsequent conversion of transuranics to fission products.

This very high efficiency of uranium ore use is:

[3] Specifically, the UNITHERM reactor requires 10.4 (kg fissile/MW(th) deployed).

$$\frac{MW(th)day}{kg\ U\ ore\ withdrawn} \approx 1000 \tag{5}$$

Identity (5) roughly corresponds to 200 MeV per fission or 0.372 (kg fission products/MW(th)-year). Such efficiency is a factor of ~150 larger than that achieved in standard open cycle LWRs, and even more compared to open cycle small reactor concepts, compare Tables 3 and 1.

Essentially, closed cycle fast reactors with a breeding ratio of slightly above 1.0 can operate using depleted uranium as fuel; and this would be the case for both small and large sized fast spectrum reactors with CR ≥ 1 in a closed fuel cycle with multiple recycling of transuranics.

As indicated by Table 2 and several design descriptions given in the annexes, achieving a CR ≥1 appears feasible in the several concepts of small fast-spectrum reactors with lead and lead bismuth coolant, especially when dense nitride fuel is employed. Should it work out in practice, such reactors will not loose to larger-size LMFBRs in the efficiency of uranium ore usage.

5.4.3. The fuel leasing option

Small reactors without on-site refuelling are being designed to meet the needs of the isolated customers or customers with small grids who cannot benefit from the economies of scale achieved in large-sized conventional LWRs or, in the future, LMFBRs. As it was shown in previous sections, long refuelling interval leads to an increased in-core specific inventory, (kg fissile loaded/MW(th) deployed), which, in turn, would lead to an increased capital cost if the fuel is purchased outright. Moreover, even if reactors with long refuelling interval achieve the same discharge burn-up as a conventional reactor, the revenue stream would be acquired over a longer burn-up cycle, so that the net present value of the revenue stream will be reduced.

Three aspects mitigate the business implications of this increase in specific fissile inventory for small reactors of long refuelling interval:

(a) Fuel cost is a small contributor ($\lesssim$ 5–20%, depending on reactor type and country) to overall busbar cost for nuclear power plants in any case, so its impact on production costs of energy products is attenuated;

(b) The power costs of non-nuclear alternatives available to many of the targeted customers for small reactors without on-site refuelling may be dramatically higher than those of traditional nuclear plants (the competition for small reactors could be a diesel with difficult fuel delivery challenges rather than a conventional large-sized LWR); and

(c) Fuel leasing is being proposed for some of the small reactor concepts that could provide the customer a “pay as you go” alternative to outright purchase of the fuel.

The leasing option may prove to be especially attractive for those small reactor concepts that design for an internal conversion (breeding) ratio of unity[4]. In that case, no loss of fissile mass occurs over the burn-up cycle because of internal breeding. Therefore, as much fissile mass is present in the discharged core as was present initially — necessity for recycle is then driven by the need to refurbish the cladding and load more fertile mass, not by loss of fissile mass.

[4] Generically, these could be fast reactors with multiple recycle of all transuranics in the uranium fuel cycle as well as high conversion thermal spectrum reactors with multiple recycle of ^{233}U in the thorium fuel cycle.

As there is no loss of fissile mass, a long-term investor might think of the fissile mass in the refuelling cassette(s) as a "principal" and find that buying a core fuel cassette can represent a safe, principal-preserving long term "investment", which he could then lease to a reactor operator at the rate of return of a long term bond.

5.5. Possible role of small reactors without on-site refuelling in the transition from an open to a global closed nuclear fuel cycle

5.5.1. Issues for a transition from open cycle to closed cycle

Many studies of global energy supply systems indicate that growing nuclear energy deployments will require closing the nuclear fuel cycle within several decades, e.g., [8, 9]. The studies reveal two principal drivers for this transition:

- Using a recycle can reduce waste management burdens per unit of energy production, a strategy already selectively implemented in some countries (e.g., France) and expected to achieve a widespread implementation within two decades;
- Using a recycle strategy can extend the resource base by breeding fissile material from fertile ^{238}U and/or ^{232}Th — a strategy that might be required at an industry scale slightly before mid century.

Once closed, the fuel cycle becomes the integrating mass flow backdrop through which an evolving mix of reactor types can exchange mass via recycle.

Most studies of the time evolution of the fuel cycle and the evolving mix of reactor types during future decades have been based on global (or national) nuclear energy demand scenario analyses which, up to now, have assumed the use of traditional reactor types, such as LWRs, pressurized heavy water reactors (PHWRs), and fast breeder reactors (FBRs). Possible implications of small reactors without on-site refuelling on the transition timing and strategy have not yet been assessed extensively.

The new issues that small reactors without on-site refuelling impose on the configuration of the global, integrated closed fuel cycle fall into three areas:

- Effect of the reduced efficiency of fuel and ore utilization of small reactor concepts in comparison to traditional reactor types;
- Effectiveness of closed cycle small reactor initial loadings as an LWR waste management strategy in comparison to MOX recycle in LWRs;
- Impact of small reactors without on-site refuelling on the dynamic response[5] capability for the transition from a once-through (open) to a closed nuclear fuel cycle.

The implications of these new issues will require dynamic scenario analyses to quantify, but the qualitative trends of expected outcomes are surveyed in brief in the following sections.

[5] Dynamic response is time-dependent response.

5.5.2. *Effectiveness of closed cycle small reactor initial loadings as a waste sequestration strategy*

The historical plan (that discussed in the 1950s and 1960s) for managing the spent fuel "waste" discharged from LWRs operating in a once-through fuel cycle was to reprocess it by partitioning it into three products:

- Uranium, to set aside for future use;
- High level waste (HLW) forms containing fission products and minor actinides with traces of uranium and plutonium; and
- Plutonium, to be refabricated into fuel to supply initial working inventories for fast breeder reactors.

In the 1980s, when growth rates of nuclear deployment declined and new uranium reserves had been found, another strategy was suggested:

- Delay fast breeder reactor deployment;
- Use the recovered plutonium to fabricate MOX fuel for one or two pass recycle in LWRs [10].

The fuel cycle outcome of MOX recycle in LWRs is not to "solve" the waste problem but only to delay by one to two decades any decision about what to do with the still present spent fuel — time to see if breeders will be needed or not. In any case, both the HLW from uranium dioxide reprocessing and the MOX spent fuel contain minor actinides, which are rated as providing challenges to their geologic disposal.

More recently, in light of the difficulties attendant to a presence of minor actinides in waste, R&D on enhanced reprocessing technologies has been directed according to the following strategy [10]:

- Delay fast breeder deployment, but
- Reprocess using new technology, which avoids minor actinide and trace plutonium content in HLW;
- Set minor actinides aside in temporary storage; and/or
- Use recovered plutonium (or transuranics) to fabricate MOX or inert matrix fuel for one or two pass recycle in LWRs.

This again delays a decision about what to do with once or twice recycled LWR spent fuel; either use it to fuel fast breeder reactors or dispose of MOX fuel in a repository. What is different, is that at least the HLW would contain no minor actinides.

While the need date for fast breeder reactors may still be uncertain, concepts of fast spectrum (more generally, high conversion) small reactors without on-site refuelling make it possible to propose yet another strategy of LWR spent fuel utilization [11]. This proposal is described in brief in the following paragraphs.

The main assumption, which still needs a further proof, is that small reactors without on-site refuelling could effectively cater to certain numerous categories of customers[6] starting from the nearer future; then:

[6] Different categories of potential customers for small reactors without on-site refuelling are discussed in Chapter 3.

- The new reprocessing technology could be used to recover transuranic elements (TRU) with necessary amount of uranium (while avoiding minor actinide and trace plutonium content in HLW); and
- These TRU plus uranium could be used to fabricate the initial working inventories of fast spectrum small reactors without on-site refuelling.

Such a strategy may have several potential advantages when compared to MOX recycle in LWRs, due to the following factors:

- The fissile specific working inventory, (kg TRU/MW(th)), of the small reactors is larger than that of MOX fuelled LWRs, see Section 5.4.2;
- The small fast spectrum reactors could fission minor actinide content of the TRU as fuel whereas in LWR MOX recycle, the minor actinides act as neutron parasitic absorbers, which either precludes their use in LWRs and necessitates them to be stored after recovery or would require a new inert matrix fuel be developed for LWRs; and
- The deployment growth rate of small reactors in developing countries could be envisaged larger than for MOX fuelled LWRs in developed countries, see the discussion in Chapter 3.

The higher specific in-core fissile inventory of small reactors with fast neutron spectrum may help ensure that not only the LWR legacy inventories already in temporary storage are worked down, but even uranium dioxide spent fuel from current and future LWR discharges is used up. As a result, minor actinide free HLW forms, which can be more tightly packed because of smaller heat loading, could be accommodated in less repository space.

Finally, for MOX recycle in LWRs, a net loss of fissile mass per unit energy delivered occurs, while the small fast reactors could deliver energy without net loss of fissile mass because the conversion ratio can be made around unity. Given a specified global nuclear deployment, this may have the effect of extending the virgin ore resource base compared to a regime of LWR MOX recycle and, thereby, could ease the time scale available for transition to a fully-sustainable ensemble of reactor types.

5.5.3. Possible impact of small reactors on the dynamic response capability of the global fuel cycle

Large-scale LWRs as well as near-term small reactor concepts are all uranium fuel based open cycle concepts; therefore, it could be possible to fuel high growth rates of their deployment simply by employing high rates of mining and deploying a sufficiently large enrichment capacity. In principle, growth rates of (8–12)%/annum, such are seen today in some developing countries (see Fig. 4 in Chapter 3), could be accommodated.

To the contrary, in the longer term when virgin ore deposits have become depleted or when low assay of remaining reserves drives ^{235}U prices out of an acceptable range, the growth rate of nuclear deployments would be constrained by the rate at which fissile mass can be created by breeding of the secondary fissile materials. Deployments of fast breeder reactors and reprocessing facilities would then be required to fuel a mix of reactor types, i.e., to provide:

- Reloads for thermal-spectrum reactors (that are net consumers of fissile mass); plus
- New working inventories for growth of nuclear deployments, including:
  - New breeder reactors; as well as
  - New non-breeder reactors.

The very best of breeder reactors have doubling times no shorter than 10–15 years, which corresponds to deployment growth rates of no more than 4.5 to 6.5%/annum [3]. This physical constraint limits the overall possible growth rate of fissile self-generated nuclear deployments, even if the entire nuclear deployment were all fast breeder reactors. When fuelling a mix of breeder and non-breeder reactors, the achievable growth rate of deployments would be even less.

Growth rates are likely to be world-region dependent; slower growth plus replacement of retiring units in developed countries compared to faster growth of new units in developing countries, see Fig. 4 in Chapter 3. However, for an integrated global fuel cycle, the world average growth rate will likely not exceed 2%/annum because world average gross domestic product (GDP) growth rates have historically not exceeded a few percent per year [12].

Furthermore, the mixes of reactor types might be region dependent; economy of scale reactor concepts may prevail where the existing infrastructure can support them and small reactors with long refuelling interval could prevail where such a concept better meets customer needs. Mass flows of fuel from one region to another would then be needed to achieve a smooth transition to a growing fissile self-generating world nuclear energy supply.

Achieving a smooth transition from a fully open cycle, fuelled from an external supply of ^{235}U, to a fully closed cycle, self-fuelled from breeding of secondary fissile material, requires a careful management of the time evolution of deployments of thermal- and fast-spectrum reactors such that the fractions of net consumers and net generators of fissile mass in the overall nuclear park can simultaneously:

- Meet a growing energy demand with the reactors targeted to certain customer needs; and
- Meet the growing demand for new working inventories from the fissile mass available from virgin ore and breeding sources, while simultaneously emplacing sufficient breeding capability early enough and sufficient enough to meet future demand for fissile mass.

The trade-offs among deployment trajectories of nuclear power plants of differing characteristics, and the resulting overall nuclear park configurations as a function of the amount of virgin ore not yet harvested, are too complex for solution without taking a benefit of dynamic simulations of energy demand scenarios [13]. These have not yet been conducted in full in application to energy systems with small reactors without on-site refuelling [11]. Nonetheless, it is evident that achieving fissile mass balance between consumption for power producing reactor emplacement and net fissile production for future deployments can support faster deployment rates:

- The greater is the virgin ore resource base;
- The greater is the overall nuclear park conversion ratio and the sooner is it increased;
- The shorter is the overall nuclear park fuel cycle turnaround time; and
- The smaller is the overall nuclear park fissile specific inventory.

The larger specific fissile working inventory of small reactors without on-site refuelling appears to present a disadvantage here. However, when viewed from the perspective of a multi-decade transition and when small reactors have a high conversion ratio CR ≥1, that would not be the case because the lifetime ore withdrawal “mortgage”, which is committed when a standard LWR is deployed, turns out to actually exceed the ore withdrawal committed to create the initial core loadings for a cluster of ^{235}U fuelled fast-spectrum small reactors without on-site refuelling of the same power rating. The reason is that over the 60-year LWR lifetime, the world supply of fissile mass will be net reduced whereas over the 60-year small reactor lifetime the world supply of fissile mass would be preserved. Therefore, for a fissile-constrained transition to a sustainable fuel cycle, high conversion small reactors without on-site refuelling may hold an advantage over standard LWRs. In addition to this, such small reactors would have a benefit to increase the effective conversion ratio of the overall energy park and to do so earlier in the transition. Both effects will be beneficial in a fissile-constrained transition.

Further examination of the trade-offs to achieve a smooth transition to a globally sustainable nuclear fuel cycle would require additional dynamic system studies [11, 13].

5.6. Fuel cycle institutional issues

5.6.1. Approaches to resolve energy security versus non-proliferation dilemma

Increasing global deployments of nuclear power plants on the one hand offers the ecological benefits and the energy security benefits of the nearly unlimited resource base of environmentally clean nuclear energy to a growing fraction of the world's population. On the other hand, it implies an increasing dispersal of reactors and of fissile material to countries which currently have no nuclear facilities, and it carries the potential for increasing dispersal of nuclear fuel cycle facilities worldwide — exacerbating the cost and complexity of safeguards measures. Member states and international organizations have acknowledged the inevitability of nuclear deployments in the coming decades to countries which currently have no nuclear facilities, and are evaluating options to confront the energy security/non-proliferation dilemma that such growth presents [14, 15].

At the time when this report was prepared, two basic institutional approaches have been receiving consideration, generally, with no relation to the concepts of small reactors without on-site refuelling:

- "Assurances of fuel supply", suggesting that "countries that do not now possess uranium enrichment and plutonium reprocessing facilities would agree not to obtain any such facilities or related technologies and materials for some extended period of time. In exchange, during this period they would receive, with attractive terms, guaranteed cradle-to-grave fuel services — specifically, fresh fuel supply and spent fuel removal — under an agreement signed by all those countries in a position to provide them" [16]; and
- "Multilateral approaches to the nuclear fuel cycle", suggesting "to limit the processing of weapon-usable material (separated plutonium and high-enriched uranium) in civilian nuclear programmes, as well as the production of new material through reprocessing and enrichment, by agreeing to restrict these operations exclusively to facilities under multinational control" [14, 17].

Both approaches comprise attempts to constrain the global dispersal of fuel cycle facilities (enrichment, fuel fabrication and reprocessing) which handle fissile material in bulk form — while at the same time not impeding the global dispersal of nuclear power plants which handle fissile material only in a discrete form (amenable to item accountability). Both approaches are essentially complementary [15, 18]; for example, reference [17] suggests a set of gradually-introduced steps, the first of which could be "reinforcing existing commercial market mechanisms on a case-by-case basis through long-term contracts and transparent suppliers' arrangements with government backing", such as "fuel leasing and fuel take-back offers, commercial offers to store and dispose of spent fuel, as well as commercial fuel banks."

Small reactors without on-site refuelling are being designed specifically for operation with the outsourced front-end and back-end fuel cycle services; they can be employed in either of the two proposed institutional approaches for providing fuel cycle support services.

5.6.2. Features of small reactors without on-site refuelling that could facilitate their deployment with outsourced fuel cycle services

Many of the nearer-term concepts of small reactors without on-site refuelling described in Annexes I through XXX indicate fuel cycle services as provided by the vendor, under a contract. This approach is clearly realistic for the near term because no multilateral fuel cycle facilities currently exist.

On the other hand, many of the longer-term and some of the nearer-term concepts identify centralized or regional fuel cycle support services as basic option and suggest that such services could, probably, be provided under an international control. Special features of small reactors without on-site refuelling that could facilitate their deployment for operation with the outsourced fuel cycle services are summarized in brief below.

First of all, as it was already discussed in Chapter 3.6.4, small reactors of long refuelling interval appear as energy sources capable of delivering energy for many years with no need of any operations with fuel during this whole period. In this way, they can relax the dependence on foreign suppliers, fuel cost changes, political and economic tensions and conflicts between countries, etc. — altogether, increasing energy security to the user.

Second, all fuel loading/unloading operations are either outsourced to a factory in the supplier-country or whole core refuelling is conducted infrequently at a site by vendor crews using equipment brought to the site and removed with the used core. In both cases, obligations of the user for spent fuel and radioactive waste management are essentially reduced.

To their users, factory fabricated and fuelled reactors may also appear more environmentally clean, more simple and safe, just because all operations with fuel are outsourced and the reactor actually appears as a long-lasting nuclear battery, perhaps, weld sealed during the whole period of its operation on the site.

Finally, small reactors without on-site refuelling could be made attractive for leasing of a fuel load or even a complete nuclear steam supply system, e.g., with the balance of plant being built by the user. This could secure lower investment costs and risks for the customer.

For the international community, small reactors without on-site refuelling could offer possibly increased non-proliferation assurances by somewhat diminishing the incentive for their users to emplace indigenous nuclear fuel cycle infrastructures. Other factors contributing to such increased assurances could be the absence of refuelling equipment on the site during plant operation and elimination of the on-site fresh and spent fuel repositories.

These and other, design-specific features of small reactors without on-site refuelling need further examination and assessment through a dialogue among possible vendors and potential customers.

5.7. Conclusion to Chapter 5

All of the nearer-term concepts addressed in this report are based on a once-through fuel cycle with enriched uranium fuel. Full scope front end and back end fuel cycle services, including spent fuel management, are supposed to be centralized. The institutional instrument guaranteeing future fuel cycle services to the user could be a commercial contract.

Many of the longer term concepts of small reactors without on-site refuelling anticipate closing the fuel cycle, and they rely on numerous facets of a symbiotic fuel cycle shared by different reactor types. Numerous of the fast reactor concepts propose to reprocess discharged LWR fuel to capture the uranium and transuranics and to use that recovered fissile-fertile mass to supply initial working inventories for small fast reactor deployments. Additionally, many of the fast spectrum small reactors are fissile self-sufficient. Waste management benefits then accrue to the multiple transuranic recycle of a feedstock recovered from once-through cycle concepts, facilitating that only fission products are consigned to the waste destined for geologic disposal.

Many of the longer-term and some of the nearer-term concepts identify centralized or regional fuel cycle support services as basic option and suggest that such services could, perhaps, be provided under an international control.

All concepts of the water cooled small reactors designed for operation in a once-through fuel cycle with enriched uranium are about two times less effective in natural uranium ore utilization per unit of energy produced, as compared to large capacity LWRs with conventional refuelling schemes.

All concepts of the fast spectrum small reactors designed for operation in a closed fuel cycle with multiple recycle of actinides have up to 2 times larger specific *overall fuel cycle working fissile inventory* per unit of thermal power deployed, as compared to the projected larger-capacity liquid metal fast breeder reactors (LMFBRs) with conventional refuelling schemes. On the other hand, several of the concepts of fast spectrum small reactors are being designed to achieve a high conversion ratio CR ≥ 1, specifically, with the use of dense nitride fuel. Should it work out, such reactors will not loose to larger-sized LMFBRs in the efficiency of uranium ore usage for energy production within a closed nuclear fuel cycle.

Higher specific fissile inventory of small reactors without on-site refuelling might facilitate considering them as an alternative in a LWR spent fuel sequestration strategy. Should the targeted customer base for small reactors be confirmed, they might be able to bind more of the LWR legacy inventories for purposeful energy production, as compared to MOX fuel recycle in LWRs. The small fast reactors would also deliver energy without net loss of fissile mass because the conversion ratio can be made around unity.

The concepts of small reactors without on-site refuelling could be configured to share a common symbiotic fuel cycle with current deployments of large capacity light water reactors and nearer-term deployments of small reactors based on a once-through cycle. The integrated symbiotic fuel cycle could be one wherein spent fuel from once-through cycle concepts will be reprocessed to fuel initial working inventories of longer-term closed fuel cycle concepts. The fast reactor members of the longer-term concepts of small reactors without on-site refuelling can be designed to be fissile self-sufficient and would self-consume their long-lived transuranic “wastes” through recycling and fissioning them as fuel, without offering a surplus plutonium breeding.

From the perspective of a multi-decade transition to an integrated symbiotic fuel cycle, small reactors with a high conversion ratio CR $\geq$ 1 offer an advantage over standard LWRs in securing a higher overall nuclear park conversion ratio. Further examination of the trade-offs to achieve a smooth transition to a globally sustainable nuclear fuel cycle requires additional dynamic system studies.

Long refuelling interval leads to an increased in-core specific inventory of small reactors, which would lead to an increased capital cost if the fuel is purchased outright. Even if reactors with long refuelling interval achieve the same discharge burn-up as a conventional reactor, the revenue stream would be acquired over a longer burn-up cycle, so that the net present value of the revenue stream will be reduced.

To cope with increased fuel load costs, fuel leasing option may prove to be effective. It could be especially attractive for those small reactor concepts that design for an internal conversion (breeding) ratio of unity, i.e., ensure that no loss of fissile mass occurs over the burn-up cycle because of internal breeding.

Being designed specifically for operation with the outsourced front-end and back-end fuel cycle services, small reactors without on-site refuelling can be employed in any of the institutional approaches considered to constrain the global dispersal of fuel cycle facilities. For the user, they can relax the dependence on foreign suppliers, fuel cost changes, political and economic tensions and conflicts between countries — altogether, increasing the energy security.

REFERENCES TO CHAPTER 5

[1] INTERNATIONAL ENERGY AGENCY, Energy to 2050; Scenarios for a Sustainable Future, IEA Publications (61 2003 26 I PI), ISBN 92-64-01904-9 2003, Paris (2003).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs 2004, IAEA-TECDOC-1391, Vienna (May 2004).

[3] WADE, D., (ANL, U.S.A.), private communication to V. Kuznetsov, 26 May 2006.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Developing Multinational Radioactive Waste Repositories: Infrastructural Framework and Scenarios of Cooperation, IAEA-TECDOC-1413, IAEA, Vienna (October 2004).

[5] Office of Nuclear Energy, U.S. Department of Energy, Advanced Fuel Cycle Initiative, http://www.ne.doe.gov/infosheets/afci.pdf.

[6] Government of Russian Federation, Concept of a Federal Target Programme: Development of Atomic Power Production Complex of Russia for 2007–2010 and up to 2015, Decree No. 101 of 15 July 2006 (in Russian), http://www.government.ru/data/news_text.html?he_id=103&news_id=21959.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Liquid Metal Cooled Fast Reactor Technology, IAEA-TECDOC-1083, IAEA, Vienna (April 1999).

[8] KOIKE, O.R., SHIBATA, Y., SANDA, T., WAJIMA, T., A semi-empirical long term scenario analysis: Global warming and nuclear energy, GLOBAL'99 (Proc. of Int. Conf., Jackson Hole, Wyoming, September 1999).

[9] Generation IV Roadmap Fuel Cycle Assessment Report, http://gif.inel.gov/roadmap/pdfs/014_fuel_cycle_assessment_report.pdf.

[10] AREVA, Operations, Back-end Division, Reprocessing and Recycling, http://www.areva.com/servlet/ContentServer?pagename=arevagroup_en/common/gotopage&assetid=1029146010170&type=Page&callingpage=1028798800945

[11] WADE, D., VAN DEN DURPEL, L., YACOUT, A., FINCK, P., KHALIL, H., A multi-regional analysis of nuclear energy systems: synergies between regions, reactors, and fuel cycles, GLOBAL'95 (Proc. of Int. Conf. Tsukuba, Japan, Oct. 9–13, 2005), paper 567 (2005).

[12] MADDISON Angus, World economy: A millennial perspective (Paris: Development Centre of the Organization for Economic Cooperation and Development, 2001, Table 3-1a.

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Methodology for the assessment of innovative nuclear reactors and fuel cycles — Report of Phase 1B (first part) of the International Project on Innovative Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1434, Vienna (December 2004).
[14] ELBARADEI, M., Towards a safer world, The Economist (October 18, 2003 issue).
[15] U.S. Department of Energy "Global Nuclear Energy Partnership", http://www.gnep.energy.gov.
[16] MONIZ, PONEMAN, KANTER, DEUTCH, Assured nuclear fuel services initiative, Survival, vol. 46, no. 4, pp. 65–80, USA (Winter 2004–2005).
[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Multilateral Approaches to the Nuclear Fuel Cycle, Expert Group Report to the Director General of IAEA INFCIRC/640, IAEA, Vienna (2005).
[18] G8 Summit 2006, Saint-Petersburg, Russian Federation (July 2006), http://en.g8russia.ru/docs/20.html.

6. PROGRAMMES IN MEMBER STATES FOR DEVELOPMENT OF SMALL REACTORS WITHOUT ON-SITE REFUELLING

6.1. Brazil

In Brazil, the concept of a modular Fixed Bed Nuclear Reactor (FBNR) of 40 MW(e) with a more than 10-year[1] refuelling interval is being developed by the Federal University of Rio Grande do Sul (UFRGS). The FBNR (ANNEX XII), which is a simplified version of the fluidized bed nuclear reactor concept[2] proposed previously, makes use of a pressurized water reactor technology and incorporates spherical fuel elements with high temperature gas cooled reactor (HTGR) type TRISO fuel. Spherical fuel elements are fixed in a suspended core by the flow of water coolant. When flow is interrupted, the spheres are relocated, driven by gravity, to a subcritical and easily cooled fuel chamber.

The FBNR concept is being developed mostly with own resources of the UFRGS. The development is currently at a design feasibility stage. An R&D programme has been planned to include the completion of the conceptual design; construction of a full size non-nuclear hydraulic facility; and studies of neutronic, thermal-hydraulic, and fuel behaviour.

6.2. India

India is developing the Compact High Temperature Reactor (CHTR, ANNEX XXIX), which is designed to have a 15-year refuelling interval. The CHTR employs liquid Pb-Bi eutectic alloy as coolant, BeO as moderator, and TRISO pin-in-block type ^{233}U-Th based fuel. The CHTR is designed to generate about 100 kW(th). By incorporating a heat pipe system, it can deliver high temperature heat (1000°C) for process applications, including hydrogen production. The design incorporates several advanced passive safety features and is also intended for autonomous electricity supply in remote areas.

The R&D programme for the CHTR is being carried out at the Bhabha Atomic Research Centre (BARC) with financial support from the Government of India. At the time of this report (2006), the feasibility study for the CHTR has been completed and a conceptual design phase has started. This phase includes a set-up of experimental facilities to carry out various tests related to liquid metals, fuel element coatings, passive safety features, and heat removal systems. Pilot testing of fuel element coatings in liquid metal coolant has already started.

6.3. Indonesia

In Indonesia, the Bandung Institute of Technology (ITB) is performing conceptual studies of two small lead bismuth cooled nuclear power reactors with fast neutron spectrum that could be operated for more than 15 years without on-site refuelling. They are the SPINNOR in a 10–20 MW(e) range and the VSPINNOR with a capacity of ~6 MW(e) (Annex XXVI); both incorporate optimum combinations of reactivity effects to secure reactor self-control in anticipated transients without scram. The SPINNOR and VSPINNOR are developed based on the concept of a long-life core reactor developed in Indonesia since early 1990s in collaboration with the Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology.

Conceptual studies are mainly carried out by the Reactor Physics Laboratory of ITB, in cooperation with the National Atomic Energy Agency of Indonesia. The projects are funded under several national research grants.

---

[1] All refuelling intervals specified in this chapter are in effective full power years.
[2] Both reactor concepts are abbreviated as FBNR.

6.4. Japan

There are 10 concepts of small reactors without on-site refuelling that are currently being developed at different stages in Japan. Three of them are light water cooled reactors: PSRD (ANNEX VII), Package-Reactor (ANNEX VIII), and PFPWR50 (ANNEX IX). There are three designs of sodium cooled small reactors. They are the 4S Toshiba Design (Annex XIV), the 4S-LMR CRIEPI Design (ANNEX XV), and the RAPID (ANNEX XVII). Another three liquid metal cooled small reactors are the Small Lead Bismuth Cooled Reactor (ANNEX XXI), the LSPR (ANNEX XXV), and the PBWFR (ANNEX XXVII), all of which employ lead-bismuth eutectic as coolant. Finally, the MSR FUJI reactor (ANNEX XXX) is a molten salt cooled reactor.

Several large industrial corporations, national research institutes and universities take a lead in the development of these innovative small reactors, in cooperation with other organizations.

Water cooled small reactors

The Japan Atomic Energy Agency (JAEA) is developing a passively safe small reactor for a distributed energy supply system (PSRD, ANNEX VII). The PSRD of 31 MW(e) is an indirect cycle, integrated design (tank-type) small pressurized water reactor with a refuelling interval of more than 5 years. Steam generator is located inside the reactor vessel. Currently at a conceptual design stage, the PSRD is designed to achieve system simplification, resulting in the reduction of costs for construction, operation and maintenance. The assessments of the plant economy are ongoing. Experimental testing has been planned to verify passive safety features and other design features.

Hitachi, Ltd. and Mitsubishi Heavy Industries, Ltd. are cooperating in the development of the Package-Reactor concept (ANNEX VIII), a 25 MW(th) (or 10–100 MW(th)) light water reactor based on a combination of pressurized water, boiling water and pressure tube reactor technologies. This concept is based on individual “encapsulated” cassette-type fuel assemblies that are coupled neutronically, and uses boiling water coolant with natural convection. The steam is directed from each of the cassettes to in-vessel secondary loop generators. The refuelling interval is 5–10 years. One of the initial objectives for the development of the Package-Reactor is to minimize the necessary costs of R&D and associated test facilities, so that the reactor could be deployed within a few years. By 2005, the feasibility of the Package-Reactor has been confirmed. A decision on whether to proceed with further R&D programme is to be made based on the consideration of market needs and other factors.

The PFPWR50 (ANNEX IX) is being developed mainly by the Hokkaido University and it is currently at a design feasibility stage. It is a pressurized water reactor of 50 MW(th) with coated particle type fuel in graphite matrix within conventional zirconium alloy tube claddings. The features of the PFPWR50, such as the use of TRISO type fuel, a refuelling interval of 7–8 years, emergency core cooling by passive injection, and large passive heat sink within containment, are expected to facilitate plant location in the immediate proximity to customers, e.g., as a plant for district heating. Reduction of operation and maintenance costs, as well as the capital cost is considered as the design objective for PFPWR50.

Sodium cooled small reactors

The RAPID concept (ANNEX XVII), developed by the Central Research Institute of Electric Power Industry (CRIEPI), is a 10 MW(th) sodium cooled reactor for autonomous, possibly unattended operation. The design offers a refuelling interval of 10 years and does not include control rods. Instead, it incorporates passive lithium expansion, injection, and release modules to enable an operator-free operation mode. The manufacturing technology has been established and a substantial amount of testing has been performed for the lithium based modules. The development of the RAPID is at a conceptual design stage. The ongoing R&D programme is supported by the CRIEPI's own resources.

There are two versions of the Super-Safe, Simple, and Small (4S) reactor concept being developed in Japan. The one targeted for a nearer-term and a once-through fuel cycle is the 4S Toshiba Design (ANNEX XIV). It is a sodium cooled reactor of 10 or 50 MW(e) with a 30-year refuelling interval, developed by Toshiba Corporation in cooperation with CRIEPI and several other organizations in Japan. The 4S-LMR concept (ANNEX XV), with an output of 50 MW(e) and a refuelling interval of 10 years or more, is being developed to operate in a closed U-Zr-TRU cycle. The latter is being developed by CRIEPI in cooperation with Toshiba Corporation. Both concepts incorporate integral design of the primary and secondary (intermediate) sodium circuits and adjust reactivity changes with burn-up using out-of-core moving reflectors. Both designs benefit from many inherent safety features, assuring long-term plant self-control in anticipated transients without scram; specifically, the 4S-LMR has a negative void reactivity effect when operated with U-Zr-TRU fuel.

The development of the 4S Toshiba Design is supported by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) in Japan. Currently the R&D is focused on the core, fuel and reflector technologies. The conceptual design and major parts of the system design have been completed. A pre-application review by the US NRC is planned in the end of 2006.

In December 2004, the Galena City Council (Alaska, USA) unanimously agreed to continue planning a 10 MW(e) nuclear power plant as a demonstration project in the small Alaskan community. City officials identified the 4S Toshiba Design reactor as their preferred option.

The development of the 4S-LMR by CRIEPI is at a conceptual design stage. Development of a new, improved design with respect to core configuration and safety and development of some key technologies, such as the driving mechanism for the reflectors, are being conducted currently under a contract with MEXT.

Lead-bismuth cooled small reactors

JAEA is also developing the Small Lead-Bismuth Cooled Reactor of 50 MW(e) with a 30-year refuelling interval (ANNEX XXI). Currently at a conceptual-design stage, this small sized tank-type reactor is designed without an intermediate heat transport system. Steam generator is located inside the reactor vessel. A corrosion test for stainless steel claddings has been conducted for 10 000 hours in stagnant lead-bismuth. Experiments on a laboratory scale have been performed to support the design of a three-dimensional seismically isolated reactor building.

The LSPR, a 53 MW(e) lead-bismuth cooled reactor with a refuelling interval of 11–12 years is being developed by the Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology, with funding from MEXT. The LSPR (ANNEX XXV) is designed as an integral type reactor with steam generator installed within the reactor vessel. After the current stage of design feasibility study, future R&D has been planned to include studies of core design incorporating the CANDLE burn-up concept (see reference [XXV-9] in ANNEX XXV), simplification of passive decay heat removal systems, identification of measures to cope with a steam generator tube rupture, and development of simplified maintenance techniques for in-vessel devices.

The direct contact boiling water small lead-bismuth cooled reactor (PBWFR) is being developed by the Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology, in cooperation with several other companies. The PBWFR (ANNEX XXVII) is a pressure vessel type reactor of 150 MW(e), in which sub-cooled water is fed into the hot Pb-Bi coolant above the core, resulting in a direct contact boiling. The resulting bubbles rise due to buoyancy force, which also works as a lift pump for Pb-Bi circulation. The refuelling interval is 15 years. The generated steam passes through the separator and the dryer to remove Pb-Bi droplets, and then flows to the turbine-generator plant. The design and technology development for the PBWFR was supported by MEXT. Feasibility studies have been conducted and a conceptual design of the PBWFR is currently being developed. The PBWFR concept is supported by a substantial amount of thermal-hydraulic and other tests, performed in the past several years.

Molten salt cooled small reactors

The MSR FUJI is a simplified molten salt reactor of 200 MW(e) intended to operate in a closed ^{233}U-Th fuel cycle. The operation cycle length is more than 30 years; however, periodical fissile-fertile feeding from an internal reservoir is necessary. The design of MSR FUJI is based on previous molten salt reactor designs developed or operated in the molten salt reactor programme at the Oak Ridge National Laboratory (USA) between 1950 and 1976. Many results of the R&D performed under that programme are therefore directly relevant to the development of the MSR- FUJI.

The principal stakeholder of the development of the MSR-FUJI is the International Thorium Molten-Salt Institute (ITHMSI), under the leadership of President K. Furukawa and Chief Manager Y. Kato. The R&D is supported by individual research groups in many Member States. The development is currently at a stage of an early conceptual design. Further need in R&D for the MSR FUJI related to the structural materials and components has been identified.

6.5. The Russian Federation

There are 11 concepts of small reactors without on-site refuelling that are currently being developed at different stages in the Russian Federation. Six of them are light water cooled reactors: the UNITHERM (ANNEX II), the ELENA (ANNEX III), the VBER-150 (ANNEX IV), the ABV (ANNEX V), the KLT-20 (ANNEX VI), and the VKR-MT (ANNEX X). In addition, there is one small gas cooled fast reactor concept which is the BGR-300 (ANNEX XIII); two sodium cooled reactor concepts: the MBRU-12 (ANNEX XVI), and the BN GT-300 (ANNEX XVIII); and one lead-bismuth cooled small reactor design, the SVBR-75/100 (ANNEX XIX). Finally, there is one non-conventional reactor concept; the MARS (ANNEX XXVIII).

The activities for small reactors in the Russian Federation are being carried out by several leading design organizations and national research institutions, in cooperation with other organizations. In several cases, the partnerships involve the organizations and institutions with the experience of design, construction, and operation of the marine reactors, e.g., the reactors of nuclear icebreakers or submarines.

Water cooled small reactors

The UNITHERM (ANNEX II) is a 30 MW(th) transportable nuclear power plant for supplying energy to urban districts and industrial enterprises in areas that are remote or difficult-to-access. The refuelling interval is designed to be 16–17 years. The design concept of the UNITHERM has been developed since the 1990s by the Research and Development Institute of Power Engineering (known both as RDIPE and NIKIET) and the Russian Research Centre “Kurchatov Institute” (RRC KI), and is based on the NIKIET experience in the design of marine nuclear installations. The development, which was carried out on the initiative of NIKIET, is now at a conceptual design stage. Some private companies and the Government of the Sakha Republic of the Russian Federation have expressed an interest as potential users. The UNITHERM requires no major R&D for technology development. However, further R&D is needed for certain innovative systems and components of the design, such as the independent circuit for heat removal and systems for the reactor equipment cooling.

The ELENA NTEP (ANNEX III) is a concept of an unattended, self-controlled nuclear thermoelectric plant of 3.3 MW(th) and 68 kW(e), with a long refuelling interval of about 22 years. The principal stakeholders of its development are RRC KI, the Federal State Unitary Enterprise (FSUE) "Krasnaya zvezda", the Joint Stock Company (JSC) "Izhorskiye zavody", FSUE "Atomenergoproekt” and FSUE VNIINM. The ELENA NTEP concept, which currently is at a conceptual design stage, is backed by the experience of the stakeholders in the design of nuclear installations for space and underwater use, and by a demonstration thermoelectric nuclear plant GAMMA. The latter has been put into operation in 1982 and is still operating. Detailed design of the fuel element has been completed.

The VBER-150 reactor (ANNEX IV), which has a capacity of 110 MW(e) and a refuelling interval of ~6 years, is being designed for a floating (barge-mounted) NPP. The VBER-150 is a two-loop modification of the VBER-300 reactor described in IAEA-TECDOC-1485 [1]. It is a small sized loop type pressurized water reactor. Modular arrangement of the main reactor components is a key feature of the reactor concept. The reactor pressure vessel, two once-through steam generators and two main circulating pumps are integrated into a single vessel system by short welded co-axial pipes for the coolant. Both the VBER-300 and the VBER-150 are thoroughly based on a successful multi-decade operating experience of marine propulsion reactors in the Russian Federation.

Design and technology development for the VBER-150 is being carried out by the Experimental Design Bureau of Machine Building (OKBM), Nizhny Novgorod (Russian Federation) in cooperation with RRC KI and JSC “Lazurit”. Collectively, the stakeholders have a unique experience in the design, construction and operation of marine nuclear reactors. Design development of a floating NPP with the VBER-150 is being financed by the companies and organizations involved in the project.

As the VBER-150 is a two-loop modification of the VBER-300, the results of the latter project are being used in the design of the former. The Rosatom supports the VBER-300 design development within the framework of a national programme[3]. The project of a floating NPP with the VBER-150 is at a conceptual design stage.

The ABV (ANNEX V) is a nuclear steam-generating plant with a small reactor of 11 MW(e) and a refuelling interval of about 8 years. The reactor is of pressurized water type and

[3] In July 2006, an agreement was reached between the Russian Federation and Kazakhstan to create a joint venture to complete design development for the VBER-300 reactor plant, and to promote NPPs with such reactors to the markets of both countries and abroad. Another small reactor suggested for joint development was the ABV, but it was agreed to discuss it in more detail at the next due meeting.

incorporates an integral design of the primary circuit, with steam generator located inside the reactor vessel. Several aspects of previous R&D performed for marine reactors and the operating experience with VVER are adopted in the ABV.

Design development for the ABV is performed by OKBM, the Institute of Physics and Power Engineering (IPPE, Obninsk), and JSC "Lazurit", which are the organizations that have a long-term experience with design development for marine propulsion reactors. At present, the R&D for the ABV reactor is financed by the companies involved in the project. The ABV design is being developed in response to the appeal of the governments of the Far North and Far East regions of Russia to the Russian Government requesting to provide small reliable power sources to support the incipient activities on development of new deposits and to cope with a shortage of power and heat for residential loads. Detailed design has been developed and licensing has been started for a preceding project — the ABV-6M, but works have been stopped after 1996. At the time of this report, the new ABV project was at a conceptual design stage.

The KLT-20 reactor (ANNEX VI) is a small nuclear power source for a floating cogeneration or power plant. It is designed to have a refuelling interval of 8 years. As a pressurized light water reactor of 20 MW(e), the KLT-20 is a two-loop modification of the KLT-40S reactor [2] with several improvements in the main equipment and a long-refuelling interval, achieved with the enrichment of less than 20% of uranium by weight.

The KLT-20 is being designed by the OKBM and RRC KI; and financed by the organizations involved in the project. A pilot floating heat and power plant with the KLT-40S reactors has been started in construction in Russia in June 2006, with its deployment scheduled for 2010. Development of a floating NPP with the KLT-20 is at a conceptual design stage.

The VKR-MT (ANNEX X) is a vessel type boiling water reactor of 300 MW(e). It incorporates an innovative core design based on a pebble bed of micro fuel elements (TRISO type coated particles of ~2 mm diameter with SiC outer coatings) directly cooled by boiling water. The refuelling interval is designed to be about 10 years. The VKR-MT is a direct successor of the VK-300 boiling water reactor [1]; the latter was developed by NIKIET for the renovation of reactor facilities previously used for weapon plutonium production. The VKR-MT also borrows from the concept of a VVER type reactor with micro fuel elements developed by RRC KI, the All Russian Institute of Atomic Machinery (VNIIAM), and the Scientific and Production Association "Luch". The organizations mentioned above are also principal stakeholders in the VKR-MT project.

The VKR-MT is currently at a design feasibility stage. The design goal is to achieve a very high level of nuclear and radiation safety by eliminating significant releases of fission products from fuel in severe accidents, including the ones caused by human actions of malevolent character. Concept development is backed by the completed out-of-pile irradiation testing of coated particle samples and the ongoing in-pile irradiation testing of micro fuel elements.

Gas cooled small reactors

A feasibility study for the concept of a fast gas cooled reactor of 300 MW(th) with a refuelling interval of 12 years (BGR-300, ANNEX XIII)) is being conducted by RRC KI. The BGR-300 is a high temperature small tank-type reactor with the secondary vessel acting as a safety system. There is no intermediate heat transport system. The reactor core uses a porous matrix fuel in the form of quasi-homogeneous heat generating blocks with cross-circulated coolant. The molten salt reflector acts as a heat sink in accidents. Through implementing a very high temperature heat exchanger in the primary circuit before the primary-to-secondary heat exchangers, the BGR-300 provides an option of hydrogen production, using thermochemical processes.

Sodium cooled small reactors

The MBRU-12 (ANNEX XVI) is a modular nuclear power plant with sodium cooled fast reactor of 12 MW(e) and a 30-year refuelling interval. The fuel assemblies are assumed to be shuffled annually under a closed guard vessel cover. The MBRU-12 incorporates the engineering solutions validated in operating practice of the power plants with fast sodium cooled reactors, such as BOR-60, BN-350, and BN-600, as well as during design development for the BN-800 reactor. The MBRU-12 uses an integral design of the primary and secondary (intermediate) sodium coolant systems. Several core arrangements have been analyzed, and the one with non-positive sodium void effect was selected.

The principal stakeholders of the MRBU-12 are OKBM, Sankt Petersburg "Atomenergoproekt" (SPb AEP) and IPPE. The development is currently at a conceptual design stage. As of 2004–2005, conceptual design studies of the MBRU-12 were performed on the initiative of OKBM specialists.

The BN GT-300 (ANNEX XVIII) is a transportable modular nuclear cogeneration plant of 300 MW(e) based on a small sodium cooled reactor with fast spectrum of neutrons and a gas turbine cycle for energy conversion. The refuelling interval is designed to be 4.5–6 years. The concept provides the possibility of having several modules of the reactor mounted in railway cars and has no intermediate heat transport system. The modules are delivered to the site by railway and fixed and connected to each other under a shelter.

The principal stakeholder for the development of the BN GT-300 is IPPE. The R&D programme is partially funded under a national industrial programme. The design phase is that of an early conceptual design. International cooperation is foreseen as an option starting from the basic design phase.

Lead-bismuth cooled small reactors

The SVBR-75/100 (ANNEX XIX) is a modular multi-purpose lead-bismuth cooled fast reactor of 75 to 100 MW(e), offering a refuelling interval of 6 to ~9 years. The design is backed by the experience of 50 years in design and operation of reactor installations with lead-bismuth coolant for nuclear submarines, available in the Russian Federation. Specifically, the marine prototypes of the SVBR-75/100 have achieved a total of 80-years of operating experience.

The SVBR-75/100 incorporates an integral design of the primary lead-bismuth circuit, with all primary circuit equipment being located in a single pool, completely eliminating valves and lead-bismuth coolant pipelines. The main reactor vessel is located in a water pool, which acts as a passive decay heat removal system and prevents lead-bismuth release to the environment in case of cracks in the reactor vessel. The technology of freezing/de-freezing of the lead-bismuth coolant has been developed and proven in operating reactors. This is also the case for the polonium (^{210}Po) treatment technology. The SVBR-75/100 incorporates a two-circuit scheme with no intermediate heat transport system. Rankine cycle is used in the power circuit. The design is flexible in fuel cycle options and applications; it can operate in a once-through as well as closed fuel cycle and allows for multi-module higher capacity plant configurations.

Design development for the prototype SVBR-75/100 plant is being carried out by the Federal State Unitary Enterprise (FSUE) "Gidropress", FSUE IPPE, and FSUE "Atomenergoproekt". A detailed design stage has been initiated by the time this report was prepared. On 15 June 2006 the Scientific and Technical Council No. 1 of the Rosatom of Russia supported the continuation of works for the detailed design of the SVBR-75/100 plant with a link to a certain deployment site.

A list of basic R&D and tests to be performed at the detailed design stage has been prepared, ANNEX XIX. In addition, in order to provide a more flexible supply of energy, IPPE and “Gidropress” also develop a smaller version of lead-bismuth cooled reactor — the SVBR-10 of 10 MW(e).

Non-conventional small reactors

RRC KI develops a concept of a high temperature autonomous micro-particle fuelled molten salt cooled reactor — the MARS. It is currently designed with a capacity of 16 MW(th) and a 15 to 60-year refuelling interval (ANNEX XXVIII). The concept incorporates a fixed bed of HTGR type spherical fuel elements with TRISO fuel and a molten salt coolant. The secondary circuit makes use of an open air-turbine cycle.

The MARS concept is developed by RRC KI on its own initiative. Feasibility of the selected approach is supported by a certain amount of tests performed at RRC KI in the 1970s. The main R&D is focussed on core design optimization to achieve the desired plant characteristics as well as on elaboration of codes for safety analyses.

6.6. The United States of America

There are 6 concepts of small reactors without on-site refuelling that are currently being developed at early stages in the United States of America. Two of them are light water cooled reactors, the MASLWR (ANNEX I) and the AFPR (ANNEX XI). There is one lead-bismuth cooled reactor concept, the ENHS (ANNEX XX). Finally, three lead cooled reactor concepts form the so-called STAR reactor family. These are the SSTAR (ANNEX XXII), the STAR-LM (ANNEX XXIII), and the STAR-H2 (ANNEX XXIV).

In the U.S.A., the activities for small reactors without on-site refuelling are carried out by national laboratories and universities, with little involvement of major industries. The financing is provided under certain national programmes and, as of 2005, was generally insufficient to move the concepts from a feasibility study or conceptual design to more advanced design stages.

Water cooled small reactors

The Multi-Application Small Light Water Reactor (MASLWR, ANNEX I) is a small pressurized water reactor of 35 MW(e) with a 5-year refuelling interval. The MASLWR has a modular design consisting of an integrated reactor vessel, steam generator, and high-pressure containment vessel. The entire reactor module would be shop fabricated and transportable to a site on most railways or roads. The design provides for a multi-module higher capacity plant construction.

At previous stages, the design and testing team of the MASLWR was comprised of staff from the Idaho National Laboratory (INL), Oregon State University (OSU), and Nexant-Bechtel; the R&D was sponsored by the Nuclear Energy Research Initiative (NERI) programme of the U.S. Department of Energy (DOE). A scaled thermal-hydraulic test facility capable of full system pressure and temperature operations has been constructed and successfully operated at the OSU. The development of the MASLWR is currently at conceptual design stage. Additional tests and design improvements may include natural circulation flow stability tests with simulated neutronic feedback and high-pressure passive containment cooling tests. Funding for the next series of tests is currently being sought.

The Pacific North-West National Laboratory (PNNL) of the USA develops the concept of a small light water reactor of 100 MW(e) with TRISO type fuel. This concept is being designed for a refuelling interval of 36 years and is tentatively named the Atoms for Peace Reactor (AFPR, ANNEX XI). The AFPR core consists of a pebble bed of the micro fuel elements (MFEs), which are uranium dioxide based small particles coated with SiC-PyC layers. These particles are in direct contact with the water coolant flowing laterally and leaving the pebble bed as steam through the perforated walls of the fuel assemblies. The outer coating layer is assumed to be manufactured of very strong and resistant protective coating materials, such as the nano-layered nitride materials like TiN/NbN or AlN/CrN. The design incorporates in-vessel storage tanks for fresh and spent MFEs, and a valve system providing for a kind of on-line refuelling of the reactor core without opening the reactor vessel cover, achieved via downward movement of the gravity-driven MFEs controlled by opening of the discharge valves. The development stage is that of a feasibility study.

Lead-bismuth cooled small reactors

The Encapsulated Nuclear Heat Source (ENHS, ANNEX XX) is a modular lead-bismuth cooled reactor of 50–75 MW(e) with a 20-year or even longer refuelling interval. The reference ENHS reactor has two coolant circuits, both being of a pool type; the primary coolant circulates inside the ENHS module while the secondary (intermediate) coolant circulates in the pool the ENHS module is inserted in. The reactor design incorporates optimum combinations of reactivity feedbacks and is fissile self-sufficient.

The Lawrence Livermore National Laboratory (LLNL), Argonne National Laboratory (ANL) and Los Alamos National Laboratory (LANL) as well as the University of California at Berkeley are collaborating on the R&D of small, lead alloy cooled battery type reactors. R&D that is specific to the ENHS is being carried out at the University of California at Berkeley and in LLNL. This work is also supported, at a relatively low level, by the US DOE Generation IV programme as part of the work done on lead alloy cooled nuclear battery-type fast reactors. The ENHS R&D is also partially financed by LLNL and the Korean Atomic Energy Research Institute (KAERI). The design stage is that of a feasibility study or early conceptual design.

Collaboration among ANL in the USA and CRIEPI and Toshiba in Japan related to the ENHS R&D is being carried out. The CRIEPI and Toshiba effort is focused on the 4S reactor design (ANNEX XIV) important elements of which were adopted for the ENHS reactor concept.

Lead cooled small reactors

The STAR[4] concept development is being conducted for a portfolio of reactor and balance of plant designs to enable an incremental market entry that is time-phased according to the degree of R&D required. The Small STAR (SSTAR, ANNEX XXII) is a 20 MW(e) lead cooled reactor to provide secure energy supply to remote small villages. It is targeted for early prototyping of the technology and institutional features of the STAR concept portfolio. The STAR — Liquid Metal (STAR-LM, ANNEX XXIII) is a lead cooled, 400 MW(th), natural circulation reactor of 565°C core outlet temperature driving a supercritical CO_2 Brayton cycle for electricity production. The STAR-H2 (ANNEX XXIV) raises the Pb outlet temperature to 800°C to drive a thermochemical water cracking cycle to produce hydrogen. All STAR concepts are designed for a 15–20 year refuelling interval.

Development of the SSTAR has been supported under the lead fast reactor element of the U.S. DOE Generation IV Nuclear Energy Systems Initiative. Development of the SSTAR small modular fast reactor under Generation IV involves liquid metal fast reactor related

[4] STAR is for Secure, Transportable, Autonomous Reactor.

funding at ANL, at LLNL, at LANL, and at INL. Development and design of the STAR-LM and STAR-H2 concepts at ANL was previously supported by U.S. DOE Nuclear Energy Research Initiative (NERI) projects. Institutions involved in the STAR portfolio research and development together with ANL are Oregon State University, Texas A&M University, and The Ohio State University. All concepts are at a feasibility study or early conceptual design stage.

As part of the Generation IV work on the SSTAR, it was proposed in 2003 that a lead cooled demonstration test reactor could be designed, constructed, and ready for operation by about 2015. There is considerable interest in a license-by-test approach that makes use of a demonstration test reactor. The demonstrator would subsequently be operated to support SSTAR commercial deployment in about 2025.

At the time when this report was prepared, funding was not available at a level sufficient to make design, construction, and initial operation of a demonstration test reactor feasible within a 2015 timeframe; and there remained substantial uncertainty as to what funding priorities the U.S. DOE would place on this concept.

REFERENCES TO CHAPTER 6

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Innovative Small and Medium Sized Reactor Designs 2005: Reactors with Conventional Refuelling Schemes, IAEA-TECDOC-1485, Vienna (2006).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs 2004, IAEA-TECDOC-1391, Vienna (May 2004).

Appendix I

OUTLINE FOR SMR DESIGN DESCRIPTION (FULL DESCRIPTION)

I.1. General information, technical features and operating characteristics

I.1.1. Introduction

- Full and abbreviated name of the nuclear installation with an innovative SMR
- Historical technical basis: identify plants/test facilities, and R&D previously performed that support this innovative SMR design
- List of principal stakeholders

I.1.2. Applications

Including: electricity generation/co-generation, district heating, seawater desalination, hydrogen production, process steam production etc, or a combination thereof

I.1.3. Special features

Such as: floating NPP option, option to use NPP as an autonomous energy source (nuclear cell), modular approach allowing for incremental capacity increase, option of prefabrication, transportability, etc.

I.1.4. Summary of major design and operating characteristics[1]

- Installed capacity (thermal and electric)
- Mode of operation (basic, load follow)
- Load factor/availability (specify targets)
- Summary of major design characteristics: types of fuel, fuel enrichment, types of coolant/moderator, types of structural materials, core type/characteristic dimensions, vessel type/characteristic dimensions, cycle type (direct/indirect), number of circuits
- Simplified schematic diagram of the nuclear installation with an innovative SMR
- Neutron-physical characteristics (temperature and coolant density reactivity effects, void reactivity effect and burn-up reactivity swing,), power flattening (peaking factors, approaches to reduce them)
- Reactivity control mechanism (burnable poisons, control rods, liquid boron, spectral shift, movable reflector, etc or a combination thereof), number of independent active reactor control and protection (RCP) systems, cumulative worth for each RCP system
- Cycle type (direct or indirect) and thermodynamic efficiency
- Thermal-hydraulic characteristics (circulation type: natural/forced, inlet/outlet coolant temperatures, flow rates and pressures in circuits, temperature limits for fuel/claddings and other ‘critical’ structural materials, maximum/average temperatures of fuel and structural materials in normal operation, DNBR, if any)
- Maximum/average discharge burn-up of fuel (% FIMA or MW day/kg)

[1] Any other relevant parameters could be added by the designer.

- Fuel lifetime/period between refuellings in effective full power days (EFPD)
- Mass balances/flows of fuel and non-fuel materials (on an annual basis per unit of thermal and equivalent electric energy produced) with a short description of how they were obtained (best estimate, reference etc)
- Design basis lifetime for reactor core, vessel and structures
- Design and operating characteristics of systems for non-electric applications, including process type, ranges for sharing energy production between different applications and specific production rate per unit of thermal and equivalent electric energy
- Economics (capital costs, estimated construction period, O&M costs, fuel costs, final product costs for a prototype and relevant projections for a final version of the nuclear installation with an innovative SMR)

I.1.5. Outline of fuel cycle options

- Standard fuel cycle (specify once-through, closed etc. for basic type of fuel)
- Alternative fuel cycle options (specify once-through, closed etc. for alternative types of fuel)
- Suggested fuel reprocessing method, if any
- Provision for fuel cycle organization (such as centralized, regional, or on-site reprocessing, etc.), if any
- SNF management and disposal planning, if any

I.1.6. Technical features and technological approaches that are definitive for nuclear installation performance in particular areas

I.1.6.1. Economics and maintainability

- Targeted markets and their specifics (for example, low labour costs and limited resources for investments in some developing countries etc)
- Provisions for reduced capital and construction costs, such as design standardisation, option of factory fabrication, transportability etc
- Provisions for low O&M costs, such as elimination of the need for on-site refuelling, long-lived core etc
- Provisions for low fuel reload costs (such as low enrichment, particular fuel or fuel cycle type etc)

I.1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

- Provisions for low consumption of non-renewable resources, including the degree of fuel utilization
- Provisions for minimum generation of wastes at the source
- Provisions for acceptable or reduced dose limits
- Provisions for low SNF and waste management costs (such as particular fuel forms, minimized specific production of waste etc)

I.1.6.3. Safety and reliability

- Safety concept and design philosophy
- Provisions for simplicity and robustness of the design
- Active and passive systems and inherent safety features
- Structure of the defence-in-depth
- Design basis accidents and beyond design basis accidents
- Provisions for safety under seismic conditions
- Probability of unacceptable radioactivity release beyond the plant boundaries
- Measures planned in response to severe accidents

I.1.6.4. Proliferation resistance

- Technical features to reduce the attractiveness of nuclear material for nuclear weapon programmes, such as isotopic content, chemical form and radiation properties
- Technical features to prevent the diversion of nuclear material
- Technical features to prevent the undeclared production of direct-use material
- Technical features to facilitate nuclear material accounting and verification

I.1.6.5. Technical features and technological approaches used to facilitate physical protection of the NPP with an innovative SMR

Such as: features that ensure enhanced protection against external impacts and sabotage due to intentional actions of the personnel, etc

I.1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of reactor installation

Such as:

- Taking into account (at the design stage) market demands and the specific needs of different market, in particular, needs of developing countries
- Provisions for leasing of fuel and/or full-scope fuel cycle service agreements
- Options for NPP leasing

I.1.8. List of enabling technologies relevant to the nuclear installation with an innovative SMR and status of their development

I.1.9. Status of R&D and planned schedule

Including:

- Information on whether R&D for this particular SMR are supported by national (State) R&D or NPP deployment programmes
- Companies/Institutions involved in the RD&D and design
- Estimate of an overall time frame within which the design could be implemented
- Information on main RD&D and licensing stages and their duration

- R&D costs needed to deploy the prototype, R&D costs needed to deploy final version of the NPP with an SMR (assume all R&D are performed in a specified country)
- Financial information, if any, including the status of funding

I.1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed,

i.e., why this particular SMR could be rated as innovative

I.1.11. List of other similar or relevant SMRs for which the design activities are ongoing

I.2. Design description and data for each nuclear installation with an innovative SMR

I.2.1. Description of the nuclear systems

- Reactor core and fuel design, including core map and cross section of fuel
- Main heat transport system, with specification of heat removal path in normal operation and in accidents
- Intermediate circuit, if any

I.2.2. Description of the turbine generator plant and systems (details to be given if innovative equipment is to be used)

I.2.3. Systems for non-electric applications

- Outline of the circuits, systems, and processes for non-electric applications
- Table of basic design data

I.2.4. Plant layout

- General philosophy governing plant layout
- Reactor building and containment layout
- Plant plot, if available

References

Appendix II

OUTLINE FOR SMR DESIGN DESCRIPTION (SHORT DESCRIPTION)

II.1. Basic summary, including:

- Full and abbreviated name of the concept, principal stakeholder(s)
- Core design summary
- Plant design summary
- Safety concept summary

II.2. Major design and operating characteristics

II.2.1. Table of major reactor (core) characteristics

- Fuel type and enrichment
- Fuel assembly type and number
- Type of structural materials
- Fuel burn-up and cycle length
- Core dimensions; reflectors, if any
- Approach to power flattening
- Average power density
- Major reactivity effects
- Breeding ratio, if applicable
- Decay heat removal systems
- Other characteristics suggested by the designer

II.2.2. Table of major plant characteristics

- Reactor type
- Thermal and electric output
- Plant efficiency
- Cycle type (direct or indirect), secondary coolant (intermediate coolant, if any)
- Circulation type (natural or forced)
- Core inlet/outlet temperature, primary circuit pressure, and primary coolant flow rate
- Turbine inlet temperature and pressure
- Containment system
- Other characteristics suggested by the designer

Figure(s) illustrating plant design scheme

II.3. List of enabling technologies and status of their development

References

CONTRIBUTIONS FROM MEMBER STATES — DESIGN DESCRIPTIONS OF SMALL REACTORS WITHOUT ON-SITE REFUELLING

WATER COOLED SMALL REACTORS

ANNEX I

MULTI-APPLICATION SMALL LIGHT WATER REACTOR (MASLWR)

**INEEL, Oregon State University and Nexant-Bechtel,
United States of America**

I-1. General information, technical features and operating characteristics

I-1.1. Introduction

The Multi-Application Small Light Water Reactor (MASLWR) design was developed through a collaborative effort sponsored by the nuclear energy research initiative (NERI) programme of the U.S. Department of Energy (DOE). The design and testing team was comprised of staff from the Idaho National Engineering and Environmental Laboratory (INEEL), Oregon State University (OSU), and NEXANT-Bechtel. The primary objectives of the project were to develop the conceptual design for a safe and economic small, natural circulation light water reactor, to address the economic and safety attributes of the concept, and to demonstrate its technical feasibility by conducting testing in a scaled integral test facility.

The design has evolved from an initial concept that employed a primary system layout similar to a traditional pressurized water reactor (PWR), with U-tube steam generators external to the reactor vessel, with the thermal centers of the steam generators elevated to enhance natural circulation. The preliminary estimates for the early design indicated that the busbar cost would be about 5.7 cents/kWh, which is far above the goal of 4 cents/kWh. It was concluded that cost reductions could be achieved by using smaller, simpler, factory-assembled units. Therefore, the focus of the project was redirected to a self-contained modular reactor design consisting of an integrated reactor vessel, steam generator, and high-pressure containment vessel. The entire reactor module would be shop fabricated and transportable to a site on most railways or roads. This sealed modular design eliminates the need for on-site refuelling.

The MASLWR concept relies on available nuclear technologies. The nuclear core and turbine generators designs were based on configurations used in a typical PWR. This allows the use of current industry expertise and manufacturing capabilities. The novelty of the system comes from the integration of the entire primary side into a single modular unit with an independent steel containment, and in the completely passive safety systems. The technical feasibility of the MASLWR concept has been confirmed by the integral system tests conducted at Oregon State University (OSU).

I-1.2. Applications

The MASLWR concept is a small (~150 MW(th)), natural circulation light water reactor that was developed with the primary goal of producing electric power, but includes the flexibility to be used for seawater desalination. Its modular design offers the flexibility of adding capacity as demand increases or as financing permits.

I-1.3. Special features

Special features of the MASLWR concept are:

- *Off-site fabrication of sealed reactor modules:* Because of their relatively small scale, each MASLWR reactor module can be fabricated, inspected and sealed in a quality controlled, secure environment located off-site;

- *No on-site refuelling*: The reactor modules are designed such that they can easily be moved (or shipped off-site) to a centralized facility for refuelling. This allows a "pull and replace" approach for refuelling of the NPP, which reduces the threat of diversion and improves proliferation resistance. This may be a significant advantage to developing countries that may not have the infrastructure required for fuel fabrication. Sealed modules can be monitored remotely and need not be opened on-site;
- *Long core life*: The core life will be five effective full power years;
- *Incremental capacity addition*: The MASLWR concept has a relatively small energy output of 150 MW(th), and each of the power generation units (see Fig. I-1) are independent. This allows flexibility in total plant power for a wide range of applications;
- *Flexible application/transportation*: The reactor module, which includes the primary vessel with the reactor and the steam generator along with the containment vessel, is 4.3 m (14 ft) diameter, 18.3 m (60 ft) long and weighs 275 metric tons (303 tons), and is transportable to a site on most railways or roads. The entire system, including balance of plant, can be shipped in 3 sections. This eliminates the requirement of a large pre-existing infrastructure to deploy the system;
- *Simplified operational licensing requirements*: The MASLWR concept utilizes natural circulation for primary coolant flow and relies on passive safety systems. This results in a reduced amount of operator training and plant certification required for plant licensing. Due to the small scale of the concept, design certification based on testing is simplified because of the relatively low cost of a full-scale prototype facility.

I-1.4. Summary of major design and operating characteristics

The layout of a single MASLWR unit for electrical power generation is shown in Fig. I-1. The unit consists of three basic modules: the reactor module, the turbine generator module, and the main condenser module. A general view of a MASLWR baseline plant comparable to a current NPP is shown in Fig. I-2. The baseline plant consists of 30 power generation units (1050 MW(e)). However, smaller plants with as few as a single power generation unit can be built if desired.

Figure I-3 shows a schematic diagram of a single MASLWR plant module with the reactor immersed in its external cooling pool. As shown in the figure, the reactor pressure vessel houses a nuclear reactor core that is surrounded by a shroud connected to a tall vertical riser. A helical coil steam generator wraps around the outside of the riser, filling the annular region between the riser and the inside of the pressure vessel wall. The top of the reactor pressure vessel contains a steam bubble that acts to control primary system pressure.

During normal operation the primary coolant enters the bottom of the core and is heated, causing the density to decrease. The density decrease provides a buoyancy force, with respect to the relatively lower temperature fluid in the downcomer. The net force drives the heated coolant up to the top of the riser section, where the coolant reverses direction and travels down the annulus/downcomer. The coolant about the steam generator tubes transfers its energy to the secondary side feedwater, and completes a cycle upon entering the bottom of the core. The secondary side cold feedwater enters the SG tubes at the bottom, and slightly superheated steam is collected at the top. The generated steam is passed to the turbine module and expanded in a turbine generator. Next, the waste steam is passed to the condenser module and condensed to a single-phase liquid and pumped back to the steam generator. All of the secondary side components are commercially available.

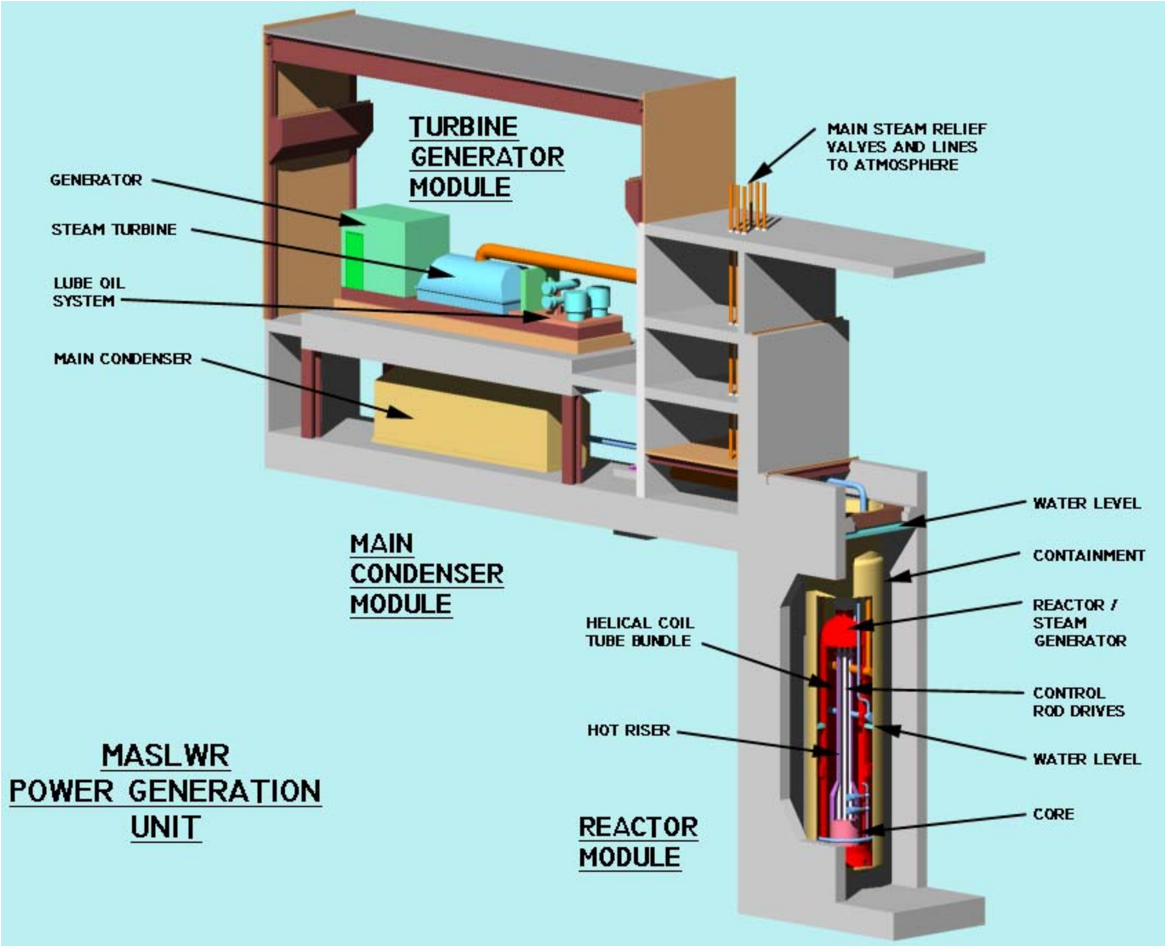


FIG. I-1. Layout of the MASLWR power generation unit.

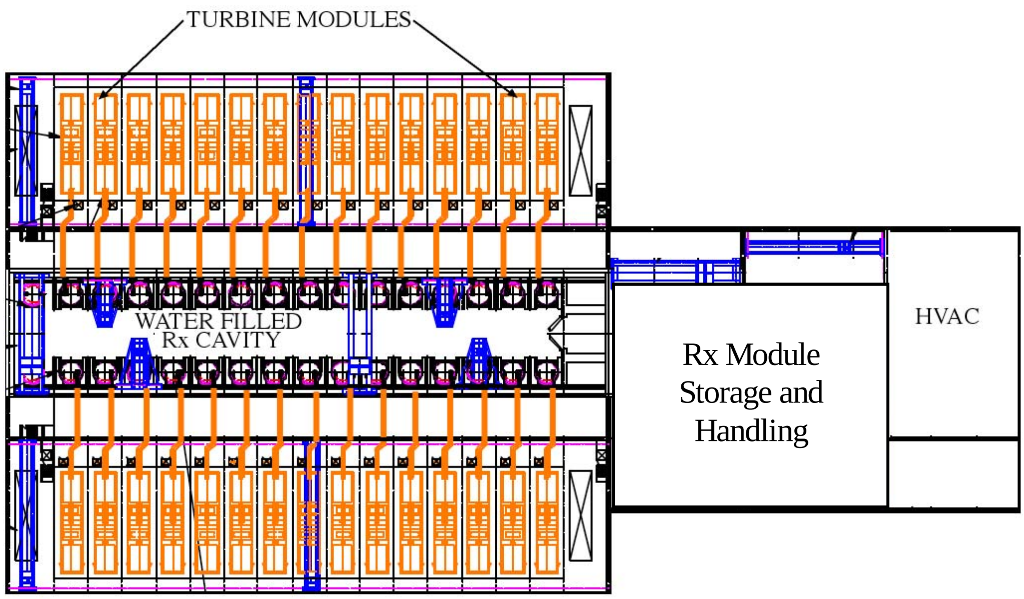


FIG. I-2. Plan view of the MASLWR baseline plant generating 1050 MW(e).

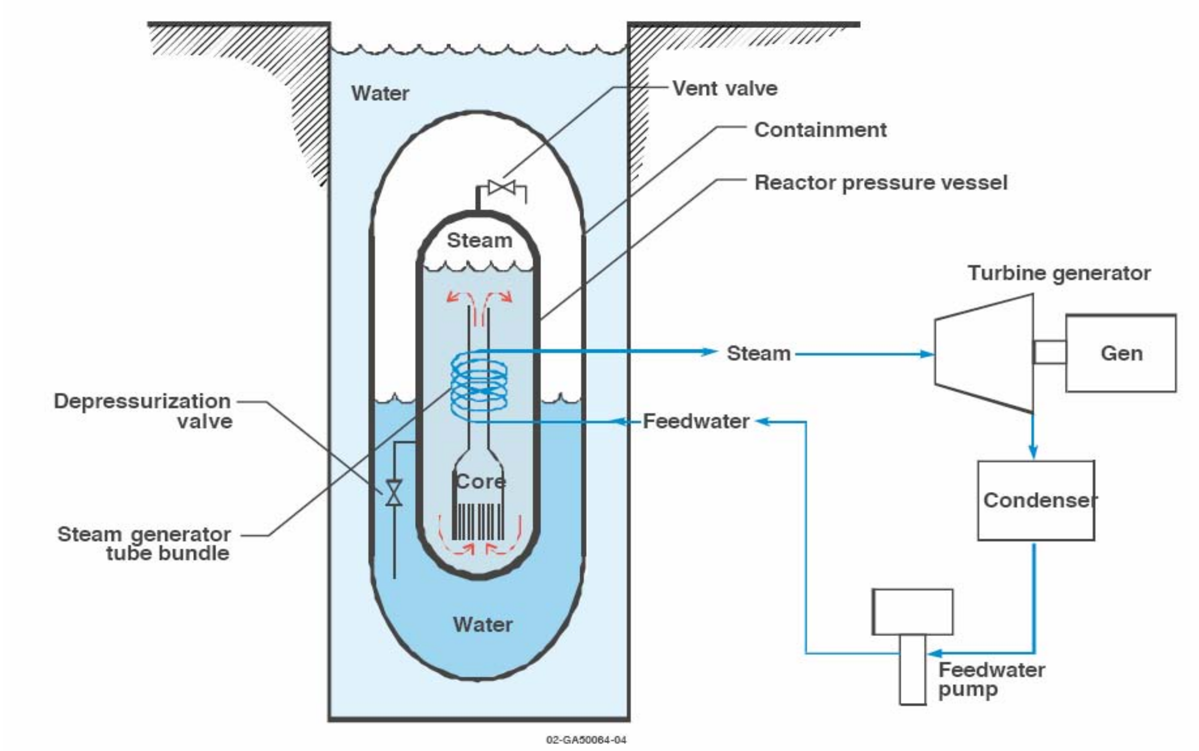


FIG. I-3. Schematic diagram of MASLWR plant (one module).

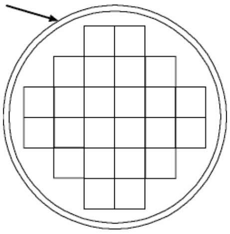
The reactor pressure vessel is housed inside of a high-pressure containment vessel partially filled with borated water. The high-pressure containment is immersed in an external cooling pool that serves as the ultimate heat sink for emergency core heat removal.

Table I-1 provides a summary of the MASLWR design and operating characteristics.

TABLE I-1. MASLWR DESIGN AND OPERATING PARAMETERS

CHARACTERISTIC	VALUE
Rated thermal power	150 MW(th)
Rated electric power	35 MW(e)
Type of Fuel	Typical PWR fuel rods: • UO_2 ceramic cylindrical pellet; • Pellet diameter — 8.259 mm; • Zircalloy-4 cladding.
Fuel enrichment	Fissile materials — 8% by weight Fertile materials — 92% by weight
Coolant	Light water
Moderator	Light water
Structural materials of reactor internals	All structural components are made of stainless steel

TABLE I-1. (continued)

Core 	24 assemblies, 2-4-6(×2)-4-2 pattern Assemblies are a 17 × 17 PWR design	
	Number of fuel rods	6336
	Heated length	1.35 m
	Equivalent diameter	1.2 m
	Power density	100 kW/litre
Reactor vessel	Cylinder with spherical upper/lower heads	
	Design pressure	8.6 MPa
	Design lifetime	60 years
	Outer diameter	2.74 m
	Overall height	13.7 m
	Wall thickness	12.7 cm
Containment vessel	Cylinder with elliptical upper/lower heads	
	Outer diameter	4.3 m
	Overall height	17.7 m
	Design pressure	2.1 MPa
Cycle type	Indirect cycle with 23% efficiency[1]	
Number of loops	1	
Nominal plant capacity factor	95%[1]	
NPP type	Modular component	
Mode of operation	Baseline with limited load following using units in on/off pattern	
Neutron-physical characteristics	Power flatting by: • Varying fuel loadings; • Placement of fixed burnable poison in fuel array.	
	Temperature reactivity coefficient	(–)0.08 $/°C
	Doppler coefficient	(–)0.005 $/°C
	Maximum axial peaking factor	~1.36
	Hot assembly factor	1.1
	Hot pin factor	1.4
Reactivity control mechanisms	• Control rods in the active core region, with one centrally located secondary shutdown magnetically held rod. • Central control rod worth (BOC) — (–)$2.05	

[1] Based on the 30-unit plant shown in Fig. I-2.

TABLE I-1. (continued)

Thermal-hydraulic characteristics	Circulation type — Natural circulation	
	Normal operating fuel temperatures:	
	Average fuel temperature	627°C
	Maximum centreline fuel temperature	~1 293°C
	Coolant inlet temperature	218.8°C
	Coolant outlet temperature	271.4°C
	Coolant flow rate	596 kg/s
	Coolant pressure	7.59 MPa
	Steam generator pressure	2.1 MPa
	Steam generator flow rate	56 kg/s
	Control rod ejection accident conditions:	
	Fuel maximum temperature	~1366°C
Fuel cycle	Core load	4 t heavy metal (HM)
	Refuelling interval	1825 EFPD/ 60 months
	Fraction of core refuelled	1
	Discharge average burn-up	67 MWd/kg
	Annual specific consumption of natural uranium, kg/(GW(e)year)	348 896, based on 2 weight % tails assay; 260 526, based on 1 weight % tails assay.
Seawater desalination	58 295 m^3/day from Reverse Osmosis for a 1050 MW(e)/3621 MW(th) plant	
Economics	Capital costs (2002 US$ dollars)	
	Lead plant: Capital costs	1458 US$/kW(e)
	N^{th} plant:	
	Capital costs	1241 US$/kW(e)
	O&M costs	0.7 US$ cent/KWh
	Fuel costs	0.00685 US$/kWh based on 2 weight % tails assay 0.01271 US$/kWh based on 1 weight % tails assay

I-1.5. Outline of fuel cycle options

The MASLWR concept utilizes a standard PWR style fuel assembly, which allows easy integration into an existing fuel cycle. The only differences between the MASLWR fuel assembly and existing PWR assemblies is that the MASLWR assembly is only about half as tall and uses an enrichment of 8%. Hence the exact characteristics of the fuel cycle will depend on the deployment, i.e., in the USA it would be a once-through fuel cycle, whereas in countries like France or Japan, the fuel cycle could include reprocessing.

In addition to the use of traditional UO_2, the possibility of utilizing thorium-based fuels has been analyzed for MASLWR. A comparison of the burn-up of the standard and alternative fuel types was performed through calculations with the MOCUP code. For a single batch core, the reactivity with burn-up is the same regardless of the specific power for similar fuel enrichments, with uranium fuels outperforming thorium-uranium fuels at a 75 weight % (ThO_2) — 25 weight % (UO_2) ratio, see Fig. I-4.

Because the MASLWR reactor module is relatively small, a novel approach is used to perform refuelling and maintenance activities. A spent reactor module is removed and transported underwater to an adjacent storage and handling facility. It is replaced with a spare reactor module. After a cooling period, the spent reactor module is shipped to a secure off-site refuelling/maintenance facility. This results in a much shorter down time for refuelling.

Once moved to the off-site refuelling/maintenance facility, the reactor module is subject to the process of disassembly to access the reactor/steam generator pressure vessel. Since the containment is entirely welded, it is necessary to cut the containment at one place in the cylindrical section. The sequence of module disassembly is shown graphically in Fig. I-5. The disassembly is performed in such a way that the reactor core remains submerged in water at all times. Reassembly of the reactor module is performed in the reverse sequence to assembly. The end of core fuel will be removed to temporary storage for a cool-off period. After decay, spent fuel is shipped off-site for disposal/reprocessing, and then moved to a final disposal location. Given the similarities of the MASLWR fuel to PWR fuel, the storage and disposal of the MASLWR fuel would be in the currently proposed deep geologic repositories (or similar facilities) alongside with the standard PWR/BWR fuel assemblies.

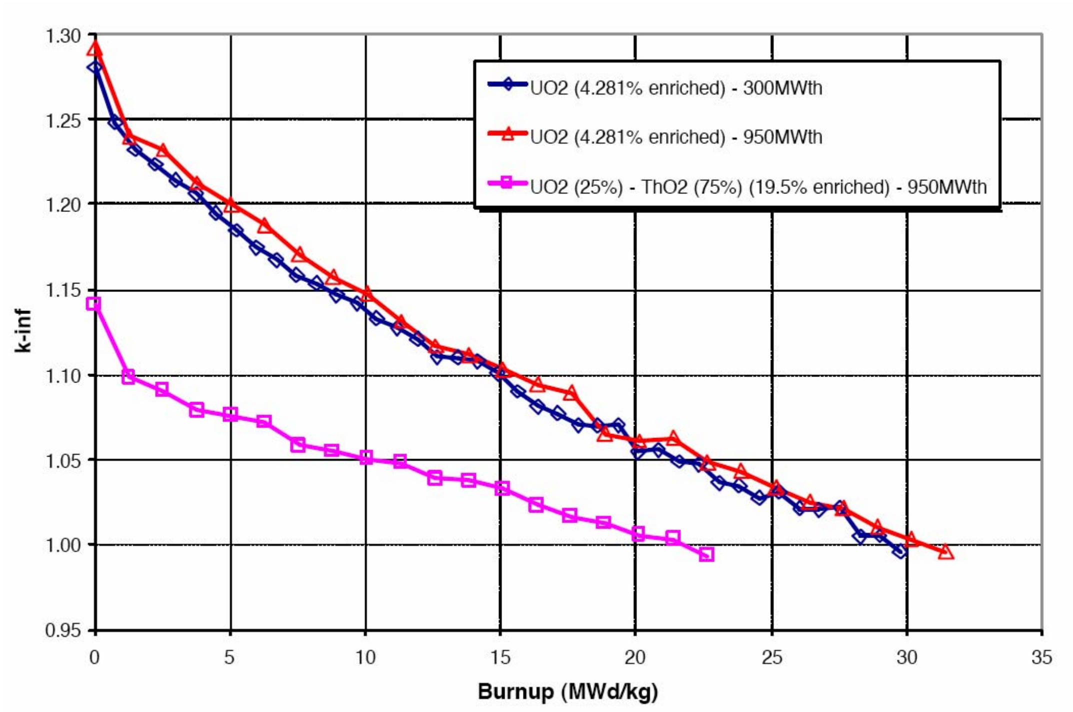


FIG. I-4. Reactivity versus burn-up of uranium and thorium-uranium fuel.

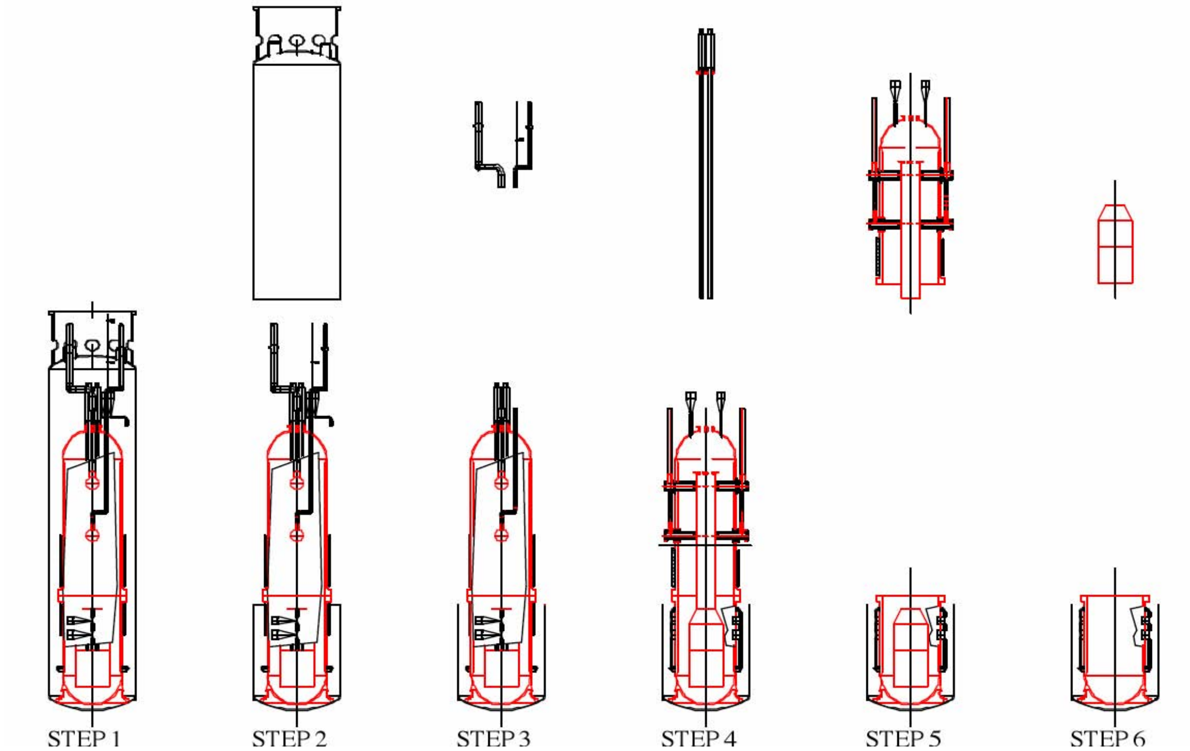


FIG. I-5. Reactor module disassembly.

I-1.6. Technical features and technological approaches that are definitive for MASLWR performance in particular areas

I-1.6.1. Economics and maintainability

Capital and construction costs are minimized in the MASLWR concept by taking full advantage of maximum modularization. This is facilitated by the small size of the modules and their ease of transport. Modularization is adopted because it is expected to provide the following advantages over conventional field fabrication:

- Lower capital costs because most components are “off-the-shelf”,
- Shorter construction schedule, better control of capital costs, and improved component quality because of factory fabrication;
- Improved productivity and quality of fieldwork because system is shipped in prefabricated modules;
- Enhanced security at a reduced cost because of sealed reactor module construction.

A reduction in O&M costs is realized in the MASLWR concept by incorporating the following design features:

- Simple design with extensive use of passive systems (in normal & accident scenarios) that eliminate expensive and maintenance intensive AC powered engineered safety systems;
- Extensive automation of control and surveillance functions that minimize the need for manual operations;

- The use of advanced diagnostics, advisory and decision making systems;
- Five-year core life and "pull-and-replace" core module reduces costs associated with refuelling outages.

The "pull and replace" core module and natural circulation core (i.e., no reactor coolant pumps) essentially reduces expensive on-site maintenance and in-service inspection requirements for the nuclear steam supply system components.

I-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The MASLWR design implements proven fuel technology and many standard components. Fuel and replacement components could be obtained from several different vendors. Additional advancements in PWR fuel design and materials would be readily applicable to MASLWR. Furthermore, the MASLWR is designed to operate at lower temperatures and pressures than a typical PWR, reducing the stress placed on these advanced materials.

The MASLWR design benefits from the efforts for PWR/BWR fuel diversification, and can exploit all the same possibilities being considered for use in PWRs. This includes the utilization of mixed oxide (MOX) and thorium fuels and advances in reprocessing as the technologies become available and find acceptance. MASLWR also takes advantage of the activities on PWR spent nuclear fuel cask design, transportation methods, and disposal technologies.

MASLWR would have a small footprint with regard to land use. Since each reactor module has its own high-pressure containment vessel submerged in a water filled underground silo, a large containment building is not required. Furthermore, since each module has a relatively small fission product inventory, emergency action plans could be simplified and the evacuation area could, potentially, be reduced.

I-1.6.3. Safety and reliability

MASLWR incorporates the following design approaches to assure plant safety:

Multiple barriers to fission product release

MASLWR will implement the state-of-the-art materials for fuel and cladding, the reactor pressure vessel, and the high-pressure containment submerged in a water-filled underground silo, all to be housed within a reactor building;

Inherent safety features

MASLWR has the following inherent safety features:

- Small fission product source term per module that is a significant factor in mitigating dose to the public in the unlikely event of a release of radio nuclides;
- Very low decay power per module. Core cooling in each MASLWR module is readily achieved by natural convection because the reactor operating power for each module is low;

- Negative temperature and Doppler reactivity coefficients. The negative coefficients of reactivity provide stable plant neutronics;
- No external primary loop piping. The use of a natural circulation driven integrated reactor vessel eliminates the potential for hot and cold leg breaks, pressurizer surge line nozzle cracking and breaks, reactor coolant pump suction line breaks, and reactor coolant pump seal breaks.

Passive safety systems

MASLWR implements safety systems that do not rely on actively powered external systems such as pumps. Analyses indicate that the passive safety systems are able to keep the peak cladding temperature below the design limit of 1200ºC (2200 ºF) for design basis accidents. The following passive systems have been incorporated into the MASLWR design:

- Primary loop natural circulation. Natural circulation/convection cooling of the reactor core via helical coil heat exchangers or via the containment pool;
- Passive containment cooling. Passive cooling of containment via steam condensation, natural convection and conduction heat transfer to the containment walls and exterior pool;
- Passive reactor scram. Passively actuated secondary reactor scram, which consists of a control rod held by electromagnetic field. The shutdown control rod, centrally located, provides –$2.05 worth and is capable of fully shutting down the reactor;
- Automatic depressurization system (ADS). The ADS provides a controlled depressurization of the plant. It is actuated passively by the pressure difference between the reactor vessel and an accumulator filled with water and nitrogen;
- Safety relief valve. The plant safety relief valve used in an overpressure scenario is designed not to reseat until primary system pressure has fallen below the set pressure for activation of the passive ADS.

The MASLWR plant reliability is enhanced relative to conventional plants by:

- Minimizing the number of active components required to operate and maintain the plant. The potential negative impacts to plant reliability due to loss of off-site power are eliminated through the use of primary loop natural circulation;
- Module independence. Shutdown of a single module does not result in a complete plant shut down; thus reducing plant downtime;
- Fabricating high quality components under controlled factory conditions. Factory fabrication of each module under controlled conditions enhances component quality and increases the likelihood of proper field assembly;
- Providing extensive automation of control and surveillance functions;
- Using advanced diagnostics, advisory and decision making systems;
- Minimizing required operator actions. Each module is designed for 5 years of continuous operation at base load conditions. The main role of the operator is one of monitoring and verifying that the plant operates as intended. The operator is only required to initiate plant start-ups, plant shutdowns, set or correct set points that control plant operation, and take corrective actions if the plant or systems do not operate as intended.

Operation of MASLWR passive safety systems

Figure I-6 illustrates the passive safety systems for MASLWR. Under accident conditions, both the ADS submerged blow-down nozzles and the ADS steam vent valves are opened. As a result, steam is vented into the high-pressure containment dome and a two-phase fluid mixture is vented into the sub-cooled containment pool. This process equalizes the pressure between the containment and the reactor vessel, effectively terminating the break that may have initiated the accident. The steam exiting the steam vent valves condenses on the containment walls transferring energy to the containment walls. Heat conduction through the containment walls and natural convection heat transfer to the external cooling pool effectively remove the decay heat generated by the core. The condensate falls by gravity down the wall of the containment to the containment pool. During an accident, the relative water levels are opposite of that shown in Fig. I-6, and this provides a driving head to flow coolant back into the reactor vessel. The circulation of this coolant provides safe core cooling for the duration of the accident.

Separate performance and safety studies have been performed for the MASLWR design using a systems code and an integral test facility for normal and accident conditions. The purpose of these studies was to evaluate the transient performance characteristics of the design and to demonstrate that the passive safety features provide adequate protection for the core under design basis accident conditions. INEEL used the RELAP5-3D thermal-hydraulic systems code in its safety analysis. The Oregon State University Nuclear Engineering Department constructed and operated a reduced scale integral system test facility to assess plant performance and passive safety system operation. The following sections present the results of those studies.

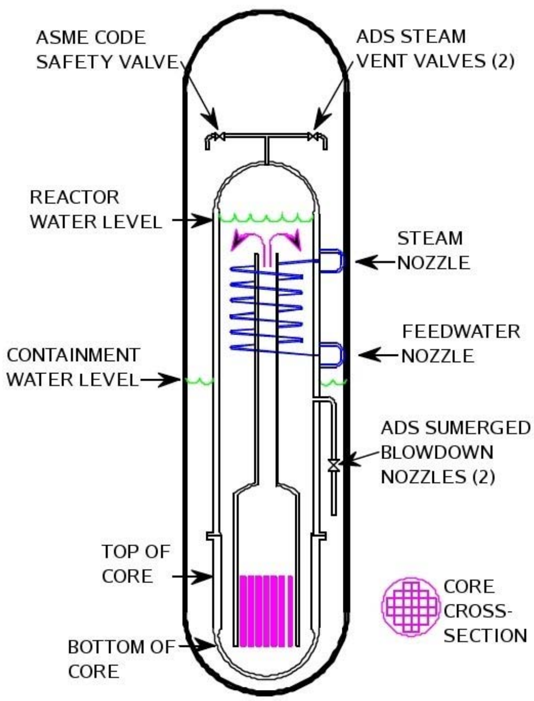


FIG. I-6. Internal components, containment water, and reactor water levels.

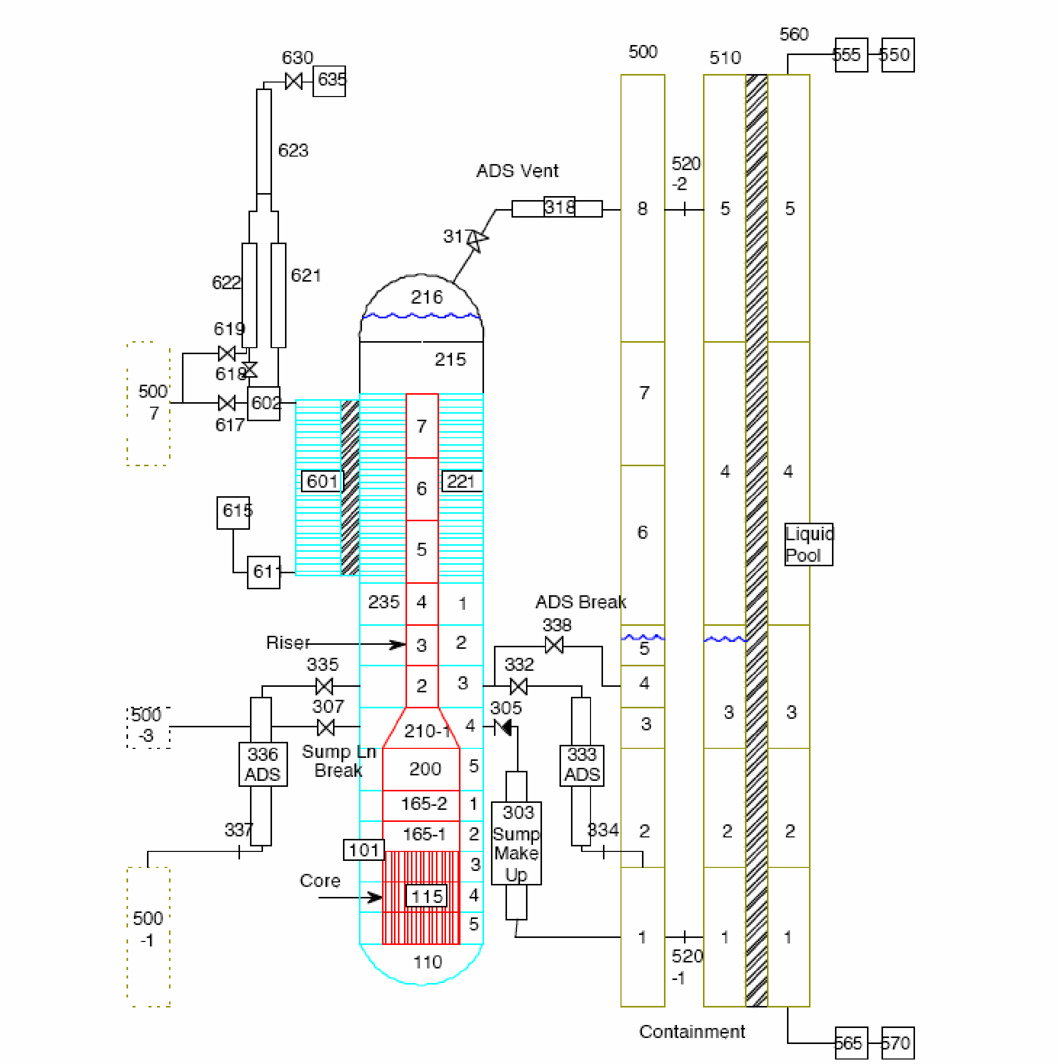


FIG. I-7. RELAP 5-3D model of the MASLWR reactor module.

RELAP5-3D performance and safety analysis

The RELAP5 nodalization diagram of MASLWR is shown in Fig. I-7. A more in-depth description of the RELAP model and related analyses are given in [I-1]. The RELAP5 performance and safety analysis studies considered a variety of hypothetical accident scenarios grouped by estimated frequency of occurrence, and included acceptance criteria for these scenarios that are typical of present day PWRs.

The steady-state simulation showed that the average reactor core channel operates in sub-cooled forced convection, with the hot core channel is in sub-cooled nucleate boiling. A minimum critical heat flux (CHF) ratio of 7.2 was calculated for the hottest location, indicating a large margin of safety during normal operation. The initial conditions obtained from the analysis of normal operation were used as the initial condition in the accident studies.

ADS and steam vent line breaks

The RELAP5-3D MASLWR model was used to calculate the sequence of events for an ADS line break and a steam vent line break. The break cases serve to bound transients that involve inadvertent opening of the ADS or steam vent valves. Fig. I-8(a) shows the core hot channel collapsed liquid level and fuel cladding surface temperature at the core hot location for the ADS line break scenario. The results show that core collapsed liquid level is sufficient to provide cooling to the fuel and that neither the fuel nor its cladding experience a thermal excursion.

Figure I-8(b) shows similar results for the steam vent line break. It is observed that heat rejection through the containment wall quickly exceeds core decay power, thus demonstrating the effectiveness of the liquid pool as an ultimate heat sink for long-term control of system pressure and removal of core decay heat. It also shows the equalization between reactor vessel pressure and the containment pressure, effectively stopping the break flow. The maximum containment pressure reached is 1.1 MPa (160 psi).

Inadvertent opening of steam vent valve with failure of reactor to scram

Beyond design basis accidents include anticipated transients with a failure of the reactor to scram. The most severe of these cases is the inadvertent opening of one of the steam vent valves without scram. Reactor power and hot fuel pin centreline temperature responses are shown in Fig. I-9. In this transient, the initial depressurization results in voiding of the core, thereby causing reactor power to decrease. Subsequently, however, the core is reflooded with cooler liquid and a power spike occurs. The vapor production in the core causes the core to void and the power again decreases. The cycle repeats until the combination of core average void fraction and system temperature results in a net negative reactivity. As shown, the power spikes have maximum amplitude of approximately 3500 MW. The spikes are very narrow, however, and significant energy deposition into the fuel does not occur. Maximum fuel enthalpy increase is about 70 Cal/g. No significant fuel heat up is noted during the transient, and there is no fuel damage.

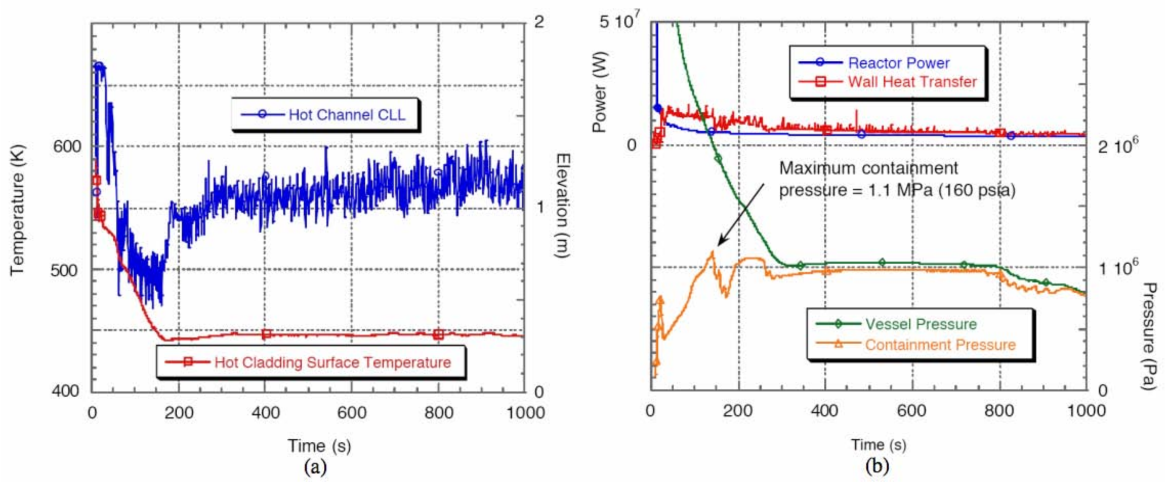


FIG. I-8. (a) Results for ADS line break (b) Results for steam vent line break.

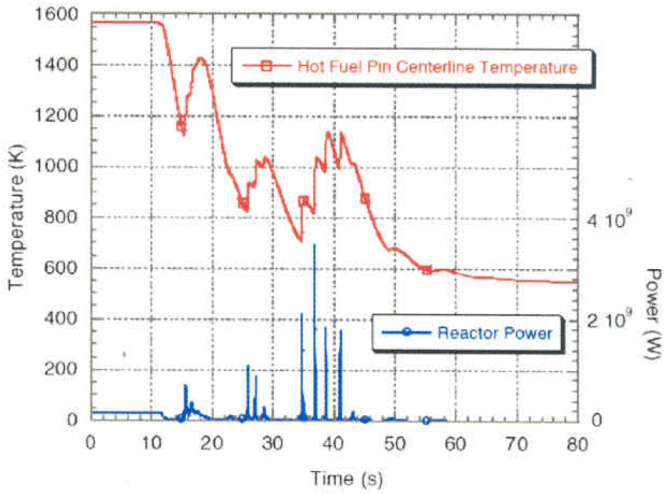


FIG. I-9. Inadvertent opening of steam vent valve with a failure of reactor to scram.

OSU-MALSWR integral system testing

A reduced scale integral system test facility has been constructed and operated by the Department of Nuclear Engineering at Oregon State University (OSU) in an effort to assess the operation and safety of the MASLWR design. This facility is designed based on a detailed scaling analysis performed using a single 150 MW(th) MASLWR reactor module as the baseline design. The scaling analysis included the development of a Phenomena Identification and Ranking Table (PIRT) for the inadvertent opening of an ADS and/or steam vent valve. The scaling analysis was used to establish the component geometry and system operating conditions for the test facility. The OSU MASLWR scaling analysis results have been issued as a separate report.

The OSU MASLWR test facility has been constructed using all stainless steel components. It is designed for operation at full system pressure and temperature. It includes a complete reactor vessel module with helical coil steam generator, an electrically heated fuel bundle simulator, a high-pressure containment vessel, and an exterior pool for passive containment cooling. All components are of 1:3 height-scale and 1:254.7 volume scale. A detailed description of the test facility is given in [I-1].

The initial OSU test programme consisted of an assessment of normal operation, an inadvertent opening of a submerged ADS line, and an inadvertent opening of a steam vent valve. The results for the inadvertent actuation of a submerged ADS vent line are shown in Fig. I-10 and Fig. I-11. These results agree with the RELAP5 calculations that indicate that the core remains covered and that the containment pressure remains below 1.1 MPa during the duration of the test.

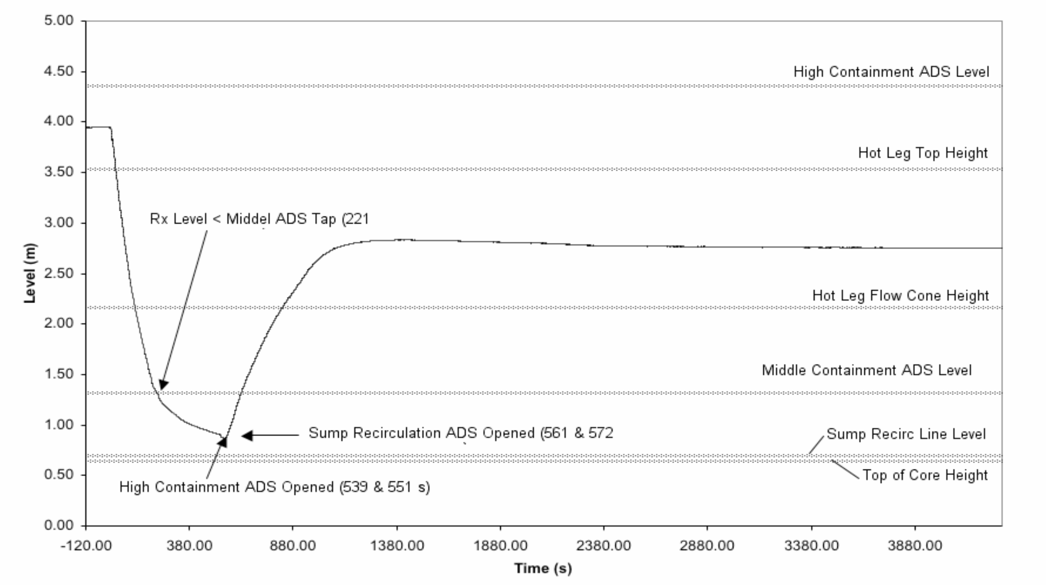


FIG. I-10. Reactor vessel liquid level during an OSU-MASLWR test simulating an inadvertent opening of the submerged ADS valve.

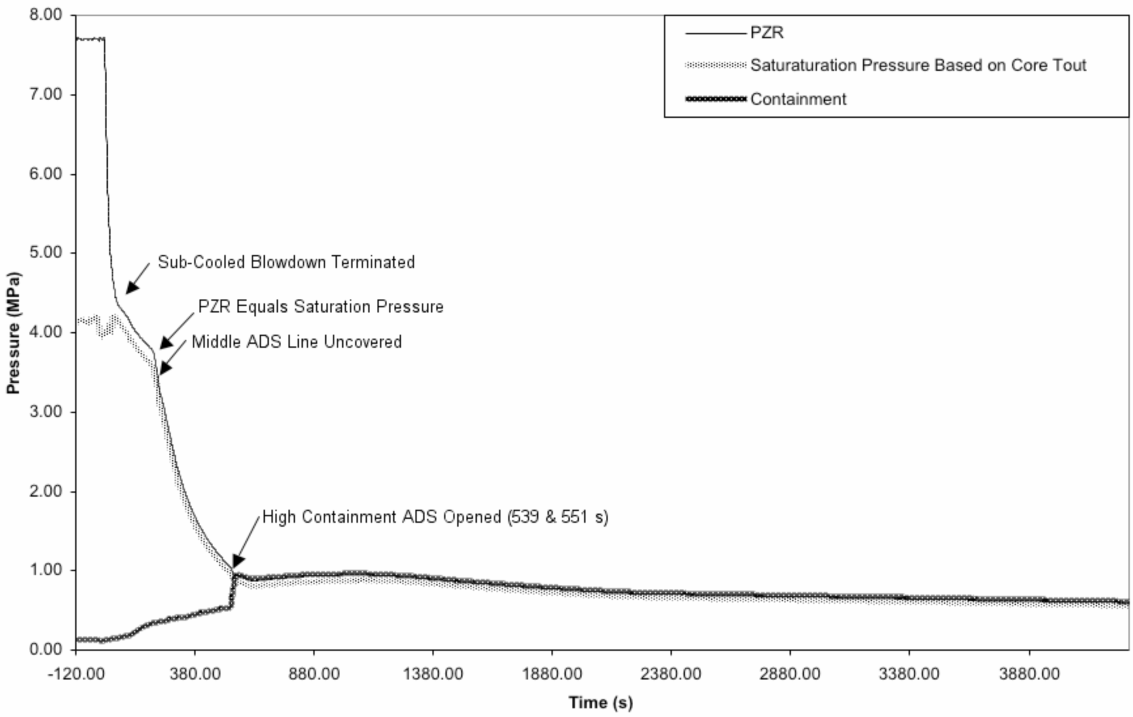


FIG. I-11. Primary and containment pressures during an OSU-MASLWR test simulating an inadvertent opening of the submerged ADS valve (PZR is for pressurizer).

The performed initial code analyses and integral system testing are very promising. Thus far, the safety analysis indicates that integrity of the fuel, the reactor structures, and the containment is maintained for several accident conditions. Additional testing and code analyses are required to analyze the impact of thermal-hydraulic/neutronic coupling on flow stability and to support probabilistic safety studies. This includes estimating potential dose at plant boundaries for a given event.

I-1.6.4. Proliferation resistance

In addition to adhering to safeguard standards for the protection of nuclear materials, the MASLWR uses the following design features and administrative controls to minimize unauthorized access to its nuclear fuel:

No on-site refuelling or fuel shuffling

The MASLWR reactor fuel is transported and operated inside a seal-welded pressure vessel. The reactor module refuelling and maintenance are assumed to be conducted by authorized personnel at a secure refuelling factory. The reactor module is never opened on-site.

Remote monitoring

Because MASLWR transport and operation is in a sealed condition, it is possible to remotely monitor an unsealed condition, indicating a breach in containment or an attempt to divert fuel.

Low enriched fuel

Even though the core module is small when considering automated refuelling, it is large when considering covert removal. In the unlikely event of diversion of a core unit, its enrichment is 8% at BOC, which is within non-proliferation guidelines. Hence, the diverted fuel would require enrichment activities that are costly and consume a large amount of energy (i.e., easy to detect), making this fuel unattractive to potential subterfuge. The highly radioactive nature of the spent fuel makes the diversion of EOC fuel assemblies even less attractive.

Sealed reactor module and isotopic content of fuel reduce the potential for weapon material production

The MASWR is a light water reactor with a conversion ratio of less than unity. The modification of the facility to operate as a fast reactor for the production of plutonium is not probable because of the isotopic content, and the small core size with poor fast neutron economy. Also, the core unit is treated as a module, preventing the reactor from being used for irradiation of fertile material by placement of a dummy fuel assembly in the core unit.

I-1.6.5. Technical features and technological approaches used to facilitate physical protection of MASLWR

In addition to meeting the standards for the physical protection of nuclear facilities, MASLWR incorporates several additional technical features to facilitate plant protection. This includes:

Low profile reactor building

Figure I-12 shows the elevation view of the MASLWR plant. It presents a significantly smaller target than conventional reactors. The placement of the doubly enclosed reactor module underwater and below ground level further reduces external threats.

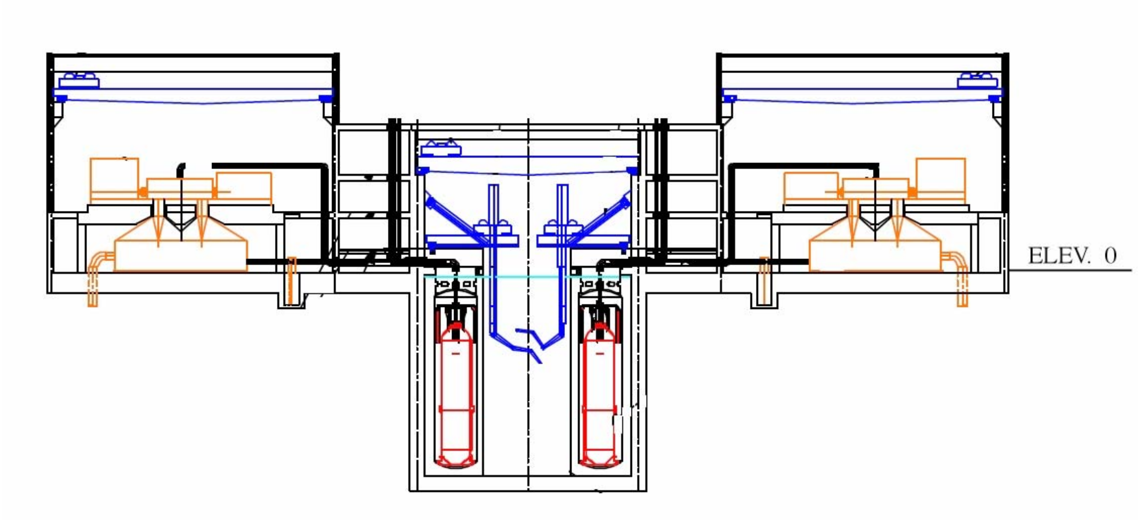


FIG. I-12. Elevation view of the MASLWR plant.

Doubly enclosed reactor fuel

In addition to the reactor building, the nuclear fuel is doubly enclosed in a reactor vessel housed within a thick-walled steel containment vessel capable of withstanding significant internal pressures (e.g., 10 times greater than conventional containments) and significant external pressures (presently not calculated.)

Remote monitoring

Actions that result in opening the weld-sealed containment vessel are remotely detected and result in an immediate response.

A detailed assessment of physical security would be part of future work for the MASLWR.

I-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of MASLWR

The MASLWR has several features that could make it a good choice for near-term development and deployment, among them:

Established technology

MASLWR implements well-established light water technology and uses off-the-shelf components and PWR fuel. In excess of US$2 billion has been invested on LWR research in the United States alone during the past 25 years. Additionally, MASLWR does not require a major computer code development effort.

Ease of safety testing

A 1/3-scale integral test facility has already been constructed and operated to assess plant safety and to benchmark codes. Because of its small size, a full-scale nuclear prototype could be built and tested at a relatively low cost, if required and accepted for certification.

Flexibility and multi-purpose use

Because the plant is modular, it can be customized to fit the specific power or water purification needs of a region. Additional modules could be added as needed.

Reduced capital outlay

Because the plant is modular, minimum capital is required to build a single or dual module plant. Revenues from the initial investment can be used to add additional modules.

Reduced staffing requirements for operation

Because there is no on-site refuelling and minimum on-site maintenance is required, the need for large numbers of specialized outage staff is essentially eliminated.

Several legal, institutional, or infrastructure changes could facilitate the deployment of the MASLWR in both developing and industrialized countries:

A. *Modular unit operating license.* New licensing requirements that would permit the automated control and operation of a multiple modular reactor system from a single control room would significantly reduce costs and personnel requirements;

B. *Licensing by single-module demonstration*. Because of its small size, it could be feasible to build and assess a full-scale MASLWR prototype to obtain regulatory authority design approval and certification;

C. *Remote monitoring requirements for the MALSWR safeguards programme*. It could be beneficial to establish requirements for safeguards programmes that implement satellite based monitoring of individual reactor modules. This includes criteria for the module tracking system selection, installation, inspection and operation. In addition, regulatory guidelines for actions in response to a loss of signal or unauthorized access or movement could also facilitate implementing a very effective safeguards programme;

D. *Design certification reciprocity.* There could be a benefit from establishing design certification reciprocity with countries that would prefer to lease MASLWR modules. Design certification reciprocity could be facilitated by the following actions:

- Providing the nuclear regulatory body of the country considering MASLWR deployment, for their independent review and approval, the full-scale prototype test data and code assessments used for design certification;
- Establishing a framework for shared liability in the unlikely event of an accident (insurance and liability limits);
- Providing a monitoring programme to assure that the modules are properly installed and operated within design limits.

E. *Lease, service and module delivery and recovery programme*. Such programme could be helpful to:

- Define the method for establishing international contracts for MASLWR module leasing, servicing, delivery and recovery. The ability of vendors to provide “turn-key” installation and full-service agreements could facilitate MASLWR deployment in developing countries;
- Establish guidelines for module transport and protection within the country considering deployment.

I-1.8. List of enabling technologies relevant to MASLWR and status of their development

The MASLWR is a light water reactor that relies strongly on existing LWR technology. The reactor fuel materials, the fuel bundle geometry, and the control rod drive system are based on the established LWR technology. Similarly, the secondary side steam turbine/generator set considered for implementation is an industry standard. However, there are several innovative features of the design that require testing and assessment. These include: the helical coil steam generator; the natural circulation mode of plant operation including plant start-up; the external cooling pool operation; the passive safety systems, the high-pressure containment; and the steam generator control system. Each of these areas requires an augmentation from the existing database or the development of new data through testing. A list of enabling technologies relevant to the MASLWR is given in Table I-2.

TABLE I-2. ENABLING TECHNOLOGIES RELEVANT TO MASLWR AND THEIR DEVELOPMENT STATUS

ENABLING TECHNOLOGY	EXPLANATION/IMPLICATIONS	STATUS OF DEVELOPMENT
Fuel technology.	Use of 7–8% enriched UO_2 fuel in standard PWR fuel bundles.	Within the capability of existing enrichment facilities.
Steam turbine and generator set technology.	Existing LWR balance of plant technology is directly applicable.	“Off-the-shelf” designs are available.
Natural circulation driven main coolant system.	A passive system for normal core heat removal.	A 1/3-scale integral system test loop has been constructed and operated at Oregon State University in conjunction with INEEL. It includes all of the primary side and secondary side passive safety systems. Initial testing completed. Current status: • Primary side stability tests being developed; • Secondary side control logic under development; • Neutronic feedback tests being developed.
Automatic depressurization system (ADS).	Passive systems for emergency core cooling.	OSU 1/3-scale test loop tests have been conducted. RELAP5 assessments have been performed. Additional tests being considered.
High pressure containment.		
External pool for containment cooling and decay heat removal.		
Replaceable containment/ reactor vessel modules. (5 year replacement cycle).	Technology to reduce O&M costs.	Requires fabrication in controlled factory environment. Potential fabricators have been identified and contacted.
Structural materials.	All stainless steel construction and material selection to minimize corrosion, specifically, to ensure 60 year vessel design life per module.	
Nuclear desalination technology.		Demonstrated technology.
Significant use of existing technology and off-the-shelf components, and factory fabrication.	Results in reduced capital cost per MW installed.	No R&D required.

I-1.9. Status of R&D and planned schedule

A 3-year nuclear energy research initiative (NERI) programme of the U.S. Department of Energy to develop the MASLWR design and conduct initial safety testing and safety analysis code benchmarks has recently been completed. A 1/3-height and 1/254.7 volume scaled thermal-hydraulic test facility capable of full system pressure and temperature operations has been constructed and successfully operated at OSU. The test facility is capable of operation at 600 kW, which represents the full-scale MASLWR power density. REALP5 assessments of the test data have been performed by INEEL.

The status of the MASLWR project as of December 2003 is described in more detail in [I-1]. The current design stage is that of conceptual design. Additional tests and design improvements are currently being considered. Potential tests include natural circulation flow stability tests with simulated neutronic feedback and high-pressure passive containment cooling tests. Methods of improving plant efficiency are also being considered. Funding for the next series of tests is currently being sought.

I-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The basic innovation of the MASLWR is a lifetime core operation without reloading and shuffling of fuel in a weld sealed reactor containment, which would require a demonstration.

Because it is based on well-established LWR technology and implements benchmarked thermal hydraulic safety analysis codes, and because each module is a relatively small reactor, it is possible that all of the information required for design certification may be obtained using a full-scale demonstration of a single module. Of significant interest for design certification would be the impact of neutronic feedback on flow stability, particularly during plant transients.

I-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The following SMRs have designs similar to MASLWR. A description of these SMRs is given in [I-2].

- CAREM — CNEA/INVAP, Argentina;
- SMART — KAERI, the Republic of Korea;
- IRIS — International Consortium led by Westinghouse, USA.

I-2. Design description and data for MASLWR

I-2.1. Description of the nuclear systems

Core and fuel design

The reactor core was scaled from a typical PWR core, and consists of 24 assemblies of standard 17 × 17 fuel design for a total of 6336 fuel rods. The heated length is approximately 1.35 m (4.43 ft) and the equivalent diameter, $dia = 2\sqrt{\frac{area}{\pi}}$ is approximately 1.2 m.

Figure I-13 shows a cross-section view of the core, and Fig. I-14 shows a cross-section of an individual fuel rod. The design average power density at 150 MW(th) is 100 kW/litre. The 17 × 17 assemblies have a pitch of approximately 21.5 cm. The core is arranged so that a

2-4-6(×2)-4-2 assembly pattern is formed. The MASLWR core concept retains typical features of the PWR core, including fuel rod diameter, lattice pitch, sub-channel geometry, parameters defining grid spacer hydrodynamic performance, and lower and upper core plate configurations. Therefore, the hydraulic performance calculated by the existing thermal-hydraulic software should represent the characteristics of the MASLWR core reasonably well. Also, the core design is consistent with existing industry experience and manufacturing capabilities. Note that optimization of the core should be performed, including changes in enrichment, fuel element configuration, etc. It is expected that this optimization will increase core thermal power to 200 MW(th). This step has not been done in the present design because the objective was to characterize overall system performance while maintaining a core design consistent within current engineering experience and manufacturing capabilities.

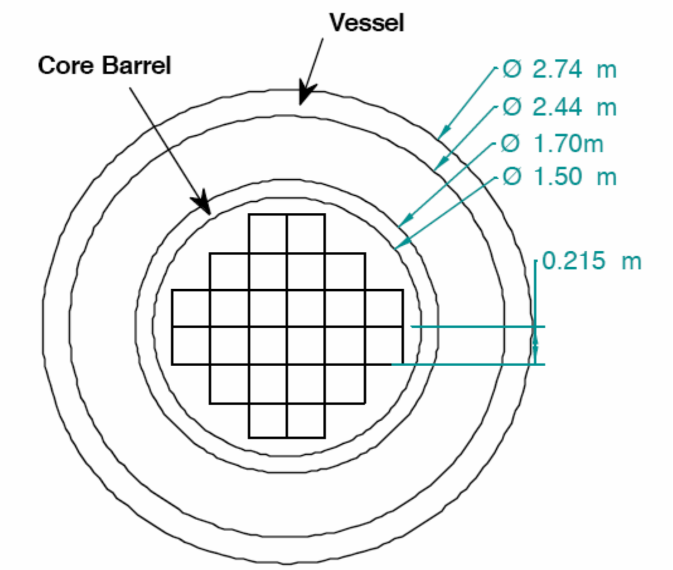


FIG. I-13. Cross-section view of MASLWR core.

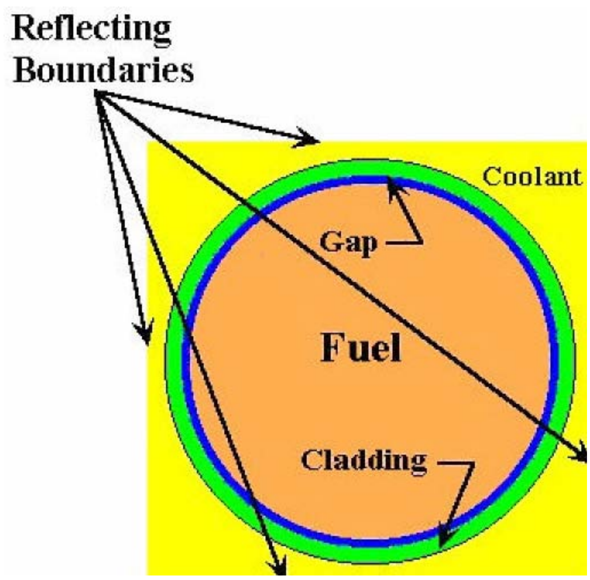


FIG. I-14. Cross section of a fuel rod.

Main heat transport system

The primary coolant flow path is shown in Fig. I-15(a). A simplified schematic of heat transport system in normal operation is given in the body of the previous part of the design description, in Fig. I-13.

Under accident conditions, the Automatic Depressurization System (ADS) depressurizes the reactor vessel, and equalizes the reactor vessel and containment vessel pressure. Feedwater flow to the steam generator is secured and a new heat transport path is established. The coolant in the core reaches sub-cooled boiling conditions and produces a steam filled region in the upper portion of the reactor vessel. The steam is vented to the containment via the upper ADS vent lines. Once in the containment, the steam condenses on the walls of the containment vessel and pools at the bottom of the containment. This pooled fluid enters the reactor vessel either by sump recirculation lines or the submerged ADS lines, completing the flow path. A simple schematic of the accident condition heat transport system is shown in Fig. I-15(b).

Steam generator

The steam generator is a helical tube, once-through heat exchanger, located in a common vessel with the reactor. A drawing of the steam generator used in the OSU-MASLWR facility is shown in Fig. I-16. The heat exchanger consists of 1012 tubes arranged in an upwardly spiraling pattern. The steam generator tubes are arranged into two tube bundles with the tubes attached to tube sheets through hydraulic expanding and welding. There are 4 tube sheets that are incorporated into the vessel. The tubes are 16 mm (0.625 in.) outside diameter with a 0.9 mm (0.035 in.) thick walls and a length of 22.3 m (73 ft) each. The tubes are arranged on a square pitch, with a transverse pitch ratio of 1.8 and a vertical pitch ratio of 1.5. The tubes occupy the space between the hot riser and the vessel cylindrical wall, and there are four rotations in the upward spiral. The material selected for the tubes is thermally treated Inconel 690.

I-2.2. Description of the turbine generator plant and systems

The novelty of the MASLWR turbine generator plant system is in its factory fabrication and modularity. As mentioned in the previous part, the turbine generator module is similar to the condenser and reactor module in that its relative size and weight allows for it to be completely factory fabricated and shipped to the plant on rail or truck. Another benefit of the small scale of the turbine module is that it can be located inside the reactor building, helping consolidate the power cycle and reducing the threat of sabotage. It can also be swapped out for maintenance or repair without a loss to downtime.

I-2.3. Systems for non-electric applications

Combining seawater desalination with power generation using a MASLWR plant was investigated in order to assess the economics of using the heat of turbine exhaust steam in the desalination process. The following proven seawater desalination technologies were considered for this evaluation:

- Multi-stage flash distillation (MFD);
- Multi-effect distillation (MED);
- Reverse osmosis (RO).

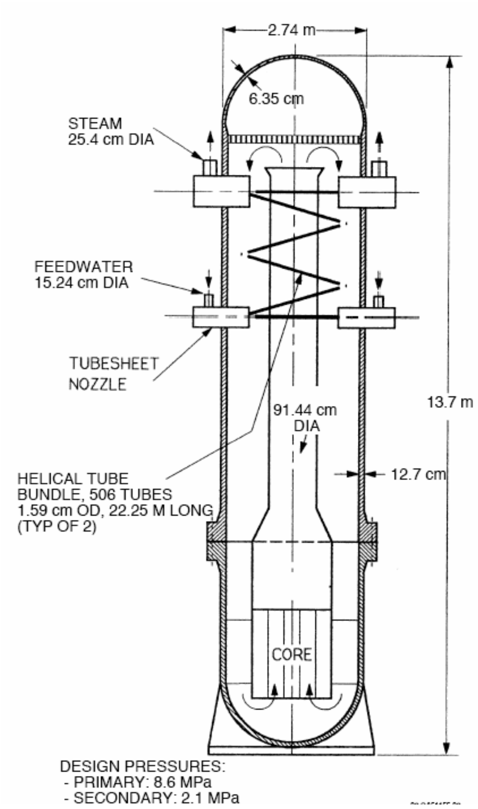


FIG. I-15. (a) Normal operation reactor module heat removal path.

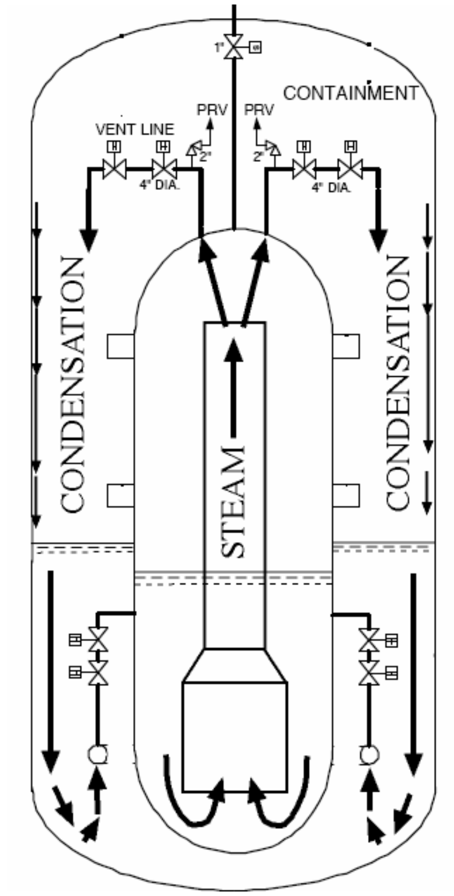


FIG. I-15. (b) Accident operation reactor module heat removal path.

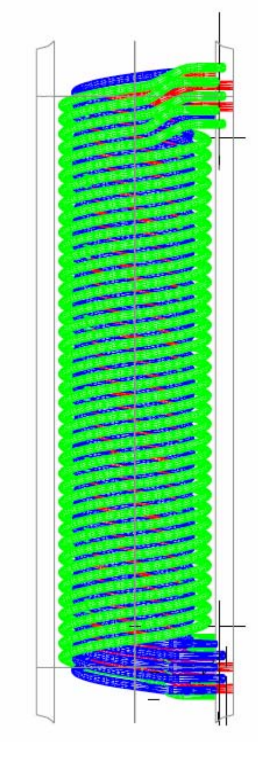


FIG. I-16. Drawing of the helical steam generator tubes used in OSU-MASLWR.

Both distillation processes require steam and electrical energy for water production. Although reverse osmosis requires only electrical energy, it was considered because recent advances in the process and, in particular, the membranes render the RO process very cost effective when compared to the distillation processes. The steam used for distillation must be provided at a pressure required by the particular desalination process, which is higher than the normal turbine exhaust steam pressure. Increasing the turbine exhaust pressure reduces the amount of power generated. A comparison of the processes was made based on the following assumptions:

- All the turbine exhaust steam from 30 turbines is available for the distillation processes (445 000 lbs/hr);
- Steam is available at the temperature/pressure required by the process;
- The seawater temperature is 72°F and the condenser outlet is 102°F with a total seawater flow of 25 000 g/min;
- The seawater salinity is 34 400 ppm;
- Product water salinity should be less than 500 ppm;
- Cost of electrical energy is US$ 0.05/kWh.

Table I-3 summarizes the results of the seawater desalination processes investigation.

Based on the comparison performed, reverse osmosis appears to be the most attractive seawater desalination process for desalination co-located with a MASLWR plant because of its lowest capital cost and the highest MASLWR net power supplied to the electrical grid.

TABLE I-3. COMPARISON OF DIFFERENT DESALINATION TECHNOLOGIES FOR MASLWR (COSTS ARE IN US$)

Process	Turbine exhaust pressure (Pcia)	Power needs (kWh/kGal)	Desalination power required (MW)	MASLWR power loss (MW)	Net power to grid (MW)	Plant production (MGal/day)	Capital cost (M$)	Annual power cost (M$)
None	0.75	—	—	—	35.0	—	—	—
MSF	30	15	6.4	25.8	2.8	10.3	63.1	13.4
MED	5	7.6	4.9	12.2	17.1	15.4	77.1	7.1
RO	0.75	15	9.6	—	25.4	15.4	62.2	4.0

A schematic flow diagram for a proposed reverse osmosis system is shown in Fig. I-17. Feedwater is supplied from the seawater intake into the power plant condensers (1). The incoming seawater is pumped (2) to a pre-treatment system (3), which contains as a minimum, a chlorinator, a filtration system, and a chemical addition system. If necessary, because of influent seawater quality, additional water treatment features can be included, such as coagulation, sedimentation, adsorption of impurities by diatomaceous earth or activated carbon, and manganese-zeolite filtration. From the pre-treatment system, the pre-treated water is pumped by high-pressure pumps (4) to a number of trains composed of RO elements housed in fibreglass pressure vessels (5). The product water is subject to post treatment (6) to make the water non-corrosive to the distribution system. The treated water (7) is then pumped to the distribution system. The reject brine (8) from the RO modules is nearly at the same pressure as the feed pressure. This brine is sent to an energy recovery device to recover energy (9). This energy can be in the form of direct drive of the feedwater pumps or electrical energy that is fed back into the system. The brine is then discharged to the sea.

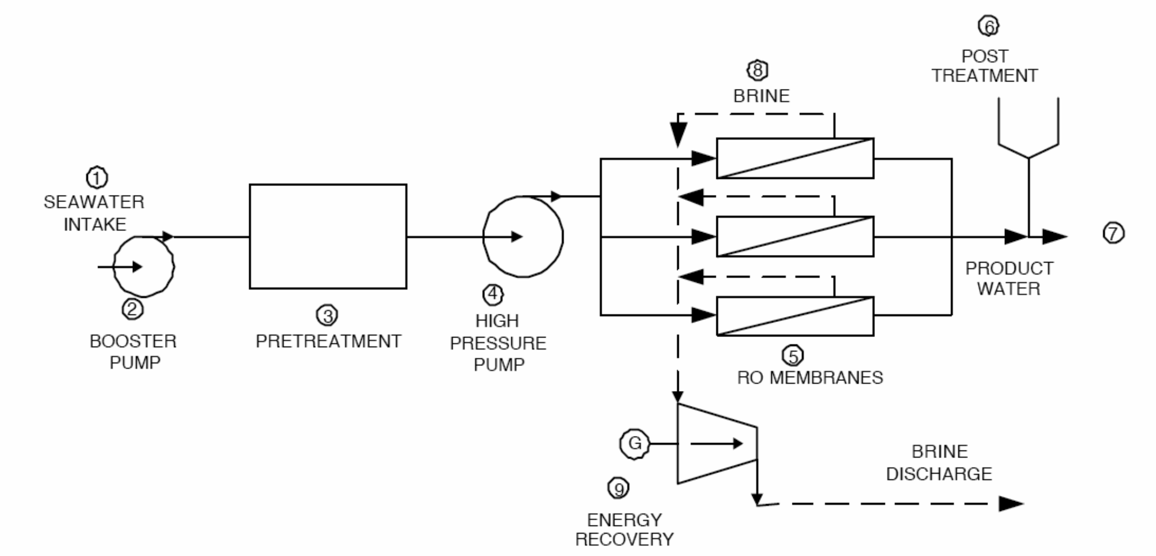


FIG. I-17. RO flow schematic.

I-2.4. Plant layout

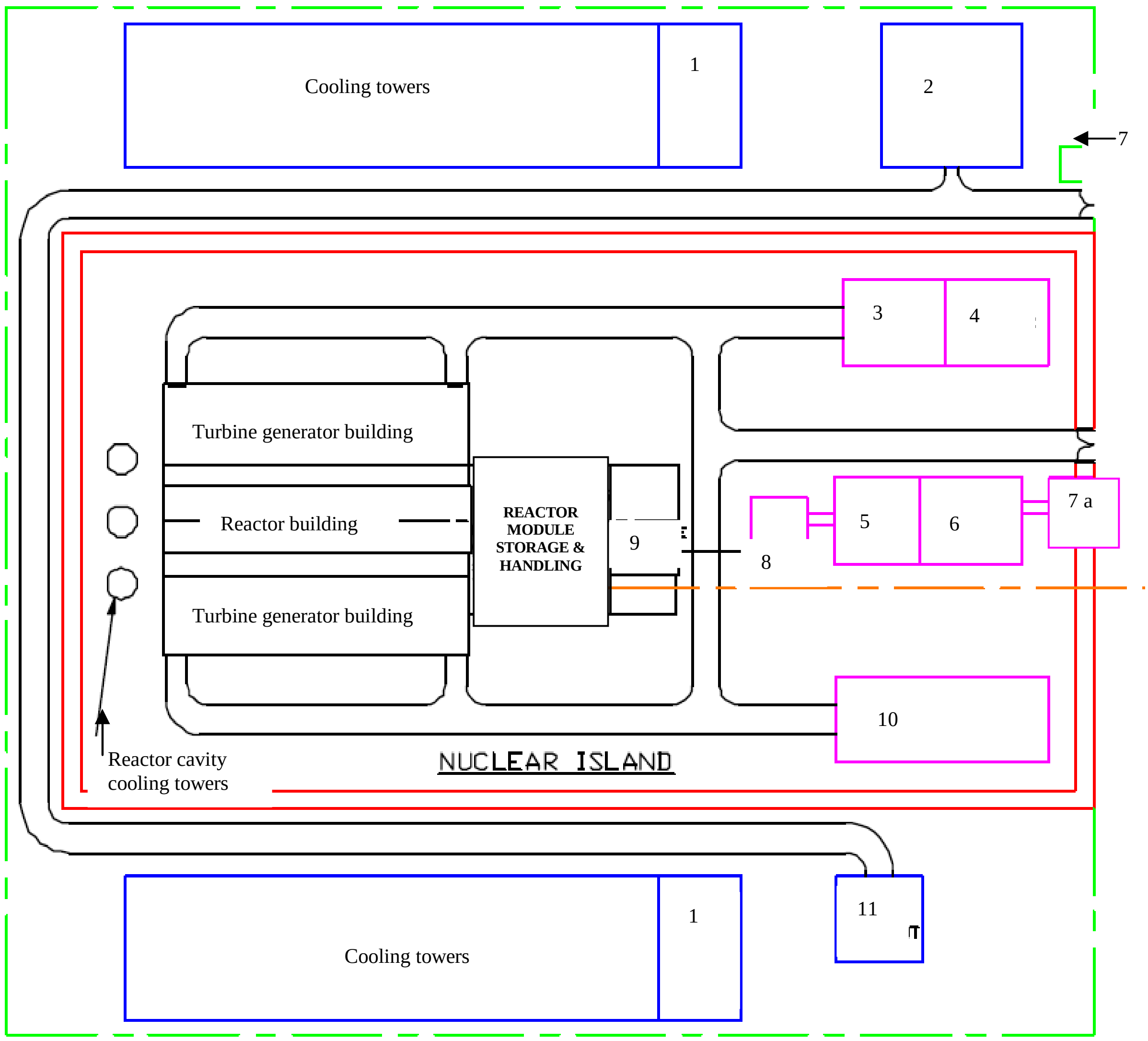
The overall plant arrangement is shown in Fig. I-18. The MASLWR plant consists of a power generation complex and common facilities. The power generation complex for the baseline plant consists of 30 power generation units (30 reactors and associated turbine generators), (and the reactor assembly, disassembly, and fuel handling equipment if onsite refuelling is desired). The plant area is about 390 m (1280 ft) by 350 m (1160 ft) or about 14 ha (34 acres). The plant net total output is 1050 MW(e).

In the case of on-site refuelling, the spent fuel is temporarily stored inside the fuel handling building before being shipped off-site to a central reprocessing facility. The plant is provided with a rail access. The rail is connected to the fuel handling and maintenance building to allow easy off-site transfer of fresh and spent fuel, receiving new reactor modules and shipping off-site radioactive waste.

The in-line arrangement of reactors provides for sequential construction of power generation units. This allows for power to be generated as soon as a unit is completed, while installation and testing may progress on the following units.

The cooling towers are located as close as practical to their respective turbine-generator buildings to minimize the length of the large-diameter circulating cooling water pipes. The switchyard should be located up-wind of the cooling towers to minimize the effect of cooling tower vapour drift on electrical equipment.

All the common facilities such as the training center, the administration building, the control building, the plant services building, the maintenance facility, etc., are located at the main entrance of the plant to provide for easy access to the various buildings without going through the entire plant. Areas on the far end of the plant are designated for use during construction. Road and rail provide transportation access to the plant. Barge access is desirable but not mandatory.



1 – W pump house
2 – Administration & training
3 – Machine shop
4 – Warehouse
5 – Control building
6 – Personnel services
7 – Guard house (7a- main house)
8 – Remote shutdown
9 – RAD waste building
10 – Plant services
11 – Waste treatment

FIG. I-18. Plan view of the MASLWR plant.

The power generation complex is an integrated structure that houses the power generation units and includes the following major facilities:

- Reactor building;
- Turbine generator buildings;
- Fuel handling and maintenance building.

A plan view of the power generation complex arrangement for the baseline plant consisting of 30 power generation units is shown in the previous part of this design description in Fig. I-2 for the above grade portion, and in Fig. I-12 for the elevation view of the below grade portion. The overall dimensions in plan view of the complex are 182 m (597 ft) long and 94 m (308 ft) wide. The building is partially located below grade. The power generation complex consists of a reactor building located longitudinally in the center of the building with two turbine buildings adjacent to each side of the reactor building. The fuel handling building and the reactor assembly/maintenance building are adjacent to one end of the reactor/turbine generator buildings. This allows for phased construction of the reactor/turbine generator buildings and power generation units. Initially, only the needed number of power generation could be constructed. Then, power generation units can be added as needed at different stages.

The reactor building consists of an embedded reactor cavity with an above-grade structure that houses hoisting and handling equipment. Also located above grade are equipment vault structures that house electrical equipment and instrumentation for the reactor control, monitoring and protection, containment isolation, and miscellaneous support and services equipment. Space is also provided in the equipment vault structures for routing of the piping, wiring and cabling, HVAC ducting, etc. The reactor cavity is about 108 m (355 ft) long, 22 m (72 ft) wide and 20 m (66 ft) embedded below grade, and is filled with water up to the grade level. The space above the reactor cavity extends about 15 m (50 ft) above grade and houses hoisting and manipulating equipment. The equipment vaults are located on each side of the reactor cavity. They are about 9 m (30 ft) wide and 15 m high (49 ft), arranged in three levels. The reactor modules are installed in a vertical position and are arranged into two rows of 15 modules each, along the external cavity walls. There are two additional spaces in each row for temporary storage of reactor modules. Vertical separation walls are provided between reactor modules for the protection of reactor modules. A space is provided in the middle between the two rows of modules to allow for moving of the reactor modules between the reactor cavity and the fuel handling and maintenance building adjacent to one end of the reactor building. Four wall-mounted travelling manipulators are provided in the reactor cavity above the water level, two on each side, for connecting and disconnecting the piping and electrical wiring and cabling during installation, replacement and removal of reactor modules. The equipment vaults are partitioned to provide separation between power generation units to the maximum extent practical. Also, the layout of safety-related electrical system provides for division separation required by the nuclear power plant regulatory requirements.

A rendition of the overall site plan is shown in Fig. I-19.

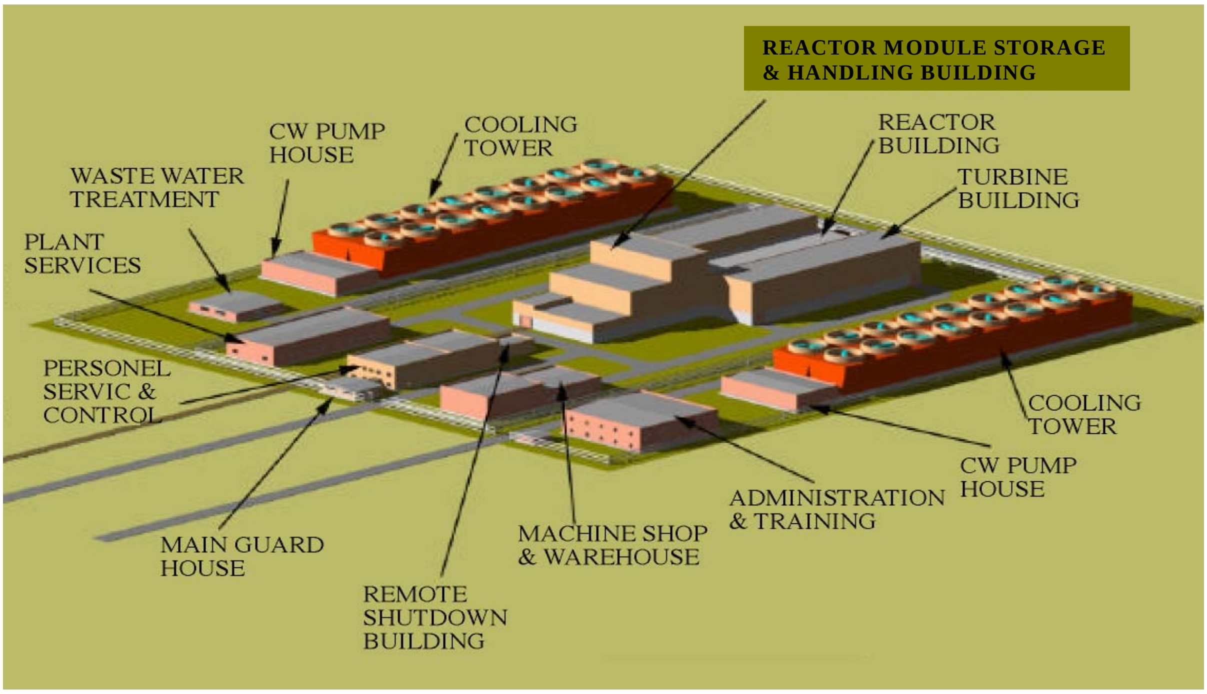


FIG. I-19. MASLWR overall site plan.

REFERENCES

[I-1] MODRO, M., Multi-application small light water reactor: Final report, Idaho National Engineering and Environmental Laboratory (INEEL) Report (December 2003).

[I-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs 2004, IAEA-TECDOC-1391, Vienna (May 2004).

ANNEX II

TRANSPORTABLE REACTOR FACILITY FOR ELECTRICITY SUPPLY IN REMOTE AREAS (UNITHERM)

NIKIET, Research and Development Institute of Power Engineering,
Russian Federation

II-1. General information, technical features and operating characteristics

II-1.1. Introduction

The UNITHERM is a transportable nuclear power plant (NPP) for electricity supply to urban districts and industrial enterprises in remote and difficult-to-access areas. UNITHERM is just a name rather than the abbreviation; it was suggested by an idea of the universal nature of the NPP as a source of thermal energy.

As estimated, the area of decentralized energy supply covers about 2/3 of the territory of Russia; here, small size nuclear power plants (SS NPP) can become a reasonable alternative to energy sources based on fossil fuel. The design concept of the UNITHERM has been developed since the 1990s by Research and Development Institute of Power Engineering (known both as RDIPE and NIKIET) of the Russian Federation and Russian Research Centre "Kurchatov Institute", and was necessitated by the conversion of the NIKIET activities to peaceful uses.

The UNITHERM concept is based upon the NIKIET experience in the design of marine nuclear installations. Successes achieved and the existing solutions in design, nuclear and thermal physics, hydrodynamics, metallurgy, water chemistry, etc., in combination with some new features, made it possible to develop a NPP design which has become the laureate of the competition for SS NPP designs established by the Russian Nuclear Society in 1994. The reactor facility has met the requirements for the so-called fourth generation nuclear reactors later formulated by the international community and in some cases, was able to meet even more strict requirements.

All activities related to design of the UNITHERM have been conducted on the initiative of NIKIET. In the first design options, the core thermal power was defined as 15 MW. Later, reactor power has been increased to 30 MW as a result of the discussion with potential users. A reactor power range from 15 to 30 MW could probably be regarded as the most reasonable. A power level below this range would cause an intolerable rise in energy cost, while the higher power would result in enlarging the units, which would complicate their transport to the site (for land-based NPPs) [II-1 to II-5]. Some private companies and the government of the Sakha Republic have expressed an interest as potential users.

II-1.2. Applications

According to the design concept, the UNITHERM can be used as a source of energy for the generation of electricity, district heating, seawater desalination and process steam production either in a complex or to meet specific demands. The purpose of the NPP would impact not only the mix of components but may also determine the characteristics of the reactor [II-5]. For instance, the use of steam at low parameters for district heating and potable water production allows the application of a turbine generator unit operated at backpressure. This sufficiently increases total plant efficiency and allows the use of the thermal siphon as an intermediate cooling circuit. As a result, the mass and size of the reactor could be significantly reduced.

The generation of electricity as the most universal form of energy requires that the turbine generator be operated in a condensation mode using the highest achievable parameters of steam. Therefore, an intermediate single-phase coolant (pressurized water) circuit could be used in the NPP. A greater difference in elevation is needed to provide natural circulation in this intermediate circuit, however, the required reheat of generated steam can be assured. This minimizes the cost of electricity production.

In general, the configuration and design of the UNITHERM are sufficiently flexible to adjust it to different target functions and user requirements, without compromising the underlying principles of the concept.

II-1.3. Special features

One of the main conceptual features of the UNITHERM is the option of shop fabrication and testing. As a result, high quality could be assured with minimum cost and time. After testing, the reactor unit would be dismantled for transport. The degree of dismantling depends on the type of an NPP and the transportation mode (either as a whole unit or in large pre-assembled parts).

Considering the constraints in distant regions of Russia, a land-based NPP could probably be the preferable option. In this, construction of several units that may be commissioned either simultaneously or individually becomes possible. The space-saving building structures of the UNITHERM do not involve a large scope of work and can be constructed by regional construction enterprises within the limited time period. Upon completion of the plant life and after appropriate cooling prior to evacuation, the building structures can be dismantled and the plant site (measuring about 1–2 hectares) can either be reused or remediated.

An alternative option to land siting is location of the plant aboard a barge if this is possible within the limits of the siting conditions. The advantage of this option is easy assembly and complete testing of the plant in shop conditions, easy transport to the site of operation and evacuation upon completion of life. The disadvantages are the higher costs of a barge compared to building structures, the need to construct a special purpose bay with facilities for energy transfer from the movable barge to the shore and possible barge overhaul in the shop during operation of the NPP. This includes transportation of the barge to and from the dockyard and replacing the plant with other sources of energy generation for a long period of time. The use of this technology is reasonable for NPP leasing to foreign users.

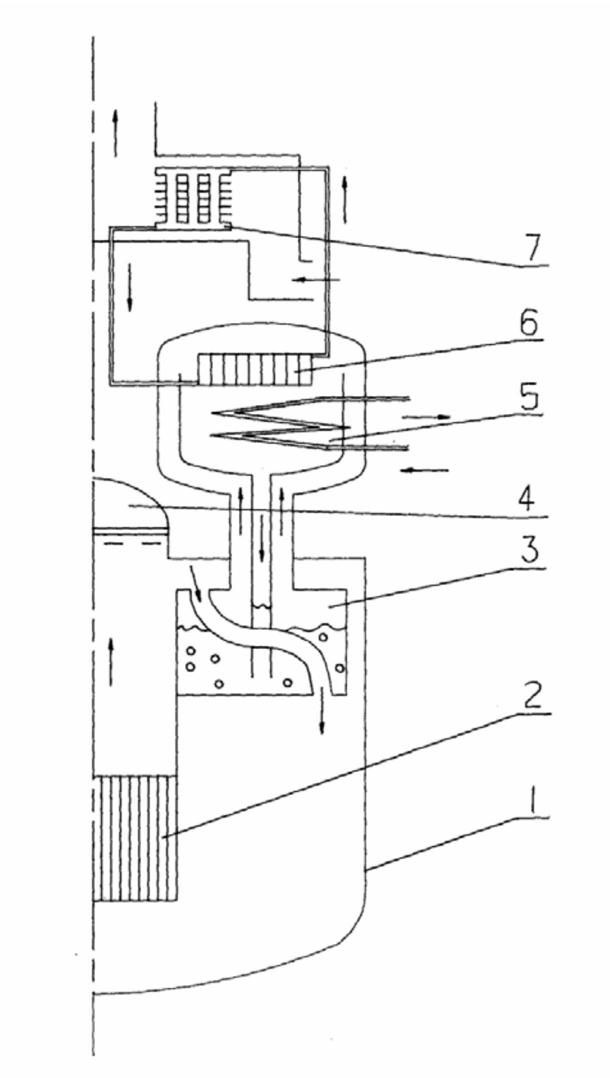
The UNITHERM plant is designed to operate without operating personnel in attendance. The intent is to provide the reactor maintenance in routine and urgent cases from regional centres common to several plants of this kind.

No refuelling of the reactor core is envisaged during the plant service life. This would eliminate potentially hazardous activities related to core refuelling, simplify operating technologies and could ensure enhanced proliferation resistance. The reactor core life can be equal to the plant lifetime and is estimated as 20–25 years at a capacity factor of 0.7.

II-1.4. Summary of major design and operating characteristics

The flow diagram of the UNITHERM is schematically shown in Fig. II-1; more details of the UNITHERM circuits are given in Fig. II-2. Figure II-3 presents the layout of the integral UNITHERM reactor.

The reactor core (2) placed in the reactor pressure vessel (1) is cooled by the primary coolant driven by natural circulation and exposed to the steam-gas environment of the pressurizer (4) In the intermediate heat exchanger (3) the energy released in the reactor core is transferred to the intermediate circuit coolant, which moves upward to flow outside the tubes of the helical coil once-through steam generator (5), Fig. II-1. After cooling on the steam generator heat exchange surfaces, the coolant is directed to the intermediate heat exchanger. The intermediate circuit and consequently, the secondary circuit, consist of several (from 8 to 16) parallel sections. In case of a leak in one of the heat exchange surfaces of the section, it is isolated from the user by isolation valves installed in the secondary circuit (the user circuit) without the need to shut down the reactor facility, Fig. II-2. Damaged sections can be repaired or replaced during scheduled preventive maintenance of the reactor.



1 – Reactor pressure vessel	2 – Core	3 – Intermediate heat exchanger	
4 – Pressurizer	5 – Steam generator	6 – Evaporator	7 – Radiator

FIG. II-1. Simplified flow diagram of the UNITHERM.

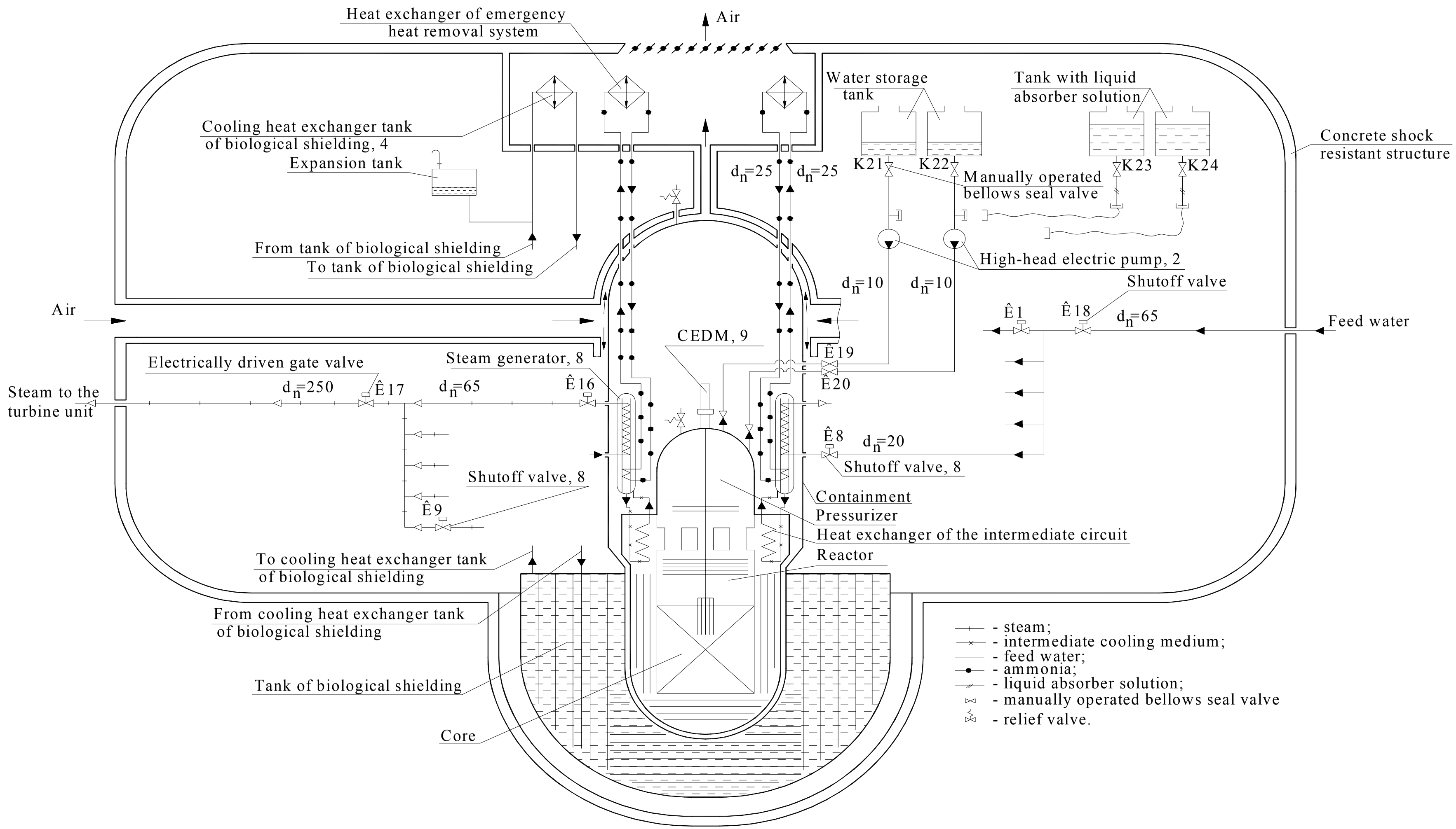


FIG. II-2. Schematic diagram of the UNITHERM.

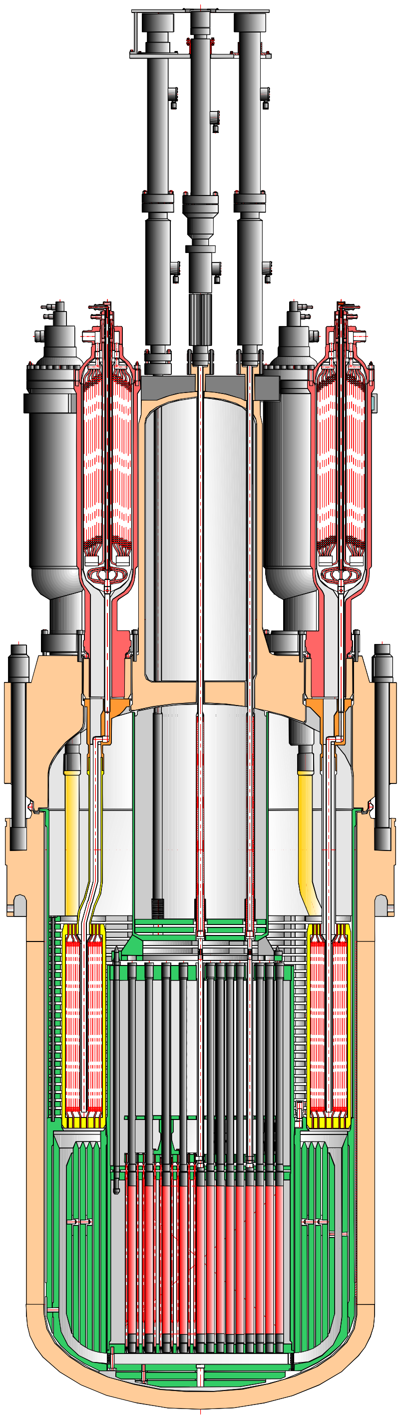


FIG. II-3. Layout of the UNITHERM reactor.

The core power varies depending on the steam load of the plant due to self-control since the temperature reactivity coefficient remains negative in the entire range of temperatures. Burnable absorbers can compensate decrease of reactivity due to fuel burn-up, by the temperature effect and by motion of the reactivity control members during periodic maintenance.

Further development of the UNITHERM layout was defined by the decision not to employ operating personnel for reactor control. A sudden reduction or even cessation of heat transfer to the user should not cause shutdown of the reactor and overshooting of the system parameters. Such situation can be mitigated through the heat exchanger- evaporator in the continuously operated independent circuit for heat dump, which is added to the intermediate circuit. In addition to the evaporator, the circuit consists of the radiator (7) connected to the evaporator and cooled by atmospheric air under natural circulation, see Fig. II-1 and II-2. The independent circuit for heat dump allows transfer of the reactor to a hot standby mode without the need for shutdown. In emergency situations the circuit acts as the decay heat removal system.

A significant seasonal temperature variation and low negative temperatures in the winter season (from plus 35°C to minus 55°C) in the candidate Russian sites for the UNITHERM NPP, located in the remote regions of Russia, predetermine the selection of a specific coolant in the independent circuit. The proposed options are, for example, ammonia, alcohol or ethylene glycol.

In different conditions water can be used as the cooling medium in the intermediate circuit.

It should be noted that shut-off and isolation valves are not used in any reactor pipelines, except for the user circuit, i.e., all systems are in continuous operation and not just available. This feature contributes significantly to the reliability and safety of the UNITHERM.

The primary coolant parameters have been chosen from the well-proven range of working pressures and temperatures typical of light water reactors. Consideration has been also made of experience in the operation of nuclear propulsion power plants in the modes of natural circulation of the primary coolant. The maximum primary coolant temperature at the core outlet (data averaged over the cross-section) can be taken equal to 330°C. Considering the above mentioned decrease of temperature due to fuel burn-up, this value would amount to 320°C at the end of life prior to maintenance shutdown. Under natural circulation of the primary coolant, the temperature difference between the core outlet and inlet will be no less than 70°C. Thus, the minimum temperature of the primary coolant will be 250°C.

As it was already mentioned, water is used as the coolant in the intermediate circuit. There are two options for the coolant flow: single-phase or two-phase fluid; the advantages and disadvantages of both options are obvious. Heat transfer with a two-phase coolant is characterized by higher values of heat transfer coefficient, while due to latent evaporation heat energy can be transferred by a lower flow rate, an important factor in natural circulation. Generally, this option is beneficial for space-saving characteristics of the design, although it poses limitations on maximum achievable temperature therein. In the case described, the temperature cannot exceed 230°C (see Fig. II-4A) and the coolant temperature in the secondary circuit cannot rise above 210°C.

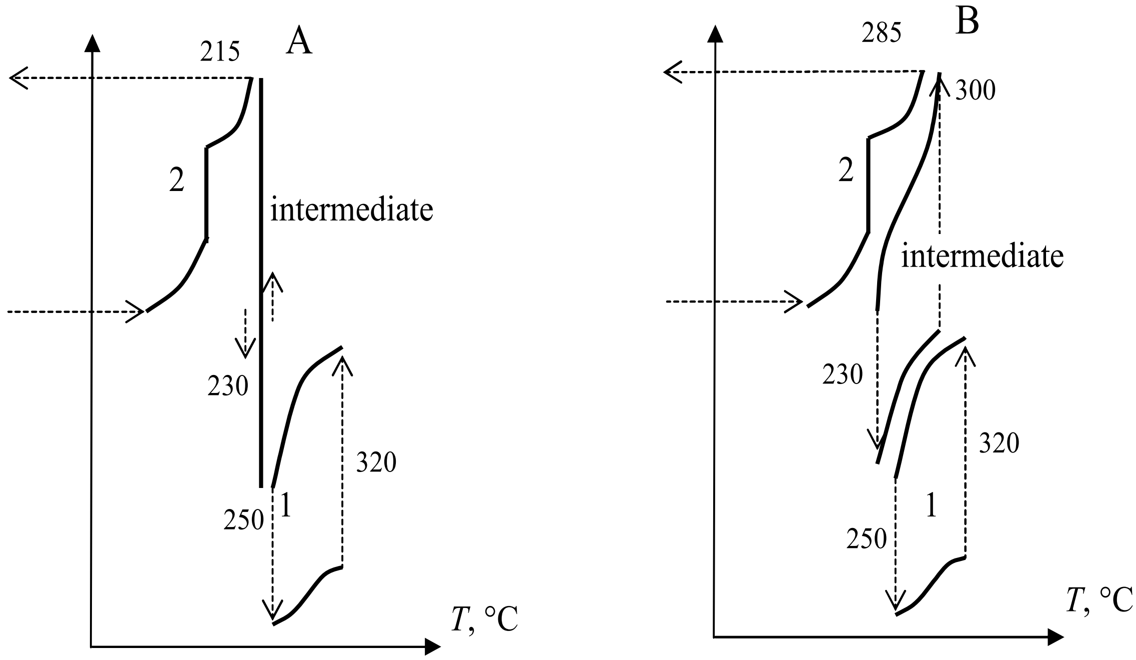


FIG. II-4. The scheme of coolant temperature distribution in the reactor primary, intermediate and secondary circuits for two-phase (A) and single-phase (B) options of the intermediate circuit.

In the single-phase option of the intermediate circuit (Fig. II-4B) much higher temperatures of generated steam can be achieved.

The parameters in the intermediate circuit impact the reactor design significantly. To ensure stable operation of the heat exchanger-evaporator of the two-phase thermal siphon, the latter should be designed as a vessel structure having the tube system inside. In this case, the primary coolant flows inside tubes, while steam of the intermediate circuit is generated outside tubes. The heat exchanger vessel should be strong enough to withstand the primary coolant pressure.

On the contrary, the intermediate heat exchanger of the single-phase system is designed as a helical casing-free structure. The primary coolant flows outside the tubes and the single-phase coolant of the intermediate circuit goes inside the tubes. The need to have a system to compensate for a changing volume in the intermediate coolant poses an additional problem.

Based on the developments and calculations, it can be demonstrated that the advantages in the parameters of generated steam due to the use of the single-phase intermediate circuit are achieved at the expense of a significant increase (by 25%) in the reactor height (see Fig. II-5), with a proportional rise of the mass and associated costs.

The reactor option for land-based siting is shown in Fig. II-6. The integral reactor is placed inside the leak-tight containment, which in turn is located within the concrete shock-resistant structure together with the biological shielding and reactor unit components. This structure enhances physical protection of the reactor unit from external impacts such as airplane crash, hurricane, tsunami, unauthorized access, etc.

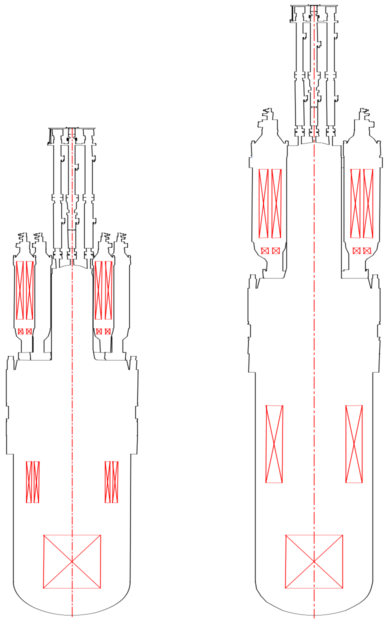


Fig. II-5. Comparison of reactor size for a single-phase (right) and two-phase (left) intermediate circuit.

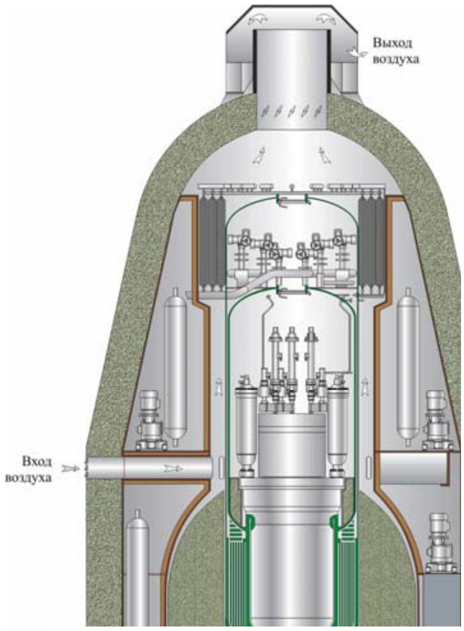


Fig. II-6. Reactor compartment of a land-based UNITHERM NPP.

An NPP with the UNITHERM may consist of a variable numbers of units depending on purpose and demand. To enhance security of supply, having at least two power units could be recommended. Each unit includes the reactor and turbine unit; the design of the latter varies depending on local demands.

The main design and operating characteristics of the UNITHERM are summarized in Table II-1.

TABLE II-1. SUMMARY TABLE OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

CHARACTERISTIC	DESIGN OPTION/VALUE	
	A	B
Installed capacity of an NPP unit, MW — electric (net) — thermal power for heating	 2.5 20.0	 6.0 —
Operating mode	Load following without limitation as to the number of power change cycles	
Capacity factor, %	70	
Design characteristics		
Fuel	Uranium-zirconium metal-ceramic (cermet)	
Fuel enrichment by ^{235}U, %	19.75	
Coolant/moderator	High purity water	
Fuel cladding	Zirconium alloy E110	

Characteristic	Design option/value	
	A	B
Design characteristics		
Type of the reactor core	Heterogeneous, channel type	
Reactor core dimensions, mm	∅1130 × 1100	
Type of reactor vessel	Cylindrical vessel with an elliptical bottom and a cover made of martensitic steel lined with stainless steel cladding.	
Dimensions of the reactor pressure vessel, mm	∅3220 × 5050	∅3220 × 6150
Thermodynamic cycle type	Direct	
NPP efficiency, %	74	20
Simplified schematic diagram of UNITHERM NPP	See Fig. II-2	
Number of main energy transfer circuits	Three-circuit scheme with water as coolant and working fluid in all three circuits: primary, intermediate and secondary (the users circuit)	
Type of intermediate circuit	Two-phase	Single-phase
Neutron-physical characteristics		
Temperature coefficient of reactivity (dρ/dt) at the working point, 10^{-5} 1/K	–30 to –60	
Reactivity coefficient on coolant density, 10^{-3} 1/(g/cm^{-3})	8.2 to 16.4	
Void coefficient of reactivity	—	
Reactivity of non-poisoned reactor at the beginning of cycle, % In the cold state In the hot state	 9.8 5.8	
Total temperature reactivity effect Δρ (without compensating rods), % At the beginning of cycle At the end of cycle	 4.8 5.1	
Average core power density, kW/l	27.3	
Maximum peaking factors in the core Volumetric Radial	 2.21 1.63	
Maximum peaking factors in fuel assembly At the beginning of cycle At the end of cycle	 1.27 1.19	
Maximum accumulation of fission products at the end of cycle, g/cm^3	1.0	
Maximum/average fuel burn-up, % FIMA	18.2/11.2	
Reactivity control mechanism)*		
Burnable absorbers, kg		
Natural boron	0.441	
Gadolinium	9.953	

CHARACTERISTIC	DESIGN OPTION/VALUE	
	A	B
Reactivity control mechanism[*)]		
Absorber rods of the reactivity control members combined into the reactivity compensation groups with the individual drives; for reactivity compensation and emergency protection	Cylindrical steel tubes ∅13.5 × 1.5 mm filled with dysprosium titanate and boron carbide	
Reactivity compensation groups: Central (CCG) Peripheral (PCG)	 1 group consisting of 54 rods 8 groups with 53 rods in each	
Change of CG rod worth $\Delta\rho$ during the cycle (BOC/EOC), %: CCG In the cold state In the hot state PCG In the cold state In the hot state	 3.7/4.5 4.6/5.7 25.0/30.0 32.6/40.0	
Emergency liquid poison injection system	Injection of boron solution in the primary circuit in beyond design basis accidents	
Thermal-hydraulic characteristics		
Type of circulation in circuits Primary Intermediate Secondary (the circuit for steam users) Independent circuit for heat dump Reactor components cooling circuit	 Natural Natural Forced Natural Natural	
Nominal temperatures of coolant in circuits (inlet/outlet), °C Primary Intermediate Secondary (the circuit for steam users)	 258/330 249 40/235	 249/330 171/314 40/310
Nominal coolant flow rate in circuits, kg/s Primary Intermediate Secondary (the circuit for steam users)	 80.5 17.8 10.7	 70.0 45.6 10.3
Nominal pressure in circuits, MPa Primary Intermediate Secondary (the circuit for steam users)	 16.5 3.9 1.35	 16.5 16.5 1.35
Maximum fuel cladding temperature, °C	355	
Maximum fuel temperature, °C	365	
DNBR	Large	

CHARACTERISTIC	DESIGN OPTION/VALUE	
	A	B
Fuel lifetime, mass balances of fuel materials, design basis lifetime of reactor core, vessel and structures		
Core lifetime, hours	145 000	
Fuel consumption/energy unit, kg/MW day	$8.7 \cdot 10^{-3}$	
Specific consumption of natural uranium, kg/GW year (kg/MW day)	$1.5 \cdot 10^{5}$ (0.413)	
Design life of the reactor, reactor core and major components of reactor system, year	25	
Design and operating characteristics of systems for non-electric purposes	No information was provided	
Economic characteristics		
Capital construction costs, thousand US $		
Single-unit NPP	16 295	18 920
Two-unit NPP	24 750	28 750
Estimated duration of construction, year	5	
Total annual operating costs, thousand US $		
For electricity	1130	3935
For cogeneration	2265	
Cost of:		
Electricity, US$/kWh	0.04	0.055
Heat, US$/GCal	11	—

*) CCG is used for reactor power and coolant temperature control during start-up and bringing the reactor to power. CCG is driven by a rotary step electric motor. PCGs are intended for compensation of temperature reactivity effect in the process of reactor start-up and heatup. During reactor operation, they are placed at limiting top position and held there by magnets. PCGs are driven by linear step electric motors. Under emergency situations the mechanisms are de-energized and the compensation groups go into the core, driven by gravity and insertion springs. If one of the groups jams, the worth of the remaining groups is sufficient to shut down and cool down the reactor. In beyond design basis accidents with several compensation groups being jammed, the reactivity is compensated by the injection of an aqueous boron solution. During reactor operation, the members compensating excess reactivity are motionless. Mechanical stops prevent them from moving upward. Fuel burn-up and core poisoning are compensated by the temperature effect. Once a year (during off-line operation) the NPP is attended for maintenance and the position of the CCG is corrected.

II-1.5. Outline of fuel cycle options

The fuel cycle foreseen for the UNITHERM differs from that used in current LWRs as comes to the manufacture of fuel elements and fuel assemblies, fuel handling during operation and fuel reprocessing. Other components of the fuel cycle are basically common with widely applied processes of enriched uranium production and the final disposal of radioactive waste.

A specific feature of the UNITHERM fuel cycle is the long and uninterrupted irradiation of fuel inside the reactor core throughout the whole reactor lifetime, with a once-at-a-time whole core refuelling.

Metal-ceramic (cermet) fuel chosen for the UNITHERM is composed of uranium dioxide particles in a metallic (silumin or zirconium) matrix. Such design is characterized by a high volume ratio of nuclear fuel and, through the use of the metallic matrix, ensures minimum swelling and high thermal conductivity. An optimally shaped cladding is formed when it is filled with the matrix composition.

Long-term fuel operation would cause the build-up of fission products at a level of up to 1 g/cm^3, with the integrity of the fuel elements being retained. Fuel storage is recommended for about 5 years upon completion of service life, which ensures a considerable decrease in radioactivity. Then the reactor will be delivered in the transporting container to the reprocessing plant for discharge and reprocessing of fuel.

Fuel with silumin matrix will be treated with a solution of nitric acid; a thermal gas chemical oxidation treatment is used for the highly resistant zirconium matrix.

Since independent phases of UO_2 and ZrO_2 are generated during fuel oxidation, uranium can be leached into the solution using nitric acid, i.e., following standard PUREX process technology.

II-1.6. Technical features and technological approaches that are definitive for UNITHERM performance in particular areas

II-1.6.1. Economics and maintainability

NPPs with the UNITHERM are designed for siting in distant and difficult-to-access regions with poor infrastructure, where qualified staff for plant operation may not be available. Moreover, the rotation of plant personnel may affect relations with the local population and quality of operation, therefore the use of the shifts is regarded as unacceptable. For these reasons, the flow diagram, design and characteristics of the plant have been chosen to minimize the scope of maintenance. This approach not only resolves staffing problems but also improves the economic aspects by reducing the expenses for salaries. It is planned to provide several UNITHERM NPPs of the same type in one region. Emergency and routine work at the plants will be provided from a common maintenance centre made accessible around the clock by satellite communication and helicopters.

Low capital costs for plant construction could be ensured by the serial production of plant equipment fully under shop conditions, including equipment assembly and testing. Next, the pre-assembled units of limited mass will be transported to the site.

The operation without on-site refuelling would eliminate potentially dangerous activities, simplify the scope of construction work and the plant design and reduce operating costs.

II-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Factors contributing to reduced environmental impacts from the UNITHERM based NPP are the following:

- A low level of radiation (comparable to natural background);
- Five barriers protecting against the release of radioactive products from the fuel to the environment;
- No continuously operated systems (such as ventilation or coolant purification) for radionuclide transfer to the environment or the sump;
- No routine activities involving depressurization of the primary circuit.

These features ensure minimum waste volume along with low exposure (comparable to natural background) of the plant personnel and population and low costs for spent nuclear fuel and radioactive waste handling.

II-1.6.3. Safety and reliability

Safety concept and design philosophy

The design philosophy for the UNITHERM is to assure that the radiation impact on personnel, population and the environment under normal operation and in design basis accidents is much lower, or at least does not exceed, the limits prescribed by current regulations. In beyond design basis accidents, the design objective is to constrain the radiation impacts as much as possible.

The UNITHERM is being designed to meet the current rules and standards and also reflects the traditional and current views on safety issues.

Provisions for simplicity and robustness of the design

The UNITHERM is a reactor of an integral design. The design of the UNITHERM is based on the technologies of water-cooled and water-moderated reactors well proven in the design, fabrication and operation of propulsion systems. As far as possible, the design relies on the use of equipment proven in series production and prototypes tested in operation.

The UNITHERM provides for no core refuelling during the whole reactor lifetime; the repair and decommissioning works are performed upon completion of the lifetime.

Active and passive systems and inherent safety features

The inherent safety features of the UNITHERM are as follows:

- Negative core reactivity coefficients in the entire operating range;
- A high flow rate of coolant in natural circulation, which assures efficient core cooling and heat removal under routine and emergency conditions, including design basis and beyond design basis accidents;
- A high heat accumulating ability of the metal structures and a large inventory of coolant in the reactor, which allows transients to proceed at a lower rate, including accidents with primary circuit leaks and other departures from normal heat removal conditions in the core.

The UNITHERM makes an extensive use of the passive systems and devices based on natural processes without external energy supply; these systems include:

- The control element drive mechanisms (CEDM), designed to provide secure insertion of rods in the core by gravity (with the reactor sub-criticality being ensured under complete cooldown with jamming of any absorber rod group of the highest worth);
- Locking devices in the CEDM to avoid unauthorized withdrawal of control rods from the core in reactor commissioning, operation and maintenance;
- An independent passive heat removal system acting as a cooldown system in emergency shutdown of the reactor;
- A containment capable of maintaining primary coolant circulation as well as providing reactor cooldown and retention of radioactive products under the loss of primary circuit leak-tightness;
- An iron-water biological shielding of the reactor, which also acts as a system of bubble tanks for cooling water storage; the shielding removes heat from the reactor pressure vessel, preventing a melt-through of the core in a postulated beyond design basis accident with reactor core voiding;
- Passive systems for heat removal from the containment and biological shielding tanks.

High reliability of the safety systems is assured through the implementation of the following principles:

- Passive operation; no active impact is required to bring the systems into action;
- Diversity of the safety systems and devices ensured by different principles of systems' action;
- Redundancy of the systems;
- Continuous or periodic in-service inspection of the equipment and systems.

Structure of the defence-in-depth

The system of safety barriers incorporated in the UNITHERM includes:

- The fuel matrix;
- Leak-tight fuel cladding;
- A leak-tight primary circuit;
- A leak-tight intermediate circuit;
- A leak-tight containment;
- Isolation valves;
- A concrete shock-resistant structure.

Design basis accidents and beyond design basis accidents

The full list of beyond design basis accidents has not been analyzed at this stage of development. It is foreseen that such list could be agreed upon with the customer.

In beyond design-basis accidents involving jamming of several compensation groups, personnel from the regional service centre may actuate an active system of liquid poison injection into the primary circuit.

Actions of the personnel may also include prevention of radioactive primary coolant spread to the attended compartments of the NPP in the event of an interfacial leak in one of the steam generator modules. In the response to a leak signal from pressure sensors, the operator makes a decision to isolate the respective module, which does not violate the overall operating mode of the plant.

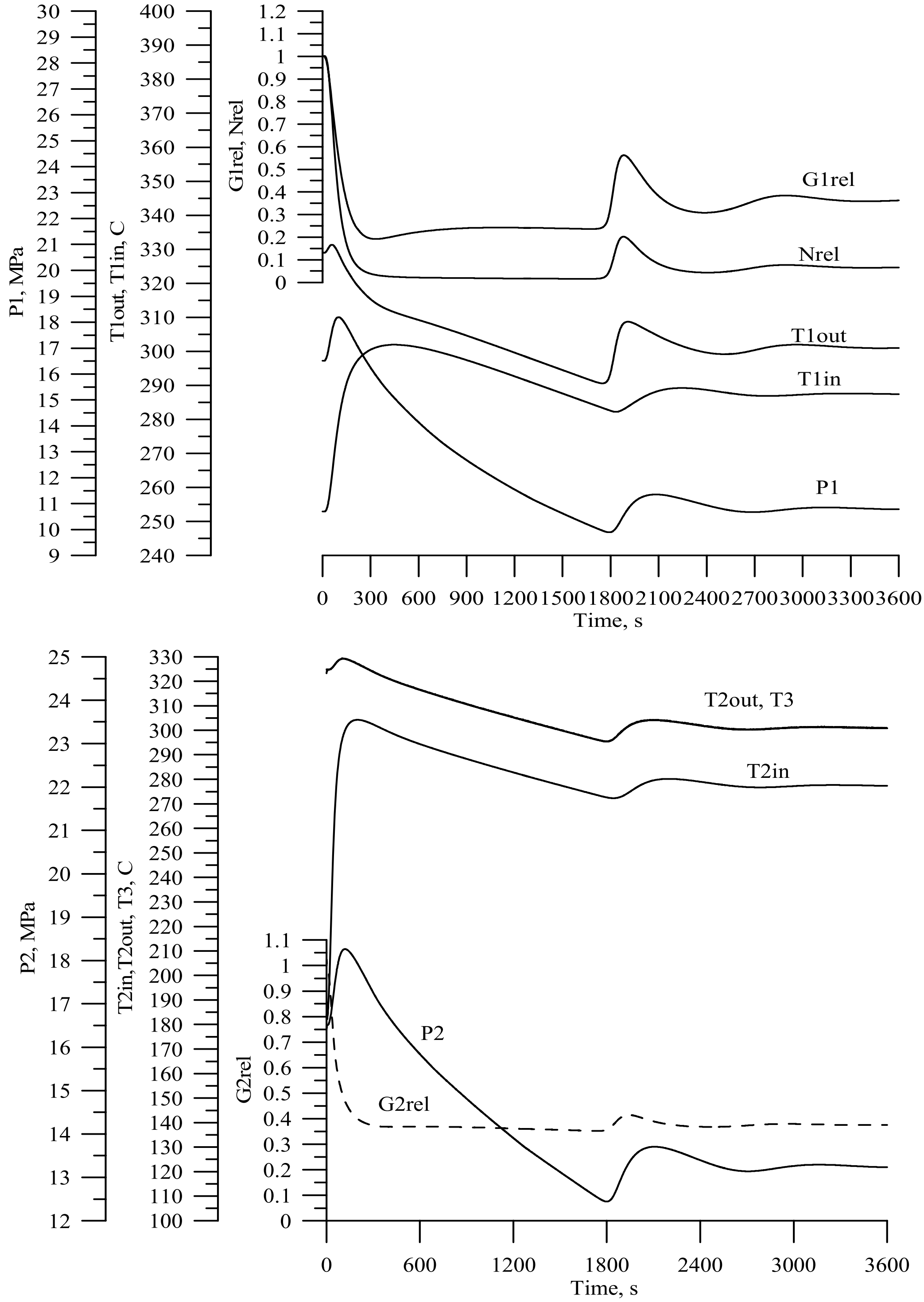
In both above mentioned cases, the response of an operator is not limited by time.

As an illustration of the UNITHERM performance in accidents, the following situations were analyzed:

(1) A sudden cessation of heat transfer to consumers;

(2) A steam pipeline break.

A sudden cessation of heat transfer to the users may occur, for example, in the event of failure in the electrical system resulting in the actuation of the fast-acting turbine isolation valve. In turn, this results in the instantaneous termination of steam take-off and feedwater supply to the reactor. According to the design concept, external events with regard to the NPP should not necessitate reactor shutdown. Therefore the emergency core protection system does not come into action, reactor power changes under the temperature effect and the independent heat transfer circuit becomes the only path for cooling (though this is a multi-train system). Figures II-7 and II-8 show diagrams of the reactor parameter changes in this situation.



G1rel – relative (with respect to nominal) primary coolant flow rate
Nrel – relative core power
T1out – coolant temperature at the core outlet
T1in – coolant temperature at the core inlet
T2out – coolant temperature in the intermediate circuit at the SG inlet
P1 – primary circuit pressure
T2in – coolant temperature in the intermediate circuit at the SG outlet
P2 – pressure in the intermediate circuit

FIG. II-7. Variations of the reactor parameters in the event of steam take-off termination.

An increase in primary coolant temperature is specific to the first stage of reactor cooldown in the event of steam take-off termination; the reactivity is conditioned by the effect of self-control and the core power decreases sharply. This process is characterized by an evident over-control: 250 s after onset of the transient, the heat rate dependent on U^{235} fission becomes overridden by the decay heat, Fig. II-8.

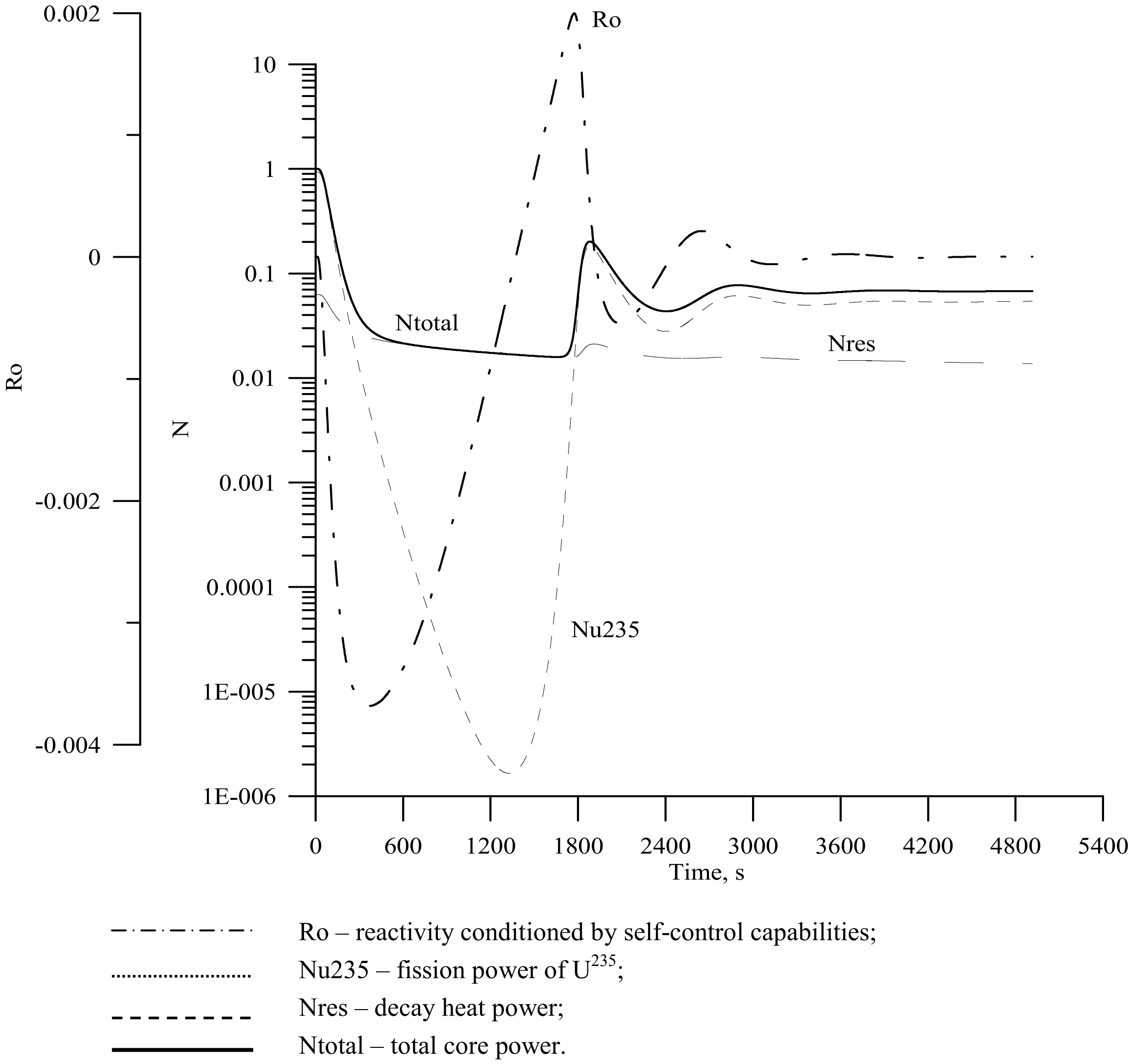


FIG. II-8. Variations in reactivity and reactor power (by constituents) in the event of steam take-off termination.

A further decrease in primary coolant temperature would cause a rise of reactivity. At the 25th minute of the transient, the reactivity would become slightly positive and the neutron power would start to increase at a high rate. Due to the low absolute value of reactivity, the first stage of this power rise is not characterized by a feedback; the latter manifests itself when the neutron power (power immediately released in U^{235} nuclei fission) becomes comparable to decay heat power. In this situation the neutron power would be subject to secondary over-control, and then all reactor parameters are characterized by damped oscillations.

The core remains subcritical for a long time. Core behaviour is similar to the situation of core start-up with varying reactivity. Therefore, safety measures provided for the start-up mode, i.e., actuation of power doubling period protection, are required. As shown by the analysis, when the core is coming out of the subcritical state, minimum power doubling periods occur at the end of the transient and make up 19 s and 13 s for the reactivity coefficients of 30×10^{-5} 1/°C and 60×10^{-5} 1/°C, respectively, while the reactivity increase rate is $7\times10^{-4}\beta$/s. This is an acceptable value that meets the current nuclear safety regulations.

The most abrupt positive reactivity perturbations, also self-controlled by the reactor, take place in accidents with a steam pipeline break. In this situation, steam pressure decreases and water accumulated in the steam generator flashes thereby removing heat, first from the intermediate circuit and then from the primary circuit. If the reactor was operated at 100% power, steam output in the initial state, and then power rise caused by this perturbation, might result in an increase of steam output up to the setpoint of the emergency protection actuation. Although not fatal, it is still in a disagreement with the original design concept, which assumes that events in systems external to the reactor should not cause actuations of the reactor emergency signals.

In the analysis it was assumed that absolute steam pressure in the steam pipeline would suddenly drop down to 1 bar whereupon feedwater supply to the steam generator is terminated. Results of the analysis in the form of diagrams are given in Fig. II-9.

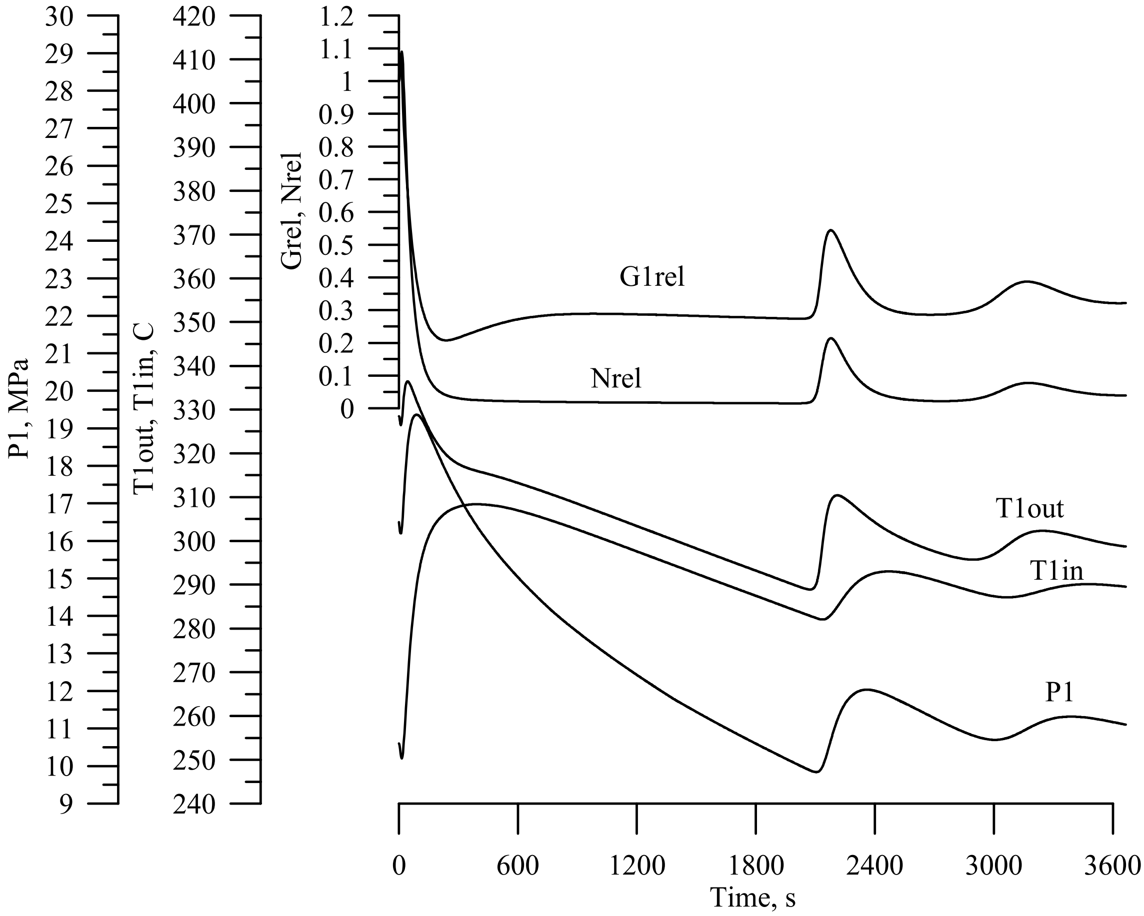


FIG. II-9. Variations in reactor parameters in the event of loss of steam pipeline leak-tightness (notation is the same as in FIG. II-6).

As it is seen from Fig. II-9, the initial power overshoot to a level not exceeding 110% of the nominal power and not initiating core emergency protection actuation, is followed by a rise of temperature and pressure in the reactor unit circuits. This is typical for the mode of energy transfer stop with a subsequent cooldown, in which the core power is reduced through self-control capability. As a result of reactor cooldown by the independent heat removal system, 35 minutes following some fluctuations, the core power again rises to a level acceptable for energy generation with some over-control. Steady state hot standby conditions are established one hour after onset of the transient. From that moment, the reactor is again ready to take up the load.

Provisions for safety under seismic conditions

The UNITHERM is designed for land-based siting. The structures of the NPP protect the reactor from impacts of extreme external events such as hurricanes, tsunami, airplane crashes, etc. The UNITHERM NPP would remain serviceable under the impact of an earthquake up to magnitude 8 per the MSK-64 scale. Under earthquakes of magnitude 8 to 9 per the MSK-64 scale or other human-induced impacts including airplane crashes, the NPP can be automatically shut down and brought to a safe state without exceeding the design limit of fuel failure.

II-1.6.4. Proliferation resistance

Nuclear materials in the UNITHERM core are not attractive for a weapon programme for the following reasons**:**

- The enrichment of fuel by ^{235}U does not exceed 20%;
- The uranium recovery from spent nuclear fuel in a cermet form is complicated;
- The reactor design ensures difficult access to nuclear fuel; the reactor is designed for operation without on-site refuelling; whole-core refuelling is assumed to be performed at a factory;
- The irradiated fuel is highly radioactive;
- A thorough physical protection is assumed for the NPP.

II-1.6.5. Technical features and technological approaches used to facilitate physical protection of UNITHERM

The details of the physical protection for the UNITHERM have not been developed since the design of the NPP itself is not available at the moment. The operating conditions chosen for the NPP, such as non-attended containment during operation, the external shock-resistant concrete structure etc., would complicate unauthorized access to the reactor.

The UNITHERM incorporates several design features and measures for protection from human errors and mitigation of the consequences of human errors or acts of malevolent character that could potentially lead to disabling of the NPP. Those include:

- Minimized scope of maintenance and on-power repair of major systems and equipment;
- Design approaches and security measures to protect against unauthorized access to the NPP systems, e.g., placing all vital systems in the containment;
- Maximum use of the fail-safe systems, for which a failure of a system component causes the system to actuate safety functions or reach a safe state;

- The use of passive safety systems and devices which do not require actuation (such as the containment, the independent heat removal system, etc.) or can be passively actuated (such as systems for the primary circuit and containment depressurization in emergencies);
- The development of plant operator support systems to provide an ad hoc evaluation of the plant status and search for the optimal responses to the warning signals;
- The use of the simulators for periodical training of the operating and maintenance personnel, including responses to accident situations.

II-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of UNITHERM

Detailed designs of the reactor and NPP could be developed to meet specific user conditions.

II-1.8. List of enabling technologies relevant to UNITHERM and status of their development

As noted above, the UNITHERM concept is based on the experience of NIKIET and other Russian institutions and enterprises in the development of marine nuclear installations. The experience is available in the form of design approaches and technologies covering many aspects of nuclear engineering, such as fuel elements, structural materials, metal treatment, welding, heat exchange equipment, water chemistry, etc. In view of this, the UNITHERM NPP may require no major technology development effort to be implemented.

II-1.9. Status of R&D and planned schedule

The UNITHERM is at a conceptual design stage.

As it was mentioned, the UNITHERM requires no major R&D for technology development. At the same time, it is reasonable at this design stage to carry out R&D for certain innovative systems and components of the design, such as the independent circuit for heat removal and systems for the reactor equipment cooling.

Once the agreement with the user is reached and the technical assignment (terms of reference) is approved, it is estimated to take 5 years to finalize design development, license, construct and commission the UNITHERM NPP, provided there are no financial or organizational constraints. The detailed design stage would include qualification of the core, heat exchangers, CEDMs, etc. At present, the time schedule for these activities is not available.

II-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

As discussed above, the UNITHERM NPP is an innovative plant both in terms of technical approaches and taking into account a specific user’s preference. The prototype (or pilot) plant is needed to:

- Demonstrate to the potential users the implementation of the whole set of the above mentioned innovative conceptual approaches;
- Qualify serviceability of the reactor core during its long-term operation without refuelling.

II-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The UNITHERM has no direct analogues being developed in the Russian Federation. Else, no information was provided.

II-2. Design description and data for UNITHERM

II-2.1. Description of the nuclear systems

Reactor core and fuel design

The UNITHERM fuel element is designed as a cylindrical rod with four spacing ribs on its outer surface, Fig. II-10. The circumscribed diameter of the fuel element is 6.95 mm, the outer diameter of the rod is 5.8 mm and the cladding thickness is 0.5 mm. To ensure self-spacing in a fuel bundle, the ribs are fixed helically with respect to a central axis, with a 400 mm pitch. The fuel composition is in the form of tiny blocks of UO_2 grains coated with zirconium and dispersed in a zirconium matrix. A gap between the fuel-containing matrix and cladding is filled with silumin. A fuel element of such design has a high uranium content and radiation resistance. These features taken together make it possible to operate such fuel elements during the whole specified core lifetime.

The fuel assembly design is shown in Fig. II-11. Burnable absorber rods (2) are used for axial and radial flattening of power density in the core and compensation of the reactivity margin throughout the operating life.

The reactor core consists of 265 fuel assemblies installed in the plates of the removable reactor screen at points of a regular hexagonal lattice, with a 72 mm pitch. The core height is 1100 mm; the core equivalent diameter is 1230.7 mm. Each fuel assembly is composed of two main parts: the upper part is a suspension and the lower part is a cassette consisting of a tight fuel element bundle, burnable absorber rods and the displacers of a plate type, Fig. II-12.

The fuel assembly's cassette and suspension are joined by a thread and fixed by pins and welding. The suspension consists of a collet head and a tie tube of 50 mm diameter. The fuel assemblies are spaced by tail parts in the lower plate of the reactor removable screen and can move axially in the plate under thermal and radial expansion. The fuel assembly design includes the materials well proven in long-term operation and manufactured based on well-explored technologies, such as:

- E-110 zirconium alloy;
- E-110B zirconium alloy with boron;
- CrNi35VTiEu alloy;
- 08Cr18Ni10Ti corrosion resistant steel.

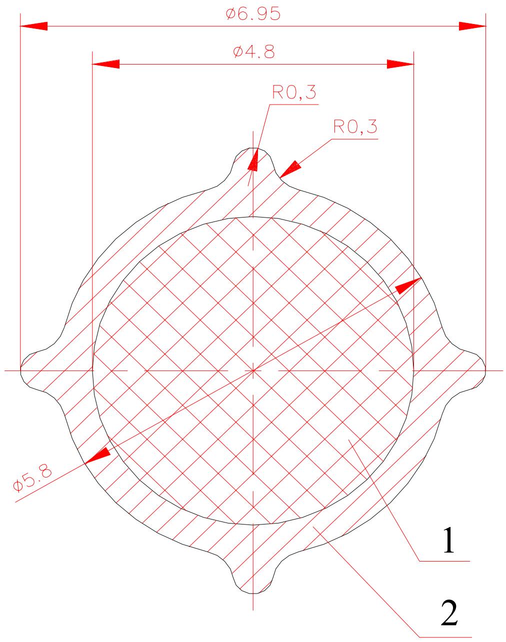
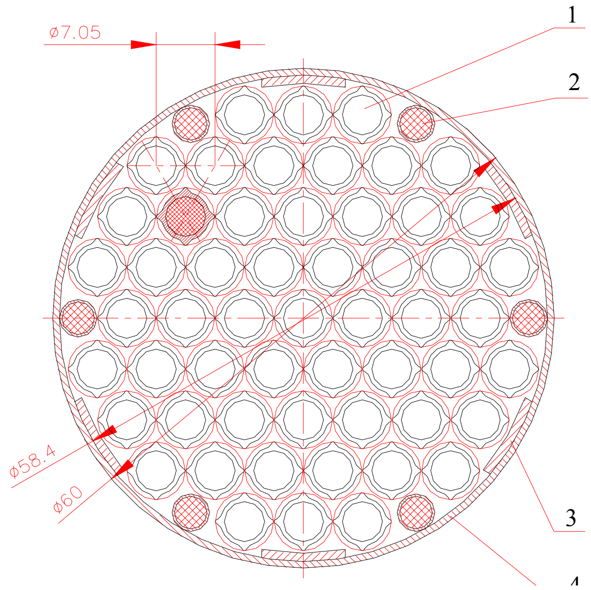
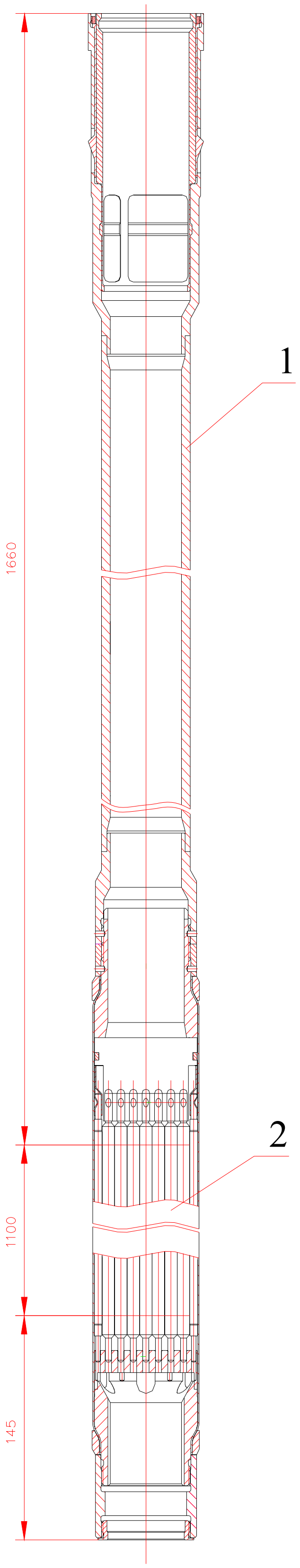


FIG. II-10. Cross-section of a fuel element.



1 – Fuel element; 2 – Burnable absorber; 3 – Plate-type displacer; 4 – Shroud

FIG. II-11. Cross-section of a fuel assembly.

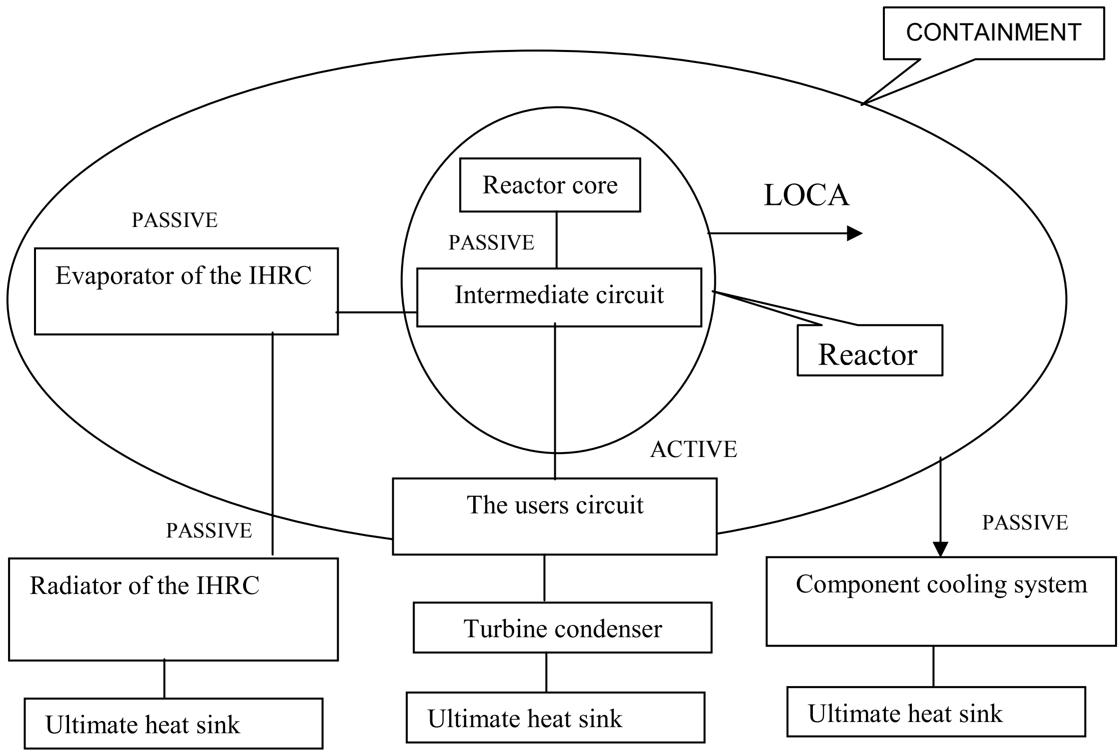


1 – Suspension; 2 – Cassette

FIG. II-12. Fuel assembly.

Main heat transport system

The scheme of the UNITHERM main heat transport system is shown in Fig. II-13. All trains shown in the diagram (except for LOCA trains) are redundant and in continuous operation.



IHRC – independent heat removal circuit.

FIG. II-13. The diagram of heat removal from the core.

The component cooling circuit is passively operated and continuously removes heat from the reactor components enclosed in the containment.

In the case of LOCA, some primary coolant and steam-gas mixture from the pressurizer are discharged to the containment. The emergency core protection system acts in response to the signals of pressure transducers. Leakage proceeds until the pressure values in the reactor and in the containment are equalized (about 2 MPa after 10 minutes of the transient). The amount of coolant remaining in the reactor is sufficient to maintain circulation in the main coolant circuit. The reactor is passively cooled via the intermediate circuit and the independent heat removal circuit without time limitation. The component cooling system removes heat from the containment.

To speed up cooling, it is possible to use the active user circuit with feedwater supplied to the steam generators and steam, steam-water mixture and water discharged to the turbine condenser.

II-2.2. Description of the turbine generator plant and systems

The choice of a candidate turbine generator plant for the UNITHERM NPP depends on the plant capacity and operation mode (electricity generation or cogeneration) as requested by its users. It is planned to use standard turbine equipment, for example, manufactured by the Kaluga Turbine Works (AO KTZ) [II-6]. These turbine-generators use steam of low parameters.

For example, maximum total efficiency of the NPP can be obtained with a two-phase intermediate circuit (i.e., in a cogeneration mode) with the use of the modular Touman-2.5 turbine unit manufactured by the AO KZT with the following characteristics:

- Output capacity, MW
  - Electric (gross) 2.5
  - Thermal 20
- Steam parameters at the inlet to the power unit:
  - Absolute pressure, MPa 0.7
  - Temperature, °C 165
  - Flow rate, kg/s 11.1
- Outlet parameters of the heating-system water:
  - Absolute pressure, MPa 0.45
  - Temperature, °C 90
  - Flow rate, kg/s 200

The turbine operates using dry saturated steam in the mode of a steam outlet backpressure of 1.2–1.05 bar. With consideration of the continuous transfer of 5 % of heat to the independent heat removal system, the total efficiency of the UNITHERM NPP would in this case amount to ~ 74 %. This very high value is achieved due to the utilization of low-parameter heat at the turbine exhaust.

The use of a single-phase intermediate circuit (i.e., the operation in electricity generation mode) allows obtaining a superheated steam temperature of 285°C under 1.35 MPa. Therefore, a suitable option for the NPP is the P-6-1.2/0.5 turbine [II-6], which supports an electric generator with a maximum output of 6.6 MW(e).

II-2.3. Systems for non-electric applications

No information was provided.

II-2.4. Plant layout

The general layout of the SS NPP with the UNITHERM RF has not been developed at this stage of design.

REFERENCES

[II-1] ADAMOVICH, L.A., GRECHKO, G.I., SHISHKIN, V.A., Self-contained co-generation nuclear power plant with integral nuclear reactor designed for remote and difficult-to-access areas, Nuclear Engineering and Design 173 (1997) pp. 175–182.

[II-2] ADAMOVICH, L.A., GRECHKO, G.I., SHISHKIN, V.A., Self-sufficient co-generation nuclear power plant with integral nuclear reactor designed for remote and difficult-to-access areas, ICONE-6 (Proc. 6th Int. Conf. on Nuclear Engineering, San Diego, CA, May 10–15, 1998) ASME.

[II-3] ADAMOVICH, L.A., GRECHKO, G.I., SHISHKIN, V.A., The use of ship-mounted reactor technologies for designing land-based nuclear power plants, ICONE-8 (Proc. 8th Int. Conf. on Nuclear Engineering, Baltimore, April 2–6, 2000) ASME.

[II-4] ADAMOVICH, L.A., EREMEEV, D.V., SHPANSKY S.B. Experimental investigation into thermal siphon used as an intermediate circuit of an integrated cooling system reactor, ICONE-10 (Proc. 10th Int. Conf. on Nuclear Engineering, Arlington, April 14-18, 2002) ASME.

[II-5] ADAMOVICH L.A., et al., A comparative analysis of the reactor designs for small nuclear power plants of various applications, ICONE-12 (Proc. 12th Int. Conf. on Nuclear Engineering, Arlington, VA, April 25–29, 2004) ASME.

[II-6] AO KTZ (Kaluga Turbine Works). Steam turbines and turbine generators (range of products), Kaluga (1996).

ANNEX III

UNATTENDED SELF-CONTROLLED NUCLEAR THERMOELECTRIC PLANT, (ELENA NTEP)

Russian Research Centre “Kurchatov Institute”,
Russian Federation

III-1. General information, technical features and operating characteristics

III-1.1. Introduction

The ELENA NTEP is an unattended self-controlled nuclear thermoelectric plant for cogeneration of electricity and heat. The plant name reflects the following major features of the design:

- Capability of power operation without the involvement of personnel;
- Compensation of burn-up reactivity swing and other external reactivity disturbances without moving the control rods;
- The use of thermoelectric energy conversion to produce electricity.

The design of nuclear power plants with thermoelectric energy conversion systems was underway in the 1960s in many countries (the Russian Federation, the USA, the United Kingdom, and France). These plants were intended as power sources for space and oceanic research.

A series of the US thermoelectric power sources of the SNAP type was developed and used in space satellites and lunar explorations. In this, radioisotopes were heat sources in plants with uneven ordinal numbers and nuclear reactors were heat sources in plants with even ordinal numbers [III-1, III-2].

Power sources of this type in use in the Russian Federation are radioisotope based thermoelectric generators on satellites and the ROMASHKA thermoelectric nuclear reactor [III-3].

General Dynamics, Westinghouse Electric and Martin developed thermoelectric nuclear plants for marine applications in the USA [III-4].

The Russian Federation developed and constructed a pilot and demonstration thermoelectric nuclear plant GAMMA, which has been put into operation in 1982 and is still operating. Over the operation period, comprehensive studies were performed at the GAMMA to validate autonomous nuclear power sources for deep-sea deployment [III-5, III-6].

A specific feature of the above mentioned plants is the capability of operator-free performance. This feature becomes very important if such plants are to be used for heat and electricity supply to small towns in the remote areas with no centralized power supply.

The ELENA NTEP project was developed using the experiences in construction and operation of marine and space power plants and the operation experience of the GAMMA reactor. The concept has been developed by the Russian Research Centre “Kurchatov Institute” (RRC KI). Participants in the development of the project were several Russian organizations, with the major contributors being Federal State Unitary Enterprise (FSUE) "Krasnaya zvezda", FSUE "Atomenergoproekt" and Joint Stock Company (JSC) "Izhorskiye zavody" [III-7].

III-1.2. Applications

The unattended ELENA NTEP plant is designed to produce heat for towns with a population of 1 500–2 000 located in remote and hard-to-reach areas where district heating is required. Since it is auxiliary in nature, the electricity generation of 68 kW could be used for the in-house power needs of the plant and to supply electricity to consumers requiring a highly reliable power supply, such as hospitals, etc.

A desalination unit can be used in combination with the ELENA NTEP plant.

III-1.3. Special features

The ELENA NTEP is a land-based plant; however, in principle, it is possible to develop its versions for underground or underwater deployment. The reactor and the main systems of the plant are assembled from factory-fabricated finished units, whose weight and dimensions enable the delivery by any transport, including a helicopter, or a delivery by sea of the completely assembled plant.

III-1.4. Summary of major design and operating characteristics

The installed capacity of the reactor is 3.3 MW(th); in this, 72 GCal/hour is being supplied for heating. The electric output is 68 kW.

The reactor is designed to operate in a base load mode. A decrease in the heat or consumer power is automatically compensated through the discharge of excess heat to the atmosphere via a dry cooling tower, with no changes in the electric power.

The availability of the ELENA NTEP plant is not less than 0.87, with regard to periodic routine checks done by a field team.

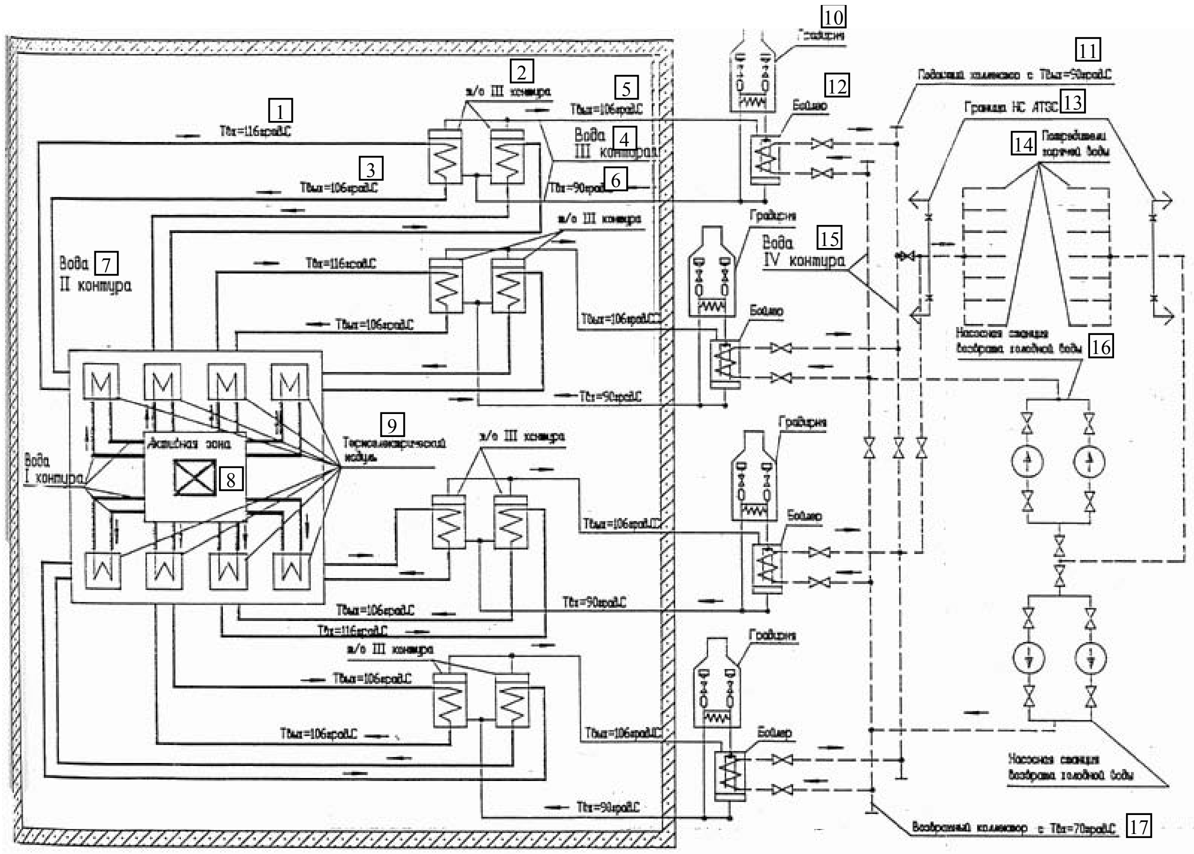
Pellet type uranium dioxide fuel is used with the average ^{235}U enrichment of 15.2%; the neutron moderator and coolant is water specially treated according to specified water chemistry. Cylindrical fuel elements with stainless steel cladding are installed in 109 fuel assemblies of 55 fuel elements each; 216 absorber rods with boron carbide based neutron absorber are divided into 6 groups. Fuel assemblies also include burnable absorbers made of Gd-Nb-Zr alloy. The ^{235}U load is 147 kg.

The cylindrical core with a height of 850 mm and the equivalent diameter of 833 mm is installed in a steel shell with a diameter of 920 mm and is encircled by an iron-water shield.

The strengthened stainless steel reactor vessel has an internal diameter of 1250 mm and a height of 3700 mm with a wall thickness of 132 mm.

A schematic diagram of the plant is presented in Fig. III-1.

The plant has three circuits with natural circulation of coolant. Heat is supplied to consumers by circuit IV with forced circulation, which is part of the municipal service lines; circuit IV pumps are installed within the town. The coolant pressure in circuit III is lower than in circuit IV to help avoid leaks from circuit III to circuit IV in case of heat exchanger failures.



1. $T_{in} = 116^oC$
2. Circuit III heat exchanger
3. $T_{out} = 106^oC$
4. Circuit III water
5. $T_{out} = 106^oC$
6. $T_{in} = 90^oC$
7. Circuit II water
8. Core
9. Thermoelectric module
10. Cooling tower
11. Header with $T_{out} = 90^oC$
12. Boiler
13. Plant boundary
14. Hot water consumers
15. Circuit IV water
16. Pump station for cold water return
17. Return header with $T_{in} = 70^oC$

FIG. III-1. Schematic diagram of the ELENA-NTEP plant.

The primary circuit (circuit I) with natural circulation of the coolant (water with a pressure of 19.6 MPa specially treated to meet specified water chemistry conditions) transports heat from the core to the thermoelectric generator (TEG) modules cooled by the circuit II coolant (water with a pressure of 0.36 MPa). The TEG consists of eight thermoelectric units, each housing 36 annular tube modules. The latter are electric current generators based on semi-conductors and operating due to a difference in the temperatures of the circuit I and II coolants.

The circuit II coolant transports heat from the external surfaces of the TEG modules to eight heat exchangers of circuits II–III. Circuit III is designed as a thermo-siphon with water or low-boiling coolant. Heat is transported to heat exchangers (4) of the circuit IV, which is for transporting heat to consumers, by the saturated steam of the low-boiling coolant condensing on the outside of the heat exchanger III–IV tubes. The internal space for heat transport to consumers is connected to an air-cooled heat exchanger enclosed in the draft tube for excess heat discharge to the atmosphere. In turn, this heat exchanger is connected to the circuit III pressurizer.

Under normal operating conditions, the internal space of the air-cooled heat exchanger is filled with gas (nitrogen) and the heat exchanger does not operate. If the load decreases in the consumer system, the pressure in circuit III increases, the gas-steam separation boundary is shifted upwards and the air-cooled heat exchanger is actuated to remove heat to the atmosphere; in this way, heat removal from circuit III is maintained at a level close to the nominal mode of operation.

A general view of the ELENA NTEP reactor installation is shown in Fig. III-2.

The reactor is installed in a caisson forming a heat-insulating gas cavity in the area of the strengthened reactor vessel and a caisson space above the reactor cover to house control and protection system (CPS) drives and to prevent radioactive substances from escaping into the surrounding space in case of a circuit I break. In turn, the caisson is encircled by the external containment, which is the next barrier to the spread of radioactivity; water that fills the containment volume is the circuit II coolant and acts as a biological shielding for the reactor. The external containment forms the cylindrical geometry of the plant with a height of 13 m and a diameter of 4.45 m.

Table III-1 provides a summary of reactivity effects of the ELENA reactor.

TABLE III-1. REACTIVITY EFFECTS, ΔK/K

Cumulative reactivity effect in the working point,		–0.0448
including:	— Density effect	–0.0374
	— Temperature effect	–0.0059
	— Power effect	–0.0052
	— Reflector density effect	0.0037
Temperature reactivity coefficient in the working point	At the beginning of lifetime (BOL)	$-42 \cdot 10^{-5}$ 1/C
	At the end of lifetime (EOL, after 627 GW-hour of energy has been produced)	$-63 \cdot 10^{-5}$ 1/C

The void reactivity effect is negative; its absolute value has not been calculated because it considerably exceeds (by dozens of times) the presented value of the density effect.

The burn-up reactivity swing during 190 000 hours of nominal power operation, without the interference of control rods, is +0.0085. This corresponds to a reduction in the average core coolant temperature by 18°C.

Figure III-3 presents the dependence of the effective coolant temperature value on the nominal power operation time. Due to the use of burnable absorbers, the change in the reactivity and therefore, in the average temperature with burn-up is of a complex oscillatory nature.

The maximum value of the core power peaking factor is 2.25.

The maximum value of the fission product accumulation at the end of the reactor lifetime is 0.581 g/cm^3 with an average value of 0.276 g/cm^3, which corresponds to 57 600 MW·day/kg U and 27390 MW·day/kg U respectively. With a low specific power rating of 7.1 kW/l, ^{135}Xe poisoning is insignificant.

The average content of nuclides in the reactor at the end of its lifetime is given in Table III-2.

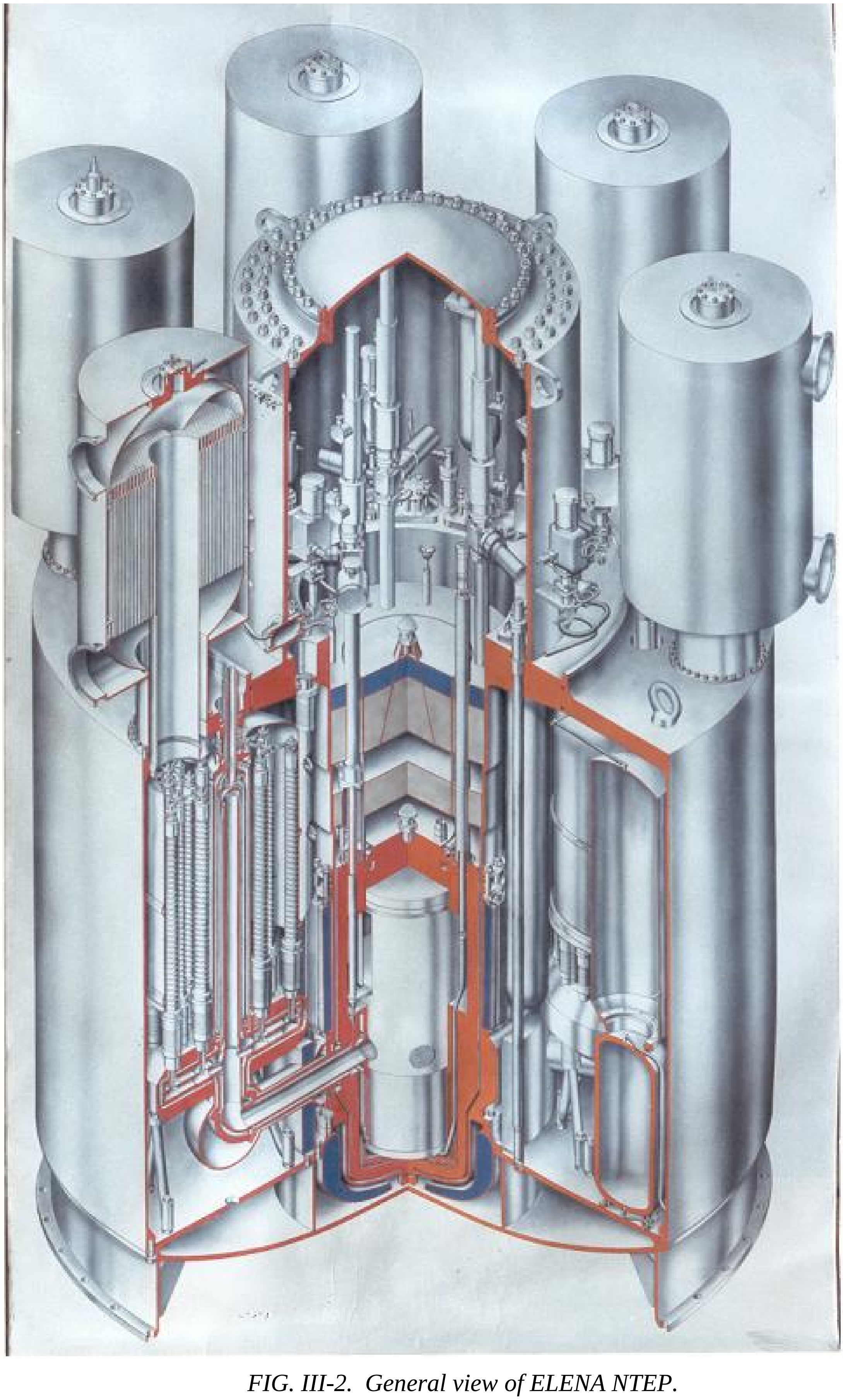


FIG. III-2. General view of ELENA NTEP.

TABLE III-2. AVERAGE NUCLIDE CONTENT AT THE END OF THE LIFE, KG

^{235}U	^{236}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	FP*-U	FP*-Pu
120.8	6.8	783.0	5.23	0.50	0.29	0.025	26.6	1.57

* FP is for fission products.

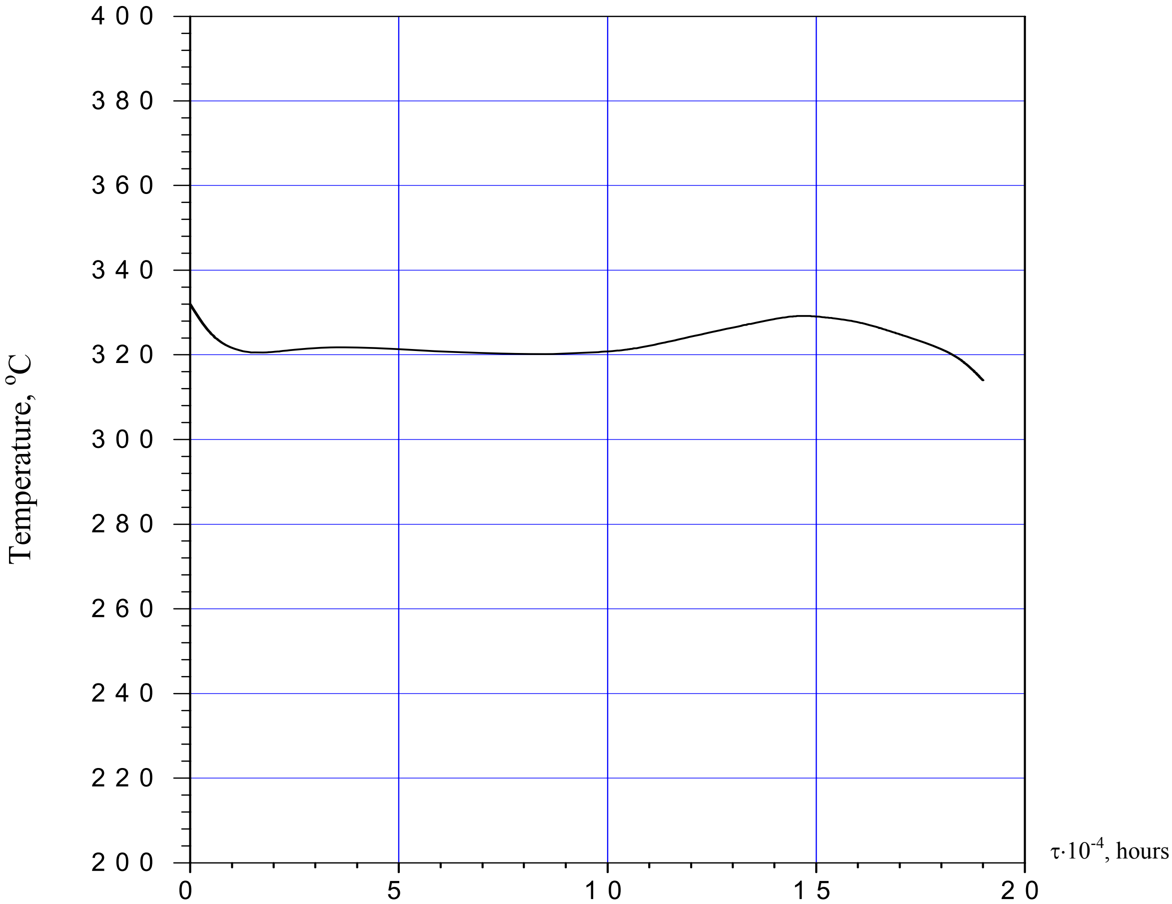


FIG. III-3. Effective coolant temperature versus nominal power operation time.

To start up and reliably shut down the reactor in any situation, a grid is included that compensates the excessive reactivity. The compensation grid consists of six groups of the boron carbide absorber rods in stainless steel claddings of 1.45 cm external diameter. Each group (34 rods) has an individual drive.

The maximum super-criticality of the reactor in the initial cold un-poisoned state with all compensation groups being withdrawn is equal to 0.0507, whereas the reactivity of the compensation groups is 0.273 for 6 compensation groups and 0.111 for 5 compensation groups. Therefore, the rods of the compensation groups also serve as emergency protection (shutdown) rods.

Due to special design of the compensation groups, the reactor is brought to a sub-critical state by the drop of any four of the compensation groups while any two groups for whatever reason remain hung up in the upper position.

Active operation of the reactivity control system relating to the compensation group movement is required only for the plant start-up and shutdown. If the compensation group drive power is off, the grid moves downwards under the action of the spring and by gravity.

After the plant is brought to the nominal parameters, the reactor changes over to the power self-control mode when the compensation group absorber elements are all located in the upper part of the core; they create an operating reactivity margin within the limits of one effective delayed neutron fraction (β_{eff}), which does not threaten reactor safety under an erroneous or even malevolent personnel action.

The main thermal-hydraulic characteristics of the ELENA-NTEP plant are presented in Table III-3; coolant temperatures in all circuits are also shown in Fig. III-1. Under normal operation conditions, the maximum temperature of the fuel is 673 K, that of structural materials (fuel cladding) is 638 K and the departure from nucleate boiling (DNB) is 15 K.

The fuel lifetime/burn-up characteristics and economic characteristics of the ELENA-NTEP plant are given in Table III-4.

III-1.5. Outline of fuel cycle options

For the ELENA-NTEP, the basic option is a once-through fuel cycle with uranium dioxide fuel. It is possible to use MOX fuel. A lower value of the delayed neutron fraction typical of MOX-fuelled reactors might not pose a problem for the ELENA-NTEP because it is designed to operate at a constant power level, resulting in essentially no transients.

If introduced, the fuel reprocessing is assumed to be based on the existing technology and performed at a specialized enterprise. After the operating life is over, the reactor is removed from the site and reprocessed in a centralized manner.

III-1.6. Technical features and technological approaches that are definitive for ELENA-NTEP performance in particular areas

III-1.6.1. Economics and maintainability

A cost reduction (up to a factor of 1.5 against the first-of-kind plant) may be achieved in the case of a serial production of the ELENA-NTEP plants. The reactor operates at nominal power in an unattended mode that does not require the involvement of operating personnel in the process control. The start-up, shutdown, routine inspection and if required, repair, are performed by personnel from the control centre on a shift basis. The site security may require auxiliary personnel residing in the plant deployment area.

III-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The design and operation of a nuclear plant with the ELENA thermoelectric nuclear facility provides for no release of radioactive waste beyond the protection vault.

The primary plant decommissioning option is disassembly of the reactor facility components and systems with the complete removal of wastes from the operation site.

TABLE III-3. MAIN THERMAL-HYDRAULIC CHARACTERISTICS

TYPE OF COOLANT CIRCULATION IN CIRCUITS I, II AND III	NATURAL
Water temperature at reactor inlet	584 K
Water temperature at reactor outlet	601 K
Average coolant temperature in the core, K	593
Circuit I water flow rate	31.7 kg/s
Circuit I pressure	19.6 MPa
Circuit II water flow rate	100 kg/s
Circuit II pressure	0.36 MPa

TABLE III-4. FUEL LIFETIME/BURN-UP CHARACTERISTICS AND ECONOMIC INDICATORS

Average fuel burn-up	27 390 MW·day/t U
Maximum fuel burn-up	57 600 MW·day/t U
Fuel lifetime (without reloading and shuffling of the fuel)	190 000 hours
Specified core lifetime at nominal parameters	21.7 years
Maximum fluence of neutrons with E>0.1 MeV with which fuel elements remain functional, n/m^2	9×10^{25}
Reactor operation lifetime	25 years
Construction cost	US$30 million*
Superstructure volume	5250 m^3
Substructure volume	5450 m^3
Concrete consumption	3400 m^3
Metal consumption	250 t
Construction period, including: — Manufacturing of components — Construction — Assembly and commissioning	4 years 1 year 2 years 1 year
Fuel cost	As per world prices
Cost of energy for district heating	US$50/GCal*
Cost of electricity	US$0.06/kW-hour*

* The ELENA NTEP is a plant of very small power designed for use in remote areas. In this connection, the construction costs would depend strongly on local conditions: the existing infrastructure, transport pathways, availability of materials for the construction, workforce, etc. Estimates performed for the Russian Federation show that site-specific conditions may change the construction cost by the factor of 1.5–2. Apart from low power, other features of the ELENA NTEP are long-term operation with a single fuel load and absence of the personnel and, therefore, labour payment and life support costs. Although the specific construction cost of the ELENA-NTEP may be high, the power generation cost could be competitive with the organic fuel for these areas.

After the reactor is shut down by control rods, cooled down and after-cooled during 30–90 days, the plant is brought to a nuclear-safe condition by removing the fuel. The heat exchangers of circuits II–III, the TEG, the drives and other components are disassembled in succession. The shielding plug and then the reactor cover are removed using a special container; they are later decontaminated and placed in special rooms at an elevation of minus 14.2.

The reactor design allows the disassembly to be performed in two ways:

(1) Removal of the core as a whole. The reactor is withdrawn from the strengthened vessel together with the fuel assemblies, placed onto a trolley with a container and transported to the scrapping room where it is disassembled into separate units and parts. The disassembled parts and units are placed in containers for subsequent recycling and disposal;

(2) Assembly-by-assembly removal. The withdrawn assemblies are placed in containers to be reprocessed. The radioactivity of each of the removed fuel assemblies in 30 days after the reactor shutdown is 300 Ci because of the fission products. The circuit I coolant is drained into special tanks and later removed to cooling and disposal areas.

Components of the stainless steel vessel structures can be scrapped and transported after 15 years of cooling. In principle, it is possible to disassemble components after one year of cooling, but this would require arrangement of a local biological shield in the area of operations.

III-1.6.3. Safety and reliability

Safety concept and design philosophy

The nuclear and environmental safety of the ELENA NTEP is assured by the following design measures:

- Reactor operation with nominal parameters in the power self-control mode that excludes the movement of the CPS control rods;
- The incorporation of the defence in depth approach based on six safety barriers preventing the primary circuit from depressurization and securing activity confinement inside the reactor during accidents;
- The operation without reloading and shuffling of fuel throughout the whole service life (potentially hazardous refuelling operations are excluded by design);
- No need for the personnel to attend the reactor installation throughout whole period of power operation (unattended or operator-free performance of the plant),
- Separation of the reactor installation from the environment (no discharges of radioactive liquid and gaseous wastes throughout the whole service life, including accidents; no systems for such discharges);
- The use of natural convection for heat removal from the core and in the intermediate circuits; elimination of circulators and active movable elements in the plant design;
- Minimum operating reactivity margin in the core to essentially eliminate the possibility of a prompt criticality;
- The use of a single-mode operation pattern, in which the reactor always operates at the rated power level; load changes in the district heating system are compensated by the excess heat discharge system;
- The use of a thermoelectric energy conversion system that does not require the presence of operating personnel;
- Special arrangement of the instrumentation and control (I&C) system intended to register parameter deviations at early stages of the accidental conditions to predict their further progression;
- A specialized field team on a shift basis performs the start-up and shutdown of the plant, and the maintenance and repairs;

— Location of the reactor installation in a leak-tight embedded compartment;
— A low specific thermal power of the reactor (7.1 kW/l) enables easy removal of the residual heat after the reactor shutdown; the residual heat is just damped naturally to the compartment, and the fuel elements do not get super-heated during the process.

Provisions for simplicity and robustness of the design

The robustness of the ELENA-NTEP design and the reliability of its operation have been already proven, to a remarkable extent, by operation experience of the GAMMA thermoelectric nuclear installation.

The following most important conceptual problems have been resolved as a result of experimental studies performed at the GAMMA installation:

(1) All operating parameters of the ELENA-NTEP plant have been confirmed, which is the evidence of a correct selection of the arrangement and process approaches.

(2) The conformity of the neutronic characteristics to the design data has been established, which may serve as a validation of the adopted calculation models and codes;

(3) The stable operation of the plant in a self-control mode with natural circulation of the primary water coolant has been demonstrated, including the impacts of the external perturbations;

(4) The reliability of the thermoelectric modules with output voltages of 28 V, 115 V and 230 V commuted in a TEG section has been confirmed and the stability of their characteristics has been demonstrated;

(6) It has been shown that, in principle, it is possible to develop and optimize the technique of a computer-based on-line diagnostics of the states of a thermoelectric nuclear installation; specifically, the noise based diagnostics methods have been elaborated to enable early detection of deviations in standard parameters while predicting the potential for their further progression;

(7) The possibility of a long-term operation of the plant in the normal operation mode without the intervention of operating personnel has been confirmed.

Active and passive systems and inherent safety features

The inherent safety features of the ELENA-NTEP are a negative temperature reactivity coefficient, a large secondary water inventory (68 m^3), a near-zero burn-up reactivity swing, a very small (near $1\beta_{eff}$) operating reactivity margin in the core, and negative coolant density and void reactivity effects.

The reactor installation is based on passive principles of heat removal (natural convection in all circuits, except for heat transport to the consumers) in normal operation and in shutdown conditions.

The active components of the protection system are scram actuators for six compensation groups of the control rods. The reactor is shut down and kept in a sub critical state at deviations of the reactor installation parameters from normal values thanks to the movement of all six compensation groups to the lower position, resulting in a sub-criticality of $\Delta\rho = -0.17$. The reactor sub-criticality is also ensured if any two compensation groups are stuck in the upper position. The control rods move down driven by a spring load and gravity.

Scram actuation signals are generated in three channels with a "two out of three" logic device. At scram actuation, heat is removed from the reactor by natural circulation of the primary, secondary and tertiary coolants.

The control safety system (CSS) consists of a control safety system for emergency shutdown and a system to input, process and transmit safety-related plant information. During normal operation the emergency shutdown CSS is permanently awaiting a scram actuation request; it also periodically provides information on the state of the plant.

The ELENA-NTEP CSS has three independent power supplies: 2 TEG sections, a diesel generator, and a storage battery.

The localizing safety systems provide the defence in depth and secure the plant safety based on inherent safety features and predominantly passive phenomena; they require no human intervention or external power sources.

Structure of the defence in depth

The safety barriers of the ELENA-NTEP are:

- The fuel elements;
- The leak-tight primary circuit;
- The caisson;
- The reactor vessel and the guard vessel designed to withstand the pressure arising within each of them at their consecutive failure; and
- An embedded silo sealed with the protective plate.

Special measures for the protection of hot water consumers ensure that radioactivity is never released into the network circuit.

Safety support systems create the conditions required for normal functioning of the safety systems; they include power supply systems and a heat removal system that transmits heat to the consumers.

Design basis and beyond design basis accidents

The list of design basis accidents considered for the ELENA NTEP is as the following:

(1) Reactivity initiated accidents:
   (a) Spontaneous withdrawal of one compensation group of the control rods;
   (b) A drop of one compensation group of the control rods with the reactor operating at nominal power;
   (c) Cold water supply to the core at an abnormal operation of one of the heat exchangers.
(2) Loss of the primary circuit leak-tightness:
   (a) Depressurization of the heating conduit of the thermoelectric module;
   (b) Depressurization of the pipelines of auxiliary systems;
   (c) Depressurization of the pressurizer.
(3) Abnormal cooling conditions on the secondary side:
   (a) Loss of the secondary coolant;
   (b) Loss of the tertiary coolant;
   (c) A coolant leak from the secondary circuit to the tertiary circuit.

Calculations have been performed for all design basis accidents; the results show their scenarios do not progress into conditions posing a nuclear hazard and do not lead to exceeding the permissible limits for the equipment.

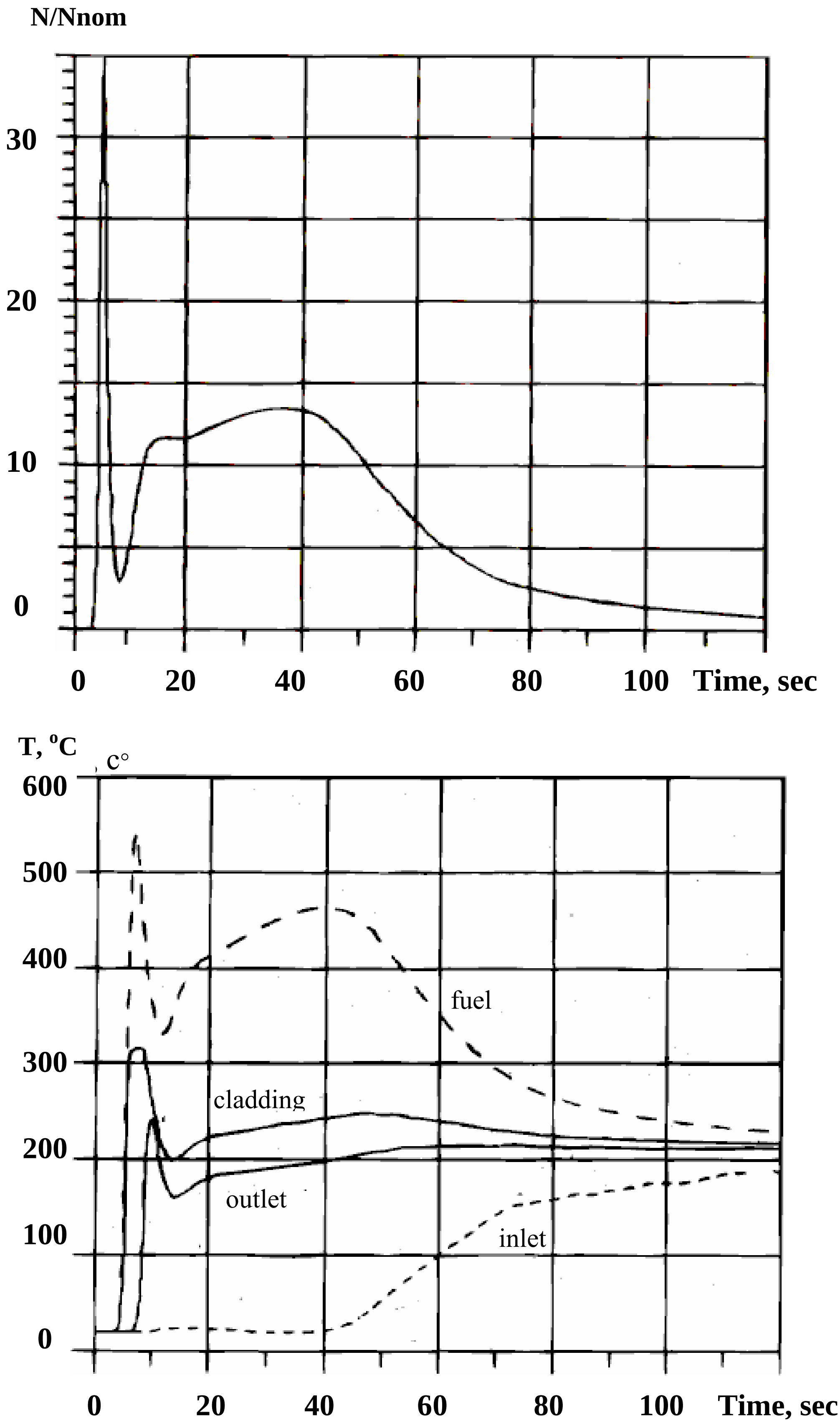


FIG. III-4. Scenario with a maximum-rate ejection of one compensation group of the control rods.

Figure III-4 shows the calculation results for the ultimate design basis accident (1a) — a maximum-rate ejection of one compensation group of the control rods, resulting from a drive break.

Such accident with a large positive reactivity introduced in several fractions of a second is potentially more hazardous in the reactor cold state; it results in two power surges (i) to the magnitude of 35 N_{nom} at the 5th second and (ii) to the magnitude of 13 N_{nom} at the 40th second, where N_{nom} is the nominal power. Doppler effect quenches the first surge; the second surge is largely quenched by the reactivity effect on coolant density. The maximum fuel and cladding temperatures are 813 K (540°C) and 593 K (320°C) respectively; they do not exceed safe operation limits.

The accident with a failure of the actuators of the shutdown system accompanied by simultaneous signal transmission to six independent drives and a trip of the interlocks that restrict the lift of a compensation group of the control rods to above the permissible step has been considered as a beyond design basis accident. In this accident, all compensation groups are withdrawn in a "cold" critical state of the reactor.

With regard to the CPS design, such an accident is hypothetical and its probability is practically negligible but, as it is the gravest of all conceivable reactivity-initiated accidents, its consequences actually define the nuclear hazard limit of the plant.

The calculation results for the above mentioned beyond design basis accident scenario are given in Fig. III-5. Doppler effect quenches the power surge; the coolant circulation develops smoothly; there is no boiling and the coolant temperature in the core does not exceed 613 K (340°C). The fuel temperature of 1173 K (900°C) and the fuel cladding temperature of 838 K (360°C) are much lower than the permissible values. The transient progression as a whole does not prevent the reactor installation from performing its functions.

Provisions for safety under seismic conditions

The reactor building of the ELENA NTEP is referred to as seismic stability category I and has been designed with regard to the ultimate earthquake with a magnitude of 8 on the MSK-64 scale.

Probability of unacceptable radioactivity release beyond the plant boundaries; measures planned in response to severe accidents

Probabilistic safety analysis has shown that the ultimate beyond design basis accident in the ELENA NTEP does not lead to core meltdown. Radioactivity release in the event of a failure of fuel element claddings and a loss of the primary circuit integrity during routine operation has a very low frequency of occurrence (10^{-11}); personnel and the population in the territory near the plant would not be exposed to hazards in excess of the existing standards.

III-1.6.4. Proliferation resistance

The ELENA NTEP is designed to operate using uranium dioxide fuel with the average enrichment of 15.2% (by weight), which is below the boundary of direct use materials (20% of ^{235}U by weight).

The absence of refuelling throughout the whole service life of the plant and plant deployment in a locality with a small population where aliens are easily detected, are factors that prevent thefts of nuclear materials.

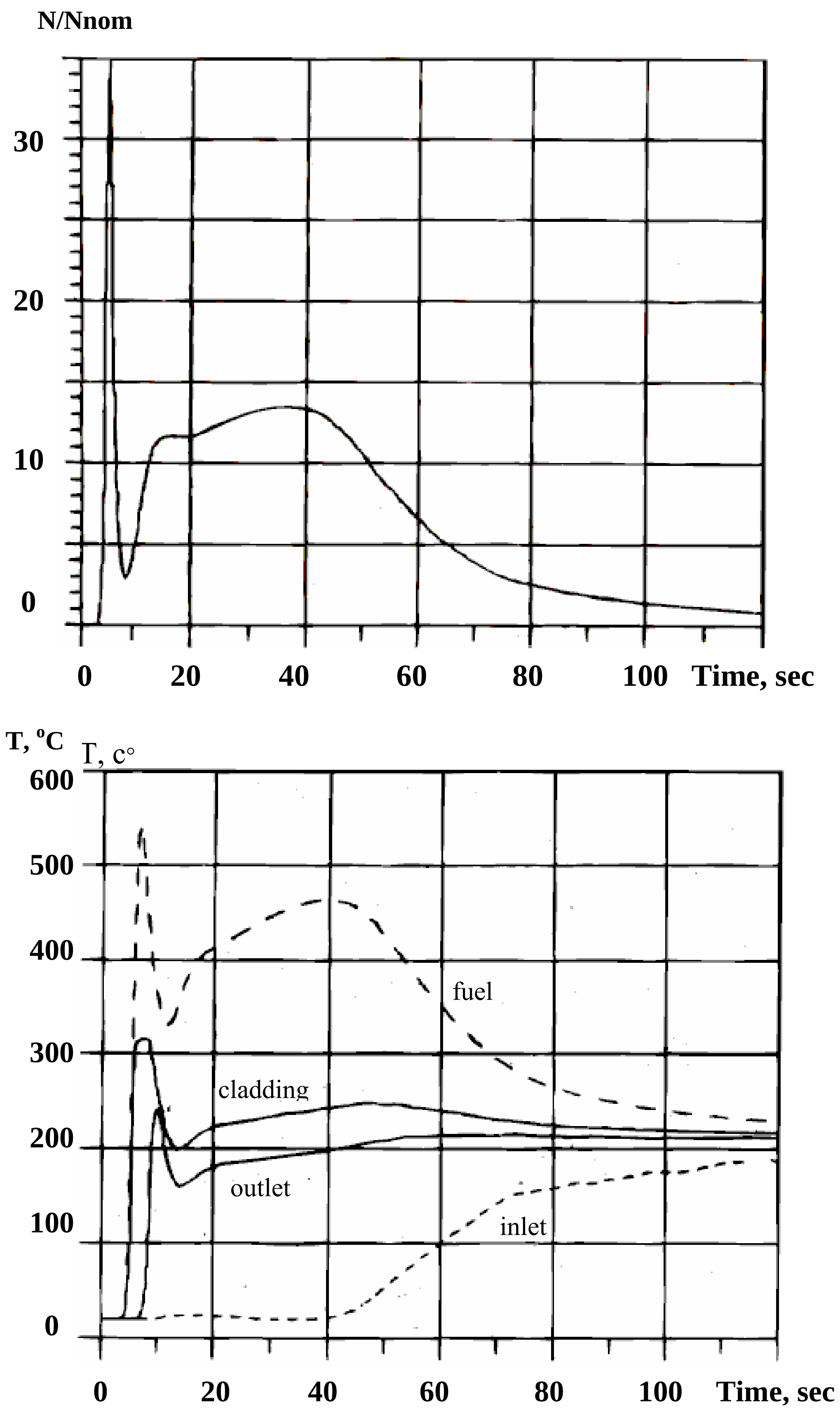


FIG. III-5. Scenario with unauthorized withdrawal of all control rods from cold critical core.

An undeclared production of fissile material at the ELENA NTEP is excluded because the reactor operates at low specific power (7.1 kW/l) with a comparatively low neutron flux and the control system does not permit the allotment of an additional reactivity for irradiation of alien materials.

III-1.6.5. Technical features and technological approaches used to facilitate physical protection of ELENA NTEP

According to the definition used in the Russian regulations, the physical protection system (PPS) of a nuclear power plant is a system of measures to exclude unauthorized interference in the operation of systems for the control and assurance of safe operation of the reactor and to prevent access to the nuclear materials.

Major requirements to the PPS are as follows:

- A preliminary comprehensive check of all persons licensed to have access to the nuclear material or the reactor facility;
- A zone principle of the PPS arrangement, providing for the segregation of the protected, internal and especially important areas;
- Provisions for a limited access to the PPS information for alien persons;
- Securing no impact of the PPS measures on the nuclear safety of the plant;
- The arrangement of security measures.

The ELENA NTEP project provides for implementation of the PPS in full compliance with the Russian regulations. In addition to this, the ELENA NTEP design ensures that any internal or external attempt to disrupt the plant systems, components and structures from normal operation would only stop the plant operation without any nuclear or radiation consequences.

III-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of ELENA NTEP

The ELENA NTEP is a plant designed primarily for low-power heating and it is reasonable to consider its use in remote areas with a heat shortage. Simultaneously, it can be used to supply arid areas with potable water. Market requirements, primarily those for northern and eastern territories of the Russian Federation, were considered at the design stage. In principle, it is possible to lease the ELENA NTEP to customers outside the Russian Federation.

III-1.8. List of enabling technologies relevant to ELENA NTEP and status of their development

Table III-5 gives a list of the enabling technologies for the ELENA NTEP, complete with the current status of their development.

III-1.9. Status of R&D and planned schedule

The main contractors for the project are the Russian Research Centre “Kurchatov Institute”, FSUE "Krasnaya zvezda", JSC "Izhorskiye zavody", FSUE "Atomenergoproekt” and FSUE VNIINM.

The total plant construction period is 4 years; components can be manufactured in 2 years following the construction license issue.

TABLE III-5. LIST OF ENABLING TECHNOLOGIES FOR ELENA NTEP

DESIGN OBJECTIVE OR SUBJECT AREA	ENABLING TECHNOLOGY	DEVELOPMENT STATUS
— Neutronic design; — Thermo-physical design; — Water chemistry.	Conceptual design approaches and calculation procedures based on the experience in development and operation of reactor installations for marine and space applications	— The experience of marine and space reactors is available; the design and operating parameters of the ELENA NTEP have been confirmed by tests performed at the GAMMA reactor; — A detailed design of the fuel element with a service life of 25 years has been completed and approved.
To assure safe operation of the plant based on natural physical phenomena without using mechanisms and active automatic devices	— Reactor self-control; — Natural circulation; — Thermoelectric conversion of energy; — Unattended operation; — Operation without on-site refuelling	
Early diagnostics	Diagnostic methods based on noise analysis of parameter variations	Design principles for the system of early diagnostics have been developed; separate system components have been validated in the GAMMA reactor

III-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Actually, a prototype of the ELENA NTEP is the pilot and demonstration thermoelectric nuclear reactor installation GAMMA since it was used to validate major concepts of the ELENA project. The current ELENA NTEP project offers an enveloping design that is not bound to a potential plant construction site. It is probable that a particular binding to a construction site will require modifying the project but such modifications are not expected to be so significant as to require the construction of a new demonstration prototype.

III-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

A Canadian project of the SLOWPOKE nuclear plant for district heating with a thermal power of 2 MW is somewhat similar, but its operation requires an off-site power supply.

III-2. Design description and data for ELENA NTEP

III-2.1. Description of the nuclear systems

Reactor core and fuel design

The design and major characteristics of the ELENA NTEP core are presented in Table III-6 and in Fig. III-6.

TABLE III-6. CHARACTERISTICS OF THE CORE

CHARACTERISTIC	VALUE
Rated thermal power, MW	3.3
Average specific power of the core, kW/litre	7.1
Core height, mm	850
Equivalent core diameter, mm	833
Number of fuel assemblies in the core	109
Number of fuel elements in fuel assembly	55
Number of burnable absorbers in fuel assembly	6
Fuel assembly duct material	Stainless steel
Fuel assembly duct dimensions, Diameter × thickness, mm	60 × 0.4
Fuel element	Rod type
External fuel diameter, mm	5.8
Fuel cladding thickness, mm	0.3
Fuel material	Sintered UO_2
U-235 load, kg	147
Average fuel enrichment with U-235, % by weigh	15.2
Maximum enrichment with U-235, % by weigh	17.0

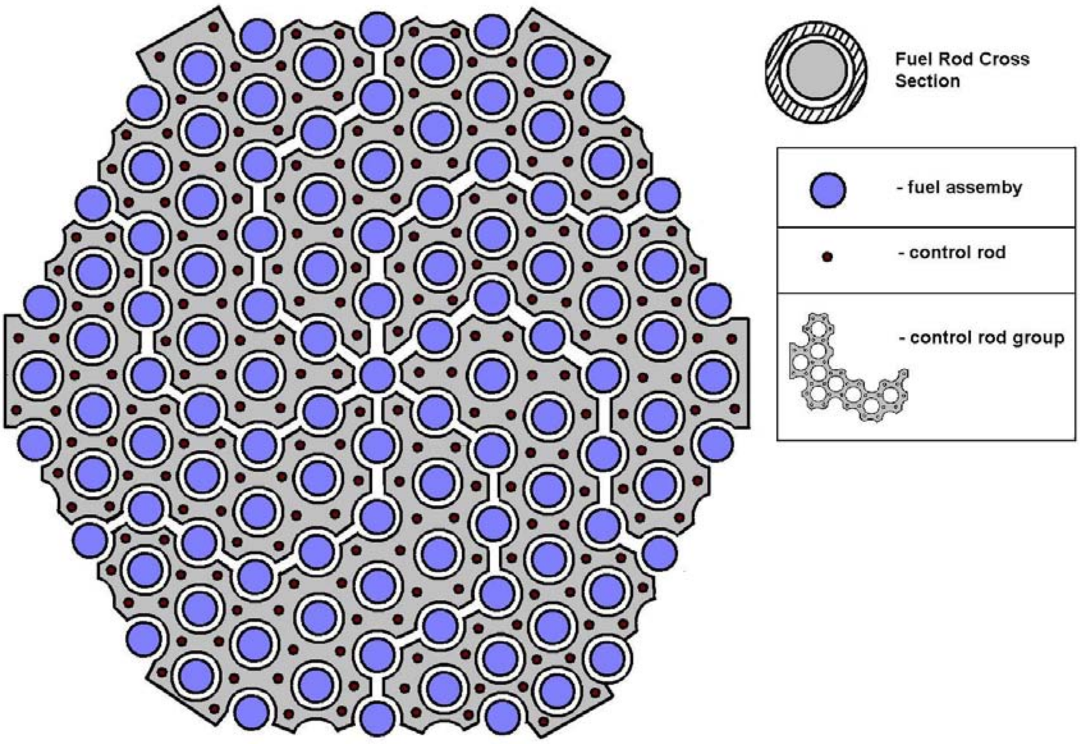


FIG. III-6. ELENA NTEP core map.

Main heat transport system

The system for coolant transport from the core to the consumer has four circuits:

- Circuit I (reactor coolant circuit) removes heat from the fuel elements and transfers it through natural circulation to the hot joints of the thermoelectric generator (TEG) thermal elements; the coolant is water specially treated according to specified water chemistry conditions;
- Circuit II (intermediate circuit) removes heat from the "cold" joints of the thermal elements and transfers it through natural circulation to the intermediate heat exchanger of circuits II–III; the coolant is specially treated water, which also acts as part of the steel-water radiation shielding;
- Circuit III transfers heat through natural circulation to the heat exchanger of the heat supply circuit; the coolant is ethanol;
- Circuit IV transfers heat from the heat exchanger of circuits III–IV to the consumers using forced circulation; the circuit IV coolant is A-60 antifreeze.

The system of heat exchangers removes residual heat from the reactor core in accidents. The major heat removal paths of the ELENA NTEP are shown in Fig. III-7.

III-2.2. Description of the thermoelectric generator

A thermoelectric generator (TEG) is used as a heat exchanger between circuits I and II; it is based on semiconductor thermo-elements enabling the generation of 68 kW of power in the reactor nominal operating mode simultaneously with heat transfer to circuit II. This power is used for the plant auxiliary needs; it could also be supplied to a small town without district power supply, partially replacing a diesel power plant.

The TEG consists of eight identical thermoelectric units (TEU). Each of them includes 36 thermoelectric modules equipped with thermoelectric packs of bismuth tellurides with electronic and hole conduction.

Table III-7 presents the TEG characteristics.

TABLE III-7. CHARACTERISTICS OF THERMOELECTRIC GENERATOR (TEG)

CHARACTERISTIC	VALUE
TEG electric power, kW	68.5
TEU electric voltage, V	29.5
Coolant temperature at TEU inlet, K	601
Coolant temperature at TEU outlet, K	387
TEG efficiency, %	3
TEU hot joint temperature, K	531
TEU “cold” joint temperature, K	419

III-2.3. Systems for non-electric applications

ELENA NTEP is essentially a district heating plant with a small share of electricity generation. Table III-8 shows heat production characteristics of the ELENA NTEP.

TABLE III-8. HEAT PRODUCTION CHARACTERISTICS OF ELENA NTEP

CHARACTERISTIC	VALUE
Reactor thermal power, MW	3.3
Installed district heating power, GCal/hour	2.72
Number of hours of installed power use per year	8000
Heat production, thousand GCal/year	24.1
Consumer system (circuit IV) coolant	A-60 anti-freeze
Circuit IV coolant temperature at the circuit III–IV heat exchanger outlet, K	363
Consumer system coolant pressure, MPa	0.45
Type of water circulation to consumers	Forced
Consumer system water flow rate, kg/s	37.7

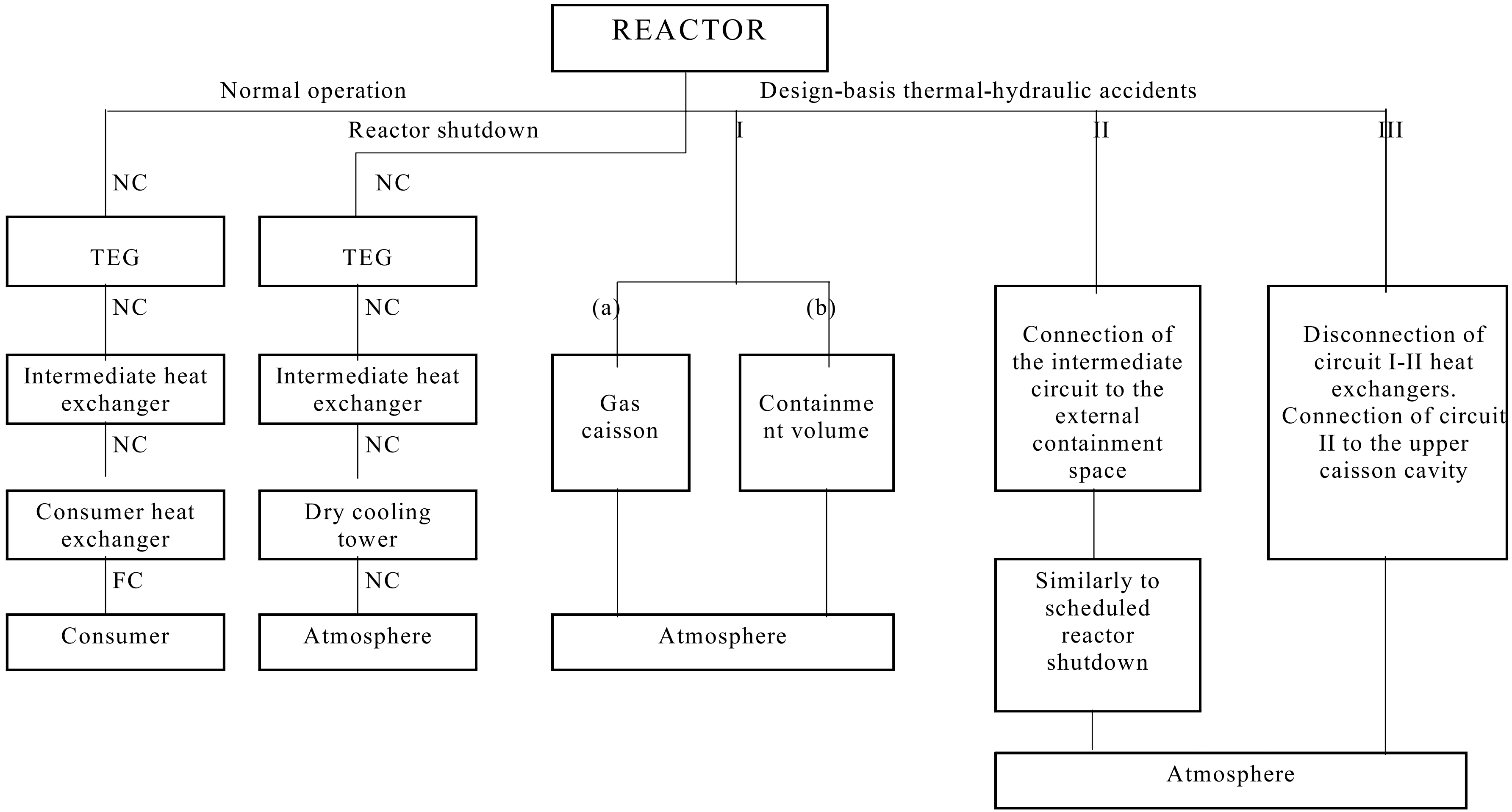
III-2.4. Plant layout

The plant building schematic is shown in Fig. III-8. The plant includes instrumentation and control systems; a system for heat removal to consumers; an auxiliary power supply system; and a radiation monitoring system, including process radiation monitoring, dosimetric monitoring, and environmental monitoring.

The plant has a main control and monitoring room accommodating the start-up and instrumentation and control equipment, as well as the equipment necessary to prepare the information to be transmitted to a monitoring centre.

The plant building has a cylindrical shape and is embedded in the ground for the entire reactor installation height with a foundation plate elevation of –19.2 m. The elevation of +0.0 has a domed ceiling. The underground portion of the structure, the walls and the overlaps are monolithic reinforced concrete.

The plant incorporates a physical protection system, has a fence and is equipped with external lighting.



I – Circuit I depressurization; II – Circuit II depressurization; III – Circuit III depressurization;
(a) before TEGs; (b) area of TEGs; TEG – Thermoelectric generator
NC – natural convection; FC – forced circulation

FIG. III-7. Major heat transport paths of ELENA NTEP.

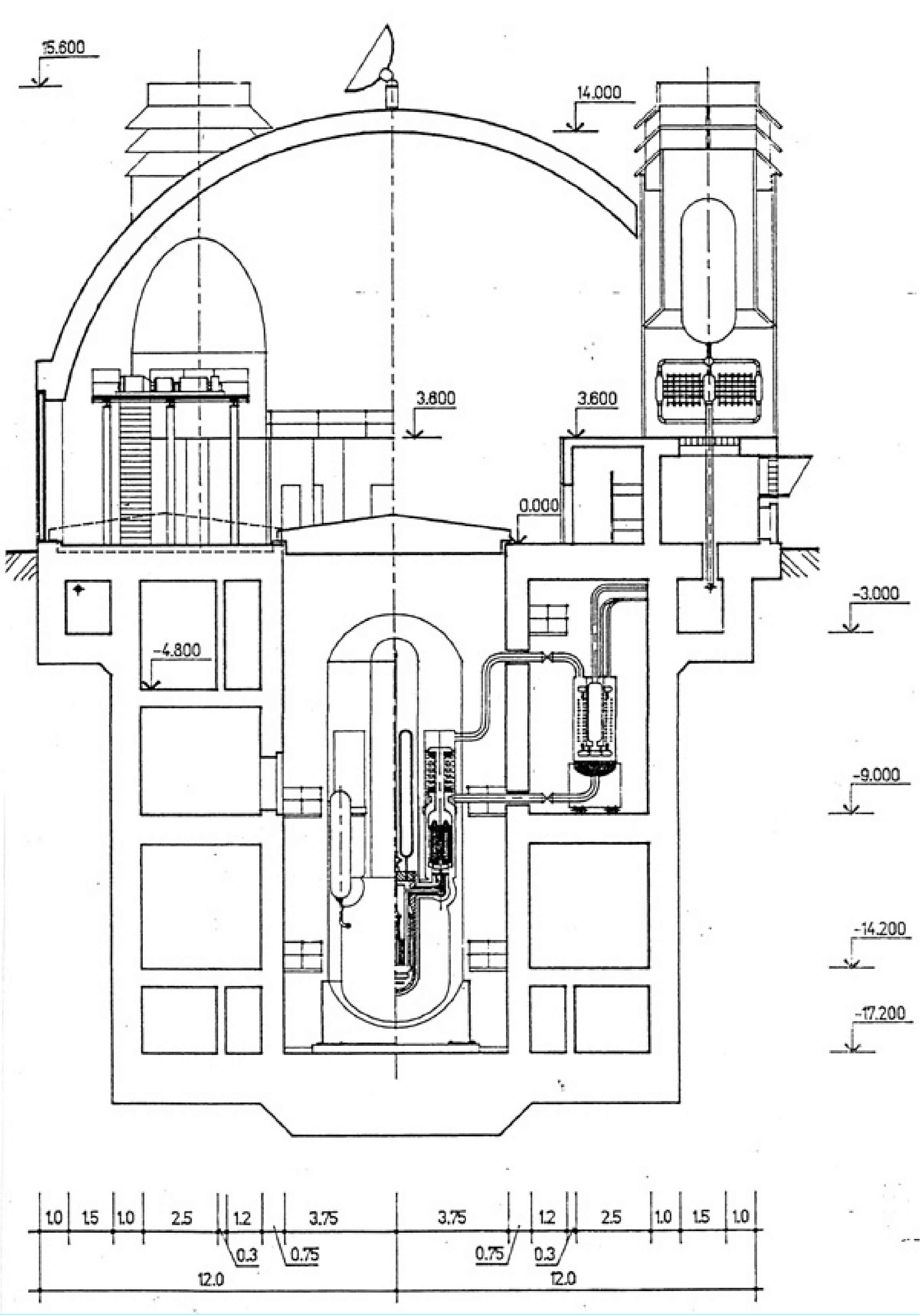


FIG. III-8. Plant general view.

REFERENCES

[III-1] BEAM B.H., Exploratory study of thermoelectric power generation for space flight application. — NASA TWD-336 (1960).

[III-2] The Atomic Energy Commission SNAP-10A Program, Nuclear Energy (August 1965).

[III-3] MILLIONSCHIKOV, M.D., et al., High-temperature conversion reactor ROMASHKA, Atomnaya Energiya, vol. 17, issue 5 (1964).

[III-4] DUFRANE, K., The 100 kWe underwater thermoelectric reactor plant. Conceptual design, Transactions of the American Nuclear Society, No 2 (1963).

[III-5] BUINITSKIY, B.A., et al., Pilot nuclear thermoelectric facility GAMMA — Prototype of an unattended self-controlled nuclear thermoelectric plant, Atomnaya Energiya, vol. 74, issue 1, p. 28 (1993).

[III-6] KAPLAR, YE.P., et al., Pilot and demonstration nuclear thermoelectric facility GAMMA (Construction and operation experience), Report of the Russian Research Centre “Kurchatov Institute”, Reg. No.31/1-600-04 (2004).

[III-7] FSUE "Krasnaya zvezda", Russian Research Centre “Kurchatov Institute", FSUE "Atomenergoproekt", Preliminary design materials for unattended self-controlled nuclear thermoelectric plant ELENA (Moscow, 1994).

ANNEX IV

FLOATING NPP WITH VBER-150 REACTOR INSTALLATION

**Experimental Design Bureau of Machine Building (OKBM),
Russian Federation**

IV-1. General information, technical features and operating characteristics

IV-1.1. Introduction

The VBER-150 is a two-loop modification of the VBER-300 reactor installation described in [IV-1 to IV-4]. It is a small sized loop type pressurized water reactor without on-site refuelling for a floating NPP with cogeneration option.

Both VBER-150 and VBER-300 use the same unified main equipment (reactor vessel and internals, steam generators (SG), main circulation pumps (MCP), control and protection system (CPS), etc. Using a two-loop modification of the VBER-300, the VBER-150 could be used to create floating nuclear power plants (NPPs) of various power output with a long refuelling interval tailored to the needs of a customer.

Modular arrangement of the main reactor components is a key feature of the reactor design concept. The reactor pressure vessel, two once-through steam generators and two main circulating pumps are integrated into a single vessel system by short welded co-axial pipes through which coolant is circulated.

Like the VBER-300, the VBER-150 is thoroughly based on a successful multi-decade experience in the production and operation of marine propulsion in the Russian Federation [IV-1].

Along with the VVER type power reactors, modular shipboard pressurized water reactors represent the most developed reactor technology, well examined and proven by successful operation. The operating experience of shipboard reactors exceeds 6000 reactor-years.

In the reactors of up to 110 MW(e), it is possible to realize long operation cycles without reloading and shuffling of fuel. The refuelling, radioactive waste management and repairs could then be provided off-site, in special maintenance centres.

Absence of off-site refuelling ensures difficult access to the fuel during the entire period of the reactor installation operation including transportation. Floating nuclear NPPs with long fuel life could, therefore, be very attractive for energy supplies in developing countries.

Design and technology development for the VBER-150 is carried out by the Experimental Design Bureau of Machine Building (OKBM), Nizhny Novgorod, Russian Federation.

IV-1.2. Applications

The VBER-150 reactor installation is designed as a power source for floating NPPs and cogeneration plants, providing for the following applications:

- Power generation;
- Heat and power cogeneration for district heating in coastal regions;
- Seawater desalination.

IV-1.3. Special features

The VBER-150 is designed for a floating NPP with power output tailored to the customer needs; it has a long operation cycle and requires no operations with fuel on the site. All refuelling operations and waste management are provided in special maintenance centres.

The VBER-150 plant is designed for operation in off-line mode or within small power grids in immediate proximity to the consumer.

IV-1.4. Summary of major design and operating characteristics [IV-1 to IV-3]

A principal scheme of the VBER-150 plant is shown in Fig. IV-1 (primary circuit and safety systems) and Fig. IV-2 (secondary circuit).

The VBER-150 is an indirect cycle water cooled and water moderated reactor. Hot coolant from the core is cooled in a once-through steam generator; on its secondary side a slightly superheated steam is produced and supplied to the turbine.

In contrast to the VBER-300, the VBER-150 makes use of a two-loop coolant flow path in the steam-generating unit (see Fig. IV-3).

Major design and operating characteristics of the VBER-150 are summarized in Table IV-1.

IV-1.5. Fuel cycle options

The VBER-150 reactor installation concept provides a possibility of core operation with a standard VVER fuel in the following two modes:

- Partial refuelling with an operation cycle of 320 effective days (the refuelling interval is about one year);
- Whole-core refuelling with a long operation cycle.

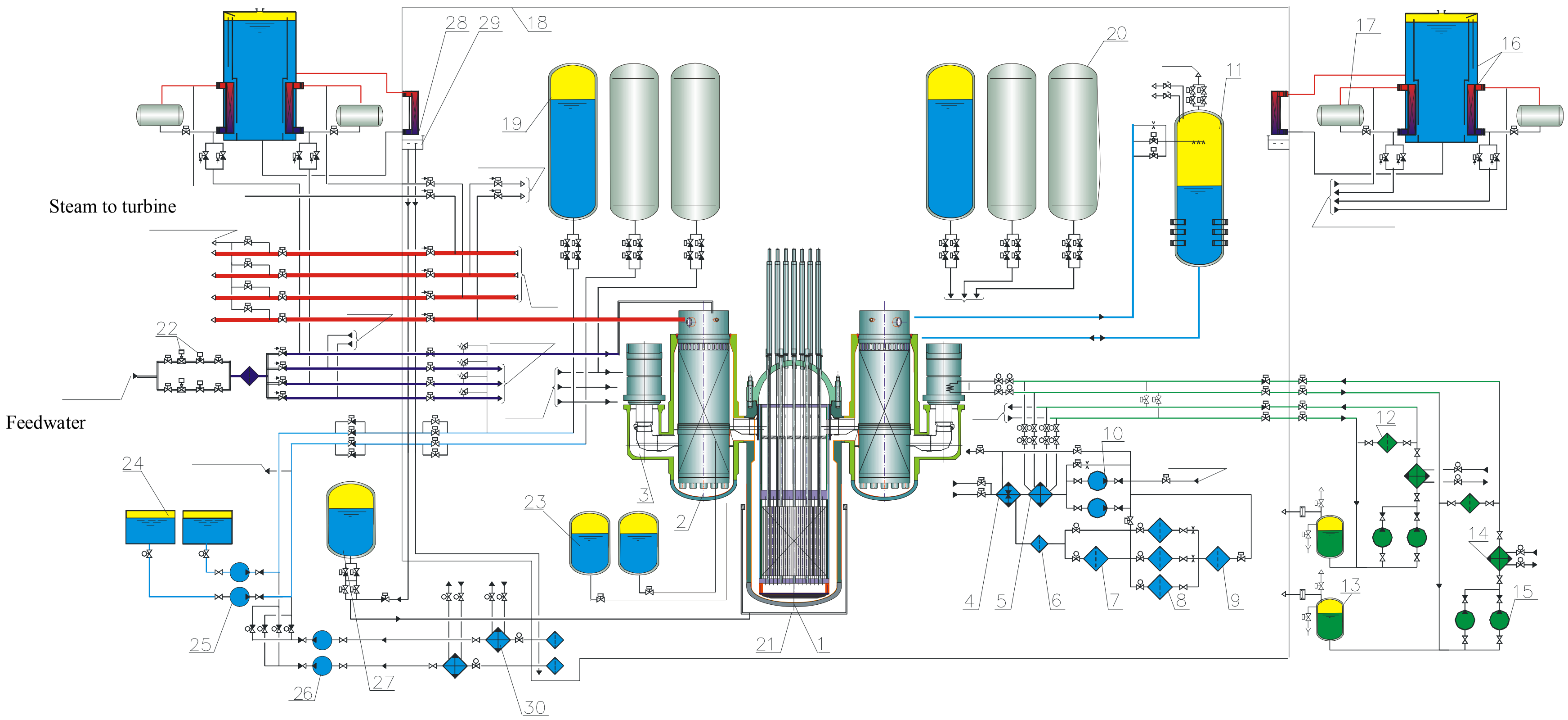
Main characteristics of the VBER-150 in these two operation modes are summarized in Table IV-2.

Maximum fuel burn-up in the VBER-150 fuel elements is 41.6 MW·d/kg U for the whole core refuelling mode with long operation cycle; this is much lower than the value of 55 MW·d/kg U proven for the VVER-1000 reactor. The margin in fuel burn-up provides an option for operation cycle extension by a factor of 1.3–1.6, when uranium enrichment above the licensed 5% value becomes possible.

The improved fuel with a reduced fuel element cladding thickness and an enlarged fuel pellet diameter currently developed for the VVER reactors will also extend core life of the VBER-150.

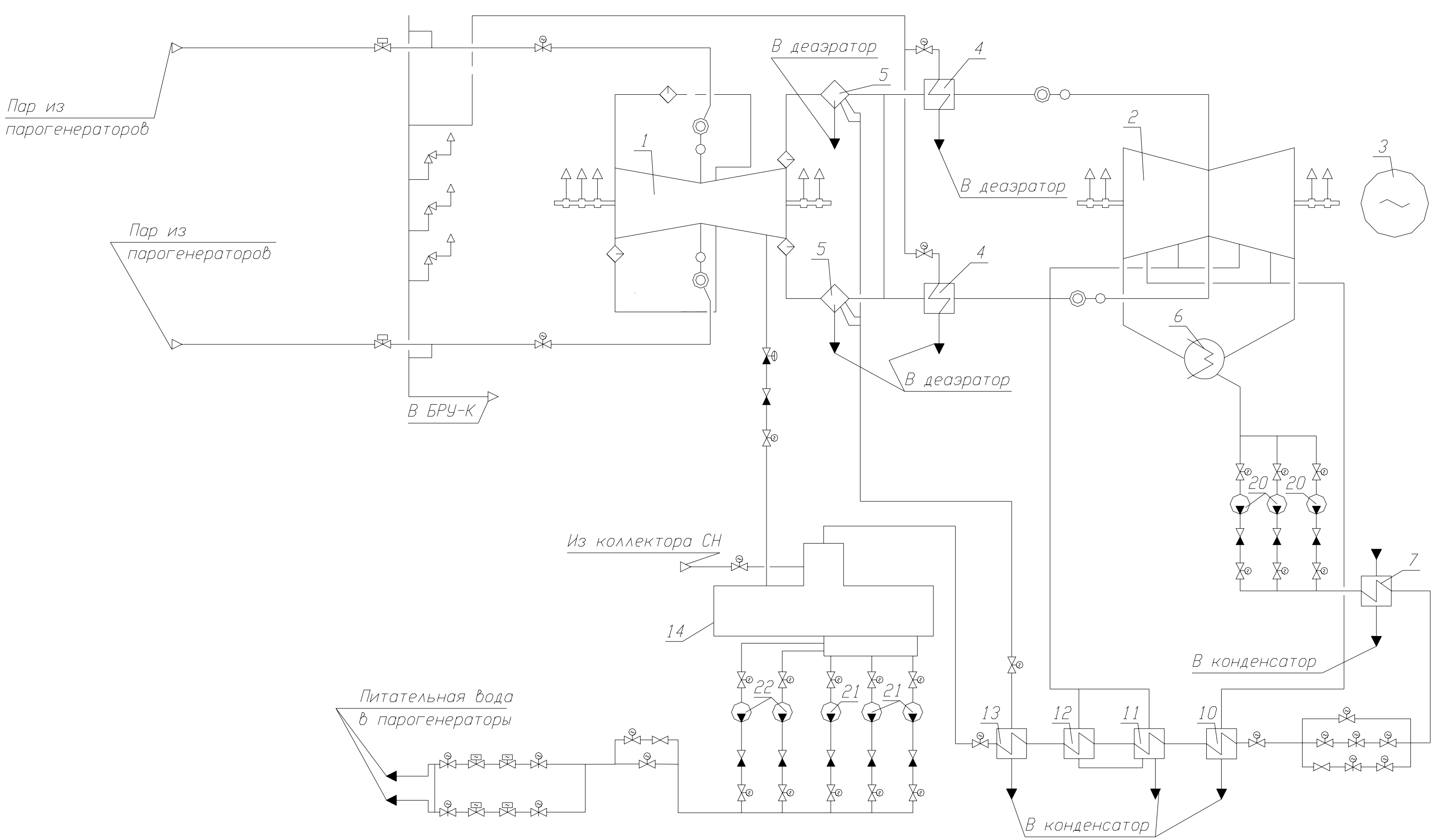
Using the above mentioned options, it might be possible to extend the operation cycle of the VBER-150 up to 10–12 years.

The fuel cycle options for the VBER-150 are the same as for the VBER-300; they include a once-through uranium fuel cycle (basic option), a uranium-thorium once-through fuel cycle to reduce specific plutonium production (Radkowsky Thorium Fuel — RTF — cycle), and a closed fuel cycle with MOX fuel, for details see [IV-1].



1 – Reactor; 2 – Steam generator; 3 – Main circulation pump; 4 – Recuperator of purification and aftercooling system; 5 – After-cooler of purification and aftercooling system; 6–9 – Filters; 10 – Pump; 11 – Pressurizer; 12 – Third-circuit filter; 13 – Expansion vessel; 14 – Third/fourth-circuit heat exchanger; 15 – Third circuit pump; 16 – Emergency heat removal system (EHRS) tank; 17 – EHRS reservoir; 18 – Protective shell (containment); 19,20 – Hydraulic accumulators; 21 – Reactor cavity; 22 – Feed valve; 23 – Storage tank of emergency boron injection; 24 – Make-up system tank; 25 – Make-up pump; 26 – Recirculation system pump; 27 – Hydraulic accumulator; 28 – Heat exchanger of containment pressure suppression system; 29 – Condesate tank; 30 – Heat exchanger of recirculation system

FIG. IV-1. Simplified schematic diagram of the VER-150 primary circuit and safety systems.



1 – High-pressure cylinder; 2 – Low-pressure cylinder; 3 – Generator; 4 – Steam super-heater; 5 – Separator; 6 – Turbine condenser; 7 – Steam condenser; 10 – 13 – Low-pressure heater; 14 - Deaerator; 20 – Condensate pump; 21 – Electric feed pump; 22 – Auxiliary feed pump

FIG. IV-2. Basic diagram of the VBER-150 secondary circuit.

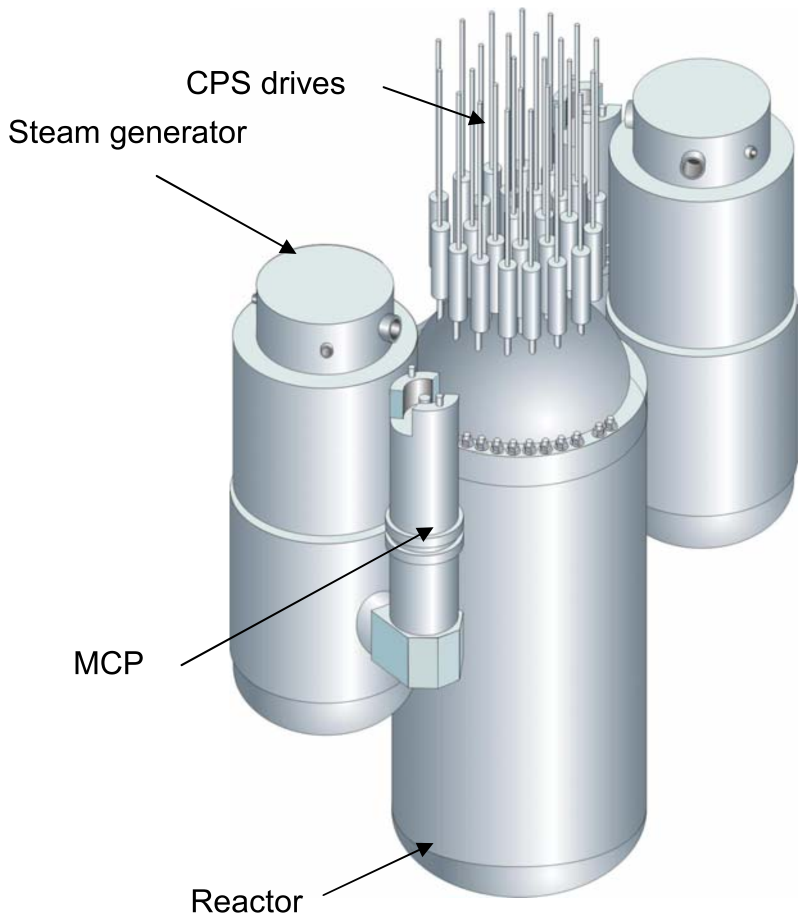


FIG. IV-3. General view of VBER-150 reactor.

TABLE IV-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF THE VBER-150

PARAMETER	VALUE
Major design characteristics	
Rated power, MW — Thermal; — Thermal power at continuous core operation without refuelling; — Electric; — Electric power at continuous core operation without refuelling	 440 350 150 110
Operation mode	Base load; it is possible to realize load follow mode to follow power changes during the day or a dispatcher mode with keeping the frequency
Availability factor	0.85–0.9

PARAMETER	VALUE
Fuel	
Fuel type	Pellets of sintered uranium dioxide
Fuel element	Fuel pin similar in design to a standard fuel element of the VVER-1000 reactor
Fuel assembly	Ductless Advanced Fuel Assembly (AFA) with a skeleton structure
Fuel enrichment	Not more than 5% by weight
Coolant	H_2O
Moderator	H_2O
Core	
Number of fuel assemblies	85
Diameter, mm	2420
Height, mm	1500; 2200 (for a whole-core refuelling mode)
Reactor vessel	
Overall height, mm	14 750
Diameter, mm	10
Operating mass, t	590
Diameter of reactor vessel shell, inner/outer, mm	3300/3700
Structural materials	
Core:	
Fuel element cladding	Zirconium alloy
Fuel assembly structural elements	Zirconium alloy
Vessel system:	
Reactor vessel	Heat-resistant pearlitic steel with anticorrosive facing
Steam generator vessel	The same
Vessel of the hydraulic chamber of circulating pump	The same
Steam generator pipe system	Titanium alloy
Reactor internals	Stainless steel 08Cr18Ni10Ti
Heat removal system	
Cycle type	Steam-turbine cycle with slightly superheated steam
Number of circuits	2
Number of loops	2

PARAMETER	VALUE
CORE NEUTRONICS FOR A WHOLE-CORE REFUELLING MODE	
Reactivity coefficients	
Reactivity coefficient on coolant temperature (taking into account coolant density changes), 1/C	-47×10^{-5}
Reactivity coefficient on coolant density (without taking into account coolant temperature), $1/(g/cm^3)$	0.21
Reactivity coefficient on fuel temperature, $1/^{o}C$	-1.8×10^{-5}
Boron reactivity coefficient, %/(g/kg)	–1.2
Reactivity effect at full core drainage, %	–60
Peaking factors	
Maximum peaking factor for fuel assembly	1.36
Maximum peaking factor for the core	1.58
Measures to reduce power peaking	In conventional refuelling mode: fuel shuffling, compensation of reactivity by liquid boron In operation without on-site refuelling: fuel enrichment profiling, burnable poison profiling
Reactivity control, emergency protection	
Compensation of reactivity margin	Fuel elements with gadolinium oxide integrated in fuel pellets; boron acid solution
Compensation of temperature and power effects of reactivity, reactivity margin for core poisoning by xenon-135 and samarium-149, operating margins for reactivity changes under reactor power changes and for core sub-criticality in the cold unpoisoned state	Electromechanical system of CPS control rods: clusters of control rods (bundles of 18 absorber rods joined by a common traverse and moving inside fuel assembly guide tubes) for reactivity compensation. Separate drive for each cluster. Primary circuit make-up and soluble poison system.
Emergency protection	All CPS control rods (48) are inserted into the core by gravity when their drives are de-energized by a protection system signal. System of emergency injection of boron acid solution

PARAMETER	VALUE
THERMAL-HYDRAULIC CHARACTERISTICS (if different, the values for whole-core refuelling mode are given in brackets)	
Primary circuit	
Circulation type	Forced circulation using canned MCP with a fly-wheel
Primary circuit coolant flow rate, t/h	7265 (710)
Coolant temperature at core inlet, °C	292 (291)
Coolant temperature at core outlet, °C	330 (322)
Coolant velocity in the core, m/s	1.3
Primary circuit coolant pressure, MPa	15.7
Maximum fuel temperature, °C	900
Average fuel temperature in the core, °C	550
Maximum temperature of fuel element cladding, °C	352
Average temperature of fuel element cladding, °C	340
Maximum acceptable fuel temperature, °C	2500
Maximum acceptable temperature of fuel element cladding, °C	700
Minimum margin to heat exchange crisis	1.44
Secondary circuit	
Steam pressure after the steam generator, MPa	6.38
Steam output, t/h	755 (600)
Steam temperature at steam generator outlet, °C	305
Feedwater temperature, °C	185
OPERATION CYCLE CHARACTERISTICS (see Table IV-2)	
Design service lifetime	
Vessel system, years	60
Steam generator pipe system, years	25–30
Main circulating pump	25–30
Economic characteristics	
Construction cost of a floating NPP, US$ million	180
Specific capital investments for construction, US$/kW(e)	1636
Fuel load cost, US$ thousand	~30
Busbar cost of generated electricity (condensation mode), cent/kW h	2.5
Payback period (from commencement of operation), years	9–10

TABLE IV-2. CHARACTERISTICS OF VBER-150 IN DIFFERENT OPERATION MODES

MODE OF OPERATION	PARTIAL REFUELLINGS	WHOLE-CORE REFUELLING
Thermal power, MW	440	350
Core height, mm	1500	2200
Number of fuel assemblies in a make-up set	15	85
Refuelling repetition factor	5.66	1
Uranium weight in a make-up set, t	2.85	23.3
Uranium-235 weight in a make-up set, kg	142	
Average mass fraction of uranium-235 in a make-up set, weight %	4.95	4.7
Operation cycle between refuellings, effective full power days	320	2 083*
Specific consumption of natural uranium, g/(MW·day)	213	339
Burn-up of discharged fuel, (MW·day)/kg U: — Average for a fuel assembly; — Maximum for a fuel assembly; — Maximum for a fuel element	 50.0 53.0 57.5	 31.3 41.6

* Corresponds to more than 7 years with a load factor of 0.8.

IV-1.6. Technical features and technological approaches that are definitive for VBER-100 performance in particular areas

IV-1.6.1 Economics and maintainability

The design features contributing to an enhancement of the economic performance and competitiveness of the reactor installation and floating power unit as a whole are as follows:

- A compact modular layout of the primary circuit main equipment, resulting in a reduction of the metal intensity in reactor installation equipment and accordingly, a reduction in dimensions of the reactor compartment;
- An increase of the reactor installation service life up to 60 years;
- An increase in plant efficiency due to the combined use of the installed capacity for electric power generation and seawater desalination;
- The possibility to locate the floating NPP in immediate proximity to the consumer with corresponding minimization of the outlay for electric power and desalinated water supply;
- The use of industrial production processes for manufacturing of the floating NPP; the NPP is assembled under shipyard conditions and delivered to the customer, tested and completely ready for operation;
- The minimum scope and cost of capital construction to arrange a floating plant location in a water area;
- The lack of need to create transportation links and energy communications and preparatory infrastructure required for land-based NPP construction;

- Flexibility in site selection for the floating NPP, the possibility of mooring in any coastal region of the world independent of seismicity;
- A possibility to relocate the floating NPPs to other regions;
- A considerable reduction in the construction period (down to 4 years) and, consequently, a shorter repayment period of a credit for the construction;
- The simplification of requirements for safety systems due to expanded use of passive safety design options in plant design;
- The adoption of fuel assemblies and structures of proven design based on the technologies of marine propulsion reactors, the VVER-1000, the AST-500, and the KLT-40S reactors;
- A reduction in the volume of solid and liquid radioactive waste due to the use of leak-tight equipment and systems and the increases in the service life of the main replaceable equipment (steam generator pipe systems, MCP replaceable elements, etc.), resulting in a reduced maintenance costs;
- A long refuelling interval with the refuelling, radioactive waste management and repairs being provided at special maintenance centres; elimination of on-site spent fuel storage; the absence of on-site refuelling, radioactive waste management and repairs simplifies operation and contributes to a reduction in the operation and maintenance costs;
- The refuelling and maintenance costs could be minimized by using the infrastructure of nuclear ship maintenance centres available in the Russian Federation; the requirements for local labour skills in developing countries could be reduced also;
- The concept of a floating NPP makes it easy to realize a “green lawn” concept on the site of the floating NPP operation or if necessary, to replace the exhausted floating plant with a new one, contributing to a reduction of the decommissioning costs.

IV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Enhanced fuel utilization efficiency and the reduction of natural uranium consumption are affected by the following engineering solutions:

- Improvements in the nuclear fuel and fuel cycle of the VVER-1000 reactors; realization of a closed nuclear fuel cycle;
- An increase in the fuel burn-up, possible due the geometric stability and improved operation reliability of the skeleton-design fuel assemblies.

A reduction in the amount of radioactive wastes for the VBER-150 based floating NPP results from the following design and conceptual features:

- The use of a standard pressurized primary circuit proven in operation for shipboard reactor plants;
- The use of a closed system of primary coolant purification;
- The use of non-waste technologies for coolant treatment;
- The use of the state-of-the-art low waste technologies for reprocessing of the radioactive wastes;
- Performance of the refuelling under controlled conditions in special maintenance centres.

Radiation safety of a nuclear power plant with the VBER-150 reactor installation meets the requirements enforced in the Russian Federation for limiting the irradiation impacts on personnel, population and the environment during operation, including the abnormal operation occurrences and accidents and a severe accident with fuel damage.

The VBER-150 design provides for a set of technical features and measures to minimize the possible level of personnel and population irradiation; the most important of them are as follows:

- Effective biological shielding;
- A closed system of primary coolant purification and boron removal that excludes leakages of the radioactive medium of the primary circuit from entering the atmosphere during plant operation;
- The use of intermediate loops of cooling water;
- A protective shell with shared roles for protection against natural and human induced external impacts and resistance to the internal accident impacts;
- Strict measures of radiation control;
- The division of plant production area into two zones: a zone of controlled access and a zone of free access;
- Establishment of a sanitary and protection area and a radiation-control area near the NPP.

IV-1.6.3. Safety and reliability

Safety concept and design philosophy

The safety design concept of a power unit with the VBER-150 was selected via the application of a system approach integrating the experience and recent achievements in safety of nuclear power plants and shipboard reactors; it meets the requirements for plant location near populated areas and incorporates an enhanced resistance to possible acts of sabotage.

Safety-related technical features of the VBER-150 correspond to worldwide trends followed by many designers of advanced nuclear power plants, such as:

- A priority of features preventing accident occurrence; design simplification;
- Incorporation of inherent and passive safety features (a self-protection principle);
- Application of the defence-in-depth strategy;
- The use of passive safety systems;
- Provision of an enhanced resistance to external impacts (including human actions of malevolent character);
- Incorporation of features limiting the consequences of severe accidents.

Active and passive systems and inherent safety features

The emphasis in the VBER-150 design is on incorporation of the inherent safety features to ensure passive reactor shutdown, to limit pressure, temperature, coolant heating rate and energy release in accidents, to reduce the scope of failures leading to depressurization of the primary circuit, to reduce the outflow rate, and to maintain the reactor vessel integrity in severe accidents.

The VBER-150 incorporates the following inherent and passive safety features:

- Negative reactivity coefficients on the fuel and coolant temperature, on the specific volume of coolant; negative steam density and power (integral) coefficients of reactivity;
- A decreased core power density compared with shipboard reactors and VVER-1000 type reactors (less than 50 kW/l);
- Stable natural circulation in all circuits, ensuring passive heat removal from the shutdown reactor;
- The primary circuit pipelines being connected to the "hot" parts of the circuit with nozzles located on the reactor vessel above the core, which limits the outflow in loss of coolant accidents and facilitates reduced requirements to flow characteristics of the emergency core cooling system (ECCS);
- The use of short and small-diameter nozzles to connect the main equipment units and elimination of lengthy large-diameter pipelines in the primary circuit, to exclude loss of coolant accidents initiated by large and medium-breaks in the primary circuit;
- The use of canned main circulation pumps;
- The use of once through steam generators, which helps limit the increase of heat removal capacity of the secondary circuit (the primary circuit cooling) under steam line breaks.

The VBER-150 incorporates the following main safety systems (see Fig. IV-1):

- The emergency reactor shutdown systems;
- The emergency heat removal systems;
- The emergency core cooling systems;
- Emergency localization systems, including a double protective shell (containment) and lock valves on the primary circuit auxiliary systems and adjoining systems;
- The reactor vessel cooling systems.

The reliability of safety systems is ensured by the following principal engineering solutions:

- Passive functioning of the systems without exceeding prescribed design limits over the whole range of operation including loss of coolant accidents (LOCAs) and loss of all alternate current (AC) sources during not less than 24 hours;
- Redundancy and diversity of reactor shutdown, core cooling and residual heat removal systems;
- Localization of the release of radioactive products provided by the use of a double protective shell and passive systems and redundant fast-acting valves;
- Separation of safety systems channels to exclude common cause failures; the incorporation of safe failure principle in the component design;
- Redundancy and diversity of control systems achieved through the use of self-actuated devices;
- The use of diagnostic devices and periodic inspections to exclude safety system component failures not detected during operation.

The application of a two-channel scheme for safety systems supports meeting the regulatory requirements for safety both deterministically and probabilistically, due to the redundancy of the elements inside the channels and safety systems redundancy.

Control safety systems provide automated control and remote control of the equipment of safety systems from independent control panels (the main control room and a standby control room).

To protect personnel and population against irradiation in the design and beyond design basis accidents, the following engineering solutions are incorporated in the design of the protective shell (primary containment):

- A system of passive heat removal from the protective shell limiting pressure in the protective shell in LOCAs;
- A system for retaining fuel in the reactor vessel in accidents with severe core damage;
- Sharing of the functions of protection against external natural and human-induced impacts and resistance to internal emergency impacts;
- A system of iodine and aerosol purification of the air of the inter-shell space (the space between the protective and guard shells) to prevent leakages of radioactivity from the protective shell in accidents with pressure increase.

The reactor installation has a pressurized cylinder-shaped steel protective shell (Fig. IV-4). It is a composite welded construction of 12 m inner diameter, and 15.9 m height; the internal volume is ~1820 m^3.

The protective shell can withstand overpressure of 1.0 MPa. Along the outer perimeter there is a stationary biological shielding, which consists of steel plates and serpentinite concrete.

The internal volume of the shell is divided by biological shielding blocks into two leak-tight rooms (a control room and the reactor compartment). A vacuum is retained in both rooms under normal operation (the residual pressure is 0.0978–0.0977 MPa).

The reactor compartment accommodating the VBER-150 reactor installation is shielded from outside by a protective guard consisting of the multi-layer ceilings of a superstructure roof, walls of the stern and bow machine rooms and board compartments of the floating NPP superstructure. Altogether, these structures constitute the external reactor compartment protection, which can withstand external impacts including aircraft crash.

Design basis accidents and beyond design basis accidents

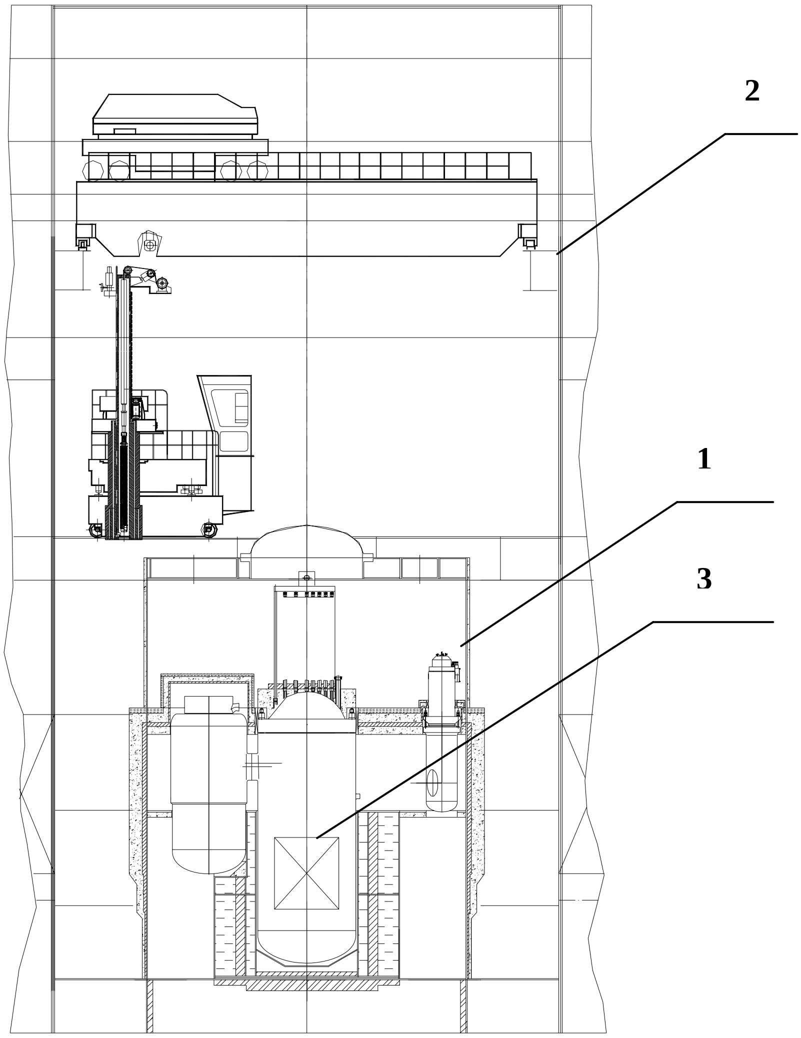
Both, deterministic and probabilistic methods are being applied to validate safety of the VBER-150.

Deterministic safety analysis is performed with the set of calculation codes developed by OKBM and proven in calculations of stationary and non-stationary modes of plant operation. The codes take into account specific features of the plant design, circulation circuit, steam generator, aftercooling systems, control system, etc., and are based on experimentally proven methods of calculation and correlations and have a long-term experience of application.

The codes were verified on the results of separate effect tests and integral experiments performed in thermo-physical test facilities, prototypes of modular reactors, as well as on the experimental data available from previous tests and operation.

All basic codes used in safety analysis are certified by the Council on Software Certification of the Russian regulatory authority.

Along with the design basis accidents, a wide spectrum of beyond design basis accidents including safety system failures combined with certain initiating events and/or human errors are analyzed.



1 – Protective shell (primary containment); 2 – Guard shell; 3 – Reactor installation

FIG. IV-4. Protective shell of VBER-150.

The list of beyond design basis accidents includes:

- Complete de-energization of a floating NPP with failure of the control safety system or channels of the emergency heat removal system;
- A break in the primary circuit pipeline with the complete de-energization of the plant (see Fig. IV-5) or failure of the core cooling systems;
- Transients with a failure of control safety systems.

Probabilistic safety analysis supports deterministic analysis regarding the elimination of ‘weak points’ of the design and effectiveness assessment of the decisions on improvement of safety features, i.e., in total, it contributes to the realization of a balanced defence in depth strategy and demonstrates that the regulatory values of probabilistic indices are achieved.

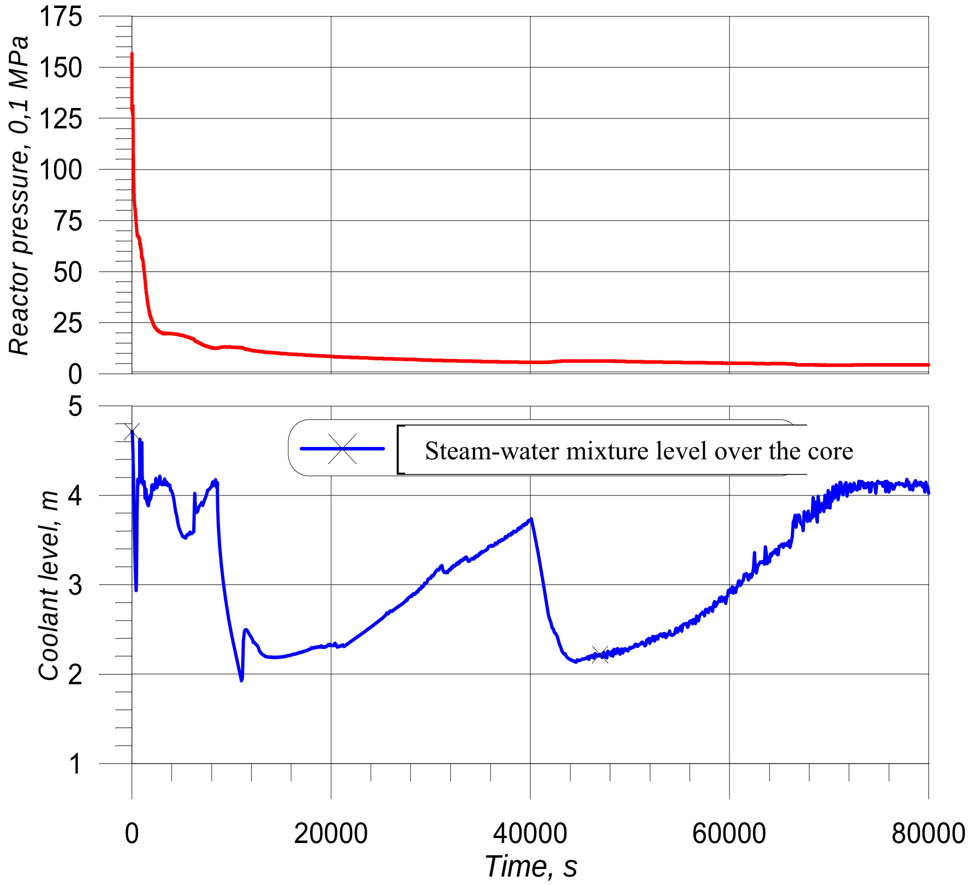


FIG. IV-5. Break of the primary circuit pipeline with complete de-energization of the plant.

Provisions for safety under natural and human-induced external impacts

Structures, systems, and components of a nuclear cogeneration plant with the VBER-150 reactor installation are developed with consideration of natural and human-induced external impacts and provide for the possibility of plant location at any suitable site meeting the regulatory requirements.

Earthquakes, wind loads, low and high temperatures, aircraft crashes, shock waves and other impacts referred to as natural and human-induced external events impacts are carefully taken into consideration starting from the original safety design concept.

The following features and measures are provided for to secure high resistance of a floating NPP to external impacts:

- Water area protected against the unauthorized access of floating objects;
- The designs of vessel structures and seawater intake systems that meet the requirements of the sea shipping register of Russia for floatability.

Measures planned in response to severe accidents

Advanced safety characteristics of the VBER-150 determine low probability of a core melt accident. Nevertheless, in accordance with regulatory requirements and taking into account the design experience of similar domestic and foreign next-generation power plants, the approaches to ensuring a high level of radiation safety in postulated severe accidents are considered in the VBER-150 design.

According to the regulatory requirements, not exceeding the allowable emergency irradiation doses to population should be ensured in accidents with severe core damage and the necessity of population evacuation should be excluded [IV-5, IV-6]. These regulatory requirements conform to the best worldwide practices and IAEA recommendations [IV-7, IV-8].

The standard approach to cope with a severe accident is applied, based on a combination of design features and accident management of two types:

- Directed toward prevention of core damages (decrease of core damage probability);
- Directed toward limitation of the consequences of severe accidents.

In-vessel retention of corium is considered a priority to limit severe accident consequences in the VBER-150, since they are to a large extent determined by the reactor vessel failure and the resulting initiation of additional loads on the containment under a release of corium.

The VBER-150 is characterized by:

- Decreased core power density, compared to large reactors (VVER, PWR);
- A relatively low level of residual heat at the stage of core degradation and corium displacement to the bottom;
- No penetrations of the reactor vessel bottom, which are potential “trouble spots” in the interaction of corium and reactor vessel;
- A smooth outer surface of the reactor vessel bottom providing more favourable conditions for steam evacuation in vessel cooling by boiling water.

The VBER-150 design incorporates a special system of emergency vessel cooling to secure in-vessel retention of corium in severe accidents; this system functions in a passive mode.

The reactor vessel is cooled by boiling water, the generated steam is condensed in the protective shell, and the generated condensate again supplied to cool the reactor vessel through the system of condensate gathering tanks.

The design calculations indicate that the problem of in-vessel retention of corium could be successfully solved for the VBER-150.

The calculations also indicate that in severe accidents, allowable emergency doses of population irradiation are not exceeded and measures for obligatory population evacuation are unnecessary. The boundary of the area of protection measures is not more than 1 km from the NPP. These results meet in full the safety requirements for next generation reactors by the NRC, US industry, the NPI consortium (EPR reactor) and the IAEA recommendations on safety of advanced reactors [IV-7].

IV-1.6.4. Proliferation resistance

The following main design features support an enhanced proliferation resistance of the VBER-150:

- The operation without on-site refuelling, which complicates unauthorized access to fuel; such operation assumes that all operations with fuel are accomplished at special maintenance centres;
- The use of uranium dioxide fuel with the enrichment not more than 5% by weight;
- The use of a standard fuel cycle of VVER reactors with the available infrastructure and mechanisms of protection against proliferation.

IV-1.6.5. Technical features and technological approaches used to facilitate physical protection of VBER-150

Technical features and technological approaches used to facilitate physical protection of the VBER-150 reactor installation are essentially the same as those for the VBER-300 reactor installation [IV-1]. In addition to them, the VBER-150 reactor installation provides for no fresh or spent nuclear fuel being stored at a floating NPP during the whole time of its operation at a site and transportation to a special maintenance centre.

IV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of VBER-150

Non-technical factors and arrangements that could facilitate effective development and deployment of the VBER-150 are essentially the same as those for the VBER-300 reactor installation [IV-1]. In addition to them, a refuelling interval of 7–8 and, later on, 10–12 years could, perhaps, add certain assurances to those users that would prefer to forego the development of an indigenous fuel cycle.

IV-1.8. List of enabling technologies relevant to VBER-150 and status of their development

The main technologies incorporated in the design of a floating NPP with VBER-150 reactor installations are listed in Table IV-3.

A more detailed list of the enabling technologies relevant for the VBER-150 reactor installation is given below:

- Modular layout of the main equipment, including the reactor, steam generator, and main circulating pumps; vessels of the equipment are directly joined by welding using short nozzles without lengthy pipelines; coaxial scheme of primary coolant circulation through connection nozzles of the main circulation path;
- Basic technologies of a vessel-type pressurized water reactor, well proven through worldwide operation experience;
- Pressurized primary circuits using welded joints, canned packless pumps and leak-tight bellows valves;
- Once-through coil-type steam-generators;
- Canned pumps, used as MCPs in shipboard plants, with a necessary increase in power, head, output and trip;
- A core with the VVER type fuel design with a decreased linear heat rate, compatible with the infrastructure of the existing VVER nuclear fuel cycle;
- Technologies of a lifetime core operation without on-site refuelling;
- The vessel with a service life of 60 years; based on metallurgical, press forging and machine-assembly processes proven in the production of vessels for shipboard reactor installations;
- Highly reliable systems of the marine nuclear reactor industry and operating NPPs;
- Proven technologies of equipment mounting, repair and replacement, including those for diagnostic devices and systems to monitor the equipment state;
- Technologies to ensure minimum radiological impacts on personnel, population and the environment; specifically, the technologies to limit consequences of the severe accidents with core degradation to a plant boundary.

TABLE IV-3. LIST OF BASIC ENABLING TECHNOLOGIES FOR VBER-150

TECHNOLOGIES	STATUS
Technologies of modular pressurized water reactors for Russian nuclear ships	Well established; the operating experience of shipboard multi-purpose reactors exceeds 6000 reactor-years.
Technologies of the VVER-1000 power reactors (core, CPS drive)	Well established; twenty VVER type reactors are currently in operation
Technologies of the AST-500 nuclear cogeneration plant (safety systems and safety design)	A safety review was performed by the IAEA during plant construction
Technologies of the KLT-40S ice-breaker reactor for the pilot floating NPP (floating NPP design)	Detailed designs of the reactor installation and floating power unit have been developed; a license for floating NPP construction has been obtained.

The technologies for production of certain structures, systems and components of the VBER-150 reactor installation are listed below; these are the technologies already mastered in commercial production:

- Welding technologies for vessel systems;
- Fabrication techniques for steam generator pipe systems of titanium alloys;
- Fabrication and assembly technologies for coaxial type reactor internals that provide the main path for the coolant circulation;
- Fabrication technologies for canned MCPs;
- Fabrication technologies for the AFA-type fuel assemblies with a rigid skeleton structure, used in the cores of VVER-1000 reactors;
- Fabrication technologies to ensure high corrosion and radiation resistance of structural materials;
- Fabrication technologies for elements of normal operation and safety system, ensuring high reliability of the self-actuated devices, pressurizers, tanks, heat exchangers, pumps, and filters.

IV-1.9. Status of R&D and planned schedule

Design development for the VBER-150 is being performed on the initiative of several Nizhny Novgorod region companies (the Russian Federation) with the use of a unique experience in the design, construction and operation of marine nuclear reactors. Main participants of the VBER-150 design development are listed in Table IV-4.

TABLE IV-4. MAIN PARTICIPANTS OF VBER-150 DESIGN DEVELOPMENT

COMPANY	AREA OF RESPONSIBILITY
OKB Mechanical Engineering (OKBM), Nizhny Novgorod	Chief designer of the reactor plant
RRC “Kurchatov Institute”, Moscow	Scientific leader of the design
Joint Stock Company «Lazurit», Nizhny Novgorod	General designer of the floating nuclear power plant

Design development of a floating NPP with the VBER-150 reactor installation is financed by the companies involved in the project.

As the VBER-150 is a two-loop modification of the VBER-300 reactor installation, the results of the latter project are being used in the design of the former. The Rosatom supports the VBER-300 design development within the framework of a national programme “Application of Nuclear Power Sources for District Heating”. The details of the VBER-300 design status are provided in [IV-1].

The project of a floating NPP with the VBER-150 is at a conceptual design stage; the estimated periods for development and deployment of this plant are given in Table IV-5.

TABLE IV-5. PROJECTED TERMS FOR REALIZATION OF THE VBER-150 PROJECT

STAGE	PERIOD
Detailed design development, including licensing	3 years
Floating plant construction (including licensing and main engineering development)	4 years

IV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The VBER-150 design uses to the maximum extent the results of design development, validation, testing and operation of many Russian shipboard reactors and also borrows from the design and operating experience of the VVER and PWR reactors. Therefore, realization of the VBER-150 project would not require wide-scope engineering developments.

The most important areas of further research and development (R&D) are: aerodynamic tests of reactor setting (flow path), development of fabrication techniques for main components of the SG; development, fabrication and testing of the MCP prototype.

To realize a core design with extended fuel lifetime, R&D is necessary on lifetime characteristics of fuel elements and fuel assemblies under low power density and extended operation interval (compared with VVER type reactors, the power density has been decreased by a factor of 2.5–3, the linear fuel heat rate - by a factor of ~4).

The construction of a pilot floating power unit is needed to master problems related to the floating location of an NPP intended for power supply to the coastal regions.

IV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The NP-300 design developed by TECHNICATOME/AREVA (France) has certain similarities to the VBER-150.

IV-2. Design description and data for VBER-150

IV-2.1. Description of the nuclear systems

Reactor core and fuel design

The VBER-150 core concept is similar to that of the VBER-300 [IV-1].

Uranium dioxide pellets of 7.6 mm diameter are used as a fuel; the uranium enrichment is up to 5% by weight (maximum licensed enrichment).

Ductless advanced fuel assembly (AFA) type fuel assemblies with rigid skeleton structure, developed by OKBM for the VVER-1000 reactor [IV-9], are used in the core of the VBER-150, as well as in the VBER-300 [IV-1]. Main characteristics of the VBER reactor core and fuel assemblies are summarized in Table IV-6, for a variant with lifetime core operation and whole core refuelling.

TABLE IV-6. MAIN CHARACTERISTICS OF THE VBER-150 CORE

Characteristic	Value
Number of fuel assemblies	85
Core diameter, mm	2420
Equivalent diameter of the core, mm	2285
Core height, mm	2200
Core volume, m^3	9.02
Power density, MW/m^3	39
Number of CPS control rod drives	48
Fuel assembly pitch, mm	236
Fuel assembly flat-to-flat size, mm	234
Fuel element pitch in fuel assembly, mm	12.75
Outer/inner diameter of fuel elements and gadolinium-containing fuel elements, mm	9.1/7.73
Fuel cladding material	E110 or E635 alloy
Number of fuel elements and gadolinium-containing fuel elements in the core	26 520
Core heat exchange surface, m^2	1668
Average heat flux from the surface of fuel elements and gadolinium-containing fuel elements, MW/m^2	0.205
Average linear heat rate of fuel elements and gadolinium-containing fuel elements, W/cm	58.2

The results of AFA fuel operation in the VVER-1000 core of the Kalininskaya NPP Unit 1 confirmed its high load-bearing capacity and resistance to deformation [IV-10]. The AFA fuel, proven by six years of successful operation at the Kalininskaya NPP, provides:

- “Soft” conditions of fuel operation; increased thermal margins;
- An extended fuel lifetime;
- An option to use uranium fuel with the enrichment of up to 5% (maximum licensed enrichment);
- The possibility of a load follow operating mode;
- Vibration strength and geometric stability of the fuel assembly during long operating periods, due to the use of a load-bearing skeleton structure.

Figure IV-6 shows the arrangement of fuel assemblies and control rods in the core.

Primary circuit

The primary circuit of the VBER-150 is a closed pressurized system intended for heat removal from the reactor core and heat transfer to the secondary circuit water and steam via the steam generator. The primary circuit includes:

- A two-loop reactor unit;
- A pressurizer system;
- A purification and aftercooling system.

Water containing a boron solution is used as the primary circuit coolant.

Reactor unit

The reactor unit is intended to generate steam of specific parameters. The reactor unit, shown in Fig. IV-7 and Fig. IV-8, includes:

- Vessel system;
- Reactor core;
- Two once-through steam generators;
- Two main circulating pumps;
- CPS drives.

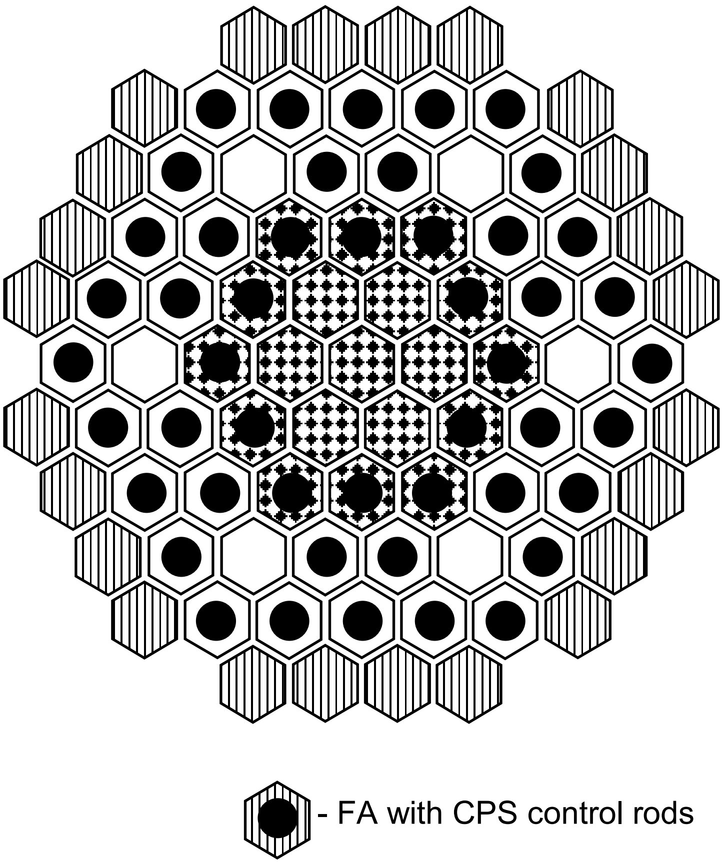
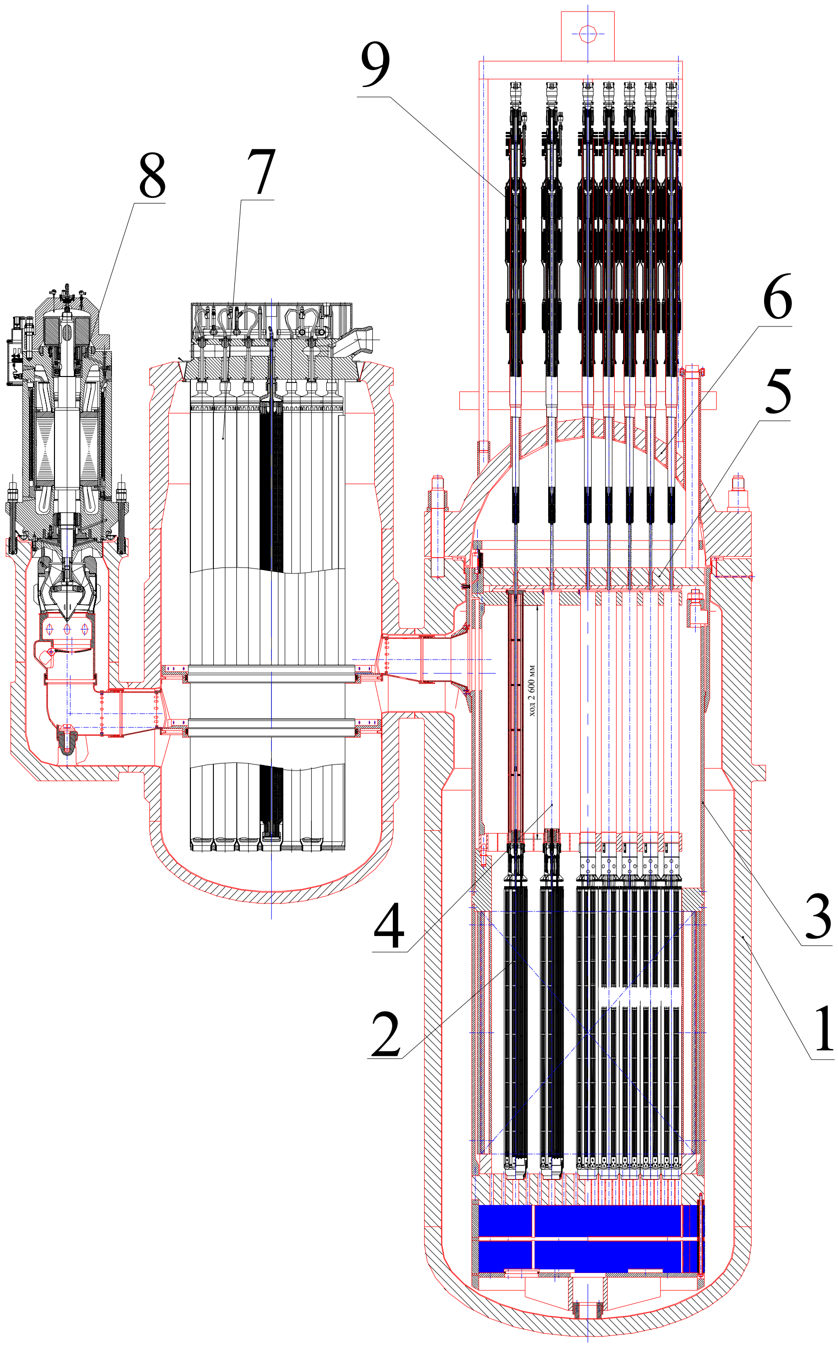
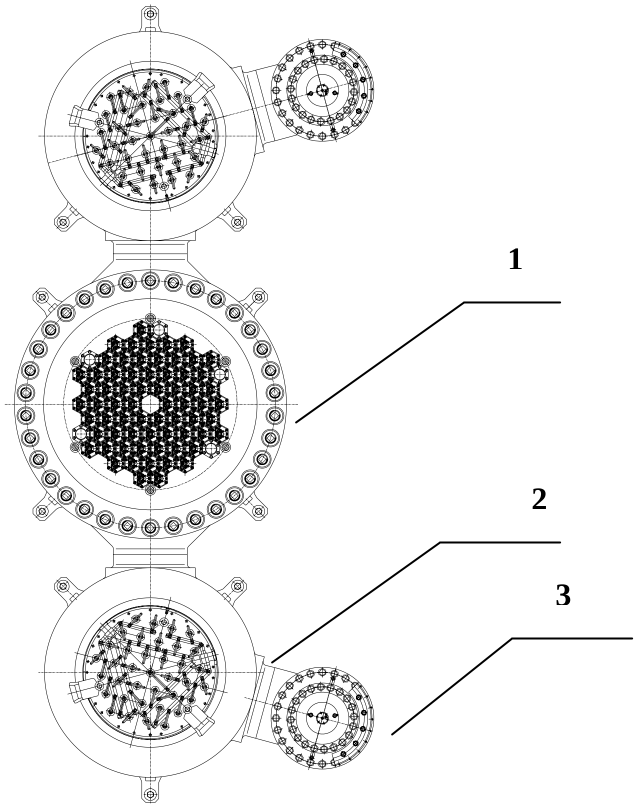


FIG. IV-6. Cross section view of the VBER-150 core.



1 – Vessel system; 2 – Reactor core; 3 – In-vessel cavity; 4 – Tube block; 5 – Probe unit of in-reactor control system; 6 – Upper unit; 7 – Two SGs; 8 – 2 MCPs; 9 – CPS drives.

FIG. IV-7. Vertical view of the VBER-150 reactor unit.



1 – reactor; 2-SG; 3-MCP

FIG. IV-8. Top view of the VBER-150 reactor unit.

The main performance data of the reactor unit are given in Table IV-7.

The vessel system consists of a reactor vessel and two units (steam generator + main circulation pump (MCP)) connected with the reactor vessel by load-bearing nozzles of the coaxial type.

The reactor vessel is intended for arranging an in-vessel unit inside. It is a welded cylindrical vessel with an elliptical bottom, two main nozzles and a flanged part.

The steam generator + MCP unit consists of a steam generator vessel connected with the hydraulic chamber of the MCP by a load-bearing nozzle.

The SGs, MCPs, and CPS drives of the VBER-150 and VBER-300 reactor installations are similar [IV-2, to IV-4]. The design and dimensions of the reactor vessels are also similar (excluding the number of main nozzles – two instead four).

Steam generator

The SG design of the VBER-150 is similar to that of the VBER-300 (see [IV-1]).

The pipe system heat exchange surface consists of 37 unified coil-type steam generating modules (Fig. IV-9) covered by a box-shaped casing. By feed water and steam, the steam generating modules are correspondingly integrated into two independent sections of 18 and 19 modules.

TABLE IV-8. MAIN PERFORMANCE DATA OF THE VBER-150 REACTOR UNIT

CHARACTERISTIC	VALUE	
	Partial refuelling	Entire refuelling
Thermal power, MW	440	350
Primary circuit pressure, MPa	15.7	
Coolant temperature,°C: — At core outlet — At core inlet	 330 292	 322 291
Coolant flow rate, t/h	7265	7100
Steam output, t/h	755	600
Superheated steam after the steam generator: — Pressure, MPa — Temperature, °C	 6.38 305	
Feedwater temperature, °C	185	

Mass characteristics and overall dimensions of the steam-generating unit (SGU) are as the following:

Overall height, mm	14 750
Operating weight of steam generator (without remote pressurizer), t	590
Diameter of reactor vessel shell, inner/outer, mm	3300/3700
SGU diameter, mm	~10 000

Main circulating pump

The MCP design of the VBER-150 is similar to that of the VBER-300 [IV-1].

Pressurizer system

An external pressure compensation system with a two-zone steam pressurizer similar to that of the VBER-300 [IV-1] is used in the VBER-150.

Purification and aftercooling system

The purification and aftercooling system is designed to maintain the primary circuit coolant at a required quality, to decrease boric acid concentration in the primary coolant, to provide normal and emergency reactor aftercooling and to inject chemical reagents to maintain water chemistry of the primary circuit (see Fig. IV-1).

To prevent coolant loss in case of pipeline or system equipment failure, the nozzles connecting the system with the reactor unit are equipped with narrowing inserts.

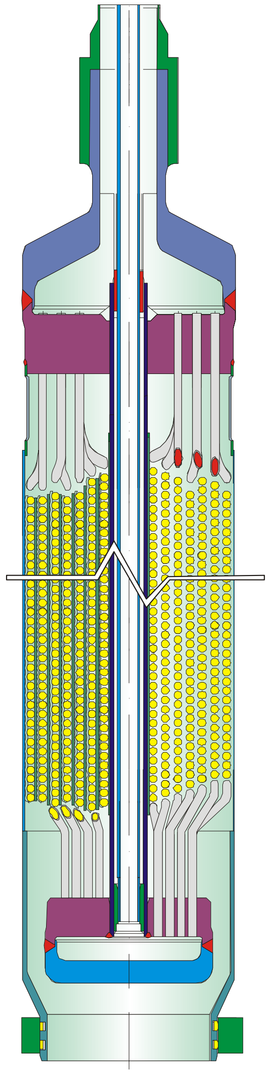


FIG. IV-9. Steam generating module.

Main heat transport system

A schematic of the VBER-150 main heat transport system with specification of heat removal path in normal operation and in accidents is shown in Fig. IV-10.

IV-2.2. Description of the turbine generator plant and systems

The floating nuclear power plant will use a turbine generator plant based on the design developed by the Joint Stock Company "LMZ" and modified to meet shipboard reactor requirements (see Fig. IV-2).

Tentatively, the turbine will have a double-stage structural arrangement, with a high-pressure and a double-flow low-pressure stage.

The main performance data of the turbine generator plant are given in Table IV-9.

TABLE IV-9. PERFORMANCE DATA OF THE TURBINE GENERATOR PLANT

CHARACTERISTIC	VALUE
Live steam pressure before high-pressure stage valves, MPa	6.03
Live steam temperature before high-pressure stage valves, ºC	300
Feedwater temperature, ºC	185
Rated electric power, MW	150
Speed of rotation, rpm	3000
Installed capacity per annum, not less than, hours	8000
Service lifetime, years	60

IV-2.3. Systems for non-electric applications

As it was already mentioned, it is possible to use floating power units with the VBER-150 reactor installations as cogeneration plants for power generation and seawater desalination.

A power and desalination complex (a cogeneration plant) based on a distillation desalinating plant (DDP) includes the reactor installation; the extraction turbine; the intermediate loop; and the desalination plant (see Fig. IV-11). Heat and part of the electric power generated by the floating power unit are used for desalination; the rest of the electric power is supplied to the consumers.

IV-2.4. Plant layout

General philosophy governing plant layout

The floating power unit of a floating NPP is a non-self-propelled autonomous floating structure classified as a harbour ship per the classification adopted in the Sea Shipping Register of Russia.

The two-vessel (catamaran) layout of the floating power unit is used providing separate vessel construction at the shipyard. The reactor plant equipment including auxiliary systems and safety systems will be installed in one vessel; the turbo-generator equipment including corresponding systems will be installed in the other vessel.

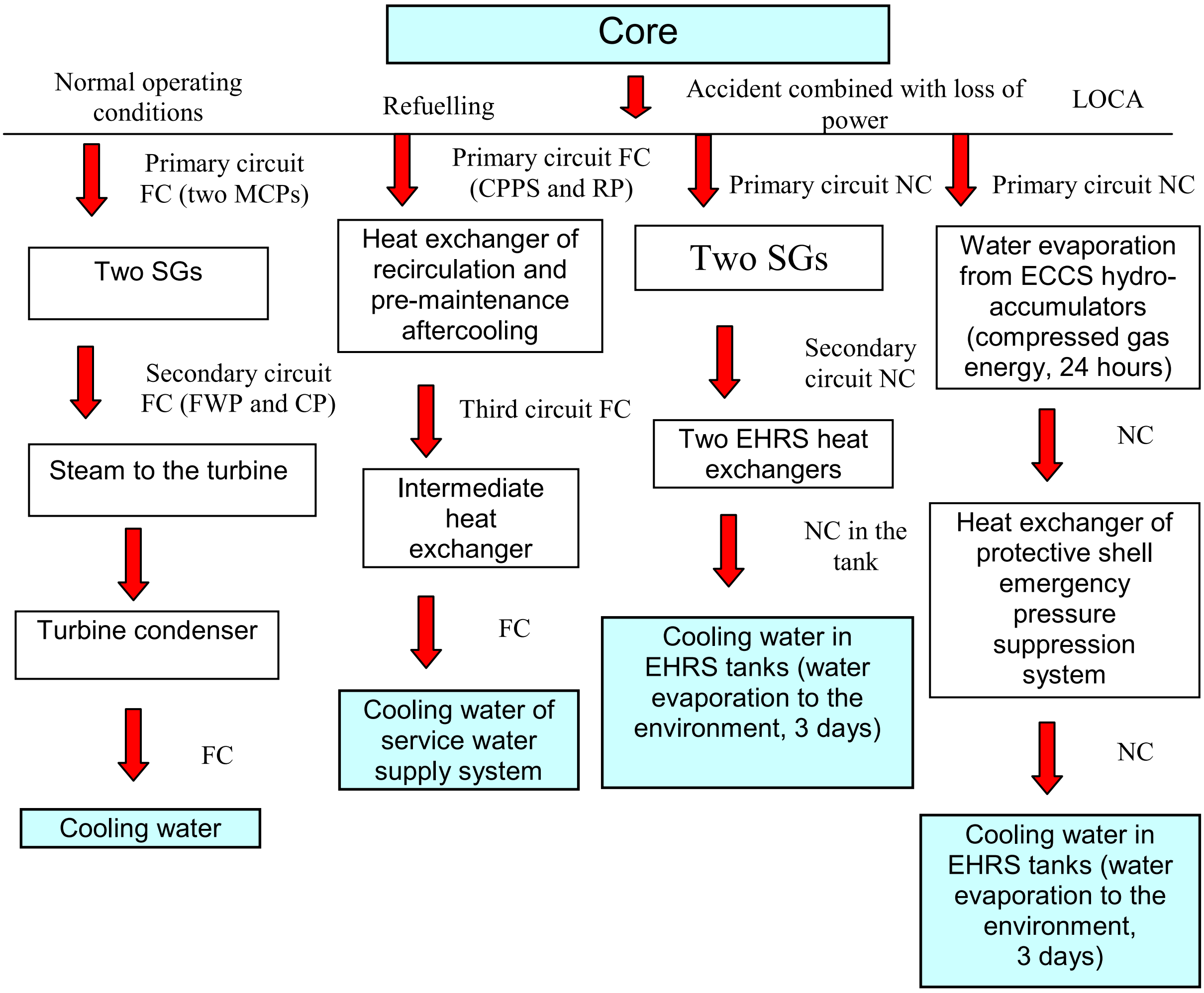
It is possible to construct a floating NPP at the interior shipyards of the Russian Federation and to transport the floating power unit to potential customers, for example, by the Volgodonsky Channel.

Two-vessel layout of the floating NPP allows separately towing each of the vessels, for example, into Sea of Azov to be joined afloat to one structure in water space of a shipbuilding facility or a shipyard located on the coast.

The main plant components (vessels, reactor installation, turbine generator plant, control consoles and systems, electric system, auxiliary and standby power supplies) are mounted, tested and commissioned at an interior shipyard.

Upper structural elements such as superstructure, protective guard, hatch covers, cranes and crane runways, components of the ventilation and air conditioning systems, beacons, antennas, etc., are fitted out on the coast.

The floating NPP design meets the requirements of the Russian sea shipping register rules for the classification and construction of nuclear ships and floating structures.



EHRS	Emergency heat removal system
ECCS	Emergency core cooling system
FC	Forced circulation
NC	Natural circulation
FWP	Feedwater pump
CP	Condensate pump
CPPS	Circulating pump of purification system
RP	Recirculation pump

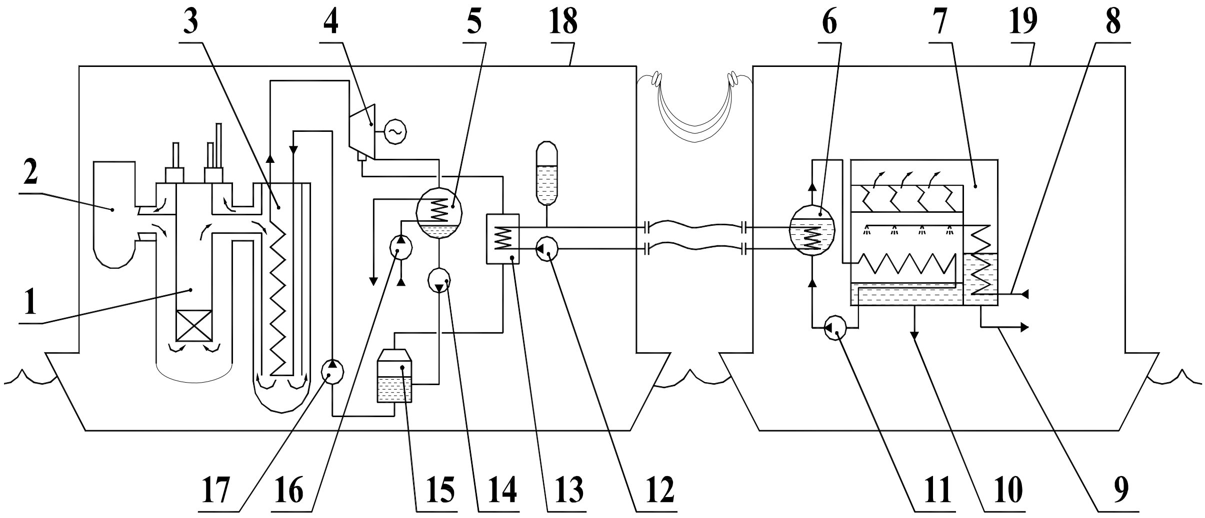
FIG. IV-10. VBER-150 heat removal path in normal operation and in accidents.

The scheme of a floating power unit with the VBER-150 reactor is given in Figure IV-12. The ship dimensions for a floating NPP with a single 150 MW(e) reactor are given in Table IV-10.

Reactor compartment (reactor unit) layout

The reactor unit of a floating NPP incorporates the reactor compartment, in which the VBER-150 is installed, Fig. IV-13.

The reactor installation has a separate leak-tight steel protective shell. The reactor compartment is closed by the protective guard consisting of multi-layer ceilings of a superstructure roof, walls of the stern and bow machine rooms and superstructure board rooms.



1 – Reactor
2 – Primary circuit circulating pump
3 – Steam generator
4 – Turbogenerator
5 – Condenser
6 – SG of desalination plant
7 – Distillation desalinating plant
8 – Seawater inlet
9 – Desalinated water outlet
10 – Brine
11 – Circulating pump
12 – Intermediate loop circulating pump
13 – Intermediate loop heater
14 – Condensate pump
15 – Deaerator
16 – Circulating pump
17 – Circulating pump
18 – Floating power unit
19 – Floating desalination unit with DDP

FIG. IV-11. Diagram of the floating cogeneration plant.

TABLE IV-10. SHIP DIMENSIONS FOR A FLOATING NPP WITH SINGLE REACTOR OF 150 MW(e)

Length, m	105
Width, m	46
Reactor unit width, m	17
Turbo-generator unit width, m	17
Board depth, m	7.6
Reactor unit draught*, m	3.6
Turbo-generator unit draught*, m	2.5
Reactor unit displacement, t	6400
Turbo-generator unit displacement, t	4430
Total displacement, t	~12 000

*When transported separately.

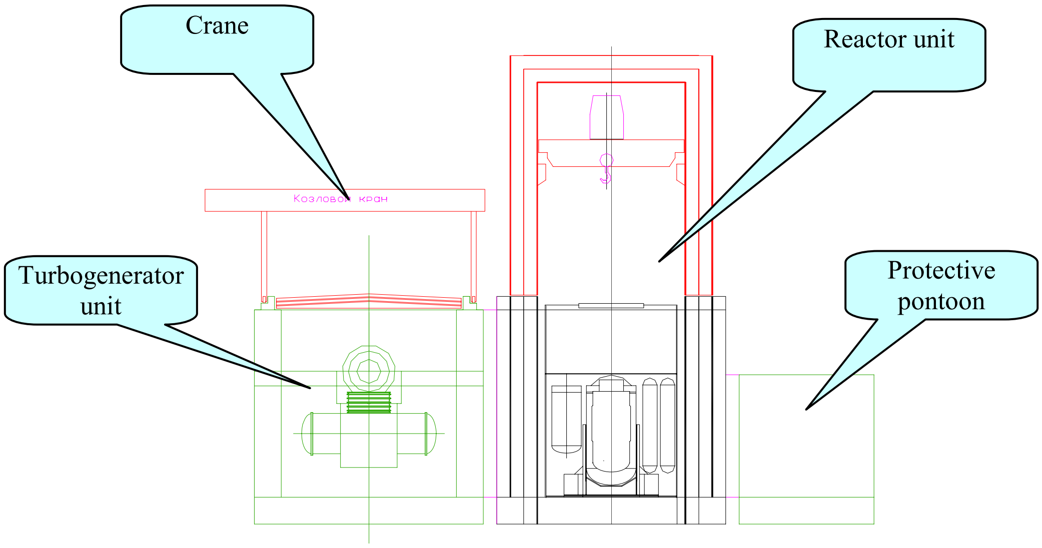


FIG. IV-12. Floating power unit with VBER-150.

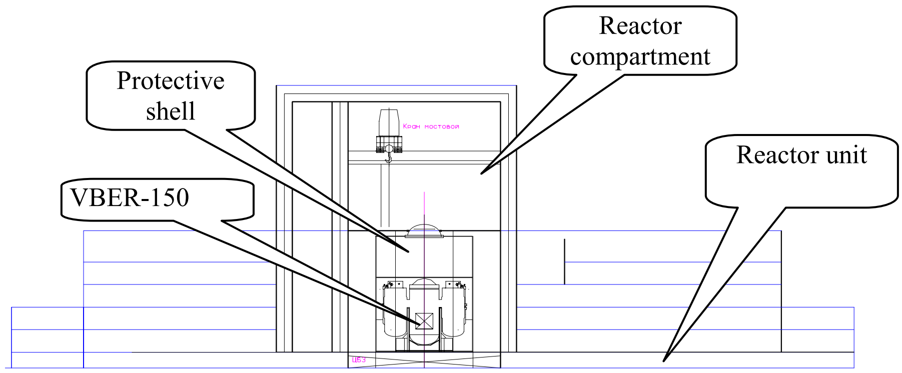


FIG. IV-13. Reactor compartment layout.

These structures constitute the external protection circuit of a reactor compartment capable of withstanding external physical impacts including aircraft crash.

Turbine island (turbo-generator unit) layout

The FNPP has a turbo-generator unit intended for mounting the turbine generator plant and auxiliary systems. The arrangement of the turbine generator plant is longitudinal.

Plant plot

Arrangement of the water space and coastal infrastructures needed for normal operation of the VBER-150 based floating NPP and its protection against external events, are similar to those for floating NPPs with the VBER-300 reactor installations [IV-1].

REFERENCES

[IV-1] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Innovative Small and Medium Sized Reactor Designs 2005: Reactors with Conventional Refuelling Schemes, IAEA-TECDOC-1485, Vienna (February 2006).

[IV-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs 2004, IAEA-TECDOC-1391, Vienna (May 2004).

[IV-3] PONOMAREV-STEPNOI, N., et al., Nuclear power plants with VBER-300 reactor installations, Journal of Russian Nuclear Society, No. 6 (December 2002, in Russian).

[IV-4] PONOMAREV-STEPNOI, N.N., et al., Nuclear power plants with VBER-300 reactor based on the technologies of modular marine reactors, Journal “Conversion in machine-building”, No. 6 (2003, in Russian).

[IV-5] Radiation safety regulations (NRB-99): Hygienic regulations, Moscow, Ministry of Health (Minzdrav) of the Russian Federation (1999, in Russian).

[IV-6] Deployment of NPPs. Major criteria and safety requirements, NP-032-01, Moscow (2002, in Russian).

[IV-7] INTERNATIONAL ATOMIC ENERGY AGENCY, Development of Safety Principles for the Design of Future Nuclear Power Plants, IAEA-TECDOC-801, Vienna (June 1995).

[IV-8] ASMOLOV, V.G., SHAKH, O.YA, Summary of foreign regulations and safety design approaches for NPPs of new generation with large-power reactors, Report of the Institute of Safety Problems of the Russian Research Centre “Kurchatov Institute”, reg. No. 90-12/1-12-98, Moscow (1999, in Russian).

[IV-9] MOLCHANOV V.L., et al., The totals of the creation and further improvement of the fuel assembly of an alternative design for VVER-1000. Electrostal (Proc. of the Int. Conf. “NPPs on the eve of the 21st century”, 8–10 June 2000), p. 412–428 (in Russian).

[IV-10] SAMOILOV, O.B, et al., Operation experience of the TVSA fuel assembly within the units of the Kalinin NPP and optimization of its design 14th annual conference of the Russian Nuclear Society “Scientific support of a safe use of nuclear technologies” (Paper presented at annual conf., 23–27 June 2003, Udomlya, Tverskaya region, Russia, in Russian).

WATER COOLED MODULAR NUCLEAR POWER REACTOR ABV
OKBM,
Russian Federation

V-1. General information, technical features and operating characteristics

V-1.1. Introduction

The ABV reactor installation is a nuclear steam-generating plant with an integral pressurized light water reactor and natural circulation of the primary circuit coolant.

The abbreviation “ABV” in Russian stands for “nuclear, modular, water”. “Modular” means it is possible to assemble a nuclear power plant of large functional factory-made units including the reactor unit and turbine generator plant.

The ABV design was developed by OKB Mechanical Engineering (OKBM, Nizhny Novgorod), a leading Russian organization in the field of shipboard nuclear installation design. The Institute of Physics and Power Engineering (IPPE, Obninsk) was a scientific leader in the ABV project.

Long-term experience in the design, construction and operation of shipboard nuclear installations and the results of previous R&D form the technological basis for the ABV design.

The ABV design was developed using operating experience of VVER type reactors and recent achievements in the field of nuclear power plant safety. The main purpose of the project is the creation of small multi-purpose power sources based on proven marine reactor technologies, providing easy transport to the site, rapid assembly and safe operation.

V-1.2. Applications

The ABV reactor installation is designed as a steam-generating plant to power civil ships and submarines or to act as a power source for land-based, underground, surface-water and underwater nuclear power plants. Possible applications of the ABV plant are the following:

- Power generation;
- Heat and power cogeneration; and
- Seawater desalination.

With corresponding improvements, the ABV reactor may also be used to develop nuclear technologies including reactor tests of fuel and structural materials and radioisotope production for commercial and medical applications.

V-1.3. Special features

The unit power, the design and operating characteristics of the ABV reactor installation offer an opportunity to create a compact power unit meeting the requirements for simple and safe operation and multi-purpose applications.

The core lifetime without reloading or shuffling of fuel is 10–12 years.

Specifically, the ABV reactor installation is intended as a universal power source for floating nuclear power plants. Depending on the needs of the siting region, the floating nuclear power plant can generate electric power or provide heat and power cogeneration or heat generation for seawater desalination, etc.

The stationary (land-based or underground) NPP is fabricated as large ready-made units; these units are transported to the site by special truck or water transport. The floating NPP is factory fabricated and commissioned.

There is a prospect of creating an underwater power unit for the oceanic shelf including the Arctic Ocean, based on the ABV as a multi-application power source.

V-1.4. Summary of major design and operating characteristics

The ABV is a pressurized water reactor (PWR); its design incorporates the following main features:

- Integral primary circuit layout with natural circulation of the primary coolant; negative feedbacks and enhanced thermal inertia;
- Passive or self-actuated safety systems;
- Increased resistance to extreme external events and personnel errors;
- The use of nuclear fuel with the enrichment of less than 20% by weight (16.5%).

On the total, the ABV design meets the requirements for next generation NPPs.

A schematic diagram of the ABV plant is given in Fig. V-1.

Core heat removal is conventional two-circuit. The core is cooled and moderated by water as the primary circuit coolant. Hot coolant is cooled in a once-through steam generator, where slightly superheated steam is generated, then supplied to the turbine.

For heat and power cogeneration, the coolant for district heating is heated by steam extracted from the turbine.

Major design and operating characteristics of the ABV are summarized in Table V-1.

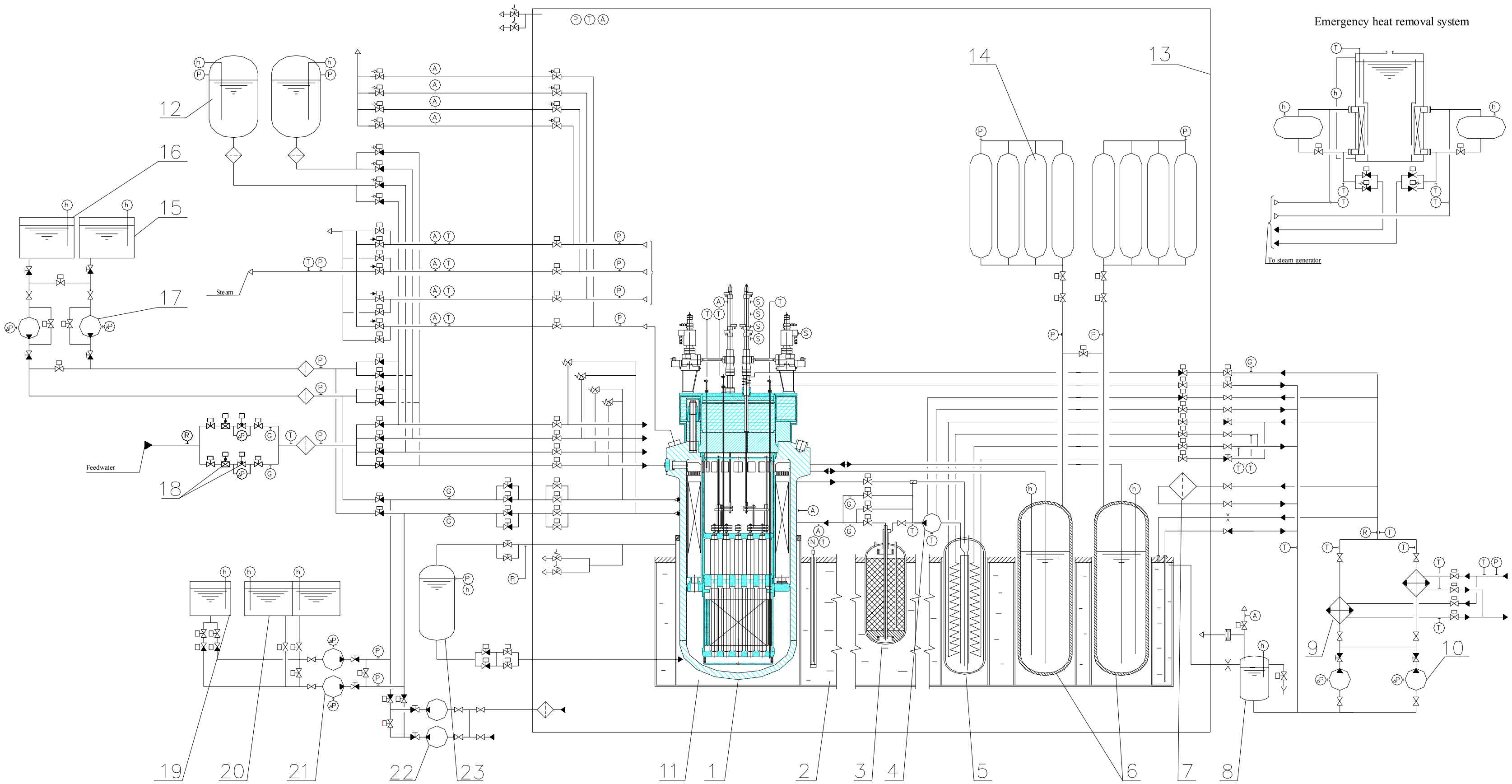
TABLE V-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF ABV

CHARACTERISTIC	VALUE
Major design characteristics	
Rated power, MW — Thermal; — Electric	 45 (reactor thermal power may be within the range of 18 to 60 MW) 11
Operation mode	Base load operation; it is possible to realize load follow mode to track daily power changes or a dispatch mode maintaining the frequency
Capacity factor	0.85–0.9
Reactor type	Integral pressurized water reactor on thermal neutrons

CHARACTERISTIC	VALUE
Number of circuits	2
Cycle type	Steam-turbine cycle with slightly superheated steam
Fuel	
Fuel type	Uranium dioxide in silumin matrix
Fuel element type	Fuel pin
Fuel assembly type	Cassette-type
Fuel enrichment by ^{235}U	16.5 weight %
Coolant	Water (H_2O)
Moderator	Water (H_2O)
Core	
Number of fuel assemblies	121
Equivalent diameter, mm	1219
Height, mm	1300
Reactor unit	
Overall height, m	7.5
Diameter, m	2.6
Operating mass, t	85.7
Diameter of reactor vessel shell, inner/outer, mm	1910/2135
Structural materials	
Fuel element cladding	Zirconium alloy
Fuel assembly structures	Zirconium alloy
Reactor vessel	Heat-resistant pearlitic steel with anticorrosive facing
Steam generator pipe system	Titanium alloy
Reactor internals	Stainless steel 08Cr18Ni10Ti
Core neutronics	
Reactivity coefficient	Rated value
Reactivity coefficient on coolant temperature (taking into account coolant density changes), $1/^{o}C$	Negative
Reactivity coefficient on coolant density (without taking into account coolant temperature $1/(g/cm^3)$)	
Reactivity coefficient on fuel temperature, $1/^{o}C$	Negative
Boron reactivity coefficient, %/(g/kg)	

CHARACTERISTIC	VALUE
Core neutronics (continued)	
Reactivity effect at full core voiding, %	Negative
Power flattening	
Maximum power peaking factor over core volume	
Measures for power non-uniformity decrease	Physical shaping of fuel in the core
Reactivity control, emergency protection	
Compensation of initial reactivity margin	Fuel elements with gadolinium oxide integrated in fuel pellets
Compensation of temperature and power effects of reactivity, reactivity margin for core poisoning by Xenon-135 and Samarium-149, operating margin for reactivity changes under reactor power changes, and provision of core subcriticality in a cold unpoisoned state	Mechanical control rod system; bundles of absorber rods joined by common traverses and travelling inside fuel assembly guide tubes; on the total, six drives of the control and protection system (CPS) are employed
Emergency protection	All control rods (six) enter the core driven by gravity at de-energization of the control rod drives, actuated by reactor protection system. System of emergency injection of boron acid solution
Thermal-hydraulic characteristics	
Primary circuit parameters	
Primary coolant circulation	Natural
Primary circuit coolant flow rate, t/h	397
Coolant temperature at core inlet, °C	247
Coolant temperature at core outlet, °C	330
Coolant velocity in the core, m/s	0.3
Primary circuit coolant pressure, MPa	15.7
Maximum fuel temperature, °C	434
Average fuel temperature in the core, °C	360
Maximum temperature of fuel element cladding, °C	350
Average temperature of fuel element cladding, °C	340
Limit for fuel temperature, °C	900
Limit for temperature of fuel element cladding, °C	700
Minimum margin to heat transfer crisis	1.5

CHARACTERISTIC	VALUE
Secondary circuit parameters	
Steam pressure after steam generator, MPa	3.14
Steam output, t/h	67
Steam temperature at the steam generator outlet, °C	290
Feedwater temperature, °C	106
Feedwater pressure, MPa	4.4
Operating cycle characteristics/mass flows of fuel materials	
Refuelling interval	10–12 years
Number of fuel assemblies in a make-up fuel set	121 (whole core)
Partial refuelling repetition factor	1
Uranium inventory, t	1.4
Operating cycle characteristics/mass flows of fuel materials (continued)	
Fuel lifetime between refuellings, effective hours	70 000
Average burn-up of discharged fuel, g/cm^3 (MW·d/kg U)	0.56 (94.5)
U^{235} specific consumption, g/MW·d	1.75
Natural uranium specific consumption, kg/MW·d	0.248
Design basis lifetime for reactor core, vessel and structures	
Reactor vessel, years	50–60
Steam generator pipe system, years	25–30
Economics	
Plant construction cost, US$ million	43
Specific capital investments for construction, US$/kW(e)	4300
Annual costs for operation and maintenance, thousands US$	
Fuel costs (initial inventory), thousands US$	
Busbar cost of generated electric power (condensation mode), cent/kW·h	3.3
Payback period, years (starting from commencement of operation)	5–7



1 – Reactor
2 – Metal and water shielding tank
3 – Primary circuit filter
4 – Purification and aftercooling system pump
5 – Filter cooler
6 – Pressurizer
7 – Third circuit filter
8 – Third circuit expansion vessel
9 – Third/fourth circuit heat exchanger
10 – Third circuit pump
11 – Reactor caisson
12 – Hydraulic accumulator
13 – Protective shell
14 – Working group of cylinders
15 – Feedwater tank
16 – Distillate tank
17 – Emergency feedwater pump
18 – Feedwater unit
19 – Soluble poison tank
20 – Water storage tank
21 – Primary circuit feedwater pump
22 – Recirculation pump
23 – Hydraulic accumulator of reactor vessel cooling system

FIG. V-1. Basic diagram of ABV plant; a simplified diagram of the emergency heat removal system is given in the right upper corner.

V-1.5. Outline of fuel cycle options

The basic fuel cycle option is a once-through fuel cycle with enriched uranium fuel.

The low power density of the core (linear heat rate of fuel is 38.2 W/cm; core power density is 33 kW/l) allow extension of the refuelling interval up to 10–12 years.

V-1.6. Technical features and technological approaches that are definitive for ABV performance in particular areas

V-1.6.1. Economics and maintainability

The following design features contribute to improving the economy characteristics of the ABV plant:

- A compact integral layout of the primary circuit, providing a reduction in the total structural volume of the reactor compartment;
- The reactor unit service life of 50 years;
- An increase in plant efficiency due to the combined use of installed capacity for cogeneration;
- An option of NPP location in immediate proximity to the consumer with the corresponding minimization of outlays for heat supply;
- Strong reliance on inherent and passive safety features and passive safety systems (passive or passively actuated), resulting in a reduced number and reduced requirements to the capability, operational speed, and power supply of active safety systems and control and monitoring systems;
- The improved operation and maintenance cost characteristics due to the exclusion of annual outlays for fuel management;
- A reduction in solid radioactive wastes and radioactive effluents, effected by the use of leak-tight equipment and systems and by an increase in service life of the main replaceable equipment, such as steam generator pipe systems.

Compared with a land-based nuclear power plant, the floating nuclear power plant option offers a considerable reduction in costs (by 1.3–1.5 times) for construction and operation achieved through the following advantages:

- The use of industrial production processes for floating plant manufacture under shipyard conditions, with delivery to the customer of the commissioned and ready-for-operation plant;
- The minimum scope and cost of capital construction needed to arrange a floating plant location in a water area, as compared with considerable areas of alienable territory for land-based NPPs;
- No need to create transportation links, energy communications and preparatory infrastructure to realize the project;
- Freedom to select the site for a floating NPP, with the possibility of mooring in any coastal region of the world independent of seismicity;
- A considerable reduction in the construction period (3.5 to 4 years) and, consequently, a shorter repayment period of the credit for construction;

- The infrastructure of nuclear ship maintenance centres available in the Russian Federation could be effectively used to minimize maintenance costs under operation as well as requirements for local construction skills; this could be especially important when exporting the plants to developing countries; and
- The concept of a floating NPP makes it easy to realize a “green lawn” concept on the site of the floating NPP operation or if necessary, to replace the exhausted floating plant with a new one, contributing to a reduction of the decommissioning costs.

V-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The integral layout and relatively low thermal power of the ABV result in a low activating neutron flux outside the vessel and, consequently, low activation of the structures, medium and air in the adjacent rooms. Off-board water radioactivity caused by the floating nuclear power plant operation is 0.1 Bq/l, which is one hundred times less than the regulatory limit for drinking water.

Heat emissions to the surrounding water caused by the absorption of penetrating radiation energy exert no influence on temperature conditions of the environment.

Leak-tight primary circuit equipment and pipelines exclude radioactive release to the environment.

In case of primary circuit depressurization, radioactivity is confined by the protective shell and protective enclosure.

Maintenance, including refuelling and radioactive waste management, is performed off-site at special maintenance centres. This excludes activities with an open reactor vessel on the site and, therefore, excludes gas-aerosol radioactivity releases to the environment. For the reasons given above, there is no spent fuel and solid radioactive wastes at the site, and most of the radioactive effluents remain onboard the floating NPP. Negligible quantities of radioactive effluents generated during operation (not more than 100 litres per year), are safely confined in special containers and removed off-site during refuelling or maintenance. Radioactive releases to the environment are, therefore, essentially excluded.

Design analysis shows that the impact of a floating NPP on the environment is insignificant and substantially lower than the allowable values. This conclusion has been confirmed by the data on samples of snow, soil, and vegetation in the berthing area of Russian nuclear ice-breakers. According to these data, relevant to a multi-year berthing of nuclear ice-breakers, there is no impact on the environment and the measured indices are on the level of background values.

After a floating NPP decommissioning, the plant is transported to special works for disposal.

The absence of radioactivity release to the water area and the coastal area during operation as well as the low level of penetrating radiation essentially excludes the environmental risk on the site. Therefore, after the plant decommissioning, “a green lawn concept” could be realized to meet the state-of-the-art Russian and international requirements for decommissioning.

The radioactivity of the major equipment of a reactor installation with expired lifetime is determined by activation of the structural materials. The accumulated activity localized inside the reactor vessel is practically maximal (more than 90%); most of the equipment and the reactor unit structure is non-radioactive and can be disassembled using conventional industrial methods.

V-1.6.3. Safety and reliability

Safety concept and design philosophy

The ABV design development is based on the following concepts of nuclear power plant safety:

- Core damage should be excluded without personnel action in any initiating event with a heat removal disturbance or uncontrolled reactivity addition followed by the failure of control and active safety systems requiring electric power, through the use of passive systems and natural heat dissipation;
- Time margins and technical features should be provided to prevent core damage in accidents with a non-localized leak of any pipe in the primary circuit. In that event, under full drainage and damage of the core, the melt should be localized in the reactor vessel so as not to exceed allowable releases beyond the plant boundaries and emergency exposure doses to personnel.

Provisions for simplicity and robustness of the design

The following features contribute to simplicity and robustness of the ABV design:

- Natural circulation of coolant in the primary circuit;
- Negative reactivity effects;
- Low power density in the core;
- High heat accumulating capacity of the circuit;
- Passive systems and safety devices of special design not requiring any intervention for a start-up, or those actuated without power consumption, by signals of changing physical parameters of the medium (without signal transformation).

Active and passive systems and inherent safety features

Inherent safety features

The ABV is an integral type PWR. The ABV design incorporates the following inherent safety features intended for the self-limitation of power, heat-up rate and coolant loss:

- Low power density in the core;
- The core with negative reactivity coefficients;
- By-design elimination of large-diameter pipelines in the primary circuit; the use of leak restricting devices for coolant outflow as well as in the absence of inter circuit leak-tightness of the steam generator;
- Large volume of coolant above the core;
- By-design elimination of main circulating pumps (natural circulation of the primary coolant in normal operation mode); and
- Reduced neutron fluence on the reactor vessel, provided by the integral design (to prevent brittle failure of the vessel during operation).

Passive safety systems

Passive systems in which the performance of a safety function is based on natural phenomena and requires no external power supply are consistently used in the ABV design. These systems include the following:

- Control and protection system (CPS) drives that provide for control rod insertion into the core under gravity, in case of a drive de-energization caused by control system signals or instantaneously, by action of a working medium through the self-actuated devices;
- An emergency heat removal system actuated by opening of the pneumatically actuated valves of normally-open design and by closing of the pneumatically actuated valves of normally-closed design under de-energizing of a pneumatic control valve occurring when the reactor installation is de-energized, or actuated by control system signals or by direct action of the primary circuit pressure through self-actuating devices;
- A reactor caisson and a system of water supply to the reactor caisson supporting the reactor vessel integrity and core melt in-vessel retention in severe beyond design basis accidents; and
- A containment and a structural unit of the nuclear island provided to protect the reactor against external impacts and also limiting the radioactive release under design basis and beyond design basis accidents.

Active safety systems

A set of active safety systems is used in the ABV design to perform all required safety functions. They include:

- A purification and cooldown system intended to remove residual heat from the core to the third circuit water and then to the process (off-board) water;
- The primary circuit make-up system supplying water to the reactor in accidents with primary circuit depressurization;
- A soluble poison injection system designed to introduce boric acid solution into the reactor to shut it down and keep it in a subcritical state under a failure of the CPS electro-mechanical system.
- Two emergency diesel-generators for the supply of electric power to the active safety systems;
- Two sets of control safety systems to provide the redundant control of the safety systems.

Self-actuated devices

One of the ways to achieve a high safety level in the reactor installation is to increase safety system reliability through the use of the self-actuated devices.

Self-actuated devices or devices of direct action are intended to activate the protection and localizing safety systems.

These devices are actuated by a change in the physical parameters of a working medium or the environment of the protected equipment. In the ABV design, the pressure in the reactor vessel is used as such a parameter.

Self-actuated devices of the ABV are the following:

- The power supply circuit breakers of the CPS drives, actuated by the excessive pressure in the reactor;
- The power supply circuit breakers actuated by the excessive pressure in the pneumatic control valve of the emergency heat removal system;

- The chosen actuating setpoints of the power supply circuit breakers are a little bit higher than those set for the actuation of the control safety systems; thus, self-actuating devices provide a redundancy in the control of the safety systems and increase the reliability of the safety systems.
- The important feature of self-actuating devices is their resistance to personnel errors or human actions of malevolent character.

Reliability of safety systems

A high reliability level of the safety systems is secured through applying the following principles:

- Passive functioning, either not requiring actions for a start up or requiring a minimal action;
- The use of self-actuated devices to start-up the operation of the protection and localizing safety systems;
- The use of diverse safety systems and devices based on different principles of operation; for example, the electro-mechanical control and protection system (CPS) and the system of soluble poison injection are used for emergency reactor shutdown; the residual heat removal system and the channel of aftercooling through the process condenser, and the channel of aftercooling by the third circuit through the cooler of the purification and aftercooling system are used for residual heat removal;
- Redundancy of the safety systems;
- Physical separation of the safety systems; and
- A continuous or periodic control of the state of the equipment and systems under operation.

Structure of the defence-in-depth

The defence-in-depth concept of the ABV is realized in two directions; it includes creation of the physical barriers preventing the potential release and propagation of ionizing irradiation and radioactive substances and arrangement of the consecutive levels of protection of these barriers against internal and external impacts.

The system of physical barriers includes:

- The fuel matrix;
- The fuel element cladding;
- The leak-tight primary circuit, including the leak-tight heat-exchanging surfaces of the steam generator;
- The stop valves;
- The leak-tight protective shell; and
- The containment.

The design of the primary circuit is leak-tight, with organized leakages being absent under operation, which provides a high level of radiation safety during normal operation.

The following features are incorporated to prevent disturbances of normal operation and provide the reliability of all safety barriers:

- The improvements in the design of the systems of normal operation important for safety, such as circuit simplification, a reduction in the number of the equipment and in the pipeline length; and

- Measures to prevent bypassing of the localizing barriers.
- To prevent deviations from normal operation, the following design solutions are used:
- A reactor design incorporating many self-regulatory features;
- The implementation of new progressive diagnostic systems and methods to forecast the residual life of the equipment and to secure early failure detection; and
- The use of highly reliable, self-diagnosing automated control systems and the system of information support for the operator.

To prevent propagation of disturbances and to limit the consequences of accidents, the following design measures are provided:

- Strong reliance on the inherent safety features ensuring the self-protection properties of the reactor installation;
- A reasonable combination of passive and active safety systems;
- Automation of the control processes in accidental conditions and in accidents; and
- The possibility to control beyond design basis accidents with both passive and active systems.

Design basis accidents and beyond design basis accidents

The design features and measures to control beyond design basis accidents include the following:

- The normal operating systems and safety systems designed to retain their operability in beyond design accidents with the possibility of both on-site and remote personnel actions;
- The use of standby safety systems, such as the system of soluble poison injection or the system of water supply in the reactor caisson, in beyond design basis accidents;
- The incorporation of a technical diagnostic system, "an adviser" for the operators providing a quick assessment of the plant state, state of the individual systems and equipment, and producing recommendations on plant control during accidents, including identification of the most effective methods to be applied to return the plant to a safe state;
- Provision of sufficient time margins for the personnel to take actions and to get assistance from the outside for the prevention of accidents or for the limitation of their consequences

V-1.6.4. Proliferation resistance

The following design features of the ABV contribute to an enhancement of proliferation resistance of the plant:

- The use of uranium dioxide fuel with the enrichment of not more than 16.5% of ^{235}U by weight, to exempt fresh fuel from the definition of direct use materials supported by the IAEA;
- A long operating period (10–12 years) without refuelling, excluding any operations with fresh or spent fuel on the site; the reactor is refuelled at dedicated factories where all fissile materials are closely controlled.

V-1.6.5. Technical features and technological approaches used to facilitate physical protection of ABV

Physical protection of an NPP with the ABV reactor installation would incorporate all conventional approaches applied for NPPs with the VVER and PWR type reactors.

The physical protection system conforms to zonal principles and includes a complex of the following technical measures:

- The system of access control;
- The system of guard engineering (supervision, signalling); and
- Organizational measures.

For a floating NPP, provision is made for a water area limited by the protective breakwaters and protective dams, a coastal technological site and the floating power unit zone, a zone of increased control.

Technical features of the ABV plant contributing to an enhanced physical protection include the following:

- The inherent safety of features (natural circulation of the primary coolant under all conditions, high thermal inertia, low power density, and high seismic resistance) and passive safety systems that altogether ensure high resistance of the plant to personnel errors and human actions of malevolent character;
- The NPP protection against aircraft crash and human-induced and natural external impacts;
- For a floating NPP, water area protection against unauthorized access of floating structures and objects;
- Elimination of all operations with fresh and spent fuel at the site; absence of fuel storage facilities on the site.

V-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of ABV

Non-technical factors and arrangements that could facilitate effective development and deployment of a NPP with the ABV reactor installation are the following:

- Construction of a pilot floating NPP with the icebreaker-type KLT-40S reactor in Severodvinsk (Arkhangelsk region of the Russian Federation) to demonstrate the advantages of a floating NPP technology; a license for the construction of this plant has been obtained; the construction is scheduled to start in 2006;
- Further decisions on the construction of small nuclear power plants in remote regions of the Russian Federation (Far North and Far East); and
- Signing of intergovernmental agreements to examine the possibility of leasing of the nuclear power plants with the ABV reactors under “Build-Own-Operate” conditions, which could decrease the presently observed political and economic restrictions on the use of nuclear power technologies in developing countries; other options, such as supply of turnkey plants or leasing of the “separate” reactor units could be considered also.

In all aforementioned options, the nuclear technology itself, including the factory-fuelled reactor installation delivery, maintenance, and spent fuel management would be controlled by the Russian Federation.

The following features of the ABV plant could help realize the above mentioned power plant supply or leasing options:

- Low construction cost (less than 50 US$ million), acceptable to many countries;
- The possibility of plant location in any coastal region of the world regardless of its seismicity;
- The floating NPP can be designed as a power-desalination complex, enlarging the number of potential customer countries as the shortage of fresh potable water throughout the world becomes more acute.

V-1.8. List of enabling technologies relevant to ABV and status of their development

Table V-2 gives a list of the enabling technologies relevant to the NPP with the ABV reactor and outlines their development status.

TABLE V-2. LIST OF ENABLING TECHNOLOGIES FOR THE ABV PLANT

ENABLING TECHNOLOGIES	DEVELOPMENT STATUS
Steam generators, control and protection system (CPS) drives, pressurizer system, reactor installation systems, biological shielding and other systems — the technologies widely used in NPPs with pressurized water reactors and shipboard reactors of the Russian nuclear fleet	Widely used reactor technologies; the operating experience of multi-purpose shipboard reactor installations exceeds 6000 reactor-years
Safety concepts and technical features of the safety design — the technologies previously developed and validated for the Russian AST-500 nuclear co-generation plant	IAEA review of the AST-500 safety design has been conducted
Technologies of floating NPPs – the technologies developed for a pilot floating NPP with the KLT-40S reactor installation	The detailed design of a pilot floating power unit has been developed; a license for the construction has been awarded; a decision on the plant construction in Severodvinsk has been adopted; the construction is to be started in 2006

The main engineering solutions of the ABV plant that were at a high level of technical validation at the time when this report was prepared are the following:

- Integral layout of the primary circuit;
- A leak-tight design of the primary circuit using welds, canned glandless pumps and sealed bellows valves;
- Core design based on fuel assemblies (cassette-type core);
- The application of once-through titanium steam generators;
- The application of the CPS drives used in Russian shipboard reactors;
- Passive safety systems providing the emergency shutdown, core cooling and aftercooling of the reactor;
- The application of proven technologies of the metallurgic, press-forging and machine-assembly production available at the Russian works for shipboard nuclear power plant manufacturing;

- The application of highly-reliable systems proven in operation of the shipboard reactor installations and contemporary NPPs; and
- The application of proven techniques of the equipment assembly, maintenance and replacement as well as diagnostic techniques and systems to monitor the equipment state.

The production of the ABV reactor installations would make use of the fabrication technologies commercially mastered in the Russian Federation, such as:

- The production techniques for reactor vessel components (cover, vessel);
- The production techniques for titanium alloys for steam generator pipe systems;
- The design and production techniques for sealed drives for CPS control rods; and
- The technologies for production of elements of normal operating systems and safety systems (self-actuated devices, compensators, reservoirs, heat exchangers, pumps, and filters).

V-1.9. Status of R&D and planned schedule

The ABV design is being developed in response to an appeal of the authorities of the Far North and Far East regions of Russia to the Russian Government requesting to provide small reliable power sources to support the incipient activities on development of new deposits and to cope with a shortage of power and heat for residential needs. The stakeholders involved in the ABV design and technology development are listed in Table V-3.

TABLE V-3. MAIN PARTICIPANTS OF DESIGN DEVELOPMENT FOR THE ABV

COMPANY	RESPONSIBILITY AREA
OKB Machine Builing (OKBM), Nizhny Novgorod	Chief designer of the reactor plant
Institute of Physics and Power Engineering, Obninsk	Scientific leader of the design
Public company “Lazurit”, Nizhny Novgorod	General designer of floating NPPs

At present, the development of nuclear power sources with the ABV reactor installations based on the technologies of nuclear shipbuilding is financed by the companies involved in the project.

The ABV design development makes an extensive use of the research and development (R&D) results obtained previously, during design development for the ABV-6M, SAKHA-92, and KLT-40S reactor installations.

The ABV design development status is characterized by the following:

- Detailed design of the original ABV-6M reactor installation was completed in 1996;
- At present, the design is in a stage of technical and economic optimization regarding the power up-rating, validation of an operating cycle with a 10–12 year refuelling interval, and cost decrease for the floating NPP.

The timeframe for the design completion and deployment of the ABV plants (under favourable conditions of financing) is outlined in Table V-4.

TABLE V-4. TIMEFRAME FOR ABV DESIGN COMPLETION AND DEPLOYMENT

STAGE	STAGE DURATION
Detailed design completion, including licensing	3 years
Plant construction (including licensing and performance of main pilot design and demonstration activities): — Land-based nuclear cogeneration plant; — Floating NPP	 5 years 4 years

V-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

As it was already mentioned, the ABV design development and validation is being carried out making maximum use of the engineering solutions and R&D results obtained previously for various PWR (VVER) type marine reactors are proven by their successful operating experience. Because of this, there is no need to perform comprehensive research and engineering, and only pilot design and demonstration of individual equipment would be required to finalize the ABV design development. The scope of the required validation and testing includes:

- Aerodynamic tests of the main flow path;
- Development of fabrication techniques for main units of the steam generator; and
- Development, fabrication and testing of equipment pilots.

In the ABV case, it would be necessary to construct a pilot floating power unit to demonstrate the production quality and reliability and cost effectiveness of the plant under a lifetime core operation. It could be mentioned that the use of floating nuclear power plants for power supply and desalination is an innovation on itself.

V-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

V-2. Design description and data for ABV

V-2.1. Description of the nuclear systems

Reactor core and fuel design

The ABV reactor core is of a unified design (see Fig. V-2); similar core has been developed and qualified for the pilot nuclear cogeneration plant with the KLT-40S reactors. Solutions developed for the ice-breaker reactor cores and used in the ABV core are the following:

- Compensation of reactivity margin for fuel burn-up by gadolinium based burnable poisons;
- Improved core power shaping achieved by optimized distribution of fuel and burnable poisons in the core;
- Two independent mechanical systems of reactivity control without a boron solution in the primary coolant; and
- Compensation of reactivity changes in power operation is effected only by the central group of shim rods.

An optimized fuel lattice with improved neutron moderation permits fuel burn-up increase in the ABV reactor core, see Fig. V-3.

Uranium dioxide granules dispersed in a silumin matrix are employed as fuel. Having a high thermal conductivity, they secure high reactor power manoeuvrability. Corrosion-proof zirconium alloy is used as fuel cladding. The nominal power change rate amounts to $0.1\%N_{nom}/s$ within the entire power operating range.

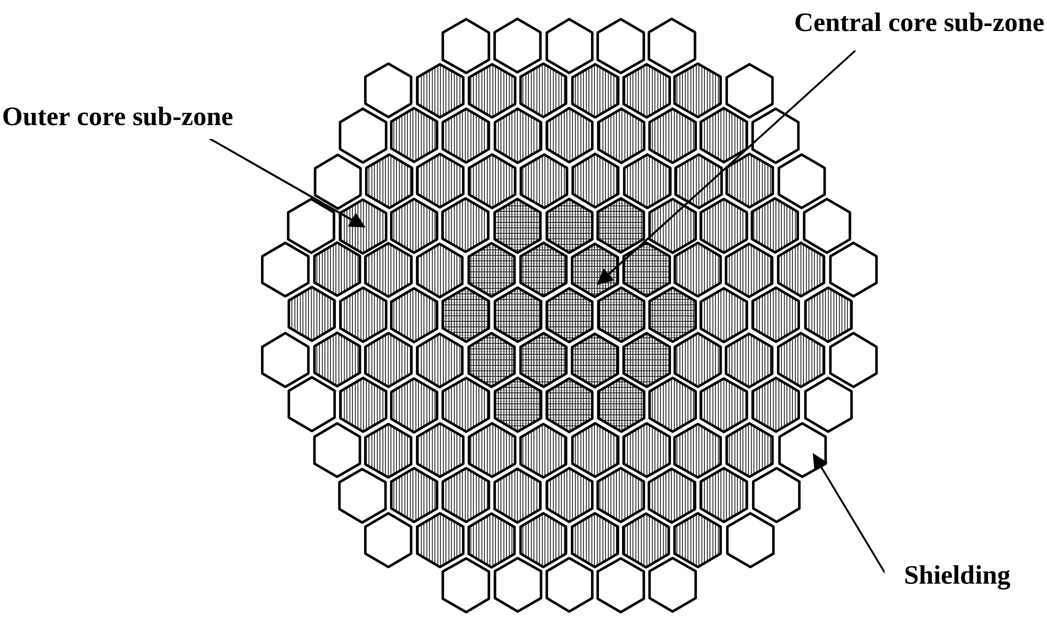
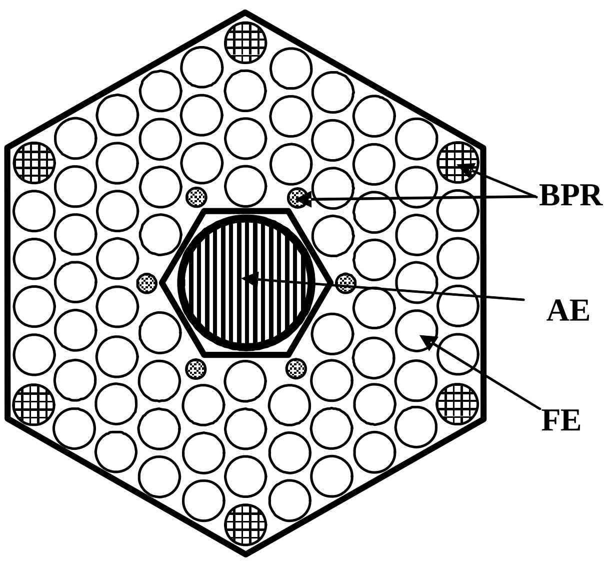


FIG. V-2. ABV reactor core.



BPR – is for burnable poison rod; FE – is for fuel element; AE – is for absorber element

FIG. V-3. Fuel assembly of the ABV reactor.

Although the uranium enrichment is less than 20%; the type of fuel used (uranium dioxide granules in silumin matrix) ensures sufficiently high uranium content.

The low core power density (33 kW/l), “mild” conditions of fuel operation and reduced fuel heat loads increase thermal margins and contribute to an extended fuel lifetime.

The elimination of soluble boron control together with the adopted parameters of the fuel lattice provide negative reactivity coefficients on the fuel and coolant temperature; negative steam and integral power coefficients of reactivity in the entire range of operating parameters, which altogether secures inherent safety features of the reactor core. These inherent safety features ensure power self-control in a steady state reactor operation, power rise self-limitation under positive reactivity insertions, self-control of the reactor power and primary coolant pressure and temperature self-limitation in transients, as well as the limitation of the heat-up rate in reactivity-initiated accidents.

Fuel fraction and burnable absorber profiling in the core, similar to those used in operating ice-breaker reactors, are employed to minimize power peaking and non-uniformities in fuel burn-up in the core, contributing to an improved fuel utilization.

The low power density of the core, the fuel characteristics and some other adopted design solutions provide a refuelling interval extension up to 10–12 years. The core lifetime is about 70 000 effective hours.

Some characteristics of the ABV reactor core, additional to those shown in Table V-1, are given in Table V-5.

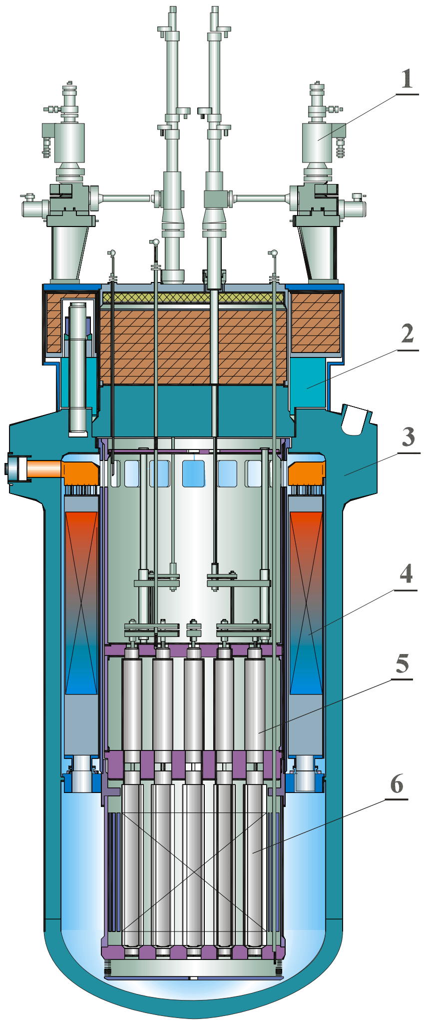
TABLE V-5. SOME CHARACTERISTICS OF ABV CORE

CHARACTERISTIC	VALUE
Thermal power, MW	45
Fuel assembly flat-to-flat size, mm	97
Core power density, kW/l	33.0
Average linear heat rate of fuel, W/cm	38.2
Fuel element diameter, mm	6.8
Burnable poison rod outer diameter, mm	6.8 and 4.5
Absorber element outer diameter × cladding thickness, mm	25 × 1.8

Reactor

The ABV is an integral type PWR with natural circulation of the primary circuit coolant at all power levels and a remote gas compensation system. The reactor core, the steam generator pipe system and the core cooling circuit are arranged within a common vessel.

A general view of the reactor is given in Fig. V-4.



1 – CPS drive
2 – Reactor cover
3 – Reactor vessel
4 – Steam generator
5 – Block of protective tubes
6 – Core

FIG. V-3. General view of the ABV reactor.

The reactor includes:

- The vessel;
- The cover;
- A steam generator pipe system;
- The removable unit consisting of the unit of protective tubes, the unit of devices, and the in-vessel shaft;
- CPS drives; and
- Primary sensors.

To control temperature, thermal converters are installed at the core inlet, in the outlet reactor chamber, and at the outlets of some fuel assemblies.

The vessel is a welded cylindrical "container" with an elliptical bottom. At the top of the vessel there are the fitting pipes for feedwater supply and superheated steam removal, as well as those for the connection of the primary circuit systems and the auxiliary process systems.

All primary circuit fitting pipes have restriction inserts (not more than DN 15), while fitting pipes of the system of makeup and emergency flushing also have ball shutoff devices.

The reactor cover consists of a load-bearing slab, a shell attached to this slab and sealed by a weld, and a top slab welded to the shell. The cavity between the top slab and the load-bearing slab is filled with serpentinite and acts as a biological shielding, and the heat insulation is located at the top.

The posts of the CPS drives and thermal converters, etc. are welded to the load-bearing slab and penetrate through the cover. Points of penetration through the top slab are sealed.

The removable unit is intended to arrange the core fuel assemblies and control rods and to route the primary coolant flow. Fuel assemblies are located in the in-vessel shaft.

The unit of protective tubes and the unit of devices provide the necessary coolant flow rate distribution between the fuel assemblies and an arrangement of the connectors combining the absorber elements of fuel assemblies into CPS control rods and connecting the CPS control rods to CPS drives. There are six CPS control rods; each CPS control rod combines the absorber elements of 9 fuel assemblies.

The steam generator pipe system, arranged in the annular space between the vessel and the in-vessel shaft shaft, is a once-through vertical surface-type heat exchanger generating steam of the required parameters from heat of the primary circuit coolant.

The pipe system is divided into four independent sections; feedwater supply and steam removal from each section is carried out through the fitting pipes in the reactor vessel.

Counter flow circulation is used, i.e., the primary circuit coolant moves downward in the inter-tube space, while the secondary circuit working medium is moved upward in the tubes.

In case of inter-circuit leaks, it is possible to cut off any section automatically or remotely. Identification of the leaking section is carried out with the use of the detection blocks of the radiation and process control system. Finding and disabling a faulty module is carried out during reactor shutdown.

CPS drives are intended to move the CPS control rods in the core at the reactor start-up, and for power control, excess reactivity compensation, and core transfer to a sub-critical state.

The drive operation is based on conversion of the rotary motion of the electric motor into reciprocal motion of a rack connected to the CPS control rods.

Main characteristics of the drive are the following:

- Travel of the rack with the CPS control rods is 770 mm; and
- Conveying speed of the CPS control rods in the control mode is 1 mm/s.

From the moment of electric motor de-energizing, the drive provides downward movement of the CPS control rods at a rate of 50–130 mm/s under the impact of gravity.

During reactor operation, the electric motor, the position sensors, the displacement pick-ups, and the limit switch sensors are energized.

Depending on the current, the drive is either in a stand-by mode or by gearing the motor rotates the rack gear imparting reciprocal motion to the rack.

In emergency conditions, when the electric motor is de-energized, the rack with the CPS control rods is inserted into the core under the impact of gravity.

The rack post is cooled with the intermediate circuit water.

There is an electric power breaker operated by pressure in the drive control system to de-energize the electric motor during emergency increases of pressure in the primary circuit medium, in case of failure of the electric control safety systems.

Pressurizer

The pressurizer system is intended to generate and maintain the primary circuit pressure within the specified range under all operating conditions of the plant. It includes:

— Two pressurizers of 1.5 m^3;
— Two groups of four gas cylinders of 0.4 m^3 each; and
— Pipelines and valves.

A gas pressurizer system is used; nitrogen is the working gas in the system cylinders.

Pressurizers are mounted in caissons of the metal and water shielding tank. Pressurizers are connected to the reactor by a DN43 pipeline, while with respect to gas; pressurizers are connected to cylinder groups by DN25 pipelines.

To prevent primary circuit coolant leakage in case of a pipeline break or equipment failure, a DN15 restriction insert is arranged in the fitting pipe penetrating the reactor vessel.

Purification and aftercooling system

The purification and aftercooling system maintains the primary circuit water at a required level of quality and removes the residual heat. The system includes:

— A cooler;
— An ion-exchanging filter;
— An electric pump; and
— Pipelines and valves.

The system is connected to the reactor by two pipelines, the DN40 coolant supply pipeline and the DN32 coolant return pipeline. To prevent primary circuit coolant leakage in case of pipeline or equipment breakage, the DN15 restriction inserts are arranged in the fitting pipes penetrating though the reactor vessel.

In the purification mode, the circulation circuit path incorporates the reactor, the cooler, the electric pump, the filter, and again the reactor. The flow of the primary circuit water supplied to ion-exchange filters for purification is selected from the calculations of balances of the corrosion products; it equals to 0.5 t/hour.

In the aftercooling mode, the circulation circuit path incorporates the reactor, the cooler, the electric pump, and again the reactor; the primary circuit filter is bypassed in that mode.

In addition to a filter with stop valve, the system has two bypass lines which are used to change the flow rate of water extracted from the reactor in the aftercooling mode –0.5 t/hour and 4 t/hour. The operating mode change is carried out by switching the corresponding valves.

The cooler over the cooling circuit is in two sections; the output of each section is sufficient to provide the aftercooling mode of the rated power reactor.

During plant operation, the purification and aftercooling system is activated depending on the results of coolant sampling in the primary circuit. The system is deactivated after coolant quality recovery. Sampling of the primary circuit coolant is carried out quarterly.

Secondary circuit

The secondary circuit supplies feedwater to the steam generator sections to generate steam of the required parameters. The system includes:

- Four steam generator sections;
- Feedwater and steam pipelines; and
- Valves.

Feedwater is supplied from the condensate-feed system of the steam-turbine plant to the steam generator sections through feed piping and then through four DN32 pipelines. Each DN32 pipeline has double pneumatically actuated valves. The first valves along medium flow are the stop valves.

Steam is removed from the steam generator sections by four DN90 pipelines equipped with double pneumatically actuated stop valves, enters the main steam line and then the turbine generator.

The valves mounted in the feedwater pipelines and steam lines are of a normally-closed type. The first valves from the reactor side are inside the protective enclosure, the second ones are outside the protective enclosure.

The emergency heat removal system is connected to steam lines and feedwater pipelines.

To prevent over-flattening of the steam generator section cut-off from the secondary circuit, each section has an automatic safety device. In both floating and land-based plants it removes steam from the steam generator sections to the process condenser in aftercooling modes.

The emergency heat removal system is connected to steam lines and feedwater pipelines before the double stop valves. A mechanical fine filter is mounted in the feed pipeline before the entry into steam generator sections.

Main heat transport system

A schematic of the ABV main heat transport system with specification of heat removal path in normal operation and in accidents is shown in Fig. V-4.

V-2.2. Description of the turbine generator plant and systems

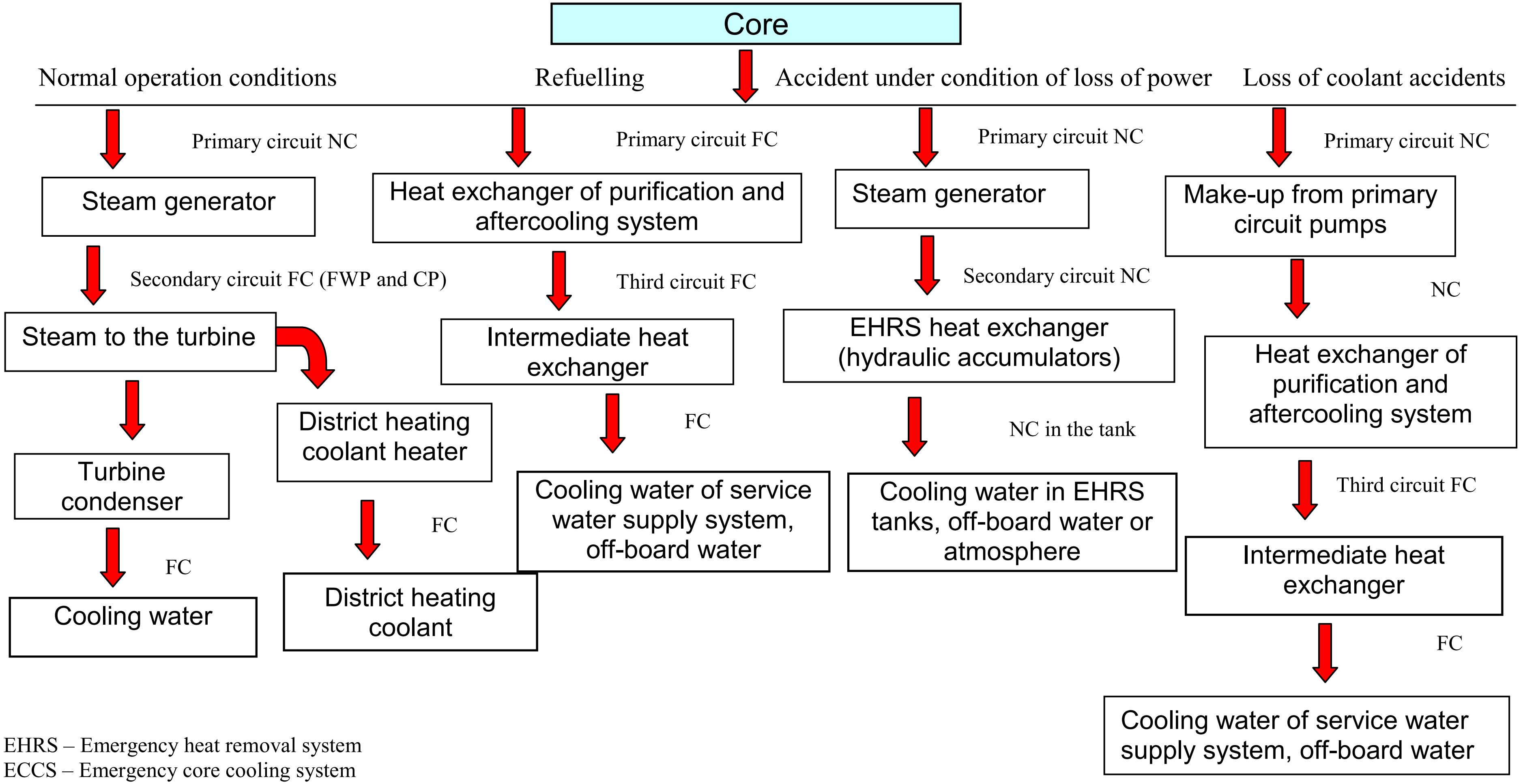
The steam turbine plant has a simplified heating circuit with feedwater heating in the de-aerator. To enhance thermal efficiency, the heat circuit provides for 0.3 MPa steam extraction from the turbine to supply the heating-system water heater. The steam pressure in a take-off is selected to secure the heating-system water heat up of 120°C.

In addition, to enhance thermal efficiency, waste steam of the feed turbo-pump is supplied to the deaerator as a heating medium.

The heat circuit provides for heat recovery of the heating steam condensate from district heating coolant heaters. The heating steam condensate is a heating medium in a water-to-water heater of the heating-system water, where water is heated by ~4°C.

Main condenser vacuum of 0.0067 MPa provides a sufficiently high efficiency of the turbine and ensures the permissible humidity of the end blades.

The following energy conversion scheme is adopted for the ABV plant.



EHRS – Emergency heat removal system
ECCS – Emergency core cooling system
FC – Forced circulation
NC – Natural circulation
FWP – Feedwater pump
CP – Condensate pump
CPPCS – Circulation pump of purification and aftercooling system
RP – Recirculation pump

FIG. V-4. Main heat transport system of the ABV plant.

Live steam generated by the nuclear steam supply system is supplied to the turbine at 3.0 MPa and then dumped to the main condenser.

The same live steam is also supplied as a heating medium to the low pressure steam generator and to the turbine of the feed turbo-pump.

The main ejector maintains vacuum in the main condenser. The auxiliary ejector draws off the steam from the seals. Both ejectors operate with a live throttle steam of 1.6 MPa.

Waste steam from the feed turbo-pump is supplied as a heating medium to the deaerator, where feedwater is deoxygenated and heated up to 106°C.

As it was already mentioned, the steam extracted from the turbine at 0.3 MPa is supplied as a heating medium to the heating-system water heaters, where the water is heated up to 120°C.

Live steam, reduced to 0.6 MPa, is supplied to the heating-system water heaters, which heat heating-system water up to 150°C.

The heating steam condensate of the heaters for district heating is supplied to water - water heaters, where it is cooled, heating the water, and then dumped to the main condenser.

Main characteristics of the turbine generator plant are summarized in Table V-6.

TABLE V-6. MAIN CHARACTERISTICS OF THE TURBINE GENERATOR PLANT

CHARACTERISTIC	VALUE
1. Rated power, MW	11.0
2. Steam pressure, MPa	3.0
3. Temperature, °C	285
4. Pressure of extracted steam, MPa	0.3
5. Extraction type	Controlled

Application of steam extraction is a feature of the turbine generator turbine. For this reason, a control grid valve is arranged before the turbine drum part, which maintains a constant pressure of extraction under primary operating conditions.

To control the grid valve, there is a servo drive in the middle of the steam exhaust part of the vessel.

The condenser is mounted separately on a turbine base; it is connected to the turbine by a fitting pipe compensating their mutual displacement.

The turbine generator condenser is a surface-type, double-flow and single-pass with respect to cooling water, with lateral suction of the air-steam mixture.

Application of a double-flow condenser enhances the reliability of the whole turbine generator plant because in case of failure of one flow channel, the second provides 100% power operation.

Application of a single-pass condenser reduces hydraulic friction of the cooling water.

Commercial Russian electric pumps EKN-80-75-R are the main condensate pumps. These pumps were specially upgraded to meet the quality standard of the main developed equipment of the ABV power plant. The upgraded electric pumps are referred to as EKN-80-75-R1.

Three feed turbo-pumps supply feedwater to the steam generators of the nuclear steam supply system. Two pumps operate constantly at half power; the third pump is under cold standby condition. In case of stoppage of one of the pumps, the second pump operates at full power.

Such an operating scheme of feed pumps (with a spinning reserve) ensures a continuous feedwater supply to the nuclear steam supply system in case of failure of one of the pumps. All other power plant pumps with electric drives, as a rule, have fully automatic 100% redundancy.

To put the power plant into operation, to remove it from operation, and to act during emergency aftercooling, commercial process-air condensers are provided, which are arranged to dump steam at a rate of 14 t/hour for each condenser.

Commercial Russian low pressure steam generators PGND 10/5 cater for the steam needs of the entire ABV plant.

V-2.3. Systems for non-electric applications

In case of a heat and power application, the ABV plant is coupled with a district heating system.

The network water system, which is a part of the plant heating system, is intended to heat water in district heating coolant heaters (the water is supplied by coastal pumps), and to control water temperatures within the range of 70°C to 150°C.

Steam and water heaters with binding pipelines and temperature regulators are the main elements of the system.

The second steam heater heats heating-system water up to 120°C, using steam heat extracted from the plant turbine. The first steam heater heats water up to 150°C, using live reduced steam.

Three-way valves control the heating-system water temperature at the heater outlet by controlling the water flow by-passing the heater.

The water heater is intended to preheat heating-system water before it is supplied to the steam heaters of the district heating coolant, by making use of heat of the heating steam condensate. The condensate, cooled in the water heater, is dumped to the turbine generator plant.

V-2.4. Plant layout

Equipment layout

The main equipment of the ABV plant is arranged as a steam-generating unit consisting of the following items:

- The integral reactor;
- The cooler of the purification and aftercooling system;
- An ion-exchange filter;
- The pressurizers (two);
- The electric pump of the purification and aftercooling system;
- A metal-and-water shielding tank;
- Biological shielding blocks; and
- Valves, pipelines and sensing devices of the control systems.

The metal-and-water shielding tank is a substantial structure for the equipment of the steam generating unit. The reactor, two pressurizers and the cooler of the purification and aftercooling system are arranged in dry caissons of the tank. A primary circuit filter is mounted without a caisson, and it is in contact with the third circuit water.

The equipment is welded to the orifices of the caissons above the cover of the metal-and-water shielding tank.

The steam-generating unit is attached to the protective shell by load-bearing horizontal flanges welded to the metal-and-water shielding tank. Doweled and keyed joints are used for the attachment, which excludes unauthorized displacement of the steam-generating unit under external impacts and provides freedom for the steam-generating unit during thermal motion.

Biological shielding blocks are mounted on the tank cover as well as at the level of the top end of the tank supporting frame.

In the sub-block space between the lower top blocks of the biological shielding, there is a piping arrangement of the reactor installation systems. Primary circuit system valves are arranged on top of the biological shielding blocks, in the o-called “valves houses” (special sections of biological shielding).

Heat-insulating material is laid on top of the biological shielding blocks to provide insulation for the hot surfaces of the block.

In the top of the biological shielding blocks, there are hatches with removable biological shielding blocks to inspect certain pipeline sections located in the sub-block space.

A general view of the ABV steam-generating unit is shown in Fig. V-5.

In the equipment room, above the top area of the biological shielding, there are:

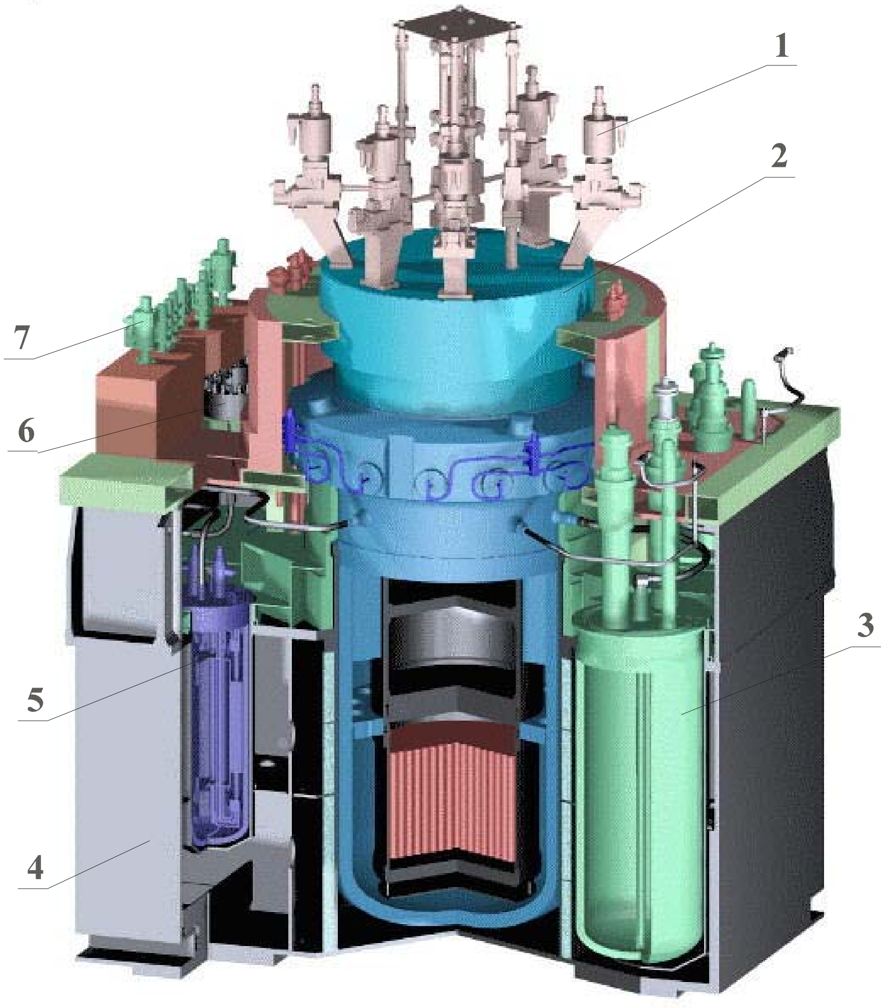
- CPS drives;
- An electric pump of the purification and aftercooling system;
- High pressure gas cylinders; and
- Pipelines, valves, working medium sensors, monitoring systems, cable routing, etc.

On the top of the equipment room, there is a manifold to supply/discharge ventilating system air.

All equipment, pipelines and valves, operating under primary circuit pressure, are arranged in the protective shell.

To prevent air leakage into adjacent rooms, the protective shell is kept at a partial vacuum maintained by differences in the air flow rates of the intake and exhaust ventilation.

The equipment arranged in the protective shell requires no maintenance during plant operation. The equipment is arranged so that it is easy to carry out routine repair and maintenance works during scheduled plant shutdowns.



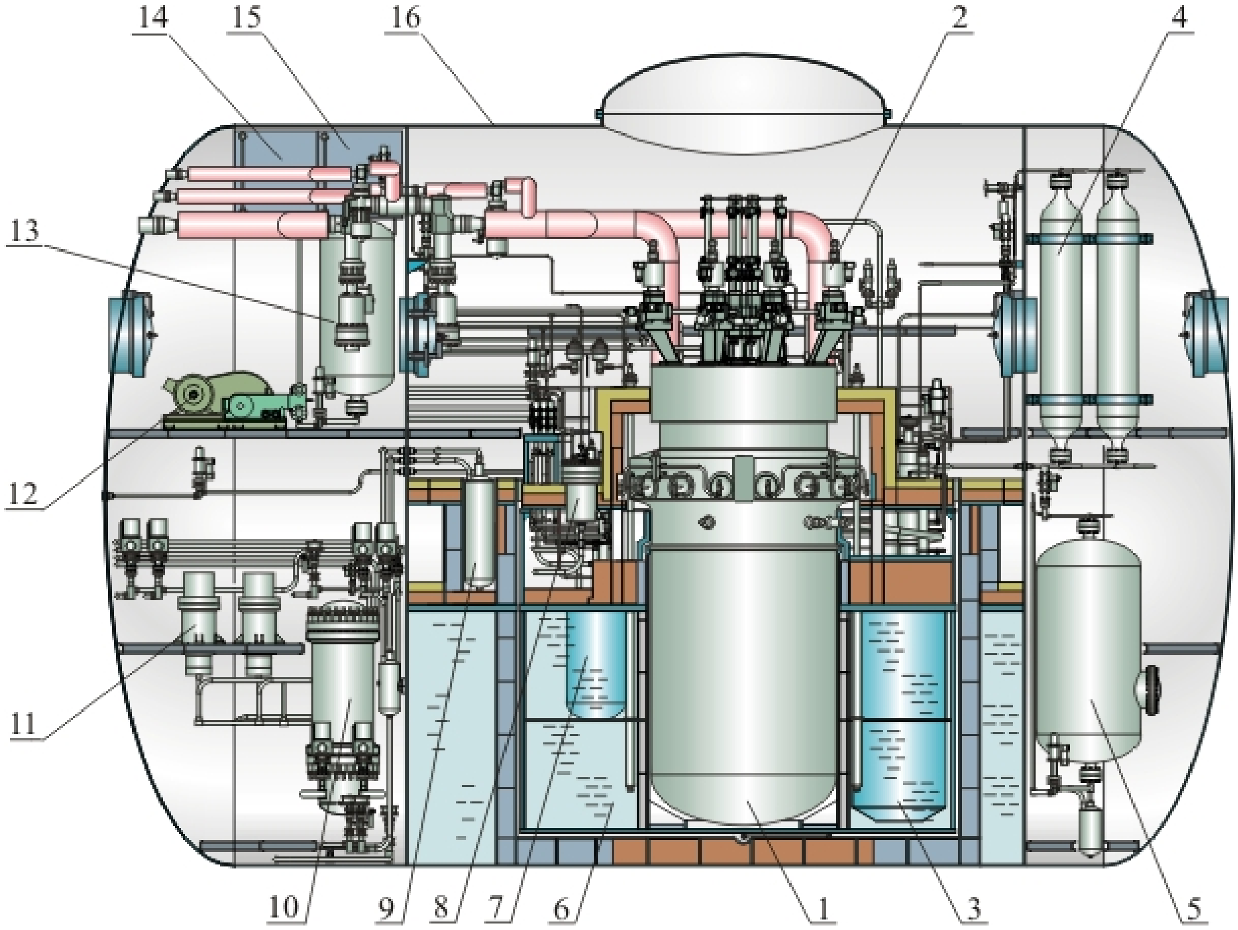
1 – CPS drive
2 – Reactor
3 – Pressurizer
4 – Metal and water shielding tank
5 – Purification and aftercooling system cooler
6 – Purification and aftercooling system pump
7 – Valves

FIG. V-5. General view of the ABV steam-generating unit.

Reactor unit and plant layout

The protective shell, where the steam-generating unit (reactor unit) is arranged, can have various design modifications depending on the purpose and destination of the NPP.

Layout of the reactor unit for a land-based NPP is shown in Fig. V-6 and V-7.



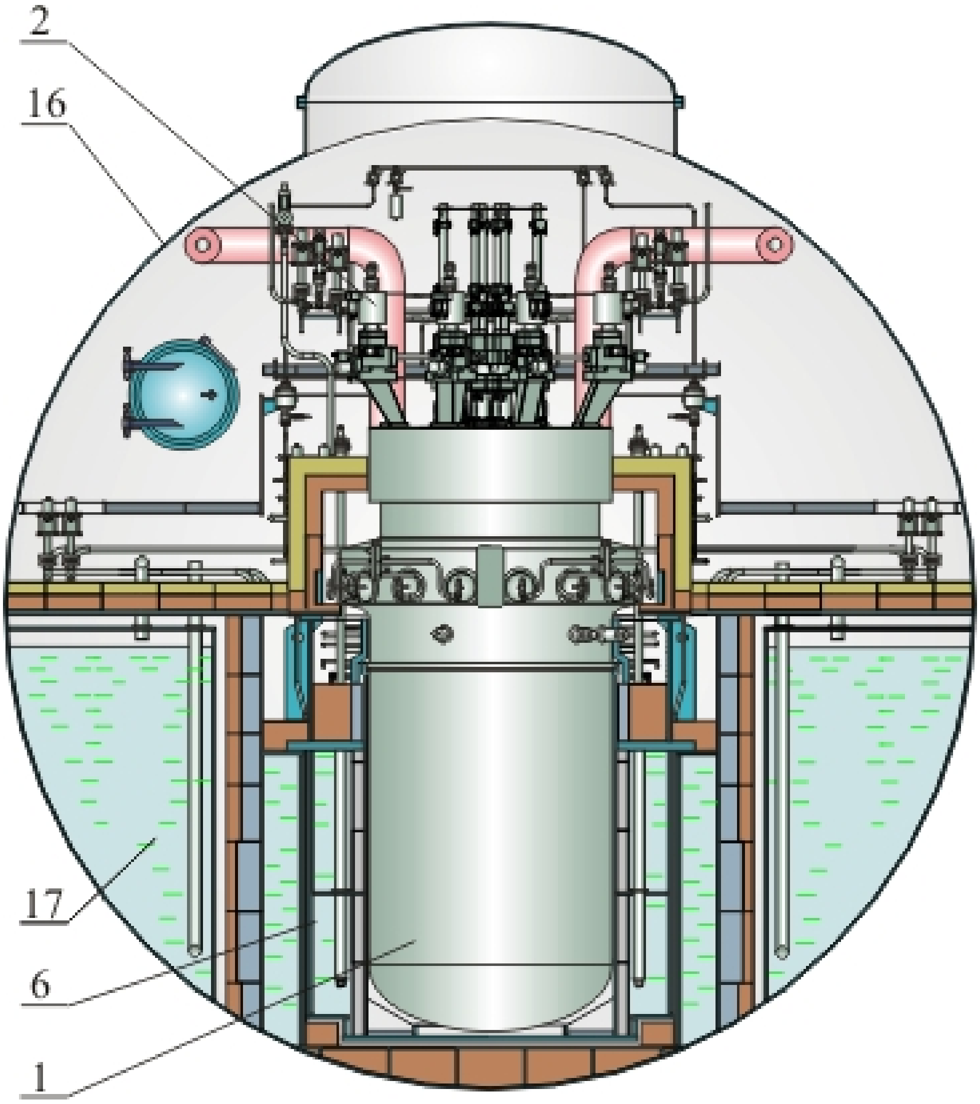
1 – Reactor
2 – CPS drive
3 – Pressurizer
4 – High pressure gas cylinders
5 – Hydraulic accumulator of EHRM
6 – Metal and water shielding tank
7 – Primary circuit filter
8 – Purification and aftercooling system pump
9 – Third circuit filter
10 – Third/fourth circuit heat exchanger
11 – Third circuit pump
12 – Make-up pump
13 – Hydraulic accumulator of reactor vessel
14 – Soluble poison tank
15 – Make-up water storage tank
16 – Protective shell

EHRM is for emergency heat removal system

FIG. V-6. ABV reactor unit of a land-based nuclear power plant (view 1).

Layout of the reactor unit for a floating NPP is given in Fig. V-8 and V-9.

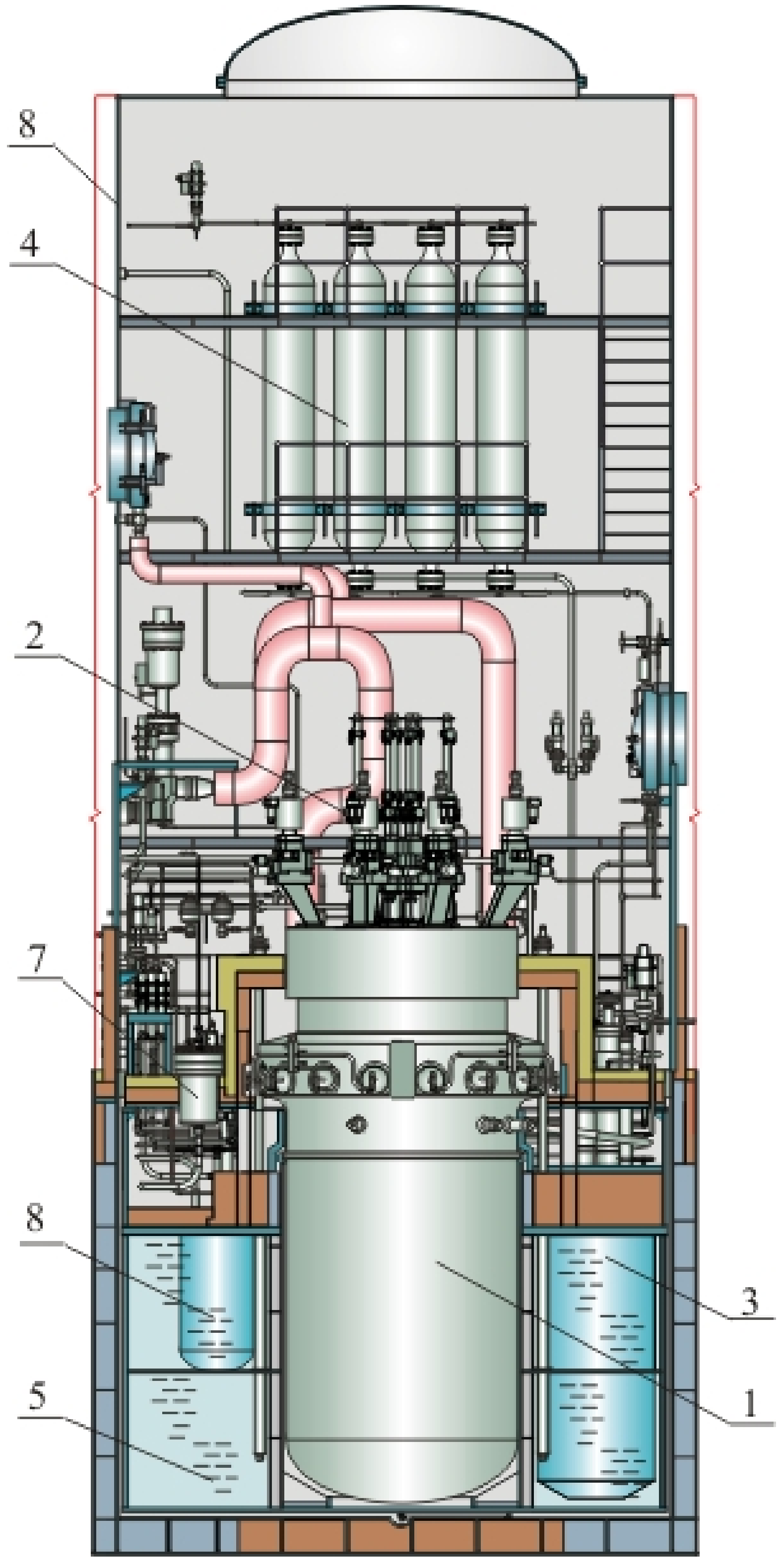
The reactor unit of a land-based ABV nuclear power plant (see Fig. V-6 and V-7) includes the steam-generating unit, the biological shielding, and equipment of the main and auxiliary systems providing heat removal from the reactor and safe reactor operation under normal and emergency conditions.



1 – Reactor
2 – CPS drive
6 – Metal and water shielding tank
16 – Protective shell
17 – Barbotage tank

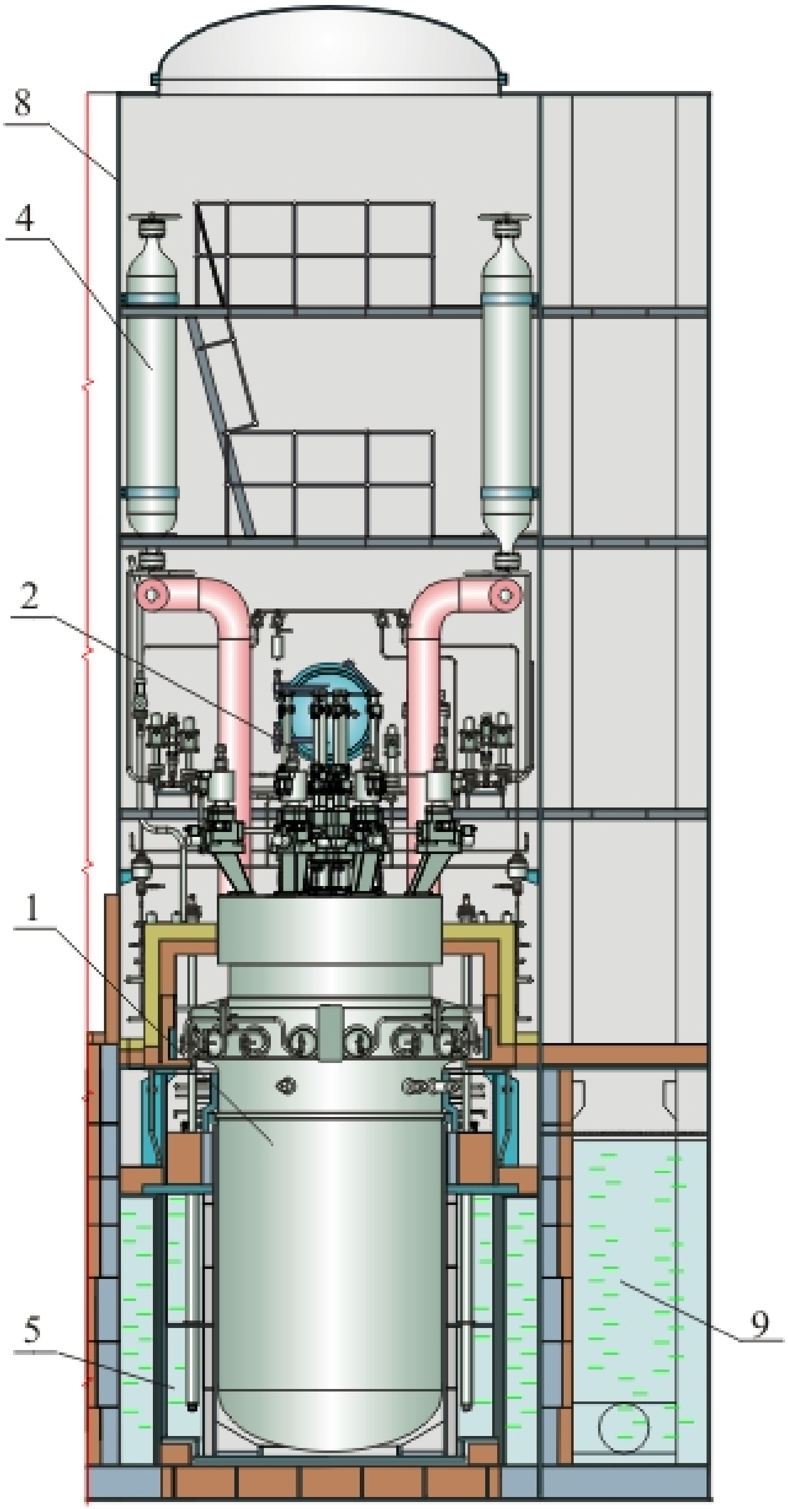
FIG. V-7. ABV reactor unit of a land-based nuclear power plant (view 2).

All equipment of the reactor unit is enclosed in a cylindrical shell intended to localize the consequences of accidents with primary circuit depressurization. The outer diameter of the protective enclosure is 8.5 m, the length is 13 m. The total mass of the reactor unit and protective enclosure is 600 t.



1 – Reactor	6 – Primary circuit filter
2 – CPS drive	7 – Purification and aftercooling system
3 – Pressurizer	8 – Protective shell
4 – High pressure gas cylinders	9 – Barbotage tank
5 – Metal and water shielding tank	10 –Ion-exchange filter

FIG. V-8. Reactor compartment of a floating nuclear power plant (view 1).



1 – Reactor	6 – Primary circuit filter
2 – CPS drive	7 – Purification and aftercooling system pump
3 – Pressurizer	8 – Protective shell
4 – High pressure gas cylinders	9 – Barbotage tank
5 – Metal and water shielding tank	

FIG. V-9. Reactor compartment of a floating nuclear power plant (view 2).

The reactor unit of a floating nuclear power plant (see Fig. V-8 and V-9) includes the steam-generating unit with part of the biological shielding and equipment of the main and auxiliary systems. The rest of the biological shielding and equipment are located in the compartments of a floating power plant.

The unit equipment is enclosed in a protective shell. The protective shell is 5.1 m long, 4 m wide and 7.5 m high.

The reactor units for both land-based and floating nuclear power plants can be transported to the site by trucks or by water. Separate equipment supply or equipment supply in the form of smaller units could be realized upon the request of a customer.

Plant layout of a land-based NPP with the ABV reactor installation is shown in Fig. V-10; that of a floating NPP is in Fig. V-11.

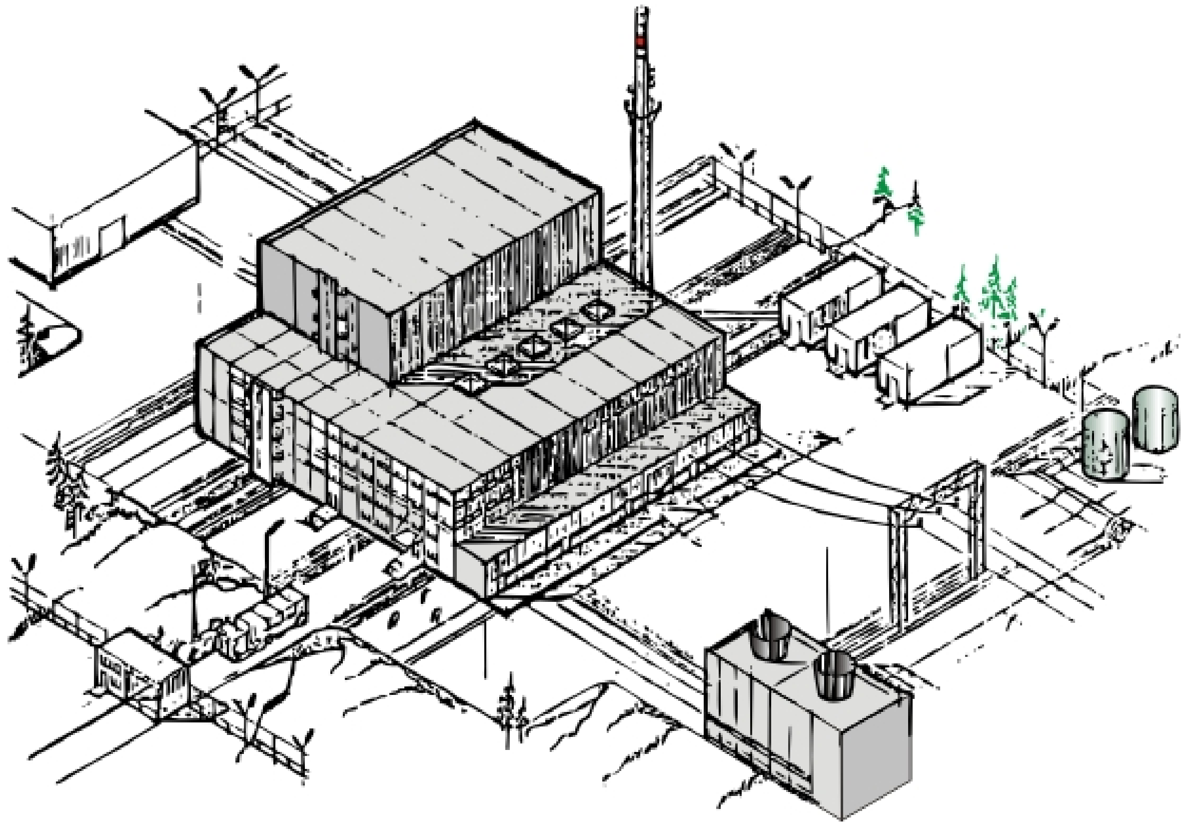


FIG. V-10. Layout of a land-based NPP with the ABV reactor.

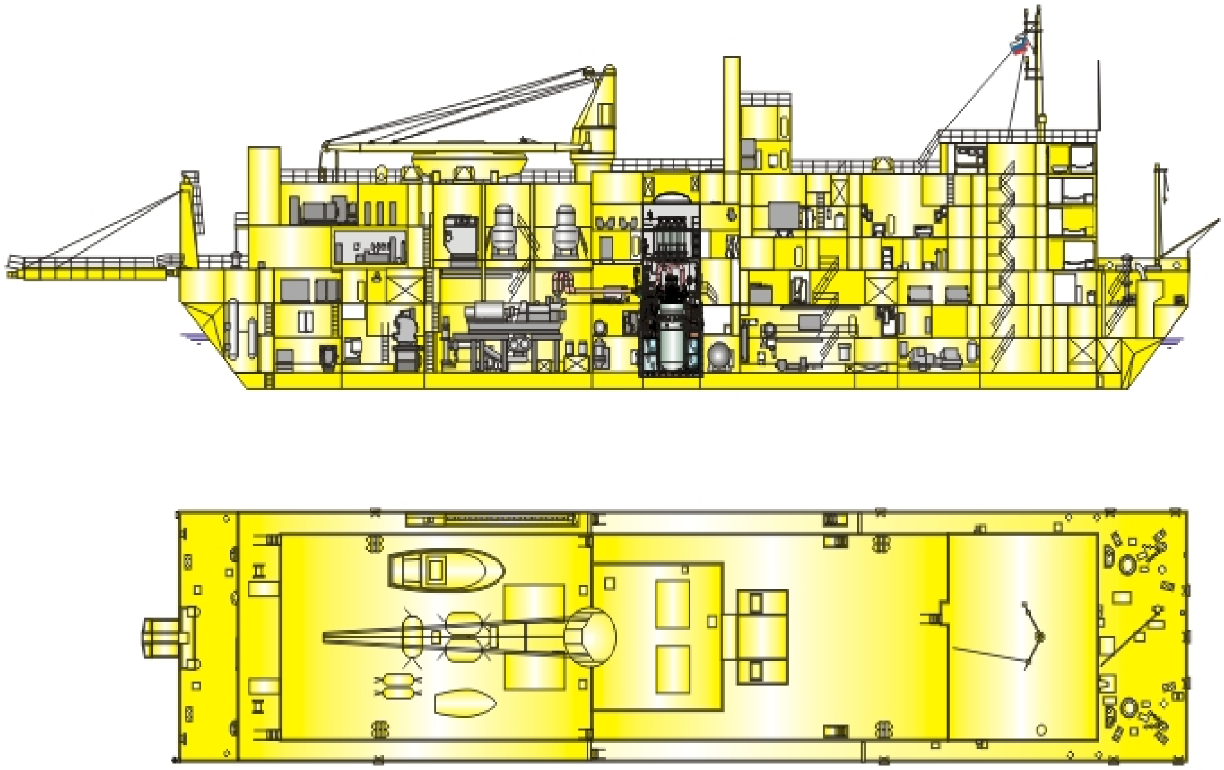


FIG. V-11. *Layout of a floating NPP with the ABV reactor.*

As shown in Fig. V-8, V-9 and V-11, for a floating NPP, the reactor and the main equipment and the pipelines and valves operated under primary circuit pressure are arranged in the reactor compartment. Auxiliary equipment is arranged in the compartment adjoining the reactor compartment on the side of the turbine compartment.

The diameter of a strong vessel for the reactor compartment is 9 m; the diameter of a light vessel is 11 m. The reactor compartment height is 11 m including a ball-ended bulkhead of 4.5 m. Auxiliary equipment of the reactor plant occupies 4.2 m of a space adjacent to the partition wall.

The reactor compartment and passage-ways can withstand pressures of 0.2 MPa, localizing radioactive products in the event of primary circuit depressurization.

Auxiliary equipment of the reactor plant is arranged in rooms adjacent to the reactor compartment. The following equipment and systems are located in these premises:

- Heat exchangers of the third/fourth circuit;
- Pumps, pipelines and valves of the third circuit;
- Pumps, pipelines and valves of the fourth circuit;
- The expansion vessel;
- The water storage tank of the reactor vessel cooling system;
- Pumps, water and boron solution storage tank, pipelines and valves of the make-up system;
- Compressed air cylinders and air distributor boards of the pneumatically actuated valve control system;

— Electrical cabinets and cable routing of a power supply system;
— The reactor compartment vacuum system; and
— The conditioning system.

Equipment (tanks and heat exchangers) of the emergency heat removal system could be arranged in the equipment compartment section, in the adjacent rooms, or in the inter-board space (for the underwater design), depending on the purpose of the plant.

REFERENCES

[V-1] MITENKOV, F.M., et al., ABV type reactor of low power level, Power Construction, pp. 21–26, (May 1993).

[V-2] PANOV, Yu.K., FADEEV, Yu.P., BARANEAEV, Yu.D., The water desalination complex based on the ABV-type reactor plant, Floating Nuclear Energy Plants for Seawater Desalination, IAEA-TECDOC-940, pp. 23–28, Vienna (1995).

[V-3] BELYAEV, A.A., KURACHENKOV, A.V., PANOV, Yu.K., SAMOILOV, O.B., Nuclear energy sources on the basis of integral reactor, Nuclear Engineering and Design, **173**, pp. 131–141 (1997).

[V-4] ANTONOVSKY, G.M., KIRYUSHIN, A.I., MITENKOV, F.M., PANOV, Yu.K., PONOMAREV-STEPNOI, N.N., PWR-Type reactors developed by OKBM, Nuclear News, ANS, pp. 33–37 (March 2002).

ANNEX VI

KLT-20 REACTOR FOR A FLOATING POWER UNIT

OKBM,

Russian Federation

VI-1. General information, technical features and operating characteristics

VI-1.1. Introduction

The KLT-20 reactor installation is being designed by the Experimental Design Bureau of Machine Building (OKBM, Nizhny Novgorod) as a power source for floating nuclear power plants (NPPs). At present, the activities are most advanced for the project of a pilot floating heat and power plant with the KLT-40S reactor installations [VI-1 to VI-5], advanced analogues of the commercial KLT-40 reactors of the Russian icebreaker fleet.

For the KLT-40S, detailed design of the reactor unit and floating power unit has been developed and approved; the Rostechnadzor of Russia license for plant siting and floating power unit construction in Severodvinsk (Russian Federation) has been obtained.

The KLT-20, based on a pressurized light-water reactor of 20 MW(e), is a two-loop modification of the KLT-40S reactor with several improvements in the main equipment and a long-refuelling interval, achieved with the enrichment of less than 20%. The reactor design with a long refuelling interval was developed based on the engineering solutions of the pilot KLT-40S reactor installation; different from it, the KLT-20 provides for no on-site refuelling.

The refuelling, radioactive waste management and repairs of a floating NPP with the KLT-20 would be performed at special maintenance centres. The infrastructure of nuclear ship maintenance centres in Russia could be used for these purposes.

VI-1.2. Applications

The KLT-20 is a small power source for floating NPPs and power desalinating complexes. Possible applications are the following:

- Power generation;
- Heat and power cogeneration for district heating in coastal regions;
- Seawater desalination; and
- Emergency source of heat and power for natural disasters.

VI-1.3. Special features

Special features of the KLT-20 are the following;

- It is a floating NPP;
- The realization of a long operation cycle with off-site refuelling.

In addition to this, a unified nuclear propulsion and power complex consisting of nuclear ships and floating cogeneration plants based on a common type of reactor installation and supported by a common maintenance infrastructure could effectively solve the problems of energy supply to autonomous consumers, such as those in the North-eastern regions of the Russian Federation, at minimum cost.

VI-1.4. Summary of major design and operating characteristics

A schematic diagram of the KLT-20 reactor installation is given in Fig. VI-1.

A conventional two-circuit scheme is used to remove core heat; the core is cooled by water of high purity, which also acts as a moderator. Hot coolant is cooled in a once-through steam generator (SG) where slightly superheated steam, supplied through the main steam line to the turbine-generator plant, is generated.

The primary circuit coolant is circulated within the two-loop reactor unit consisting of high-pressure vessels connected by short nozzles. Removable parts of the main equipment (SG pipe system, main circulating pump (MCP), reactor core) are located in the vessels.

Steam lines of the auxiliary steam system extend from the main steam line; they supply steam to heat exchangers of the district heating system, to the process condenser, oil heater and for own needs of the plant.

After the turbine, spent steam is supplied to the turbine condenser; steam condensation is achieved by seawater cooling. Condensate is then supplied to the condensate pump intake whereupon, driven by pump head, it is supplied to the feedwater system and, specifically, to the feedwater pump intake. Feedwater is supplied through the feed valve to the SGs of the secondary circuit. Major design and operating characteristics of the KLT-20 are summarized in Table VI-1.

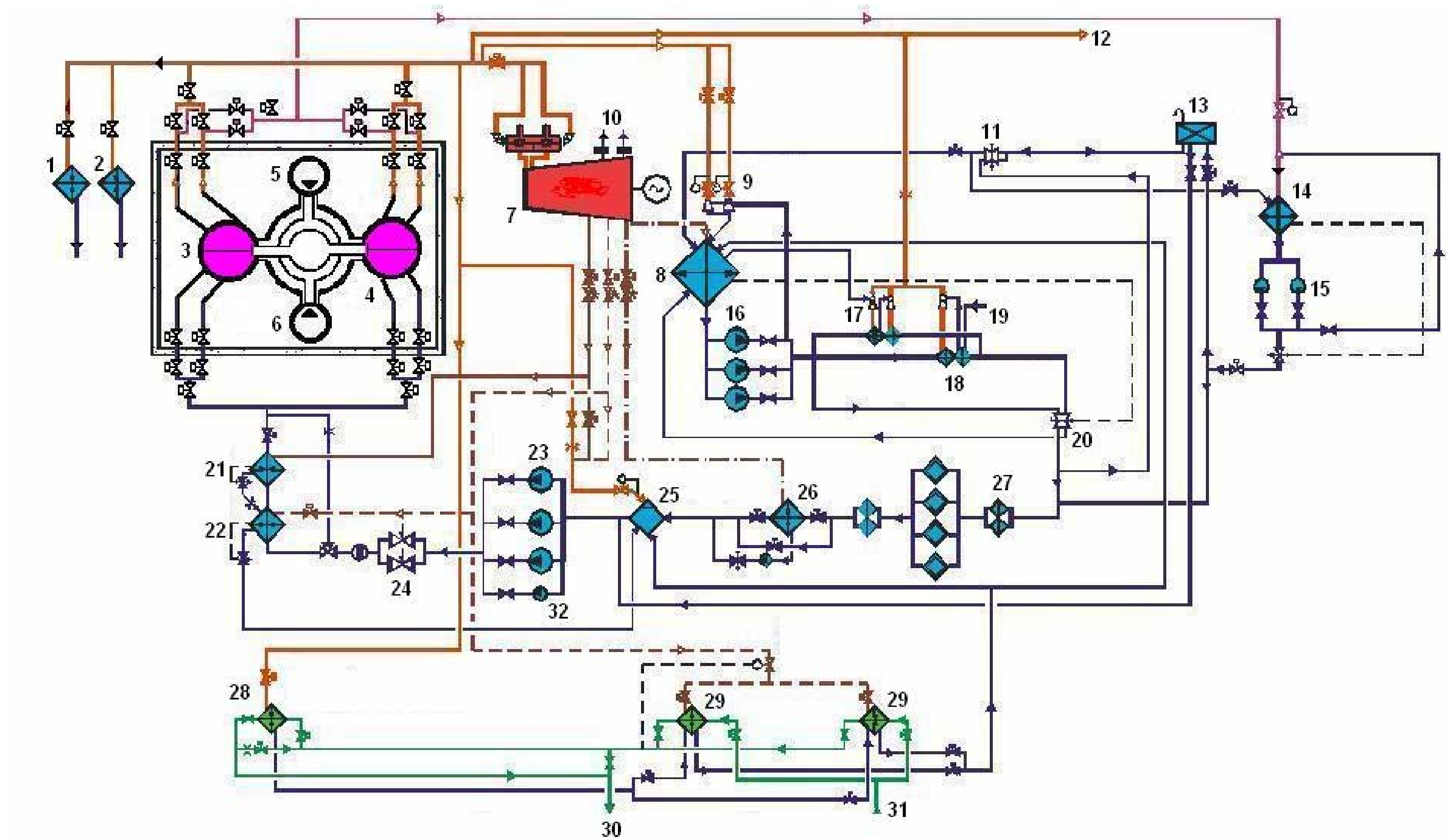
TABLE VI-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF THE KLT-20

CHARACTERISTIC	VALUE
Major design characteristics	
Rated power, MW — Thermal; — Electric;	 70 20
Operation mode	Load follow, including daily power changes
Capacity factor	0.85–0.9
Fuel	
Fuel type	UO_2 pellets in inert matrix
Fuel element	Fuel pins similar in design to standard fuel elements of nuclear ice-breakers
Fuel assembly	Hexagonal fuel assemblies with ducts
Fuel enrichment	Not more than 20% by weight
Fuel type	Metal ceramics (cermet); UO_2 + silumin dispersed composition
Coolant	Water (H_2O)
Moderator	Water (H_2O)

CHARACTERISTIC	VALUE
Core	
Number of fuel assemblies	121
Fuel assembly flat-to-flat size, mm	97
Diameter, mm	1219
Height, mm	1800
Vessel system	
Overall height, mm	~9000
Diameter, mm	6.5
Operating mass, t	~150
Diameter of reactor vessel at core level, inner/outer, mm	1920/2176
Structural materials	
Core	
Fuel element cladding	Zirconium alloy
Fuel assembly structural elements	Zirconium alloy
Vessel system	
Reactor vessel	Heat-resistant pearlite steel with anticorrosive facing
Steam generator vessel	The same as above
Vessel of the hydraulic chamber of circulating pump	The same as above
Steam generator pipe system	Titanium alloy
Reactor internals	Stainless steel 08Cr18Ni10Ti
Power conversion cycle	
Cycle type	Steam-turbine cycle with slightly superheated steam
Number of circuits	2
Reactor type	Modular pressurized water thermal reactor
Number of heat removal loops	2
Neutron-physical characteristics	
Reactivity coefficients	Rated values
Reactivity coefficient on coolant temperature (taking into account coolant density changes), 1/°C	(40–50) 10^{-5}
Reactivity coefficient on coolant density (without taking into account coolant temperature), $1/(g/cm^3)$	0.25–0.30
Reactivity coefficient on fuel temperature, 1/°C	–(1.8–2.0) 10^{-5}
Total void reactivity, %	~ – 60

Characteristic	Value
Peaking factors	
Maximum for fuel assembly, over the whole core lifetime	~1.35
Maximum for the core	~2.0
Approaches used to reduce peaking factors	Enrichment zones and burnable poisons
Reactivity control	
Compensation of initial reactivity margin	Fuel elements with burnable poison based on gadolinium
Reactivity control (continuation)	
Compensation of temperature and power effects of reactivity, reactivity margin for core poisoning by xenon-135 and samarium-149, operating margin for reactivity changes under reactor power changes and for maintaining core subcriticality in a cold unpoisoned state	Electromechanical reactivity compensation system, control elements placed in central displacers of fuel assemblies (one rod per fuel assembly) are integrated into eight CPS control rods (compensation groups)
Emergency protection	Six scram rods in leak-tight sleeves placed in the centres of six fuel assemblies and eight control rods of the compensation group (101 absorber elements)
Core sub-criticality in a ‘cold’ state under a stuck event of the most effective control rod at the upper limit stop switch, with other control rods being at a lower limit stop switch, %	≥1.0
Thermal-hydraulic characteristics	
Primary circuit parameters	
Circulation type	Forced circulation, canned MCPs
Primary circuit coolant flow rate, t/h	1650
Coolant temperature at core inlet, °C	289
Coolant temperature at core outlet, °C	317
Coolant velocity in the core, m/s	1.6
Primary circuit coolant pressure, MPa	12.7
Maximum fuel temperature, °C	390
Average fuel temperature in the core, °C	360
Maximum temperature of fuel element cladding, °C	330
Average temperature of fuel element cladding, °C	300
Maximum acceptable fuel temperature, °C	550

CHARACTERISTIC	VALUE
Maximum acceptable temperature of fuel element cladding, °C	700
Minimum margin to heat transfer crisis	1.4
Secondary circuit parameters	
Steam pressure beyond the steam generator, MPa	6.0
Steam output, t/h	115
Steam temperature at steam generator outlet, °C	305
Feedwater temperature, °C	170
Operating cycle (operation without on-site refuelling)	
Refuelling mode	Once-at-a-time reloading of all fuel assemblies in the core
Refuelling interval, years	~10
Number of reloaded fuel assemblies	121 (single loading)
Fuel lifetime between refuellings, effective hours	70 000
Uranium inventory, kg	1680
Operating cycle (operation without on-site refuelling)(continuation)	
Uranium-235 inventory, kg	322.5
Average uranium enrichment in the core, %	19.2
Average fuel burn-up, g/cm^3	0.56
Specific consumption of ^{235}U, g/MW·day	1.8
Design service lifetime	
Lifetime of the vessel system, years	40
Lifetime of steam generator piping system, years	20
Lifetime of the main circulating pump	20
Economics	
Construction cost of a floating NPP, US$ million	~100
Specific capital investments for construction, US$/kW(e)	~2500
Projected primary cost of generated electricity (condensation mode), US$ cent/kW·h	~4
Payback period, years (beginning from the commencement of operation)	12–15



1 – To low pressure SG; 2 – To desalination plant; 3 – SG 1; 4 – SG 2; 5 – Primary circuit circulation pump 1; 6 – Primary circuit circulation pump 2; 7 – Turbine generator unit; 8 – Main condenser; 9 – Exhaust valves; 10 – Release to the atmosphere; 11 – Pressure level regulator; 12 – Steam to glands; 13 – Equalizing tank; 14 – Process condenser; 15 – Process condensate pumps; 16 – Electric condensate pump; 17 – Main ejector; 18 – Glands exhaust system ejector; 19 – Steam from glands; 20 – Condensate level regulator; 21 – High pressure heater 2; 22 – High pressure heater 3; 23 – Electric feed pump; 24 – Feed valves; 25 – Deaerator; 26 – Low pressure heater; 27 – Ion exchangers; 28 – Peak heater; 29 – Main heater; 30 – Process water to the coast; 31 – Process water from the coast; 32 – Standby feed pump.

FIG. VI-1. Schematic of the KLT-20 reactor installation.

VI-1.5. Outline of fuel cycle options

Ensuring maximum operating period between refuellings was a priority task during design development of the KLT-20; considering this factor, a once-at-a-time core loading concept has been accepted. The increase of natural uranium consumption related to once-at-a-time core loading as compared to partial refuelling is compensated by the capacity factor increase as well as by the specific cost decrease for refuelling.

The IAEA recommendations on proliferation resistance were accepted as obligatory; therefore, the initial uranium enrichment was selected to be not more than 20 % by weight.

To provide maximum fuel burn-up and considering the accepted limitation on uranium enrichment, the fuel lattice parameters that affect water-uranium ratio have been optimized.

To provide a maximum fuel lifetime at a given fuel loading and to meet the selected criteria on power peaking, a scheme of the operating reactivity margin compensation by central group of absorbing rods, previously developed for icebreaker reactor cores, was applied.

To decrease gadolinium under-burnup and provide acceptable power peaking during the lifetime, the profiling of allocation of fuel zones and burnable poisons in the core similar to that developed for icebreaker reactor cores is used. A minimum reactivity swing during fuel and poison burn-up is provided through an appropriate selection of burnable poison rod parameters and by radial and axial profiling of their allocation in the core. The parameters of cermet fuel selected for the KLT-20 core ensure the load follow mode of plant operation without a limit on customer requirements.

A closed fuel cycle with radiochemical reprocessing of spent fuel would be used when floating NPPs with the KLT-type reactors will be widely deployed. For icebreakers with nuclear reactors of the KLT type, the reprocessing of spent fuel is currently performed at the existing reprocessing plants. For the deployment of floating NPPs with the KLT type reactors it would be necessary to upgrade the technological line for nuclear icebreaker dispersed fuel reprocessing to include the reprocessing of a UO_2+silumin fuel composition.

VI-1.6. Technical features and technological approaches that are definitive for KLT-20 performance in particular areas

VI-1.6.1. Economics and maintainability

The design features contributing to an enhancement of the economic performance and competitiveness of the reactor installation and the floating power unit are as follows:

- Full factory readiness of a floating power unit based on the use of industrial production processes for manufacturing of the floating NPPs; the NPP is assembled under shipyard conditions and delivered to the customer already tested and completely ready for operation;
- The minimum scope and cost of capital construction needed to arrange a floating plant location in a water area;
- The absence of a need to create transportation links and energy communications and preparatory infrastructure required for land-based NPP construction;
- A considerable reduction in the construction period (down to 4 years);
- A long refuelling interval with the refuelling, radioactive waste management and repairs being provided at special maintenance centres; elimination of on-site spent fuel storage;

- The absence of on-site refuelling, radioactive waste management and repairs simplifies operation and contributes to a reduction in the operation and maintenance costs;
- The refuelling and maintenance costs could be minimized by using the infrastructure of nuclear ship maintenance centres available in the Russian Federation; the requirements for local labour skills in developing countries could also be reduced;
- The concept of a floating NPP makes it easy to realize a "green lawn" concept on the site of a floating NPP operation or, if necessary, to replace the exhausted floating plant with a new one, contributing to a reduction of the decommissioning costs.

VI-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Enhanced fuel utilization efficiency and a reduction of natural uranium consumption is effected by the following engineering solutions:

- Improvements in the nuclear fuel and fuel cycle of the VVER-1000 reactors; realization of a closed nuclear fuel cycle;
- An increase in the fuel burn-up, possible due the geometric stability and improved operation reliability of the skeleton-design fuel assemblies.

A reduction in the amount of radioactive wastes for the KLT-20 based floating NPPs results from the following design and conceptual features:

- The use of a standard pressurized primary circuit proven in operation for shipboard reactor plants;
- The use of a closed system of primary coolant purification;
- The use of "wasteless" technologies for coolant treatment;
- The use of the state-of-the-art low waste technologies for radioactive waste reprocessing;
- Performance of the refuelling under controlled conditions in special maintenance centres.

Radiation safety of a nuclear power plant with the KLT-20 reactor installation meets the requirements enforced in the Russian Federation for limiting the irradiation impacts on personnel, population and the environment during operation, including the abnormal operation occurrences and accidents and a severe accident with fuel damage.

The KLT-20 design provides for a set of technical features and measures to minimize the possible level of personnel and population irradiation; the most important of them are as follows:

- Effective biological shielding;
- A closed system of primary coolant purification and boron removal that excludes leakages of the radioactive medium of the primary circuit into the atmosphere during plant operation;
- The use of intermediate loops of cooling water;
- A protective shell with shared roles of protection against natural and human induced external impacts and resistance to the internal accident impacts;
- Strict measures of radiation control;

- The division of plant production area into two zones: a zone of controlled access and a zone of free access;
- Establishment of a sanitary and protection area and a radiation-control area near the NPP.

VI-1.6.3. Safety and reliability

Safety concept and design philosophy

The KLT-20 safety concept provides for [VI-6]:

- Incorporation of the state-of-the-art safety requirements and safety principles developed by the world engineering community and summarized in the IAEA safety standards and national regulations in Member States;
- Incorporation of the engineering solutions and equipment proven via long operating experience of shipboard reactor installations, also taking into account the disadvantages revealed in the operation;
- Incorporation of experience in the design and validation of next generation nuclear plants; and
- Incorporation of features to minimize radiation impact during radioactive media treatment.

The general safety objective is to ensure the protection of staff, population and environment against radiation hazards by effective application of the engineered safety features and protection measures. It addresses all lifecycle stages of a nuclear cogeneration plant in all its operational states.

The technical safety objective is to realize effective measures for the prevention of accidents and limitation of the radiation consequences of design basis and beyond design basis accidents and to ensure that the probability of severe accidents is reduced to a very low value.

Active and passive systems and inherent safety features

The KLT-20 safety during operation is ensured to a considerable extent by the inherent and passive safety features of the plant, i.e., by the realization of the so-called self-protection principle.

Self-protection of a nuclear power plant is expressed in its ability to prevent initiation and to limit the development and consequences of the initial events which could lead to accidents, achieved via natural feedback mechanisms and passive processes with no requirement for operator intervention and external power supply over a reasonably long period, which could be used by personnel to evaluate the situation and take necessary corrective actions.

The KLT-20 reactor installation incorporates the following inherent and passive safety features:

- Negative reactivity coefficients on the fuel and coolant temperature, and on the specific coolant volume; negative steam density and power (integral) coefficients of reactivity — in all reactor states and at any moment during the core lifetime;
- Passive reactor aftercooling achieved via natural circulation of coolant in the primary circuit and in the passive channels of the residual heat removal system (RHRS);
- High heat conductivity of the fuel composition defining a relatively low fuel temperature and correspondingly, low stored energy;

- High heat capacity of the reactor installation due to a high heat capacity of the primary coolant and metal structures;
- The use of a "soft" pressure suppression system under all operating modes;
- Large safety margin for depressurization pressure of the primary system under pressure increase in accidents;
- The drop of emergency control rods accomplished with the force of springs when the hold-up electromagnets of the emergency protection actuators are de-energized;
- Effective elimination of core uncovery in loss of coolant accidents due to the primary pipeline arrangement above the core;
- The elimination of large-diameter primary pipelines through applying a compact design of the steam-generating unit with short nozzles between the main equipment;
- The use of narrowing devices in the nozzles connecting the primary circuit systems to the reactor, to limit blowdown flow rate of the water coolant, and selective placement of these connecting nozzles to provide a fast transition to the steam blowdown of the primary coolant under depressurization of the pipelines;
- Arrangement of the conditions to realize the concept of "leakage before breakdown" in application to primary system components.

Active and passive safety systems of the KLT-20 (see Fig. VI-2) perform the following safety functions:

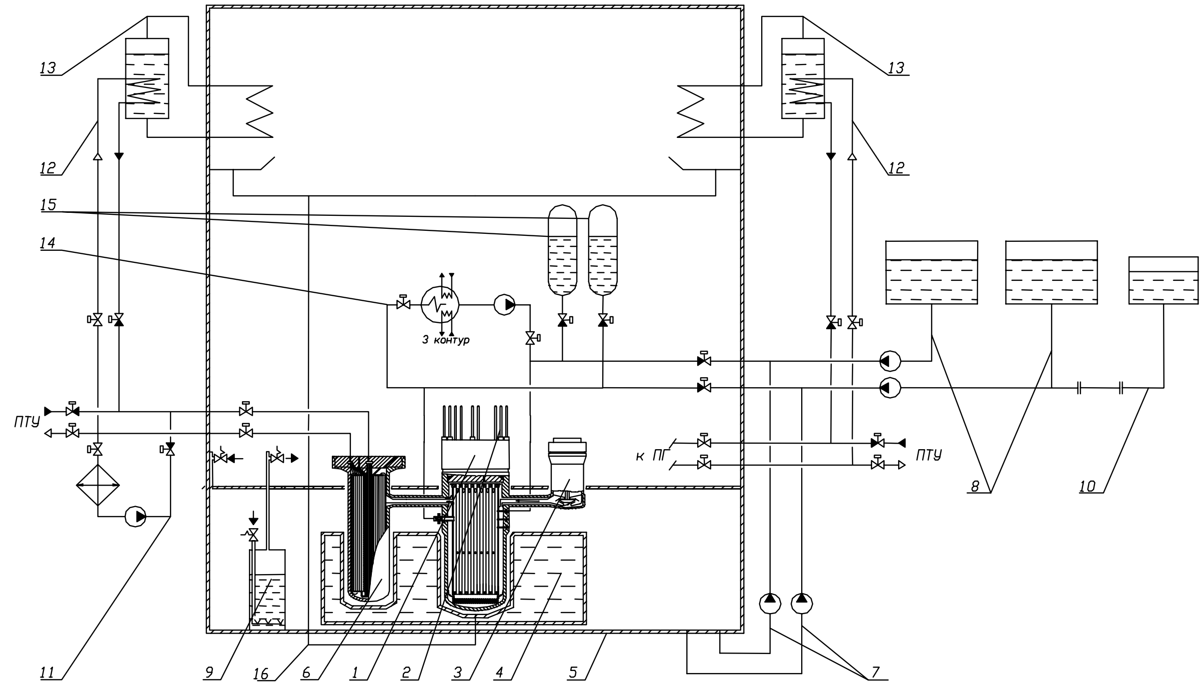
- Emergency reactor shutdown;
- Primary circuit emergency heat removal;
- Emergency core cooling;
- Localization of released radioactive products.

Active safety systems are the following:

- System of reactor shutdown via insertion of the shim control rods in the electromotive mode;
- System of emergency aftercooling via the steam generator with steam dump to the process condenser;
- System of emergency aftercooling via the primary/third circuit heat exchanger;
- System of emergency water supply from the emergency core cooling system (ECCS) pumps and recirculation pumps;
- System of filtration of the release from the protective enclosure.

Passive safety systems are the following:

- System of reactor shutdown via gravity-driven insertion of the shim control rods;
- System of the emergency insertion of control rods driven by the pressing force of the accelerating springs, actuated when the hold-up electromagnets are de-energized;
- Passive system of emergency aftercooling through the steam generator (natural circulation in all heat removal circuits, tank water evaporation);
- System of emergency water supply from the hydraulic accumulators;
- The protective shell and stop valves (normally in a closed position) in the auxiliary systems of the primary circuit and adjoining systems;
- Passive system of external aftercooling of the reactor vessel.



1 – Reactor; 2 – Control rod actuators; 3 – Primary circuit pump; 4 – Metal-and-water shielding tank; 5 – Protective shell; 6 – SG; 7 – Recirculation channels of ECCS; 8 – Active channels of ECCS; 9 – Barbotage sub-system for emergency pressure suppression in the protective shell; 10 – Soluble poison injection system; 11 – Active channel of RHRS through SG; 12 – Passive channel of RHRS through SG; 13 – Condensation sub-system for emergency pressure suppression in the protective shell; 14 – Active channel of RHRS through primary/third circuit heat exchanger; 15 – Passive channels of ECCS; 16 – System for reactor caisson fill-up with water

FIG. VI-2. Safety systems of KLT-20.

Performance of the safety system functions is provided in the scope required considering external natural and human-induced events and internal events caused by accident conditions. Functioning of the safety systems is provided considering potential failures such as a single failure or a common cause failure resulting from a single failure, or an impact of a personnel error. To ensure reliability of safety systems, the principles of redundancy, diversity and physical separation are applied, as well as certain measures such as:

- The use of systems combining the principles of passive and active operation and elements meeting the principle of a safe failure to the extent possible;
- Automation of the control functions and redundancy of the protection systems achieved through the use of self-actuated devices (direct-action devices);
- The application of a conservative approach in the design of protective barriers and safety systems and in the selection of the scope of the initiating events, accident scenarios, key accident parameters and characteristics, and design margins.

A two-channel scheme with the internal redundancy of active elements such as valves and pumps is provided for the majority of the KLT-20 safety systems. The use of a two-channel scheme of safety systems under specific conditions typical of floating structures (and resulting in the necessity to save room space and equipment weight compared with land-based NPPs) makes it possible to reduce the number of bulky equipment such as tanks and heat exchangers.

According to the defence-in-depth principle, the KLT-20 design provides for a protective shell, which is a leak-tight metallic structure containing the reactor and the equipment and systems with radioactive coolant, Fig. VI-3.

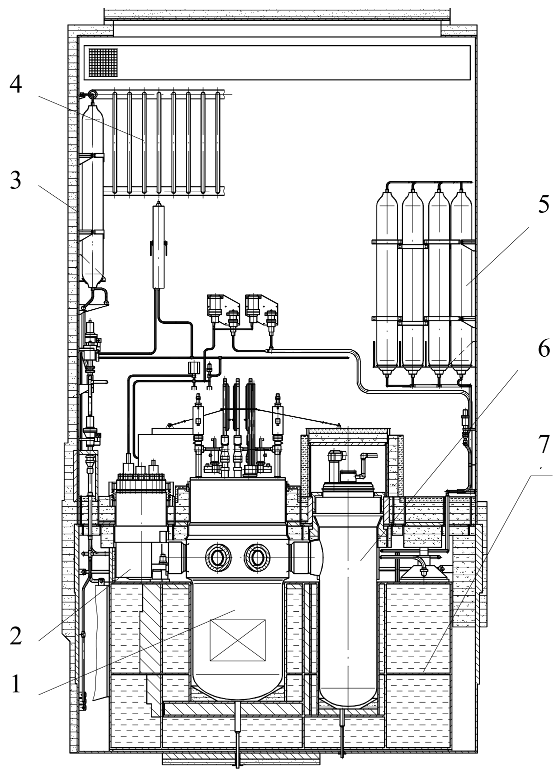
The protective shell is designed for internal pressure corresponding to the design and beyond design basis accidents and taking into account temperature stresses arising from accidents. The allowable leakage accepted in the design is based on the consideration of reducing the emergency-planning zone around the plant.

Taking into account ship requirements, the protective shell is designed to withstand external events including sinking of a floating power unit.

Systems that make use of seawater are not present in the protective shell, which aims to eliminate corrosive damages to the equipment and pipelines that might otherwise result from contacts with seawater.

The protective shell is equipped with a passive emergency pressure decrease system with heat removal to the ultimate heat sink. This system includes the following:

- A barbotage sub-system; including the barbotage tank in the reactor compartment of the protective shell and by-pass channels for the gas-steam mixture with safety devices;
- A condensation sub-system; including two channels, the heat exchangers of which are placed in the equipment room and use water of the emergency aftercooling tanks for their operation (heat removal is provided during not less than 24 hours with no external power supply).



1 – Reactor; 2 – Main circulating pump; 3 – Protective shell; 4 – Pressure suppression condensation system; 5 – High pressure gas cylinders; 6 – Steam generator; 7 – Metal and water shielding tank

FIG. VI-3. Schematic of the protective shell.

Protection of the safety-related systems against external impacts is provided by a protective enclosure. The protective enclosure is a water and gas-proof structure built as a part of the ship hull; it includes protective shells for the plant and the storage of liquid and solid radioactive waste, and additionally limits the leakage of radioactive substances into other parts of the floating power unit and into the environment, in case of an accident.

Design basis and beyond design basis accidents

Analysis of the design basis and beyond design basis accidents for NPPs based on a floating power unit with the KLT-20 reactor is being performed using a set of calculation codes developed by OKBM and proven in calculations of stationary and transient modes of ice-breaker reactor operation.

The results of the analysis of design basis accidents with inadvertent reactivity addition and heat removal failure show that the emergency processes evolve without posing a threat to reliability of heat removal from the core.

In design basis accidents with coolant loss, the core remains covered and fuel elements are not heated above nominal temperatures. The protective shell localizes the coolant flowing out of the reactor. Personnel intervention is not required for 8 hours after the initiating events. The irradiation doses for population are lower than the allowable levels requiring protective measures.

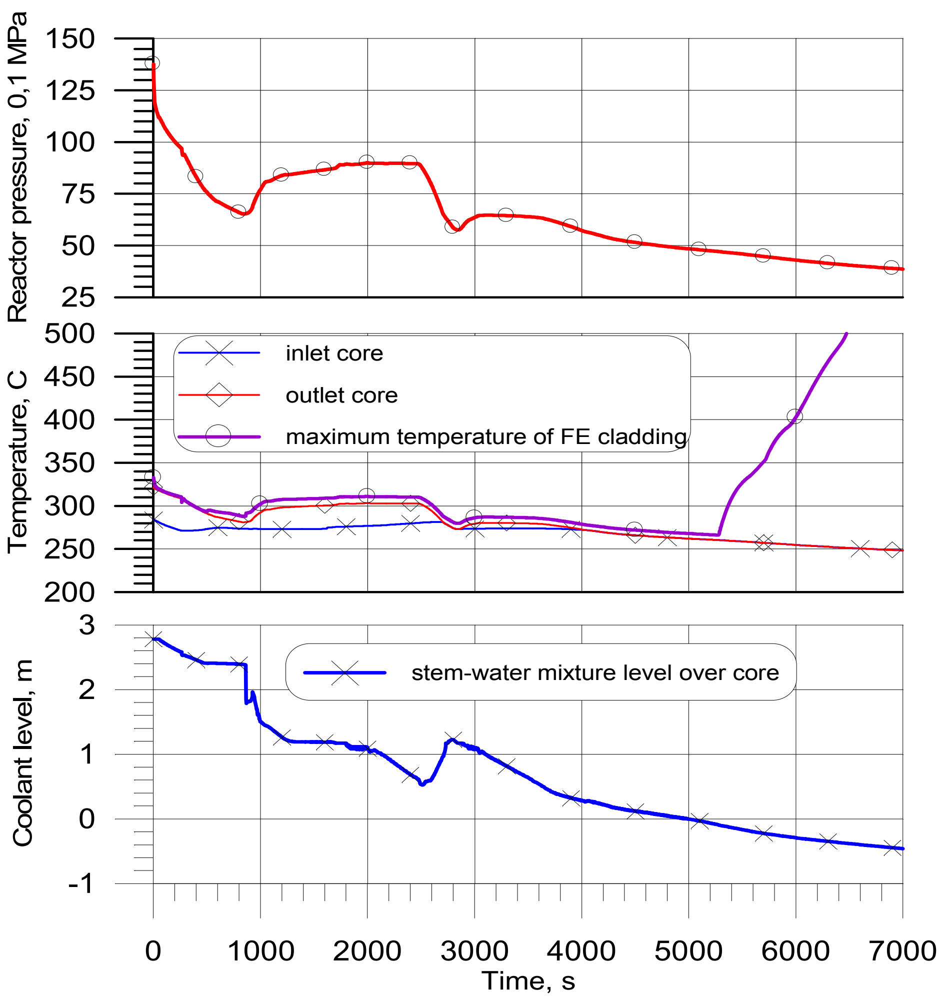


FIG. VI-4. Primary pipeline break accident.

In beyond design basis accidents with inadvertent reactivity addition and heat removal failure, the ultimate allowable value of the primary circuit pressure is not exceeded and the temperature of the fuel element claddings does not exceed values at which the integrity becomes challenged.

In beyond design basis accidents with primary coolant loss and the failure of all ECCS pumps, provision is made for the core to remain covered during a period of not less than 1 hour, Fig. VI-4.

Provisions for safety under external natural and human-induced impacts

Structures, systems, and components of a nuclear cogeneration plant with the KLT-20 reactor installation are developed with consideration of natural and human-induced external impacts typical of a floating NPP location site and transportation, and comply with the OPB-88/97 safety regulations of Russia, the Sea Shipping Register of Russia and other regulations.

The floating NPP safety design incorporates features to cope with natural external events that may affect a floating power unit and the reactor installation, including design basis and maximum postulated earthquakes and design basis human-induced events, such as fuel explosion in moored tankers or service vessels, underwater contact or non-contact explosion, e.g., due to human actions of malevolent character, etc..

The equipment, machinery and safety systems can withstand an acceleration shock load in all directions of not less than 3g and remain operable in the conditions of tilt and heaving, typical of a floating power unit.

The following features and measures are provided for to secure high resistance of a floating NPP to external impacts:

- Protection against wind loads and earthquakes is achieved via an appropriate choice of the structural components for mooring rods and shock absorbers to ensure safe mooring;
- In the area of the middle compartment and the solid radioactive waste storage compartment on the deck, there is a collision protection consisting of steel plating and a framework of required parameters to prevent crashed helicopter penetration into the reactor vessel;
- There is a side collision protection consisting of reinforced plates in the hull and deck plating adjacent to the board and longitudinal stiffening plates of the board to protect against ship collision;
- Protection from grounding is achieved by the separation of the bottom cover from the protective shell structures by horizontal corrugations in the bulkheads;
- The vessel of a floating power unit guards the internal compartments against air shock waves.

Measures planned in response to severe accidents

In recognition of the defence-in-depth principles, the strategy of severe accident prevention is first realized in the plant design and includes measures to control these accidents and limit their consequences.

Measures of severe accident control are aimed at the following:

- Limitation of the scope of core damage;
- Prevention of core melting under high pressure in the primary circuit;

- Maintaining the reactor vessel integrity and retaining core materials inside the vessel;
- Maintaining the protective integrity of the shell with consideration of the impacts that may accompany a severe accident; specifically, there are provisions for hydrogen safety;
- Limitation of radioactive product release into the environment.

In the processes of fuel assembly destruction and melting and fuel performance in severe accidents, features of the KLT-20 such as a relatively low operating temperature of fuel, small quantities of materials and low core power density as compared to typical large capacity power reactors, are important.

A small amount of core melt and lower decay heat release determine the relatively low heat fluxes from the melt to the vessel bottom; in this, the problem of keeping the melt inside the reactor vessel could be solved by external vessel cooling, i.e., by filling the reactor caisson with water in emergencies. Keeping the melted core inside the vessel reduces the consequences of accidents and eliminates some uncertainties associated with maximum loads on the protective shell.

The KLT-20 plant design incorporates a dedicated passive system of emergency vessel cooling (Fig. VI-2) designed to secure the in-vessel retention of corium in severe accidents. The reactor vessel is cooled by boiling water, the generated steam is condensed in the protective shell, and the generated condensate is again supplied to cool the reactor vessel through the system of condensate gathering tanks and pipes.

The calculations indicate that in severe accidents, allowable emergency doses of population irradiation are not exceeded and measures for obligatory population evacuation are unnecessary. The boundary of the area of protection measures is not more than 500 m from the NPP. These results meet in full the IAEA recommendations on safety of advanced reactors [VI-7].

VI-1.6. Proliferation resistance

The following main design features support an enhanced proliferation resistance of the KLT-20 plant:

- The operation without on-site refuelling, which complicates unauthorized access to fuel; such operation assumes that all operations with fuel are accomplished at special maintenance centres;
- The use of a fuel (uranium dioxide in a silumin matrix) with the enrichment by ^{235}U not more than 20 weight %;
- The use of a standard fuel cycle of nuclear icebreaker reactors with the available infrastructure and mechanisms of protection against proliferation**.**

VI-1.6.5. Technical features and technological approaches used to facilitate physical protection of KLT-20

The technical features to support physical protection of NPPs with the KLT-20 reactor correspond to those provided for floating power units with the KLT-40S reactors [VI-2]. Physical protection systems of the plant include the following technical measures:

- Alarm, supervision, and hot link systems;
- Access control system;

- Engineered safety features; and
- Administrative measures.

The physical protection system uses zoning principles; for floating NPPs there is a water area, limited by seawalls, coastal service areas and the floating power unit area, which is a high control zone.

In addition to them, the KLT-20 reactor installation provides for no fresh or spent nuclear fuel being stored in a floating NPP during the whole period of its operation at a site and transportation to a special maintenance centre.

VI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of KLT-20

Non-technical factors and arrangements that could facilitate effective development and deployment of a floating NPP with the KLT-20 reactor installation are similar to those for floating nuclear cogeneration plants with the KLT-40S reactors; they are the following:

- A decision has been made on the construction of a pilot floating NPP with the KLT-40S reactor in Severodvinsk (Russian Federation), to demonstrate the advantages of this technology;
- Floating NPPs offer flexible applications; they can be configured as cogeneration plants to produce potable water, which could considerably enlarge the number of customers as the shortage of potable water becomes more and more acute in many regions of the world;
- Floating NPPs could serve as emergency cogeneration sources in regions of natural disasters;
- Floating NPPs offer the possibility to be based in any coastal region of the world irrespective of seismicity, they could be based near coastal cities; they could be located in remote regions;
- Floating NPPs could be leased under “construct-own-operate” conditions, which may considerably reduce the political and economic restrictions toward nuclear technology use in developing countries;
- In addition to the above mentioned, a refuelling interval of ~10 years, characteristic of the KLT-20 reactor installation, could perhaps add certain assurances to those users that would decide to forego the development of an indigenous fuel cycle.

Last but not least, floating NPPs secure the possibility of an increased local labour involvement for a wide range of countries with shipbuilding and energy-machinery production facilities.

VI-1.8. List of enabling technologies relevant to KLT-20 and status of their development

The main technologies incorporated in the design of a floating NPP with the KLT-20 reactor are listed in Table VI-2.

TABLE VI-2. LIST OF BASIC ENABLING TECHNOLOGIES FOR KLT-20

TECHNOLOGIES	STATUS
Technologies of modular pressurized water reactors for Russian nuclear ships	Well established; the operating experience of multi-purpose shipboard reactors exceeds 6000 reactor-years.
Technologies of the KLT-40S reactor installation developed for a pilot floating NPP to be constructed in Severodvinsk	The design of the reactor installation and floating power unit has been developed; a regulatory body license for construction has been obtained.
Technologies of the AST-500 nuclear cogeneration plant (safety systems and safety design)	A safety review was performed by the IAEA during plant construction

A more detailed list of the enabling technologies relevant for the KLT-20 reactor installation is given below:

- Modular layout of the main equipment, including the reactor, the steam generator, and the main circulating pumps; primary coolant circulation is performed through the connection nozzles according to a coaxial scheme;
- Basic technologies of a vessel-type pressurized water reactor;
- Once-through coil-type modular steam-generators;
- Cermet fuel (uranium dioxide in silumin matrix);
- A long refuelling interval in operation with off-site refuelling;
- Proven metallurgical, press forging and machine-assembly production technologies established in the production of reactor installations for nuclear icebreakers;
- Proven technologies of the equipment mounting, repair and replacement, including those for diagnostic devices and systems used to monitor the equipment state;
- Technologies to ensure minimum radiological impacts on personnel, population and the environment; the reliability of these technologies is proven by long operating experience of nuclear icebreakers and, specifically, by the fact that no accident has resulted in any significant consequences.

The technologies for production of certain structures, systems and components of the KLT-20 reactor installation are listed below; these are the technologies already mastered in commercial production:

- Welding technologies for vessel systems;
- Fabrication techniques for steam generator pipe systems of titanium alloys;
- Fabrication and assembly technologies for coaxial type reactor internals that provide the main path for the coolant circulation;
- Fabrication techniques for canned MCPs;
- Fabrication technologies for fuel elements with dispersed fuel;
- Fabrication technologies to ensure high corrosion and radiation resistance of structural materials;
- Fabrication technologies for elements of normal operation and safety systems, ensuring high reliability of the self-actuated devices, pressurizers, tanks, heat exchangers, pumps, and filters.

VI-1.9. Status of R&D and planned schedule

Design development for the KLT-20 is being performed on the initiative of the Experimental Design Bureau of Machine Building – OKBM (Nizhny Novgorod, Russian Federation) — in cooperation with the Russian Research Centre "Kurchatov Institute" (Moscow, Russian Federation), as indicated in Table VI-3.

TABLE VI-3. MAIN PARTICIPANTS OF THE KLT-20 DESIGN DEVELOPMENT

COMPANY	RESPONSIBILITY AREA
OKB Machine Building (OKBM), Nizhny Novgorod	Chief designer of the reactor installation
RRC "Kurchatov Institute", Moscow	Scientific leader of the project

The development of a floating NPP with the KLT-20 is financed by companies involved in the project.

As the KLT-20 is a modification of the KLT-40S reactor installation with the equipment such as vessel system, main circulating pump, safety system components, and control rod actuators being common to both designs, the results of design development for the KLT-40S are being widely used.

The Federal Agency for Atomic Energy of the Russian Federation (Rosatom) has made a decision to start in 2006 the construction of a floating barge-mounted heat and power cogeneration plant based on the KLT-40S reactor of 150 MW(th). Severodvinsk-city (Arkhangelsk region) in the Russian North-West was selected as location site for the first unit. Detailed design of the plant is completed and the site is licensed. It is planned to construct and commission the plant within a four-year period.

Development of a floating NPP with the KLT-20 reactor installation is at a conceptual design stage. The estimated period for development and deployment of this plant is given in Table VI-4.

TABLE VI-4. PROJECTED TERMS FOR REALIZATION OF THE KLT-20 PROJECT

STAGE	ACTIVITY PERIOD
Detailed design development, including licensing	2 years
Floating power plant construction (including licensing and main engineering development)	4 years

VI-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The KLT-20 design uses to the maximum extent the results of design development, validation, testing and operation of many Russian shipboard reactors and, specifically, borrows from the design experience of a floating NPP with the KLT-40S reactor installation.

The most important areas of further research and development (R&D) are development and validation of fabrication technique for cermet fuel; development of a long-life once-at-a-time refuelled core and validation of its safe and reliable operation over the whole fuel lifetime;

and advancement of the steam generator design. Specifically, R&D on lifetime characteristics of fuel elements and fuel assemblies is required.

For any floating NPP, the construction of a pilot floating power unit is needed to master problems related to the location of a floating NPP intended for power supply to the coastal regions.

VI-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

VI-2. Design description and data for KLT-20

VI-2.1. Description of the nuclear systems

Reactor core and fuel design

The KLT-20 core [VI-5] is a fuel assembly based structure providing for the maximum possible number of fuel elements and maximum possible fuel volume to increase uranium content and keep the fuel enrichment with ^{235}U below 20% by weight. The KLT-20 core dimensions make it possible to allocate it within an icebreaker-type reactor vessel of increased height. Vertical cut of the KLT-20 fuel assembly is given in Fig. VI-5.

Fuel elements have smooth cylindrical claddings of corrosion-resistant zirconium alloy (∅6.8 mm) structurally similar to that of icebreaker reactor fuel elements. The fuel concept is innovative; it is based on the use of uranium dioxide granules in an inert matrix.

High operating parameters of the used cladding alloy have been verified by pilot fuel element testing in an icebreaker reactor core.

Compared with icebreaker reactor fuel, the KLT-20 fuel has very similar physicochemical, operating and processing characteristics; however, it provides for the uranium content increase by two times, making it possible to use fuel of lower enrichment.

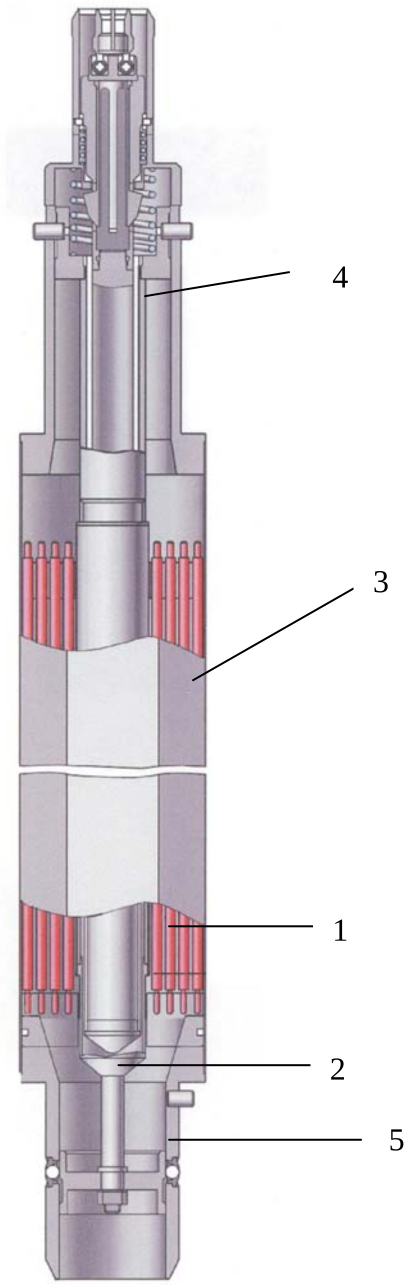
A plain view of the KLT-20 fuel assembly is given in Fig. VI-6; a cartogram of the KLT-20 core is given in Fig. VI-7.

The core consists of 121 hexagonal fuel assemblies with ducts placed in a regular triangular lattice with a pitch of 100 mm. The fuel assembly height is 1800 mm; the flat-to-flat size is 97 mm.

Fuel elements are placed in fuel assemblies with a regular triangular lattice pitch of 9.95 mm. The fuel element outer diameter is 6.8 mm; the cladding thickness is 0.5 mm

Burnable poison rods are based on the gadolinium; like in icebreaker reactors, they provide a near-complete compensation of the burn-up reactivity swing.

The electromechanical reactivity compensation system incorporates the absorber rods placed in the centres of fuel assemblies (one rod per assembly) and integrated into eight compensation groups of the control and protection system (CPS). These groups are used to compensate the temperature and power effects of reactivity, the reactivity margin for core poisoning by xenon-135 and samarium-149, the operating margin for reactivity changes corresponding to a reactor power change during lifetime, and to provide core subcriticality in a reactor shutdown.



1 – Fuel elements; 2 – Absorber element; 3 – Duct; 4 – Upper end; 5 – Lower end.

FIG. VI-5. Vertical view of the KLT-20 fuel assembly.

On the total, the core has 101 absorber rods. In the centres of six fuel assemblies, sealed sleeves are welded to the reactor cover where scram rods are moved. The scram rods are similar to those used in icebreaker reactors.

Two fuel assemblies accommodate the beryllium-antimony start-up neutron sources, the technology for which has been developed within the icebreaker reactor programme.

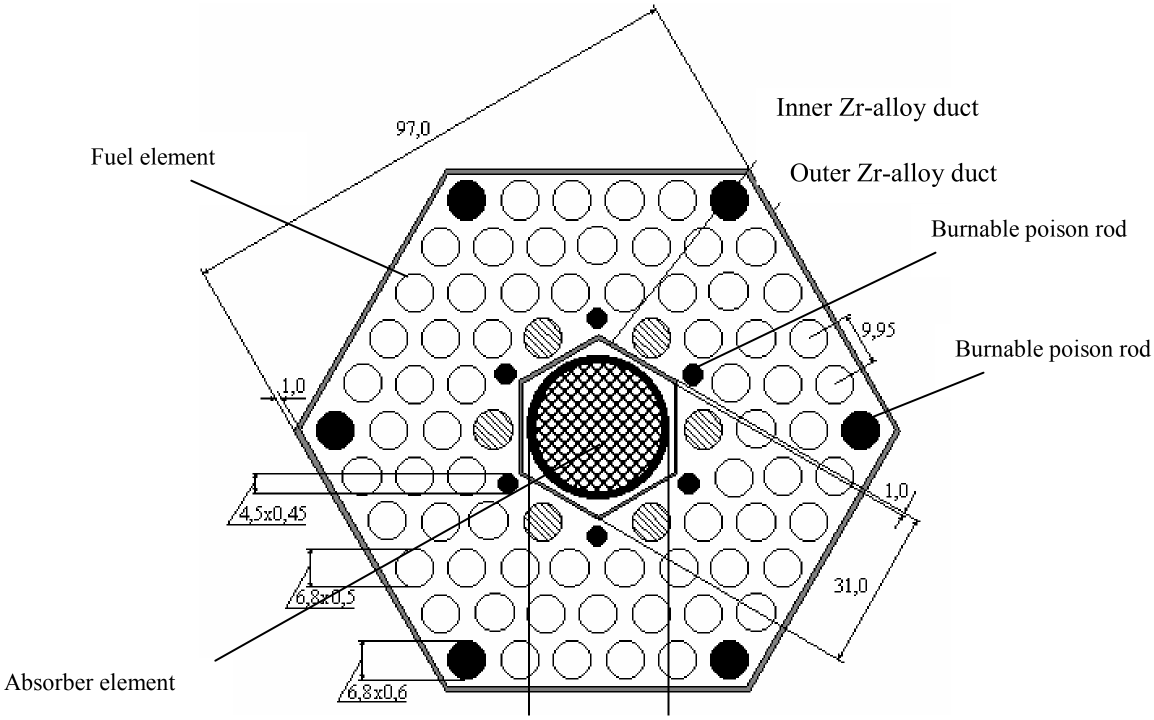


FIG. VI-6. Cross-section of the KLT-20 fuel assembly.

Reactor unit

The reactor unit (Fig. VI-8, VI-9 and VI-10) is intended to convert nuclear power into thermal power and to generate steam of the required parameters.

The unit is an interconnected high-pressure vessel system with removable parts in the main (replaceable) equipment, Fig. VI-8, VI-10.

The unit consists of the reactor, housing the core with actuators of the scram and compensation control rod groups, two steam generators that are connected to the reactor vessel with short nozzles, and two hydraulic chambers accommodating the electric pumps.

In all main connections the outer nozzles house the inner thin-walled nozzles; they are connected to form a coaxial structure. The inner thin-walled connecting nozzles direct the primary coolant flow and are fitted to minimize leakages between areas different in pressure and temperature.

Inside the reactor unit there is a main circulation path to transfer heat from the reactor to the steam generators. Coolant enters the pressure chamber of the reactor from the electrical pumps through two inner connecting nozzles; then it goes to the reactor core where it removes heat from the fuel elements.

From the discharge chamber of the reactor, the coolant enters the steam generators through two inner connecting nozzles; in the steam generators it transfers heat to the secondary coolant to generate steam in the power circuit. Thermal-hydraulic characteristics of the primary and secondary circuits of the KLT-20 plant are given in Table VI-1.

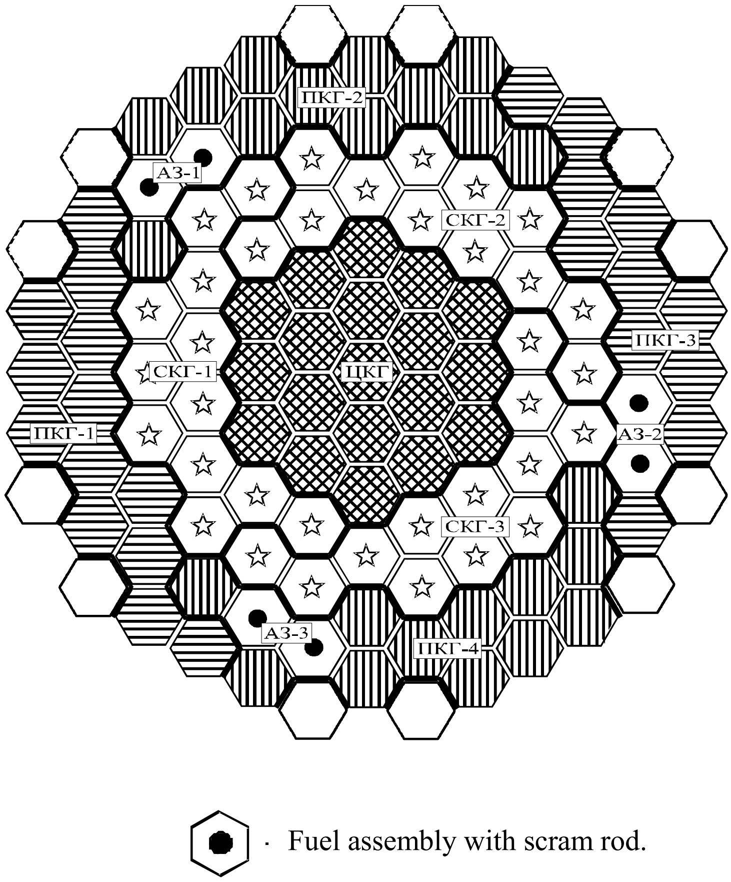


FIG. VI-7. Cartogram of the KLT-20 core

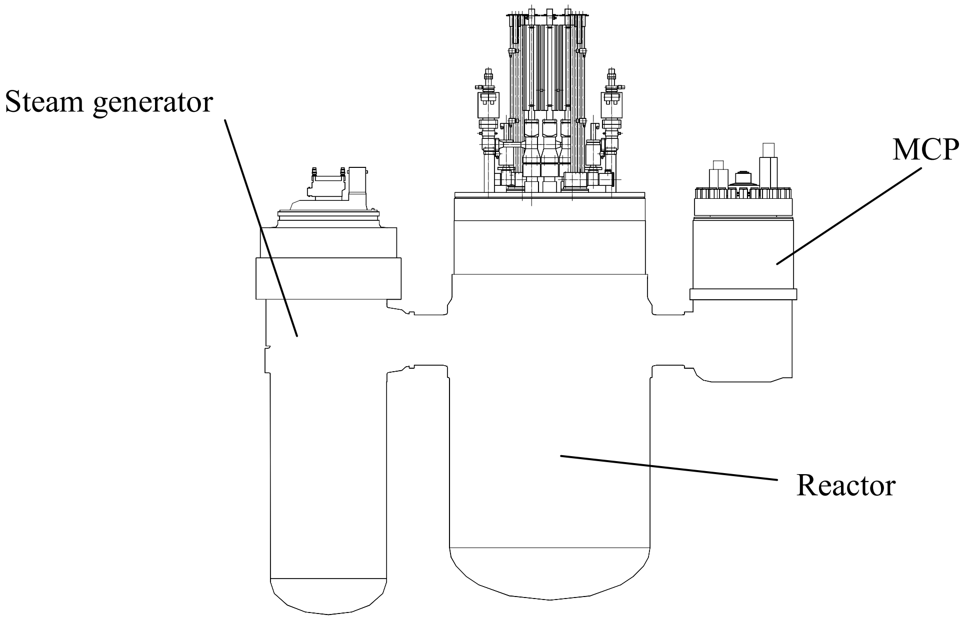


FIG. VI-8. Reactor unit (side view).

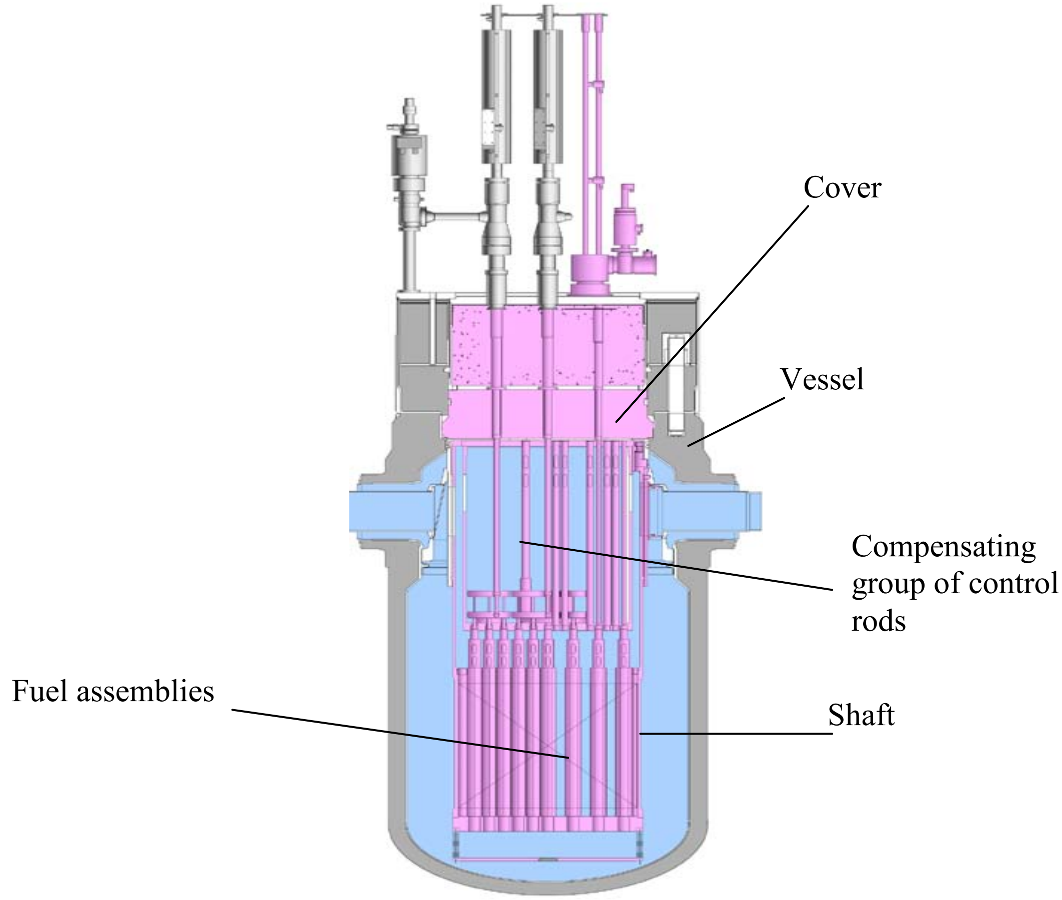


FIG. VI-9. KLT-20 reactor.

From each steam generator, through the annular channel between the inner and outer nozzles, primary coolant enters one of the two chambers of the annular space formed by the cone shaped barrel and the reactor vessel. Each chamber, separated by vertical dividing walls, is a suction cavity of the corresponding electrical pump.

Then coolant flows through two annular channels of connecting ducts into the electrical pump to finish circulation.

The maximum possible scale of depressurization in case of a primary circuit pipeline rupture is not more than DN 25 mm.

Main mass and overall dimensional characteristics of the reactor unit are the following:

— Overall height	≈9 m
— Operating weight (without remote pressurizer)	≈150 t
— Circumscribed diameter of the reactor unit	≈6.5 m

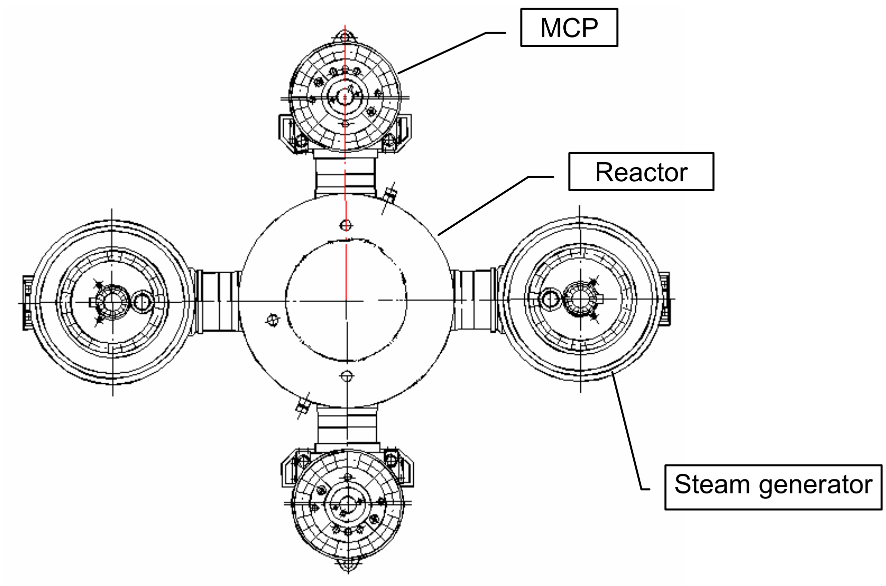


FIG. VI-10. Reactor unit (top view).

Steam generator

The pipe system of the steam generator (SG) together with its casing appears as a modular coil-type vertical-cylindrical heat exchanger of a surface type, in which the heat exchange between the primary coolant circulating in the tube space and the secondary working medium circulating in the in-tube space is performed, Fig. VI-11.

The pipe system heat exchange surface of either of the SGs consists of 19 unified coil-type steam generating modules covered by a box-shaped casing. The steam generating modules are integrated into two independent sections on feedwater and steam.

The heat exchange surface of a module consists of five rows of cylindrical multiple-thread coils wound on the central tube of the module. Coils are spaced in longitudinal and lateral directions using spacing combs and supporting strips.

Main circulating pump

The main circulating pump is a canned, centrifugal, single-stage, vertical pump with a shielded, double-speed (double-winding) asynchronous electric motor.

The electric pump consists of an electric motor and a centrifugal, single-stage pump integrated into a single unit.

The pump has an impeller and a guide device with check valves to exclude coolant circulation through a non-operational electrical pump.

The electric motor consists of the stator placed in the vessel, pipe cooler, bearings, and rotor.

The stator winding space is separated from the rotor space by a leak-tight thin-walled stator partition.

The stator is covered with a lid and sealed by a lens gasket.

The stator windings, partition, rotor and bearings are cooled with water of an autonomous circuit that includes space under the upper cover, rotor space, and holes in the vessel and cover and tube space of the cooler. To control water temperatures of the autonomous circuit, there is a seat for a thermal converter in the upper cover. This autonomous circuit is cooled by water circulating in the cooler tubes.

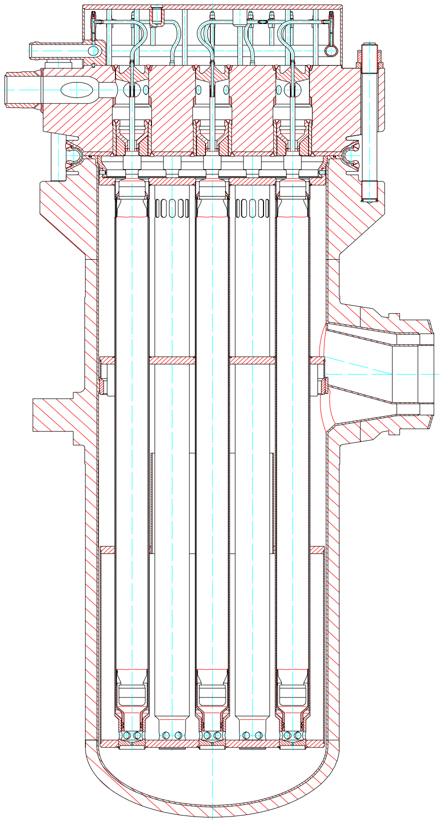


FIG. VI-11. Steam generator of the KLT-20 reactor installation.

During electric pump operation, gas accumulated in the space under the upper cover is discharged together with water to the impeller intake along an axial drilling of the rotor.

Pressurizer

The KLT-20 primary circuit incorporates a gas pressurizer system [VI-5]. This system, which is remote to the reactor unit, includes:

- Four pressurizers;
- Two groups of gas cylinders (6 cylinders in each group);
- A stand-by cylinder group (6 cylinders);
- A gas compressor;
- Piping;
- Valves; and
- Measurement instrumentation.

Technical characteristics of the pressurizer system are given in Table VI-5.

TABLE VI-5. TECHNICAL CHARACTERISTICS OF THE PRESSURIZER SYSTEM

CHARACTERISTIC	VALUE
Pressure, MPa	12.7
Water weight, t	6.2
Gas volume, m^3	~7.7

A spider mixer in the area of the reactor nozzle has a restriction (DN25) to prevent primary circuit outflow in case of a pipeline rupture.

Purification and aftercooling system

The purification and aftercooling system maintains the primary circuit water at a required quality and removes residual heat from the core to the third circuit water. The system includes:

- A primary-third circuit heat exchanger;
- A primary circuit filter;
- Two electrical cooldown pumps;
- Piping;
- Valves; and
- Measurement instrumentation.

The system maintains the quality of the primary circuit water during power operation in accordance with the regulatory requirements.

Main heat transport system

A schematic of core heat removal under the conditions of normal operation and in accidents is shown in Fig. VI-12.

VI-2.2. Description of the turbine generator plant and systems

The KLT-20 turbine generator unit is being developed based on the turbine unit of the TK-35/38-3.4 type used in the pilot floating NPP with the KLT-40S reactor installation to be constructed in the town of Severodvinsk of Russia.

Main characteristics of the KLT-20 turbine generator plant are given in Table VI-6.

TABLE VI-6. MAIN CHARACTERISTICS OF THE KLT-20 TURBINE GENERATOR PLANT

CHARACTERISTIC	VALUE
Live steam pressure before high-pressure cylinder valves, MPa	5.7
Live steam temperature before high-pressure cylinder valves, ºC	300
Feedwater temperature, ºC	170
Maximum electric power, MW	22
Speed of rotation, rpm	3000
Installed capacity per annum; not less than, hours	8000

The turbine generator unit consists of the following main equipment:

- The steam turbine with a bleed-off system, barring motor, steam sorting, stop valves and the control and protection system;
- An electric generator;
- A surface type double-circuit condenser with a reamer tank, hotwell and safety diaphragm;
- Three main electrical condensate pumps (each of 60% capacity);
- Two coolers of fresh water of the generator cooling system;
- Piping, valves and controllers; and
- Operating floors.

VI-2.3. Systems for non-electrical applications

As it was already mentioned, it is possible to use floating power units with the KLT-20 as part of a cogeneration plant for power generation and seawater desalination (similar to the floating power unit with the KLT-40S reactor installation [VI-3, VI-8, and VI-9]). A possible way to couple the reactor installation of a floating power unit with the desalination plant is described in ANNEX IV, using an example of the VBER-150 reactor installation.

VI-2.4. Plant layout

General philosophy governing plant layout

A floating power unit with the KLT-20 is a non-self-propelled flat-deck vessel classified as a harbour ship with a multi-deck superstructure, Fig. VI-13. The reactor compartment is located in the middle part of the floating power unit; the turbine generator and electric equipment compartments are located toward the bow from the reactor compartment, and the compartment for auxiliary devices and the living block are located toward the stern.

Such a layout meets the safety requirements and provides an optimal arrangement for laying the pipelines and electric cables.

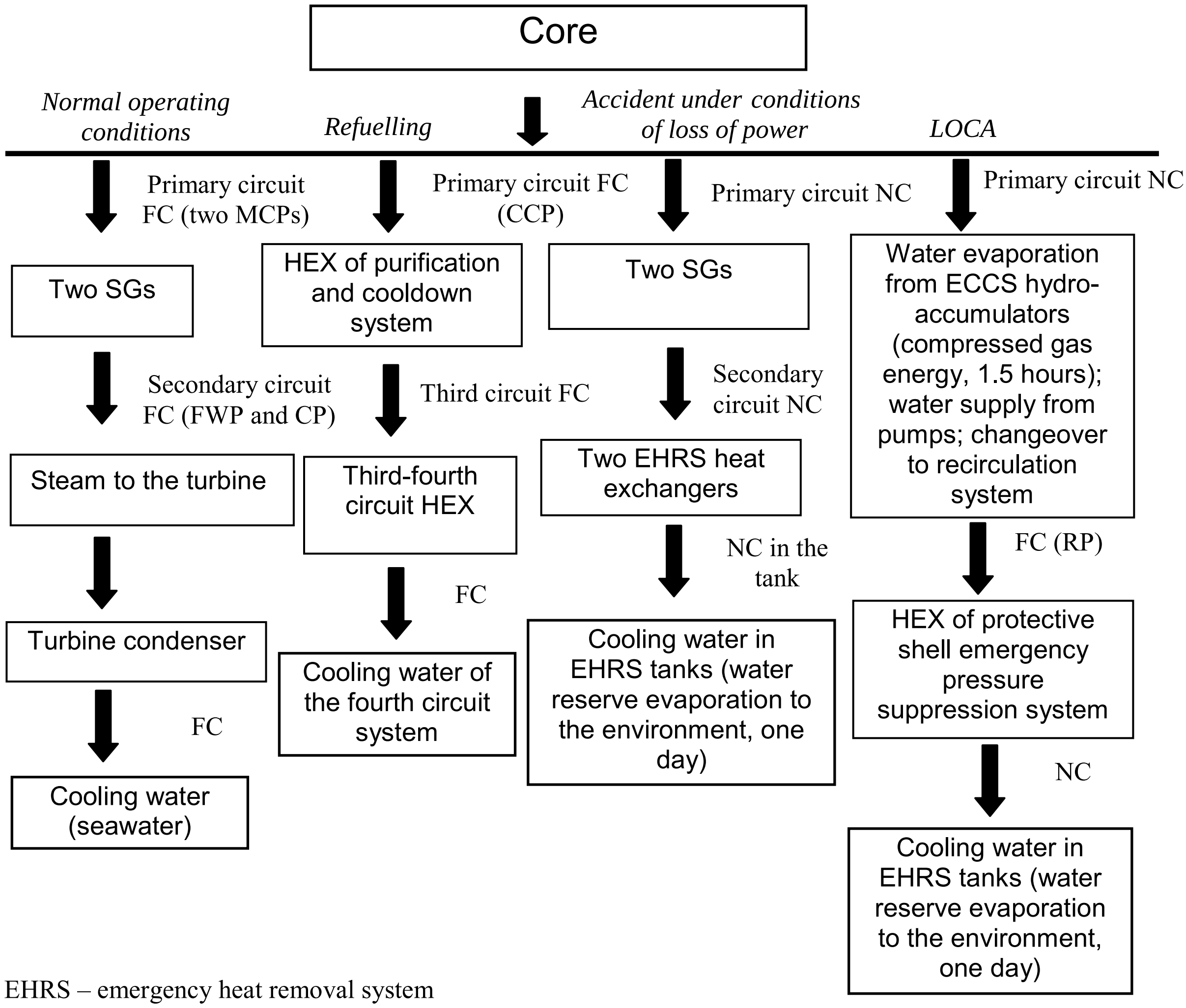
The all-welded vessel of the floating power unit has ice reinforcement and special facilities for towing and mooring. The floating power unit vessel is divided into tight compartments by watertight bulkheads reaching the upper deck.

The floating power unit floatability is secured under flooding of any two adjoining watertight compartments under all design basis loads and load combinations and meets in full the requirements of the Sea Shipping Register of Russia.

Reactor compartment

As it was mentioned, the reactor compartment is located in the middle part of the floating power unit, which is typical of all such unites designed by OKBM.

The reactor installation has its own steel leak-tight protective shell. The reactor compartment is closed by a protective guard consisting of multi-layered ceilings of the superstructure roof, walls of the stern and bow machine rooms and the superstructure premises. Altogether, these structures constitute the external protection of a reactor compartment capable of withstanding external physical impacts including an aircraft crash.



EHRS – emergency heat removal system
FC – forced circulation
NC – natural circulation
CCP – cooldown (aftercooling) circulating pump
FWP – feedwater pump
CP – condensate pump
RP – recirculation pump
HEX – heat exchanger

FIG. VI-12. Schematic of the KLT-20 main heat transport system.

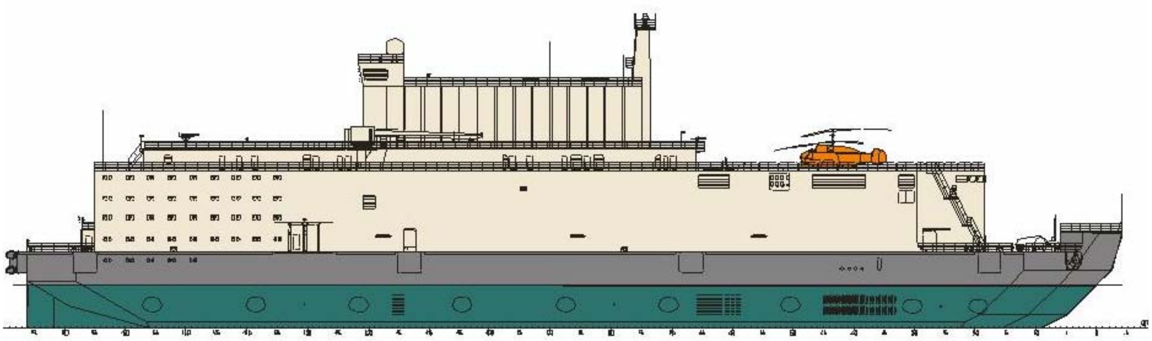


FIG. VI-13. General view of a floating power unit with KLT-20.

Turbine island

The floating NPP has an autonomous machine room for mounting the turbine generator unit and relevant auxiliary systems. The machine room is located toward the bow from the reactor compartment and is separated by cross walls of the reactor compartment protective enclosure.

General layout

A certain arrangement of water space and the creation of coastal infrastructure are needed for normal operation of the floating NPP. The coastal infrastructure includes the following:

- Hydraulic engineering structures (jetties, beacons, boom barriers);
- Waterfront structures (sea-walls, piers, etc.);
- Power line supports for transmission of generated electricity to the consumers; and
- Coastal structures.

Hydro-engineering structures are intended for safe location and mooring of a floating power unit near the coast. Technical communication with the coast is carried out through wharfage. Service vessels may be moored to the floating power unit. Coastal infrastructure and special devices are intended for electric power and heat transfer from the floating power unit to the consumers.

REFERENCES

[VI-1] KHLOPKIN, N.S., Atomic Energy for the Development of the North, the book “IV Alexandrovskie Chteniya” (Moscow, 2000, in Russian).

[VI-2] MITENKOV, F.M., PANOV, YU.K., POLUNICHEV, V.I., BELIAEV, V.M., The KLT-40S reactor system for the floating unit of the thermal power plant, The Strategy of Nuclear Power Development in Russia and Countries of the Asian-Pacific Basin (Paper presented at the scientific and technical conf., Moscow, 1998).

[VI-3] MITENKOV, F.M., POLUNICHEV, V.I., Small nuclear heat and power co-generation stations and water desalination complexes on the basis of marine reactor plants, Nuclear Engineering and Design, Vol.173, Nos. 1–3, pp. 183–191 (1997).

[VI-4] ANTONOVSKY, G.M., et al., PWR-type reactors developed by OKBM, Nuclear News, A publication of American Nuclear Society (March, 2002).

[VI-5] POLUSHKIN, A.K., KUZIN, E.A., VOROBIOV, V.M., KLYKOV, D.M., PANOV, J.K., Implementation of the project for the construction and operation of a nuclear heat and power plant on the basis of a floating power unit with KLT-40S reactors, Small Power and Heat Generation Systems on the Basis of Propulsion and Innovative Reactor Technologies, IAEA-TECDOC-1172, pp. 53–66, Vienna (2000).

[VI-6] BELIAEV, V., POLUNICHEV, V., Basic safety principles of KLT-40C reactor plants, IAEA-TECDOC-1172, Vienna (2000).

[VI-7] INTERNATIONAL ATOMIC ENERGY AGENCY, Development of Safety Principles for the Design of Future NPPs, IAEA-TECDOC-801, Vienna (June 1995).

[VI-8] HUMPHRIES I.R., DAVIES K., A floating generation system using the Russian KLT-C reactor and Canadian reverse-osmosis water purification technology, Floating Nuclear Energy Plants for Seawater Desalination, IAEA-TECDOC-940, Vienna (1997).

[VI-9] POLUNICHEV, V.I., Prospects for the utilization of small nuclear plants for civil ships, floating heat & power stations and power seawater desalination complexes, Status of Non-electric Nuclear Heat Applications: Technology and Safety, IAEA-TECDOC-1184, Vienna 2000.

PASSIVELY SAFE SMALL REACTOR FOR DISTRIBUTED ENERGY SUPPLY SYSTEM (PSRD)

Japan Atomic Energy Agency (JAEA),
Japan

VII-1. Basic summary of PSRD

The Japan Atomic Energy Research Institute[*] is developing A passively safe small reactor for distributed energy supply system, PSRD.

Core design

The PSRD has a long-life single-region homogeneous core, using conventional UO_2 fuel. The core lifetime is approximately 5 years, within which the reactor operates without reloading or shuffling of fuel.

Plant design

The PSRD is an indirect cycle light water cooled tank-type small reactor with an integral design of the primary circuit. Steam generator is located inside the reactor vessel. The PSRD is designed to achieve system simplification, resulting in the reduction of costs for construction, operation and maintenance. The assessments of the plant economy are ongoing.

Safety design

The main and auxiliary cooling systems are based on natural circulation of water coolant. The containment vessel (CV) is water-filled, preventing activity release to the environment and acting as a radiation shield. The control rod drive mechanism (CRDM) is in-vessel type, with no penetrations in the reactor pressure vessel (RPV). No chemical and volume control system is used during reactor power operation. The PSRD has a passive reactor shutdown system.

VII-2. Major design and operating characteristics

Main characteristics of the reactor core are summarized in Table VII-1. Major characteristics of an NPP with the PSRD are given in Table VII-2. A section view of the PSRD module is given in Fig. VII-1. The PSRD passive systems are shown in Fig. VII-2.

[*] Currently Japan Atomic Energy Agency (JAEA).

TABLE VII-1. CORE CHARACTERISTICS

ITEM	SPECIFICATION
Fuel type	UO_2
Core type	Single-region homogeneous
Number of fuel pins	17 × 17 per assembly, Zircaloy-4 cladding
Enrichment by ^{235}U	<5%
Fuel burn-up	26 000 MW·day/t
Operation cycle duration	>5 years
Cladding outer diameter/lattice pitch	9.5 mm/13.9 mm
Equivalent core diameter	1.62 m
Active core height	1.5 m
Average core power density	32.3 W/cm^3
Maximum linear heat rate	68 W/cm
Burn-up reactivity swing	1.02%Δk/k

TABLE VII-2. PLANT CHARACTERISTICS

ITEM	SPECIFICATION
Reactor Type	Integral design pressurized water reactor, tank type
Electric output	31 MW, with an option of incremental capacity increase through modular approach
Thermal output	100 MW
Primary coolant inlet/outlet temperature	270.4°C/311°C, saturated water
Primary coolant pressure	10 MPa, self-pressurization
Main steam temperature/pressure	289°C/4.6 MPa
Feedwater temperature	185°C
Cycle type	Indirect
Plant efficiency	31%
Circulation type	Single phase natural convection
Steam generator (SG)	Two once-through helical coil type steam generators (SGs), located inside the RPV
Decay heat removal systems	Decay heat is transported from RPV via SG and emergency decay heat removal system (EDRS) to the containment water, and then, via containment water cooling system (CWCS), to the environment
Containment system	RPV is immersed in water-filled CV; the cushion gas is Ar (highly effective RPV thermal insulation is needed)
Emergency core cooling system	Passive system, no emergency core cooling (ECC) pumps or accumulators

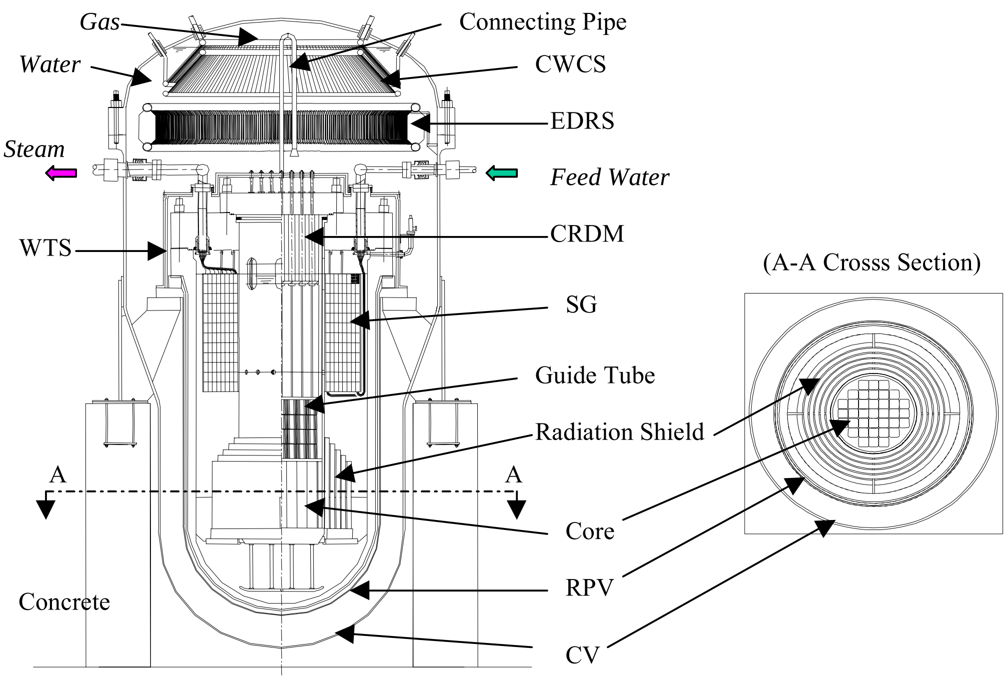


FIG. VII-1. Section view of PSRD.

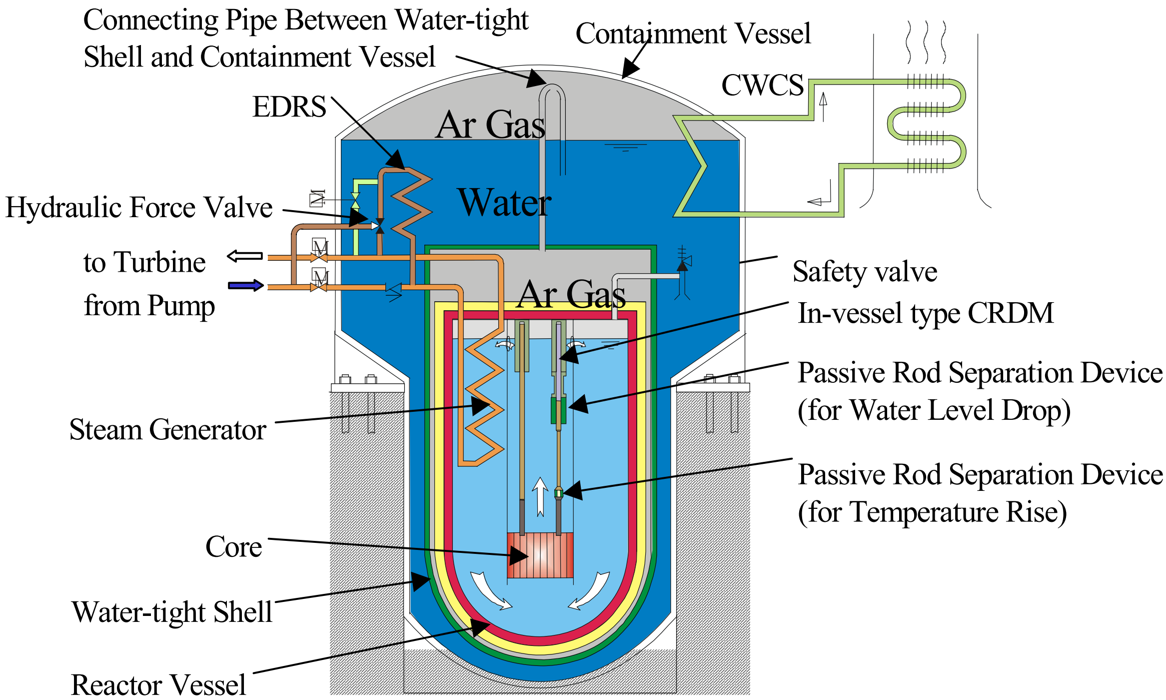


FIG. VII-2. Passive systems of PSRD.

VII-3. List of enabling technologies and status of their development

The enabling technologies for the PSRD are listed in Table VII-3, with their development status being indicated.

TABLE VII-3. LIST OF ENABLING TECHNOLOGIES

ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Use of conventional UO_2 fuel.	Fuel fabrication technology is available.
Essential elimination of loss of coolant accident (LOCA) by reducing the amount of piping penetrating the RPV, with only steam and feedwater pipes and safety valves being left.	Provided by design
All safety systems are passive, including the water-filled CV, the emergency decay heat removal system (EDRS) with hydraulic force valves, the steam generators (SGs) and the containment water cooling system (CWCS).	The integral test has not been performed yet; only the main functions of the water-filled CV and pressure suppression in case of a LOCA were verified by test
An additional passive reactor shutdown system based on the gravity driven operation of control rods actuated by a decrease in the magnetic flux in the magnetic control plate due to increased surrounding temperature, resulting in the release of control rods.	The basic design analysis has been completed. Tests to confirm the function are planned.
System simplification: no pumps in the primary and emergency core cooling systems, no accumulators and no external pressurizer or steam generator.	

PACKAGE-REACTOR

**Hitachi, Ltd. and Mitsubishi Heavy Industries, Ltd.,
Japan**

VIII-1. General information, technical features and operating characteristics

VIII.1.1. Introduction

The Package-Reactor is an ultra small reactor with light water cooling and a reference output of 25 MW(th). The design target of this reactor is to compete with fossil fuel power generators in economy, ease of operation and maintainability for markets requiring stand-alone energy supply systems. The Package-Reactor generates electricity and process heat, and extra energy is either used or stored in the energy conversion system. To achieve the target, this reactor has no active systems to control burn-up reactivity swing and no active devices in the high-pressure boundary because of the adoption of a reactivity control-free system and a core cooling system based on natural circulation [VIII-1, VIII-2].

In 2002, Hitachi, Ltd. (Hitachi) and Mitsubishi Heavy Industries, Ltd. (MHI) started the conceptual design study of the Package-Reactor. The design has been selected considering the technologies for both, current pressurized water reactors (PWRs) and boiling water reactors (BWRs), and limitations for commercial fabrication and reprocessing facilities of the current fuel cycle. The goal was to ensure minimum costs of necessary research and development (R&D) activities and small necessary modifications of the existing facilities, so that the reactor could be developed and deployed within a few years, subject to availability of the financing.

The design and development of this reactor has been fully funded by Hitachi and MHI (Japan).

VIII-1.2. Applications

The Package-Reactor is primarily designed to supply electricity and thermal power for non-electric applications as a stand-alone energy supply system for remote regions, where it is difficult to construct long-distance power transmission and distribution facilities.

VIII-1.3. Special features

The Package-Reactor is a land-based power station unit; as the unit is very small and light, it could be easily transported by a small ship and also shipped overseas.

The core is divided into several sub-cores and each sub-core is housed in a pressure tube called the “cassette”. The capacity of a power station can easily be changed by adjusting the number of cassettes and by constructing additional units.

Each part of the unit is compact and transport by land is feasible.

Another special feature is in the refuelling design. The reactor is capable of operating without reloading and shuffling of fuel in the sub-cores for a period of 5–10 years. After the operating period expires, all irradiated cassettes with spent fuel are exchanged once-at-a-time with new cassettes containing fresh fuel.

VIII-1.4. Summary of major design and operating characteristics

Installed capacity

The reactor is designed to produce from 10 to 100 MW(th). The ratio between the electric and process heat output is flexible.

Mode of operation

The Package-Reactor is designed for base load power generation, but its electric output can be changed according to demand. The extra energy is then used for non-electric purposes or stored in a chemical heat pipe system, described later in this ANNEX.

Load factor/Availability

For a 5-year cycle operation, the target load factor is estimated at 97%, taking into account that the reactor stops for one or two months to exchange cassettes and maintain components after each operation cycle.

Some major aspects of design and operating characteristics of the Package-Reactor are given in Table VIII-1.

Figure VIII-1 shows a simplified schematic diagram of the nuclear steam supply system with the Package-Reactor. The concept resembles a calandria-type pressurized heavy water reactor (e.g., the FUGEN advanced thermal reactor (ATR) or CANDU reactors) since all these employ pressure tubes. But the Package-Reactor is somewhat different from the ATR or the CANDU. The Package-Reactor employs natural circulation with two-phase flow for core cooling and has no recirculation pumps. The height of the pressure tubes of the cassettes is required to be as low as possible to attain a compact unit. Two-phase flow with high void fractions similar to BWRs is adopted to attain natural circulation with a cassette height of 6 m and a fuel rod length of 3.65 m.

Natural circulation cooling leads to low cost power generation. The core outlet temperature of 345°C and the pressure of 15.5 MPa have been selected for the core cooling system to obtain higher thermal efficiency. The coolant system does not use chemical shim for reactivity control. To compensate for burn-up reactivity swing, a perfectly passive reactivity control system is under study, based on burnable poisons in fuel and using moderator void feedback.

In the current design, the whole core is horizontally separated into twelve sub-cores and each sub-core is enveloped in a pressure tube. The pressure tube containing the fuel is called a “cassette”. The core shutdown system is placed outside the pressure boundary, which contributes to the reduction of costs of control rods and the control rod drive system. A steam generator (SG) is connected to the cassettes by riser pipes and these pipes transfer heat from the core to the energy conversion system. In each cassette, water flow areas are divided into two regions: the fuel region and the downcomer. Cooling water flows upward accompanied by boiling in the fuel region and is separated into steam and saturated water above the chimney exit. The separated steam flows into the SG while the saturated water flows into the downcomer where it is mixed with condensed water from the SG. The mixed water moves downwards in the downcomer driven by natural circulation.

Simplified schematic diagram

Figure VIII-2 shows the Package-Reactor system configuration inside the containment vessel. To reduce construction costs, the prototype currently being developed adopts twelve nuclear cassettes and three steam generators as one package (unit). Every four cassettes are connected to one steam generator. Heat transfer tubes in the SG are separated into two groups in view of a heat transfer tube rupture event. During normal operation, pressure boundaries are maintained by the cassettes and steam generators, while the containment vessel is kept under atmospheric pressure. The containment vessel is made of stainless steel with a height and diameter of about 10 m and 5.5 m, respectively.

TABLE VIII-1. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

ITEMS	SPECIFICATIONS
Thermal output	25 MW
Core characteristics:	
— Effective core height	3.7 m
— Average void fraction	~30%
— Hydrogen to uranium ratio	~6
— Power density	8.1 kW/l
— Average linear heat rate	0.7 kW/ft
— Void coefficient	$\sim-15 \times 10^{-4} \Delta k/k/\%V$
Fuel:	
— Material	UO_2
— ^{235}U enrichment	Maximum 5.0% by weight
— Uranium inventory	5.5 tons
Cladding:	
— Material	Zircaloy
— Outer diameter	9.5 mm
Pressure Tube:	
— Material	Zircaloy
— Thickness	30 mm
Shroud:	
— Material	Zircaloy
— Thickness	5 mm
Burnable absorber material	Gd_2O_3
Chemical shim	Not used
Core cooling system:	
— Circulation type	Natural circulation
— Pressure	15.5 MPa
— Inlet temperature	340°C
— Outlet temperature	345°C

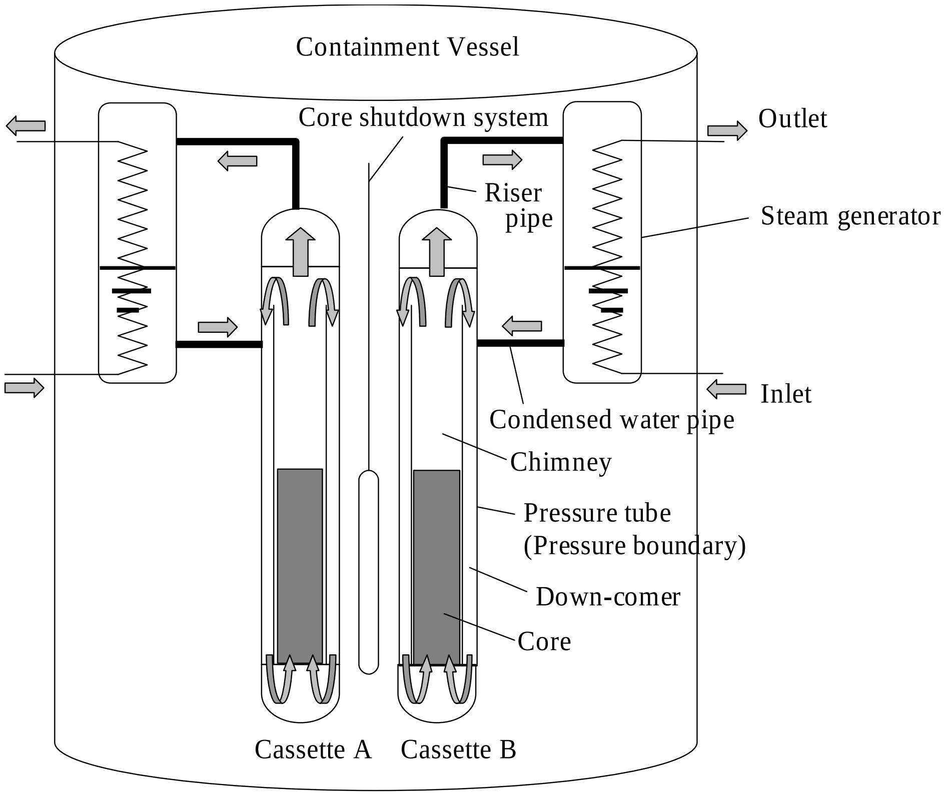


FIG. VIII-1. Simplified schematic diagram of Package-Reactor.

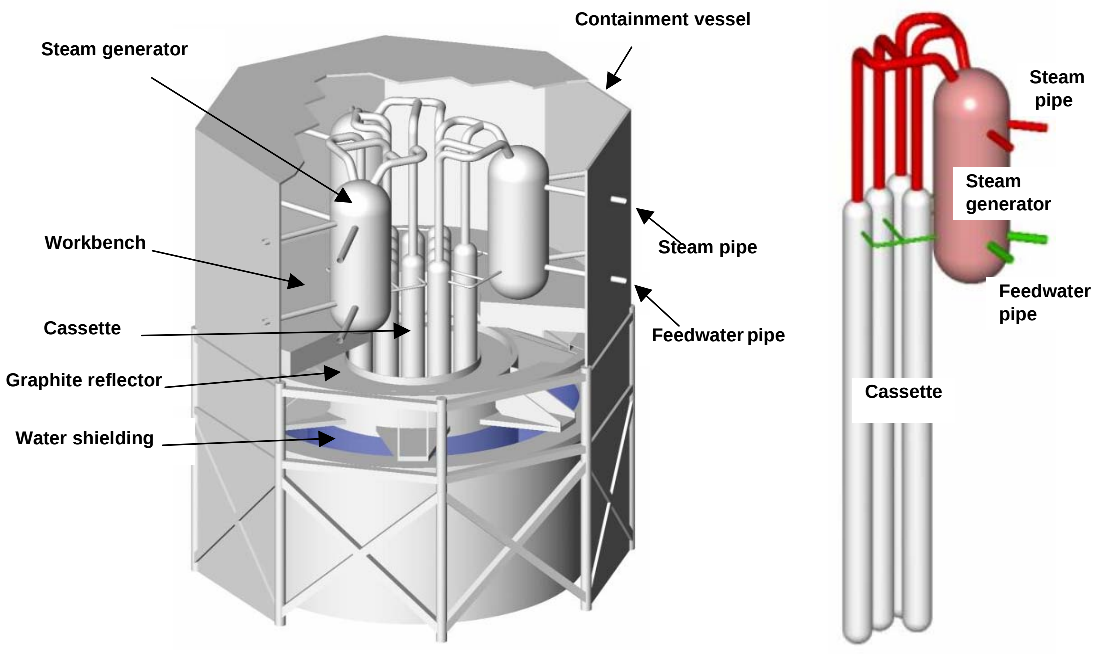
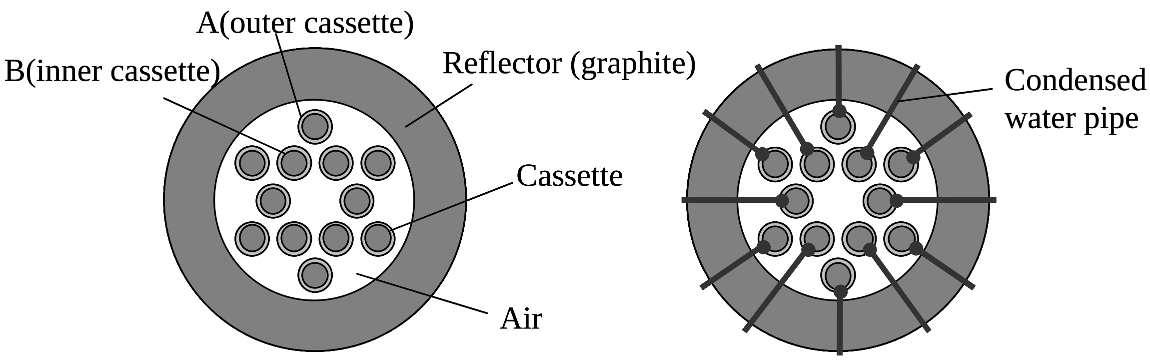


FIG. VIII-2. Outline of nuclear power generation system and the configuration of multi cassettes/single steam generator.

Neutron-physical characteristics

Fuel technology of current PWRs can be used for the fuel design of the Package-Reactor as the pressure in its cassette is almost the same as in current PWRs. Figure VIII-3 (a) shows the core layout with 12 cassettes. The arrangement takes into consideration the factor of easy access to each cassette for replacement and maintenance. Cassettes are arranged so that each of them has an access route not overlapping with other cassettes in the radial direction. This allows the condensed water pipes from the SG to be brought to the side of each cassette along a straight line (Fig. VIII-3 (b)), which simplifies the installation and maintenance of the condensed water pipes. Each SG is connected to four cassettes. A graphite reflector surrounds the cassettes to improve the neutron economy.

Figure VIII-4 shows k_{eff} change with burn-up for the core with and without the burnable poison (Gd_2O_3). A fuel material is UO_2 with the enrichment by ^{235}U less than 5.0% (by weight). An operating period of about 10 years can be achieved with some of the UO_2 fuel rods containing Gd_2O_3. It could be noted that the k_{eff} change in 10 years is less than 1% $\Delta k/k$, see Fig. VIII-4.



(a) Cassette layout (b) Cassette accessibility

FIG. VIII-3. Schematic of the cassette arrangement (horizontal view).

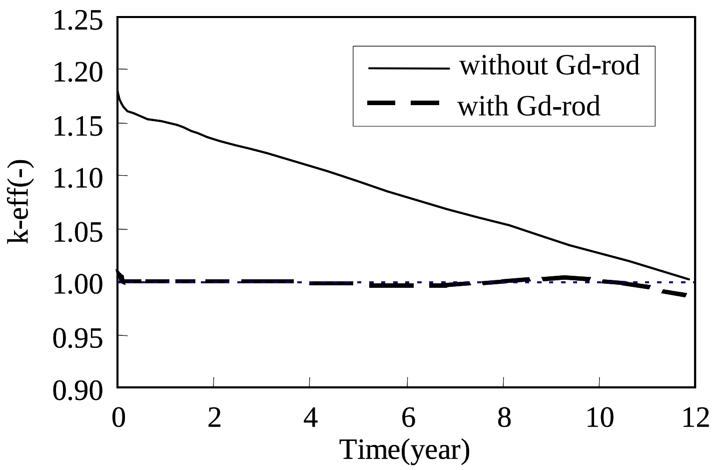


FIG. VIII-4. K_{eff} change with burn-up with and without Gd-containing fuel rods.

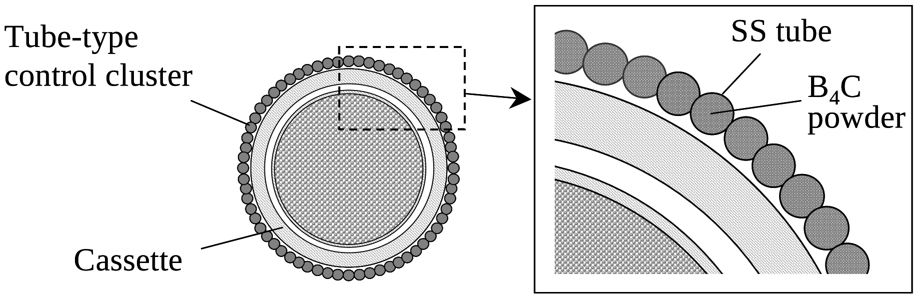
Reactivity control mechanism

Reactivity control in normal operating conditions

The burn-up reactivity swing is compensated by burnable poison (Gd_2O_3) and the void reactivity feedback arising from changes in the core thermal power. In normal operation, no control rods are used. With the use of the burnable poison, the reactivity change to be controlled is less than 1% (see Fig. VIII-4) and this small change could be controlled by changing the core thermal power and making use of the corresponding void reactivity feedback. The change of thermal power is accomplished via the energy conversion system. In this way, it may be possible to avoid control rod use for reactivity control during normal operation.

Core shutdown mechanism

The Package-Reactor is equipped with tube-type control clusters to shut down the core. Figure VIII-5 shows a schematic of the control rod cluster. The cluster consists of about 60 absorber rods, each of which appears as a stainless steel tube of about 20 mm diameter containing natural B_4C powder. When the core is shut down, all control rod clusters are inserted.



SS – stainless steel

FIG. VIII-5. Schematic of the control rod cluster.

Licensing procedures for light water reactors (LWRs) require complying with a “one rod stuck margin” criterion. The one rod stuck margin means the margin for the core criticality in the cold shutdown condition with one control rod, which has a maximum control rod worth, not fully inserted. The one rod stuck margin is also called the cold shutdown margin (CSDM) in the BWR core design. Because of a symmetrical layout of 12 cassettes, evaluation can be limited to two one-rod-stuck conditions; the first case is where the control cluster for cassette A (the outer cassette shown in Fig. VIII-3 (a)) is not inserted and the second case is where the control cluster for the cassette B (the inner cassette shown in Fig. VIII-3 (a)) is not inserted. The corresponding increase in k_{eff} against the state when all rods are inserted is less than 0.5% $\Delta k/kk'$ in both cases, which is smaller than that in a conventional BWR. The reason is that there are two “layers” of control rods (corresponding to 2 control rod clusters, one for each cassette) between neighbouring cassettes in the shut down Package-Reactor core. In a one-rod-stuck condition, one “layer” of the control rods remains fully inserted and this keeps the reactivity low. As a result, the one rod stuck margin (or CSDM) is more than 14%Δk in the reference core design of the Package-Reactor.

Possibility to use stainless steel as a pressure tube and shroud material

The reference Package-Reactor design uses Zircaloy as a material for the pressure tube and shroud to decrease neutron absorption and improve the neutron economy. If stainless steel is used as a pressure tube and shroud material, the neutron absorption increases in comparison with the Zircaloy. This leads to a decrease in the control rod worth, and the cold shutdown margin (CSDM) will be decreased also. The use of a stainless steel leads to a lower cost of the pressure tube and shroud because the Zircaloy is more expensive. As the CSDM is excessively high in the reference Package-Reactor core using Zircaloy (the requirement for a BWR is 1%Δk only), there is a possibility that the core system with the pressure tube and shroud made of stainless steel would meet the design criteria for the CSDM. If the stainless steel is used, the capital cost of the Package-Reactor could be significantly reduced.

Cycle type and thermodynamic efficiency

The Package-Reactor is designed to operate in an indirect cycle; it uses the backpressure steam turbine with a target of about 20% thermodynamic efficiency.

Thermal-hydraulic characteristics

The thermal margin of the Package-Reactor core is higher than that in a conventional BWR core because the power density and the linear heat generation rate are essentially smaller (see Table VIII-1).

Maximum/average discharge burn-up of fuel

The average discharge burn-up of fuel is 16 GW·day/t·U. The peaking factor of the package-reactor cassette is less than 1.05, so it is assumed that the maximum discharge burn-up of the cassette is about 17 GW·day/t·U.

Fuel lifetime/period between refuellings

The fuel lifetime and the period between refuellings are both about 1800 effective full power days (EFPD), or more than 5 years. From the viewpoint of the neutronics, a 10-year (about 3650 EFPD) refuelling interval is possible but the integrity of the fuel cladding is not confirmed for more than 5 years of operation in the conditions similar to those of a PWR core.

Mass balances/flows of fuel materials

The mass flow of UO_2 is 207 kg/year/MW(th) based on a 10-year operating cycle.

Design basis lifetime for reactor core, vessel and structures

The target for the fuel lifetime is about 10 years but the integrity of the fuel element cladding for such period of continuous operation has not been confirmed so far. All components including cassettes can be replaced at the site, so the concept of an increased lifetime of the reactor vessel and structures is not applicable to the Package-Reactor.

Design and operating characteristics of systems for non-electric applications

Figure VIII-6 shows the conceptual diagram of a stand-alone energy supply system with package-reactor operating in a cogeneration mode.

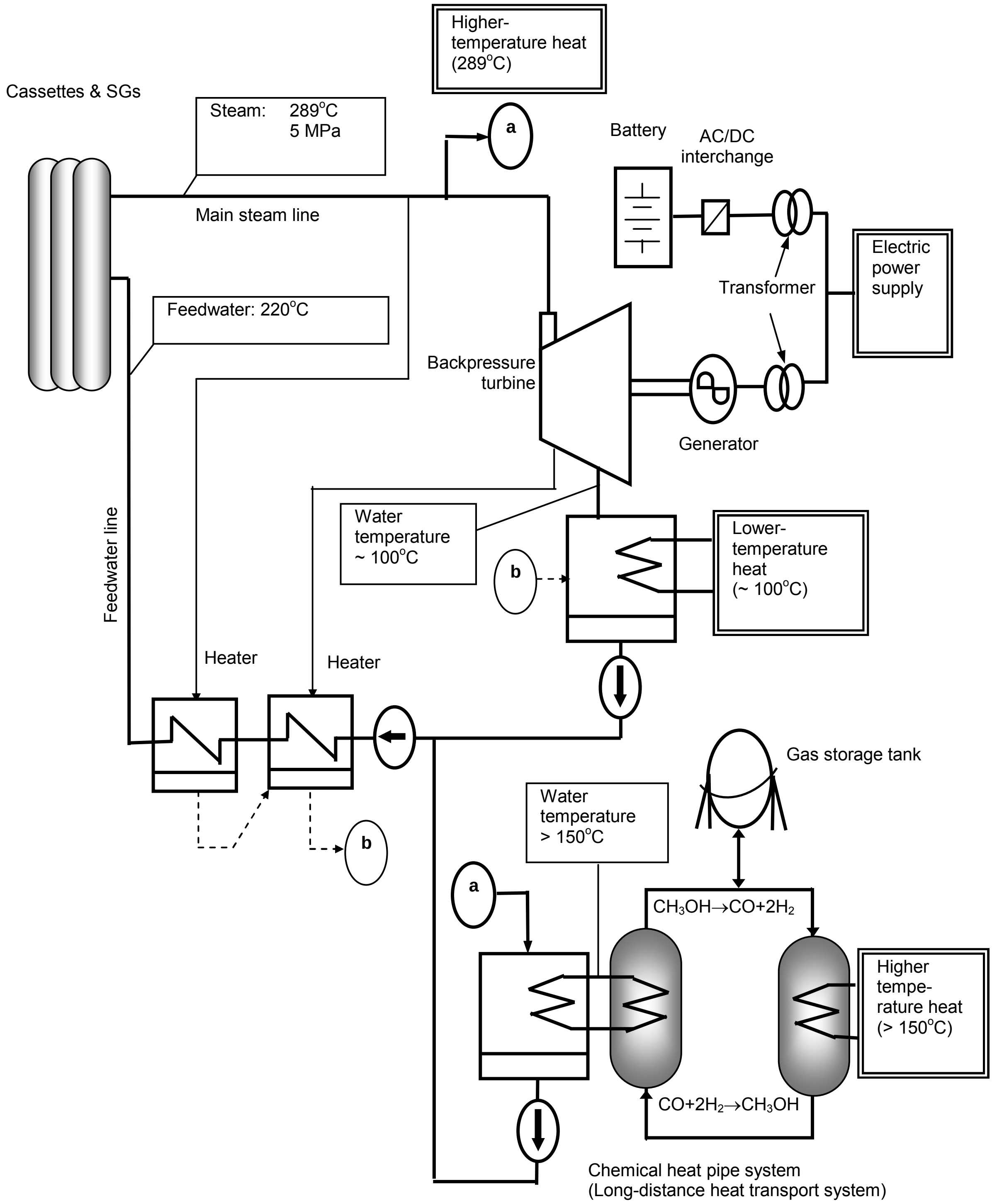


FIG. VIII-6. Conceptual diagram of a stand-alone energy supply system with the Package-Reactor.

Steam generated by the cassettes (steam generators) is merged in a steam header and sent to a steam turbine transforming the steam into electric energy. A backpressure turbine is currently under consideration for the turbine generator unit of the Package-Reactor. Compared with the condensing turbine, the backpressure turbine is inferior in energy transformation efficiency; however, since the steam exhaust from the backpressure turbine has a temperature higher than 100°C, it could be effectively used as a heat source for non-electric applications. Chemical heat pipe system that would provide long-distance transport of energy with small losses is being considered as a heat transport system for the Package-Reactor. The chemical heat pipe system conveys the gas incurred by an endothermic reaction to the heat demand area (via a pipeline), and then the heat generated through an exothermic reaction is supplied to this area. With a buffer tank installed on a pipeline, storage of energy becomes possible by temporarily holding the gases there. The coaxial tube design employing a service pipe as the inner tube and steel jackets for the outer tube might be effective in improving the energy transport reliability, but no detailed evaluation has been performed so far.

The heat for warming water to over 150°C required for an endothermic reaction (see Fig. VIII-6) can be provided by main steam generated by the cassettes. This steam being taken off to heat water, the turbine steam supply and the electric output would decrease. As electricity demand often changes during the day, a load follow mode of operation could be organized by slowly adjusting the flow rate of steam extracted from the main steam supply system for the above mentioned purpose. A momentary small load change could be followed by automatic operation of the accumulation-of-electricity/electric discharge control device using a storage battery. This device could initiate the adjustment of the flow rate of steam taken off to support the endothermic reaction. Thus, a stand-alone energy supply system that is capable of autonomous operation in a load follow cogeneration mode with an option of thermal energy storage and long-distance transport might be created.

An example of electricity demand change is shown in Fig. VIII-7. Figure VIII-8 illustrates possible variations in the production rate of energy of three types (electricity, low-temperature heat energy of the turbine exhaust steam, and higher temperature heat energy of the main steam) by such stand-alone system, as a function of the fraction of main steam taken off for non-electric applications.

Other cogeneration options for the Package-Reactor would be considered in the course of further design development.

Economics

The economic evaluation has not been completed so far.

VIII-1.5. Outline of fuel cycle options

The fuel cycle concept of the Package-Reactor is basically similar to that of BWRs and PWRs because of essentially the same type of fuel.

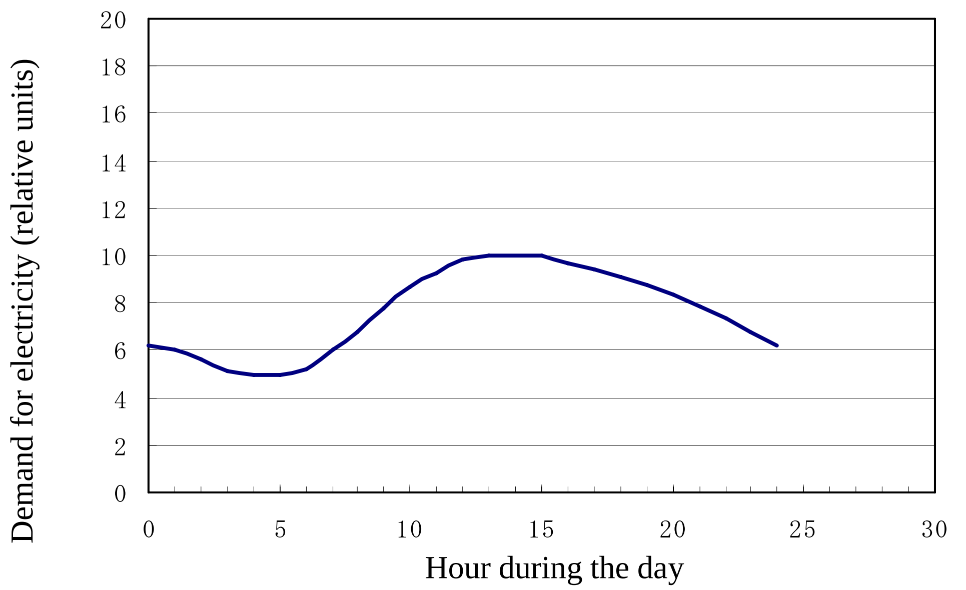


FIG. VIII-7. An example of electricity demand change over the day.

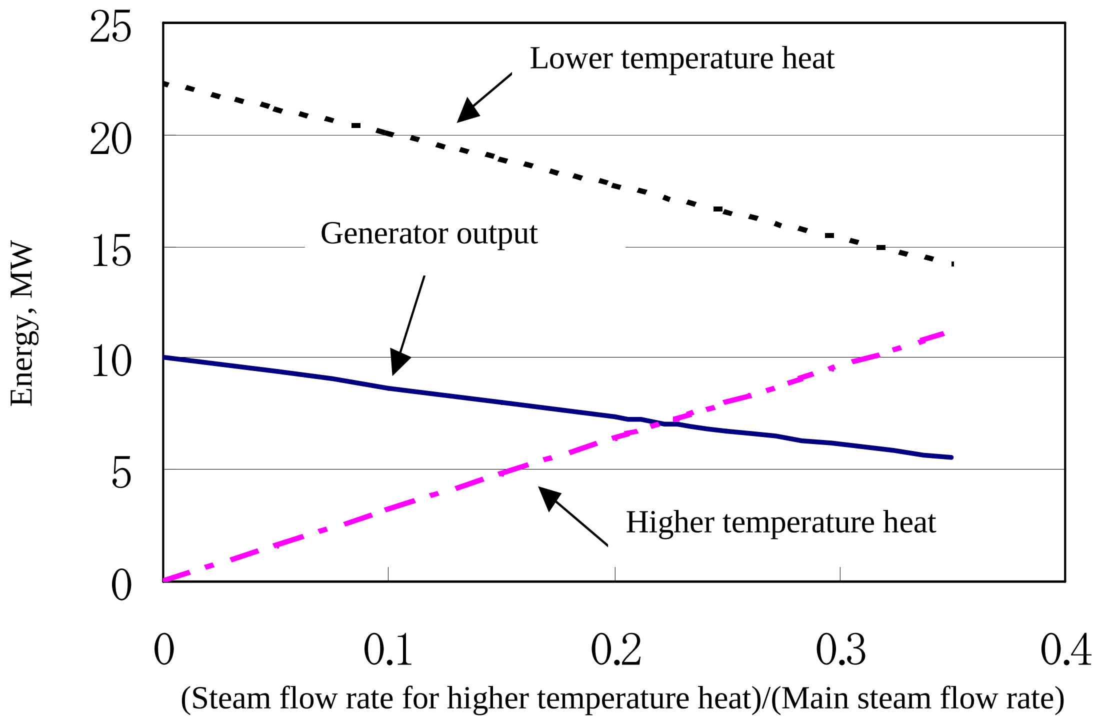


FIG. VIII-8. Variations of the production rates of three types of energy.

VIII-1.6. Technical features and technological approaches that are definitive for Package-Reactor performance in particular areas

VIII-1.6.1. Economics and maintainability

The advanced safety features, completely standardized design and low capital cost of the Package-Reactor are expected to facilitate its deployment in developing countries with limited financial and technical resources.

The design features contributing to a reduction in capital and construction costs are:

- Natural circulation based core cooling, which results in the elimination of main coolant pumps;
- A simplified reactor shutdown system with its critical components operating outside the cassettes under atmospheric pressure;
- The use of passive cooling for the cassettes and the containment vessel, which facilitates elimination of the Elimination of the emergency core cooling system (ECCS);
- Small volume of the high pressure boundary area; small containment vessel, and the absence of a large reactor pressure vessel, fuel-storage and maintenance buildings;
- Rationalization of the design, manufacturing and construction to be achieved by completely standardized and compact design;
- A short construction period achieved by the package design and factory fabrication of most of the components;
- A short decommissioning period with no decommissioning waste left on the site, achieved by a completely replaceable system design.

The design features contributing to reduced operation and maintenance (O&M) costs are:

- Reactivity control-free design with operator-free performance during accidents, which could facilitate reducing the number of operational staff;
- Simplified design that eliminates certain equipment items and provides for a 5-year operating cycle, resulting in reduced maintenance requirements;
- A completely replaceable system design, resulting in a practically infinite design lifetime;
- The use of a backpressure turbine with simple mechanisms and easy maintenance.

No new fuel fabrication or reprocessing facilities would be required for the Package-Reactor as it is based on conventional UO_2 fuel with the enrichment of less than 5% (by weight), similar to that used in operating PWRs.

VIII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The average discharged burnup in the Package-Reactor is 16 GW·day/t·U, i.e., about one third of that in a conventional LWR. However, the Package-Reactor attains a double or triple energy efficiency compared to the conventional LWR due to the use of multiple cogeneration options; the efficiency of energy generation per weight unit of natural uranium consumed is similar to that of conventional LWRs. The quantity of solid, liquid and gas waste from the equipment and systems is reduced through the application of a reactivity control-free design; simplification of the core cooling system contributes to a reduction of radioactive waste generated during the operation and maintenance.

The radiation exposure of workers is reduced through the reduction of the number of equipment items related to the core cooling system and a smaller core cooling volume.

The Package-Reactor is a reactor without on-site refuelling, that is, a reactor which operates without reloading or shuffling of fuel in the core for a reasonably long period; after this, all cassettes are withdrawn and moved to an off-site factory and new cassettes are brought from the factory and installed. The concept of “cassette exchange” without withdrawing fuel reduces the quantity of on-site waste arising from the core cooling system. The reduction of waste and the elimination of spent fuel storage are highly advantageous for the reduction of waste management costs.

VIII-1.6.3. Safety and reliability

Safety concept and design philosophy

The basic propositions of the Package-Reactor safety concept are as follows:

- By-design elimination of initiating events that might cause fuel failure; ensuring a safe operation without active systems;
- Provision of a broad allowance for design margins by adopting a low power density core design;
- Diversity and redundancy of the systems of heat removal and reactor shutdown;
- Incorporation of inherent safety features into the core design, specifically, ensuring a large negative void reactivity feedback;
- Ensuring very small energy release from the core in accidents;
- Ensuring that there is no need in operator actions or external support such as water, power, etc., in accidents.

By adopting a “multi-cassette” concept with the cassette inner diameter of around 30 cm, it is possible to remove decay heat from a cassette in a loss of coolant accident (LOCA) just by cooling the outer surface of the cassette. As shown in Fig. VIII-9, when heat removal becomes necessary due to steam generator stoppage and rising temperatures, water coolant is passively injected around the cassette from the shielding water tank, actuated by a direct-action device based on low melting-point metal film.

A rod ejection event does not occur because the shutdown rods are under atmospheric pressure and fully withdrawn during operation.

Incorporation of a cogeneration design in the energy conversion system and a large design allowance for fuel failure ensures that there is no reactor trip in the event of a turbine trip.

Provisions for simplicity and robustness of the design

Some important provisions for simplicity and robustness of the Package-Reactor design are as the following:

- Incorporation of several passive safety features;
- Very low excess reactivity without active reactivity controls;
- Low power density in the core; and
- A multi-cassette concept incurring very small energy releases in accidents.

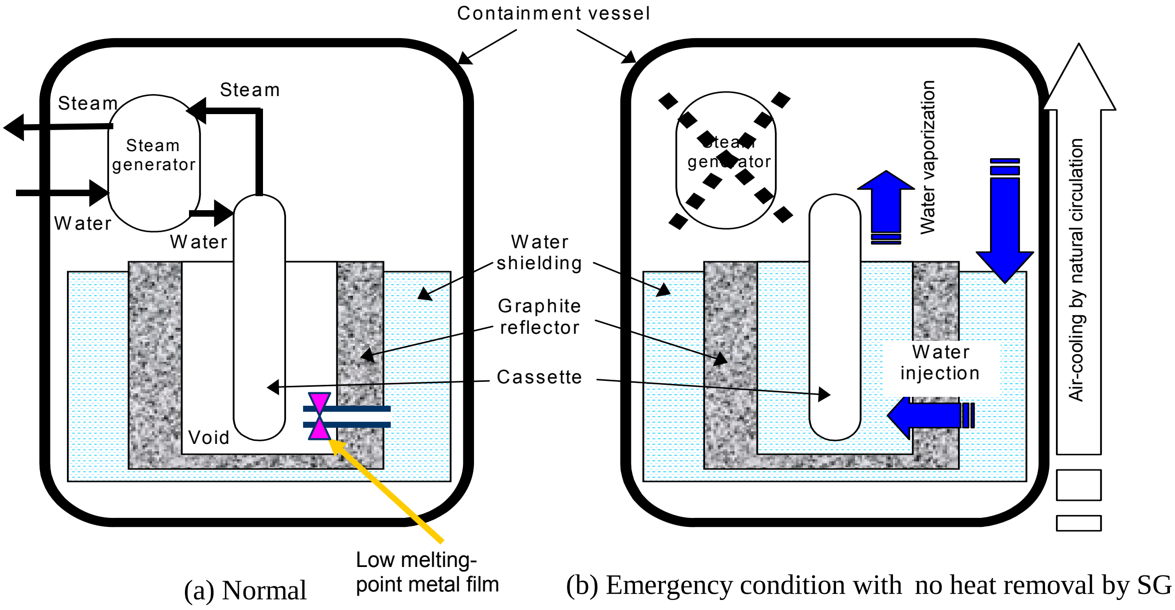


FIG. VIII-9. Heat removal system.

Active and passive systems and inherent safety features

Inherent safety features of the Package-Reactor are as the following:

- Large negative void reactivity coefficient;
- Broad design margins for fuel failure achieved through very low power density (no requirement for power distribution measurement);

The passive safety systems are:

- Natural circulation driven heat removal during normal operation and shutdown conditions;
- Passive coolant injection around the cassettes actuated by a low melting-point metal film;
- Ultimate heat removal by natural air-cooling around the containment vessel.

The active safety systems are:

- A gravity driven control rod scram system;
- A boron ball insertion system with a rupture disk (also driven by gravity drop);
- A multi-path cogeneration system for energy conversion (turbine, chemical heat pipe system, etc.).

Structure of the defence-in-depth

This Package-Reactor is assumed to be designed, manufactured, constructed and operated with the same quality and reliability and based on the same philosophy as conventional LWRs. Additionally, as the design basis, this reactor accepts a multi-cassette concept to eliminate the causes of initiating events which might result in fuel failure.

The incorporation of “no fuel failure” criterion in the original design concept remarkably decreases the possibility of radiological release. In addition, the cassette and the containment vessel act as barriers to the release even if fuel failure occurs.

Design basis accidents and beyond design basis accidents

The Package-Reactor has fewer accidents to be considered as design basis accidents: accidents with riser and cassette-inlet pipe breakage between the cassette and steam generator; breakage of steam and feedwater pipes between the steam generator and the energy conversion system; breakage of a heat tube in the steam generator; and drop-of-a-cassette accident during the cassette exchange.

In design basis accidents, when abnormal conditions are detected, the shutdown rods are automatically injected between the cassettes, driven by gravity. The shutdown rod worth is sufficient to maintain cold shutdown conditions with a one-rod stuck margin. If shutdown by the rods fails, boron balls stored in the workbench fall between the cassettes, a system similar to that adopted in high temperature gas cooled reactors (HTGRs).

Beyond design basis accidents have not been addressed for this reactor.

Provisions for safety under seismic conditions

Safety under seismic condition has not been evaluated, but the design criteria adopted for structures are the same as those for conventional LWRs.

Probability of unacceptable radioactivity release beyond the plant boundaries

It is expected that the probability of unacceptable radioactivity release beyond the plant boundaries for the Package-Reactor will be less than that in conventional LWRs. Even if radioactivity is released, its quantity would be very small because the fuel inventory of the cassette is very small.

Measures planned in response to severe accidents

It is not easy to analyze the performance of the Package-Reactor in severe accident conditions because the causes for severe accidents are essentially eliminated. Cassettes are separated into several groups and each group has a steam generator; if a single pipe break accident occurs, the accident impact is limited to cassettes belonging to the cassette-group where the accident occurs. This design concept targets to retain plant integrity in severe accidents.

VIII-1.6.4. Proliferation resistance

The Package-Reactor is an LWR with moderation ratio similar to conventional LWRs, so the properties of the fresh and spent fuel are also very similar. Therefore, the proliferation resistance features are also similar to those of conventional LWRs, that is, the enrichment of fresh fuel by ^{235}U is less than 5% (by weight) and it is difficult to convert it to a weapon-grade material.

VIII-1.6.5 Technical features and technological approaches used to facilitate physical protection of Package-Reactor

The safety concept of the Package-Reactor is based on inherent safety features and passive safety systems, which do not depend on operator actions or external supports such as water and power supplies; this enhances the protection against external impacts and sabotage.

VIII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of Package-Reactor

The Package-Reactor is an ultra small reactor with small initial capital costs favourable to a reduction of the investment burden and risk. Simple and sound safety features could also be helpful for public acceptance around the construction sites. The power scale is suitable for energy demand in remote regions where it is difficult to construct long-distance power transmission and distribution facilities.

This reactor does not require multiple operation staff or large areas for siting and could be constructed easily. Essentially, the reactor might be installed everywhere.

VIII-1.8. List of enabling technologies relevant to Package-Reactor and status of their development

Basically, confirmation tests are needed for the stability of a two-phase flow between the cassettes and a SG and for the safety system based on natural air circulation cooling of the cassette. Tests on core criticality will be done during the operation of the first-of-a-kind package-reactor.

VIII-1.9. Status of R&D and planned schedule

The design and development of this reactor is performed by Hitachi and MHI, Japan. The basic design feasibility of the Package-Reactor has been confirmed. Regarding the R&D, the Hitachi and MHI will make a decision based on consideration of the market needs and other factors.

VIII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The design will be qualified through analytical and experimental work, and the technologies would be demonstrated during the operation of the first-of-a-kind Package-Reactor.

VIII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No other similar SMRs are under design elsewhere.

REFERENCES

[VIII-1] HIBI, K., TAKIMOTO, H., CHAKI, M., MATSUURA, M, MORIYA, K., Development of package-reactor (1) Reactor for pioneer of new markets, Progress in Nuclear Energy, Vol. 47, No.1–4, pp 115–122 (2005).

[VIII-2] HINO, T., et al., Development of Package-Reactor (2) — Core characteristics, Progress in Nuclear Energy, Vol. 47, No.1–4, pp 123–130 (2005).

ANNEX IX

PARTICLE FUEL PRESSURIZED WATER REACTOR (PFPWR50)

Hokkaido University,
Japan

IX-1. General information, technical features and operating characteristics

IX-1.1. Introduction

The PFPWR50 is a particle fuel pressurized water reactor of 50 MW(th).

It has been acknowledged that nuclear energy has a strong role to play in the global energy market, particularly if the goal is sustainable development. However, it is also true that deployment of nuclear energy can be said to have stagnated on a global scale except in a few countries. It has been clearly shown in an opinion survey recently carried out in Japan that the most important issue of nuclear energy is fear of radiation exposure. Radiation cannot be seen or felt directly and there is a strong belief that radiation dosage should be as low as possible. Such beliefs might be changed in the future; however, it will take a long time. The other reason for stagnation is the “NIMBY; Not In My Back Yard” concept referred to by the public, pointing that some technical aspects of conventional nuclear systems are still insufficiently understood. This might have been caused by the lack of knowledge about the innovations in nuclear power. In such situation, certain research and academic circles reflecting public concerns about nuclear power could indicate pathways to improve the situation. It could be very important for the smooth deployment of nuclear energy on a global scale to change NIMBY into “CIMBY; Come Into My Back Yard or Construct In My Back Yard.” However, the CIMBY on its own cannot solve problems of public acceptance; it is the reactor concept that must satisfy the requirement of zero radiation release under any conditions. Only when this is guaranteed there would be no difference between nuclear power plants and conventional industrial plants built close to residential areas.

The problem of public acceptance is not easily solved by technical discussion. Explanations of the total system must allow people to understand the safety, reliability, design philosophy and so on, to their satisfaction. A very good way to satisfy people is to show rigid proofs of excellent operating experience with regard to the technologies to be applied. From this point of view, the operating experience of light water reactors all over the world is a good example. They have shown excellent performance over the decades. We cannot forget that these operating records have been supported by a large amount of investments for research and development of fuel, materials and maintenance technology by relevant governments and industries. Maximum use should be made of these precious data for the future development of nuclear energy.

Based on these considerations, proposed is a reactor concept based on a small, pressurized water reactor (PWR) using coated particle fuel within conventional fuel rod claddings. As it is widely known, coated particle fuel has an excellent capability of fission product confinement up to about 1600°C for long periods and up to 2100°C within short periods [IX-1]. The reason for using the claddings is to take advantage of the long operating experience of the fuel rods and to avoid fire and eliminate the option for the particles to contact air or oxygen directly during accidents. This could help assure zero radiation release. The particles are loaded into a fuel rod with graphite to make the elements compact and to facilitate neutron moderation. The graphite also contributes to higher thermal conductivity of the fuel, thus keeping the fuel temperature lower than in conventional ceramic pellets.

Since the fundamental technologies have been already developed, there are no requirements for large scale R&D except for the technology to manufacture larger sized kernels of particle fuel.

The proposed reactor concept also targets to operate for as long a period as possible without refuelling. Specifically, the PFPWR50 could achieve a 10-year period of operation without on-site refuelling with the initial enrichment of uranium fuel not exceeding a moderate 5% by weight. Since the volume ratio of C to UO_2 is about 9 for current high temperature gas cooled reactor (HTGR) fuel and this is insufficient for a long operating life; it is, therefore, necessary to increase fuel loading in the core. Since a small reactor is economically "handicapped" by size, the system design should be as simple as possible. For example, a soluble boron system required for reactivity control during burn-up could be eliminated by introducing the burnable poison (BP), B_4C. The design and development of the PFPWR50 has been carried out by Hokkaido University, Japan with the cooperation from Mitsubishi Heavy Industries, Ltd., and Nuclear Development Corporation, Japan.

IX-1.2. Applications

The PFPWR50 nuclear power plant is designed to produce 50 MW of thermal power with the main usage for district heating and hot water supply.

IX-1.3. Special features

The PFPWR50 is a land-based nuclear power station.

IX-1.4. Summary of major design and operating characteristics

Installed capacity

The reactor is designed to produce 50 MW(th).

Mode of operation

The PFPWR50 based plant can be operated in base load and load follow modes.

Load factor/Availability

The target lifetime load and availability factors for the PFPWR50 are 70% and 90%, respectively.

Some major design characteristics of the PFPWR50 are given in Table IX-1.

TABLE IX-1: MAJOR DESIGN CHARACTERISTICS OF PFPWR50

ITEM	SPECIFICATION
Thermal output	50 MW(th)
Fuel type	TRISO type (UO_2 kernel)
Enrichment	5% ^{235}U by weight
Coolant and moderator	Light water
Linear power density	87.2 W/cm
Heavy metal load	5.2 t
Fuel rod pitch	~33 mm
Fuel rod diameter	~29 mm

ITEM	SPECIFICATION
Cladding material	Zircaloy-4
Guide tube material	Zircaloy-4
Lattice type	Hexagonal
Number of fuel rods in an assembly	37/31 (with guide tubes)
Number of fuel assemblies	85
Number of fuel assemblies with guide tubes	31
Equivalent core diameter	~2.0 m
Effective core height	~1.8 m

A schematic diagram of the PFPWR50 plant is shown in Fig. IX-1.

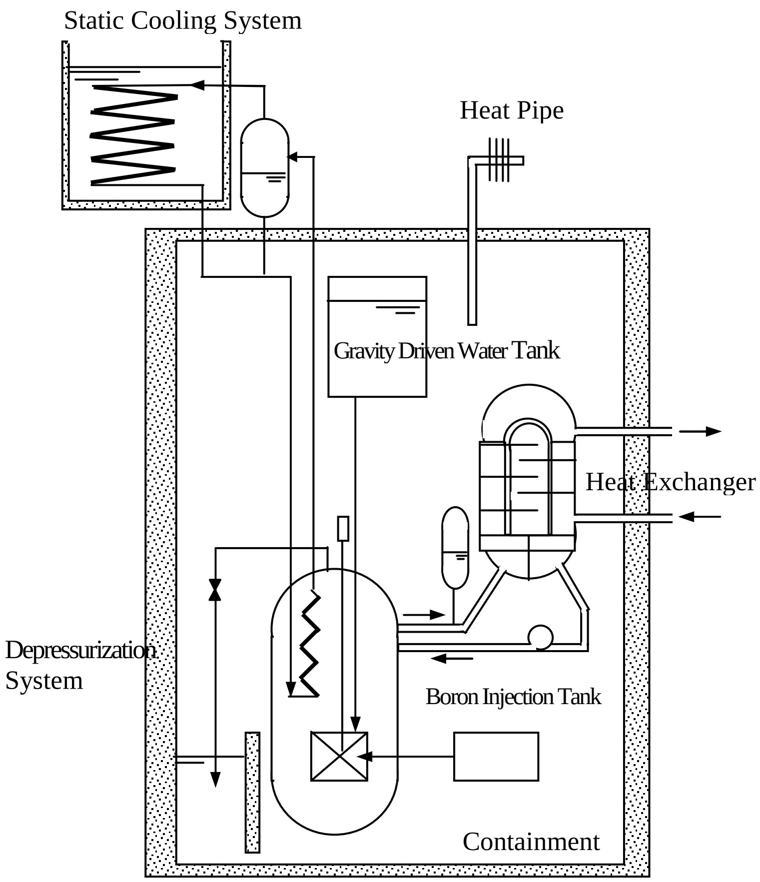


FIG. IX-1. Schematic diagram of the PFPWR50 nuclear power plant.

The reactor core is housed in a cylindrical stainless steel pressure vessel containing light water as a coolant; it also acts as a moderator and reflector. The coolant circulation is driven by natural convection or by a conventional pump (to be determined in the near future).

The energy generated in the core is used to heat-up the secondary loop water to be used for district heating and hot water supply.

A passive containment cooling system is incorporated to remove heat from the containment following LOCA. This system removes the energy released into the containment through heat pipes. The reactor is cooled by water supplied from the Gravity Driven Water Tank (GDWT), which can supply water directly into the reactor, driven by gravity, with the help of a depressurization system located in the containment. The heat from the reactor is removed by natural convection through a hydraulic communication with the Static Heat Exchanger located outside of the containment.

Neutron-physical characteristics

The reactor physics design of the PFPWR50 has been optimized to achieve the following objectives:

- Long core life, such as 10 years, without refuelling;
- Negative coolant temperature reactivity coefficient;
- Negative void reactivity coefficient;
- Minimization of burn-up reactivity swing during the core life;
- High fuel burn-up.

To flatten the excess reactivity through a cycle of one batch, 54 BP assemblies with 5 different quantities of burnable poison (B_4C) are required, Fig. IX-2. Reactivity control systems could be simplified with this loading pattern. With this, the PFPWR50 core can achieve an average discharge burn-up of 25 400 MW day/t and is capable of operating for 7.3 effective full power years (EFPY) without reloading and shuffling of fuel. Also, analysis of the moderator temperature coefficient (MTC), Doppler reactivity coefficient, and void reactivity coefficient calculated at BOL and EOL confirmed that they all are negative, Table IX-2.

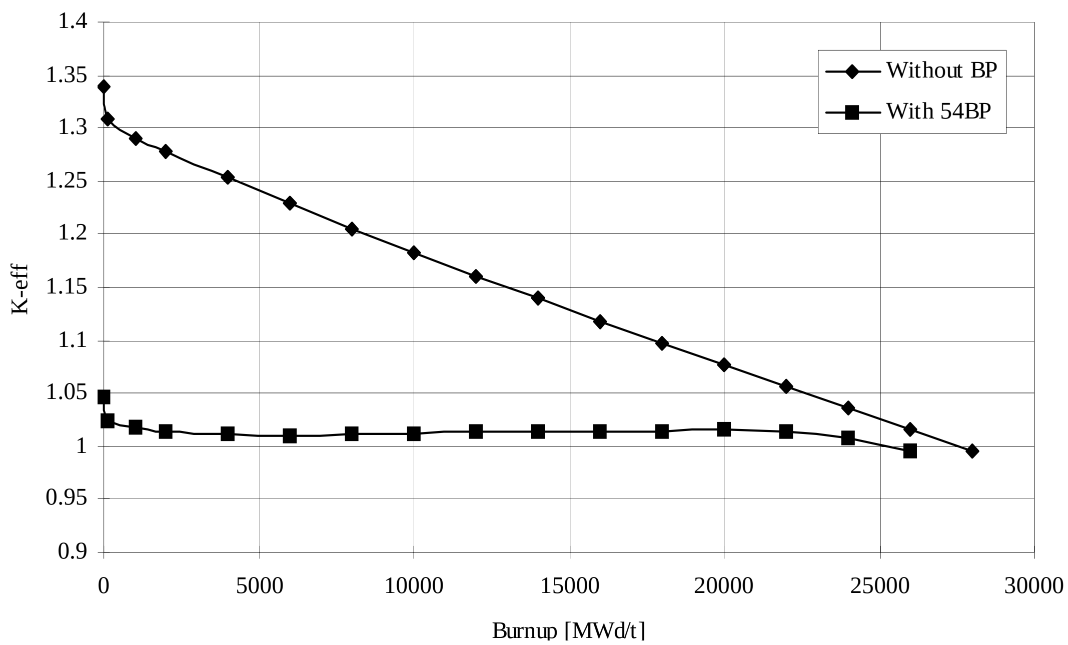


FIG. IX-2. Burn-up characteristics of PFPWR50.

TABLE IX-2. REACTIVITY COEFFICIENTS AT BOL AND EOL

	BOL	EOL
MTC [$10^{-5}\,\Delta k/k$ /K]	–32.4	–26.2
Void reactivity coefficient [$10^{-5}\,\Delta k/k$ /%]	–160	–151
Doppler coefficient [$10^{-5}\,\Delta k/k$ /% power]	–3.4	–7.3

Reactivity control mechanism

It is necessary to load 31 fuel assemblies with guide tubes for control rods (GT assemblies) to gain a satisfactory shutdown margin under cold zero power shutdown conditions. The Reactor Protection System comprises two independent fast acting shutdown systems. Shutdown System-1 (SDS-1) is based on mechanical shutdown rods with boron carbide based absorbers in 31 fuel assemblies; it provides sufficient negative reactivity with all rods inserted, with one maximum worth rod not available, in the cold shut down condition. Shutdown system-2 (SDS-2) is based on liquid poison injection into the moderator.

Cycle type and thermodynamic efficiency

The PFPWR50 is designed to supply primary loop water of 260°C and 8.6 MPa to a heat exchanger of a system of hot water supply for district heating. No energy conversion is foreseen; therefore, thermal efficiency is not applicable.

Thermal-hydraulic characteristics

The main thermal-hydraulic characteristics of the PFPWR50 are given in Table IX-3.

TABLE IX-3. THERMAL-HYDRAULIC CHARACTERISTICS OF PFPWR50

CHARACTERISTIC	VALUE/DESCRIPTION
Type of circulation	Forced flow (current design)
Coolant conditions	Pressurized non-boiling water
Average primary coolant temperature	250°C
Core outlet temperature and pressure	260°C and 8.6 MPa
Fuel temperatures during normal operation	To be evaluated
Secondary water temperature at heat exchanger inlet	130°C (current design)
Secondary water temperature at heat exchanger outlet	200°C (current design)

The maximum critical heat flux ratio (MCHFR) is expected to be sufficiently high due to the low power density and low operating temperature; detailed analysis is yet to be performed.

Fuel lifetime/period between refuellings

The period of reactor operation without refuelling may be more than 7.3 EFPY with 5% enriched UO_2 fuel; all fuel assemblies are to be replaced at the end of life.

Mass balances/flows of fuel and non-fuel materials

The annual consumptions of materials, based on the annual load factor of 70% (the output is thermal energy only), are as follows:

Natural uranium:	9970 kg/GW(th)/year
Zircaloy-4:	6120 kg/GW(th)/year
Graphite:	7010 kg/GW(th)/year

Design basis lifetime for reactor core, vessel and structures

The design basis lifetime of all non-replaceable structures is 60 years; components and equipment with a lower design lifetime will be easily replaced during routine shutdowns.

Design and operating characteristics of systems for non-electric applications

The plant supplies hot water at about 200^{o}C to a conventional heat supply system, which could be a thermal supply grid or some centralized heat consumer.

Economics

No information was provided.

IX-1.5. Outline of fuel cycle options

A once-through fuel cycle is planned until fuel reprocessing of coated particle fuel can be effectively and economically achieved. Once available, centralized or regional reprocessing would be expected.

IX-1.6. Technical features and technological approaches that are definitive for PFPWR50 performance in particular areas

IX-1.6.1. Economics and maintainability

The advanced safety features and low anticipated capital cost of the PFPWR50 are expected to facilitate its deployment in developing countries with limited financial and technological resources.

The main design features of this reactor leading to reduced capital costs are:

- Elimination of a boron chemical system;
- Direct use of heat energy for district heat supply; and
- Shop-assembled reactor cores.

The reactor has two important provisions to reduce O&M costs: the first is the elimination of on-site refuelling with a long-life core; the second is simplification of reactivity regulation during the core lifetime by optimizing the distribution of burnable poison in the core and fuel assemblies.

Through optimizing the ratio of light water and graphite, the PFPWR50 achieves a core average discharge burn-up of 25 400 MW day/t, which is lower than in current light water reactors. One reason for lower fuel burn-up is that the core is operated in a single-batch refuelling scheme. The resulting economical disadvantages could be reduced by simple operation and fewer system components.

IX-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The fuel is composed of small UO_2 kernels coated with a few layers of PyC and SiC, and the particles are further shaped into compacts within a graphite matrix. The compacts are cladded by Zircaloy-4, ensuring the confinement of radioactive materials in-depth (together with the high confinement capability of the coated particle fuel). In addition, large heat capacity of the graphite moderator mitigates the reactor transients, with the target being to eliminate the consequences of accidents and relevant off-site emergency measures.

The major design provision of the PFPWR50, aimed at reducing dose levels to the population in the vicinity of the plant, is eliminated on-site refuelling. In addition to this, no reprocessing of spent fuel is currently planned.

IX-1.6.3. Safety and reliability

Safety concept and design philosophy

The emphasis in design has been to incorporate inherent and passive safety features to the maximum extent as a part of the defence in depth strategy. The main objective has been to establish a case for the elimination of evacuation planning following credible accident scenarios in the plant. Another major objective has been to provide a grace period for the absence of any operator or powered action in the event of a credible accident scenario.

Provisions for simplicity and robustness of the design

Several passive safety features have been incorporated to ensure simplicity and robustness of the PFPWR50 design, such as low power density in the core, coated particle fuel with low operating temperature, and graphite moderator with large heat capacity. The SiC layer of each fuel kernel is a containment for itself, an important prerequisite to simplify the plant design.

Active and passive systems and inherent safety features

The main inherent safety features of the PFPWR50 are:

- Negative void coefficient of reactivity;
- Negative moderator temperature coefficient;
- Perfect confinement of fission products at high temperatures, 1600–2100°C.

The main passive safety features and passive systems of the PFPWR50 are:

- Natural circulation driven heat removal during hot shutdown conditions;
- Passive injection of low-pressure emergency core coolant, driven by gravity;
- Passive residual heat removal based on heat pipes and atmospheric water heat exchanger.

As described above, the reactor protection systems (control rods and liquid boron injection) are active.

Structure of the defence-in-depth

Some major highlights of the PFPWR50 design, structured in accordance with the various levels of defence in depth are brought out below:

Level-1: Prevention of abnormal operation and failure

Characteristics of the PFPWR50 design which help to reduce the extent of an overpower transient are the following:

- Negative void coefficient of reactivity;
- Negative moderator temperature coefficient;
- Low core power density,;
- Low excess reactivity, due to the extensive use of burnable poison.

Conditions in all important equipment and components will be continuously monitored on line. For example, the annulus gas monitoring system is incorporated to monitor postulated leakage from the primary loop.

Level-2: Control of abnormal operation and detection of failure

Characteristics of the PFPWR50 design, which contribute to this level, are the following:

- Large coolant inventory in the main coolant system;
- Increased reliability of the control system achieved with highly reliable digital control using advanced information technology based on proven technologies;
- Increased operator reliability achieved with advanced displays and diagnostics using artificial intelligence and expert systems.

Level-3: Control of accidents within the design basis

The following features contribute toward the achievement of this objective:

- Increased reliability of the emergency core cooling (ECC) system achieved through depressurization of the primary system and passive injection of cooling water from the gravity driven water tank (GDWT) directly into the core;
- Increased reliability of a shutdown achieved by two independent shutdown systems, one comprising mechanical shut-off rods and the other employing injection of liquid poison into the low pressure moderator; each system is capable of an independent shutdown of the reactor;
- Increased reliability of decay heat removal achieved through a passive decay heat removal system, which transfers decay heat to a static cooling system by natural convection;
- The large heat capacity of the graphite moderator (the water in the GDWT provides prolonged core cooling, meeting the requirements for an increased grace period).

Level-4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents

Use of a graphite moderator as a heat sink facilitates the achievement of this objective.

Level-5: Mitigation of the radiological consequences of significant releases of radioactive materials

The following features of the PFPWR50 help in passively minimizing the releases from the containment following a LOCA:

- The particle based fuel elements with high fission product confinement capability;
- A large heat capacity of the graphite moderator;
- Passive containment cooling.

Design basis accidents and beyond design basis accidents

Due to the negative reactivity coefficients, large heat capacity of the graphite moderator and reliable high temperature performance of the particle fuel, together with low power density and low operating temperature, any design basis accident is expected to be terminated in much a milder way than predicted for current light water reactors. In addition to this, perfect confinement of fission products at temperatures up to 1600–2100°C would be assured.

Provisions for safety under seismic conditions

The PFPWR50 structures, systems and equipment will be designed for high level and low probability seismic events such as a design basis maximum earthquake and a design basis limiting earthquake. Seismic instrumentation is planned in accordance with national and international standards.

Probability of unacceptable radioactivity release beyond the plant boundaries

No analysis has been performed so far.

Measures planned in response to severe accidents

One of the important design objectives for the PFPWR50 is significant enhancement of the level of safety to eliminate the need for intervention in the public domain, beyond the plant boundaries, as a consequence of any postulated accident within the plant.

IX-1.6.4. Proliferation resistance

Some of the important technical features of the PFPWR50 to reduce the attractiveness of its spent fuel for clandestine nuclear weapon programmes are the following:

- Fissile materials are contained in small particles coated with three or four layers of PyC and SiC, thus making reprocessing to extract pure fissile materials by aqueous methods difficult;
- At the end of the core life, spent fuel assemblies are removed and transported to centralized fuel storage or reprocessing facilities. This will be required only every 7.3 EFPY and, therefore, fuel assemblies would be not easy to steal or transport. It is assumed that the reactor vessel cover is not opened during the whole period of reactor operation.

IX-1.6.5. Technical features and technological approaches used to facilitate physical protection of PFPWR50

The PFPWR fuel assemblies are not handled separately except for whole-core replacement at the end of each core lifetime, every 7.3 EFPY. Each coated particle has a high temperature corrosion-resistant SiC coating layer, which acts as a containment on its own, providing a large margin to fuel failure (~1300°C) in all conceivable accidents, including those initiated by human actions of malevolent origin.

IX-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of PFPWR50

The PFPWR50 is a concept developed in Japan with a local community perspective; it may also suit the needs of local communities in some developing countries.

IX-1.8. List of enabling technologies relevant to PFPWR50 and status of their development

The enabling technologies for the PFPWR50 are summarized in Table IX-4.

TABLE IX-4. ENABLING TECHNOLOGIES FOR PFPWR50 AND THEIR DEVELOPMENT STATUS

OBJECTIVE	ENABLING TECHNOLOGIES	STATUS OF DEVELOPMENT
Use of a larger diameter coated particle fuel	New fabrication technology	Conceptual design completed.
Negative void coefficient in all modes of operation	Optimized hydrogen-to-carbon (H/C) ratio	Feasibility demonstrated.
		Critical facility experiments being planned.
Optimum use of passive systems for core heat removal	Large passive heat sink within containment	Several experiments are ongoing and future experiments are under planning.
Increased fuel burn-up	Optimized H/C ratio Use of graphite moderator	Validation in a prototype reactor would be needed
Reduced O&M costs	Elimination of on-site refuelling	Fuel handling and storage system to be designed.
	60-year design lifetime of main components	To be achieved through material selection and selection of design approaches to facilitate easy management of ageing. No major R&D required.
Reduced capital cost per MW installed	Full use of heat energy achieved through a combination of different applications	R&D to be performed.
Enhanced safety following LOCA	Passive ECCS (Emergency Core Cooling System) of enhanced effectiveness	R&D to be performed.
	ECCS injection directly into the core	R&D to be performed.
	Passive containment cooling	R&D to be performed.

IX-1.9. Status of R&D and planned schedule

The design and technology development for the PFPWR50 are mainly carried out by the Hokkaido University, Japan in cooperation with Mitsubishi Heavy Industries, Ltd. and Nuclear Development Corporation (Japan). The R&D for the PFPWR50 has been and is being carried out without any financial support from the Government of Japan but only through the private research activity of the Hokkaido University.

The current design stage is that of a feasibility study, with only some developments matching a conceptual design stage. It is projected that by the end of 2010 the design could be sufficiently complete to enable initiation of construction related actions, subject to the availability of funds and cooperation with the industry.

IX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Elaboration and validation of fabrication technology is required to produce larger UO_2 or MOX fuel kernels with multi-layer PyC and SiC coatings. This kernel enlargement is essential for longer core life without refuelling. The fuel based on coated particles in a graphite matrix cladded by Zircaloy is essentially new for PWRs. The performance characteristics of such fuel are not precisely known; therefore, a substantial amount of R&D and a demonstration in a prototype reactor would be required.

IX-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

IX-2. Design description and data for PFPWR50

IX-2.1. Description of the nuclear systems

Reactor core and fuel design

The proposed reactor uses HTGR type fuel, i.e., TRISO coated particle fuel packed in a graphite matrix. TRISO fuel consists of a UO_2 sphere covered with 4 coating layers: porous PyC, dense PyC, SiC, and dense PyC again. It is proven that these layers all play significant roles in confining fission products at high temperatures and high fuel burn-ups. Recent experimental studies in Russia have proven the long-term corrosion resistance properties of non-irradiated coated particles with an outer SiC coating layer in water under the operating parameters of a PWR. These tests also demonstrated acceptable short-term corrosion resistance of SiC in steam and steam-air mixtures at very high temperatures, up to 1000°C [IX-2].

The volume ratio of C to UO_2 is usually 9.0 for HTGR fuel. A study has been conducted with this fuel composition, to confirm the feasibility of a PWR with such fuel being cladded by Zircaloy-4 [IX-3 to IX-5]. It concluded that excess reactivity can be effectively suppressed, power flattened and control rod programming substantially simplified by mixing Gd_2O_3 in the UO_2 fuel kernel. However, in this study, the volume of UO_2 was intentionally doubled, i.e., UO_2:C = 1:4, to achieve a longer operation period. Actually, such increase can be achieved by applying a larger size UO_2 kernel, in other words by changing the current kernel size of about 0.6 mm in diameter to the diameter of around 2 mm. The burnable absorber in the fuel was also changed from Gd_2O_3 to B_4C to control reactivity for a longer operating interval. B_4C burnable poison (BP) is loaded in the graphite matrix. The compatibility of graphite with Zircaloy-4 in normal operation is quite good and both fuel and cladding are chemically stable. The fuel concept is shown in Fig. IX-3.

The PFPWR fuel assembly can hold a total of 37 fuel rods in a tight hexagonal lattice, making the core size as small as possible. Two types of special fuel assemblies are present in the core: guide tube (GT) assemblies and burnable poison (BP) assemblies, Fig. IX-4. The GT assembly has control rod guide tubes at each corner and 31 normal fuel rods in the remaining space. The BP assembly has 18 fuel rods that contain B_4C in their graphite matrix and are placed along the periphery. The fuel rod pitch was selected as approximately 3.3 cm to obtain the highest average discharge burn-up.

A total of 85 fuel assemblies are loaded in the core of approximately 2 m diameter, surrounded by a light water reflector (Fig. IX-5). The PFPWR50 thermal output is 50 MW therefore, the average linear power density is half that of typical PWRs. Light water with an average temperature of 250°C is circulated up through the core, acting as a coolant and moderator.

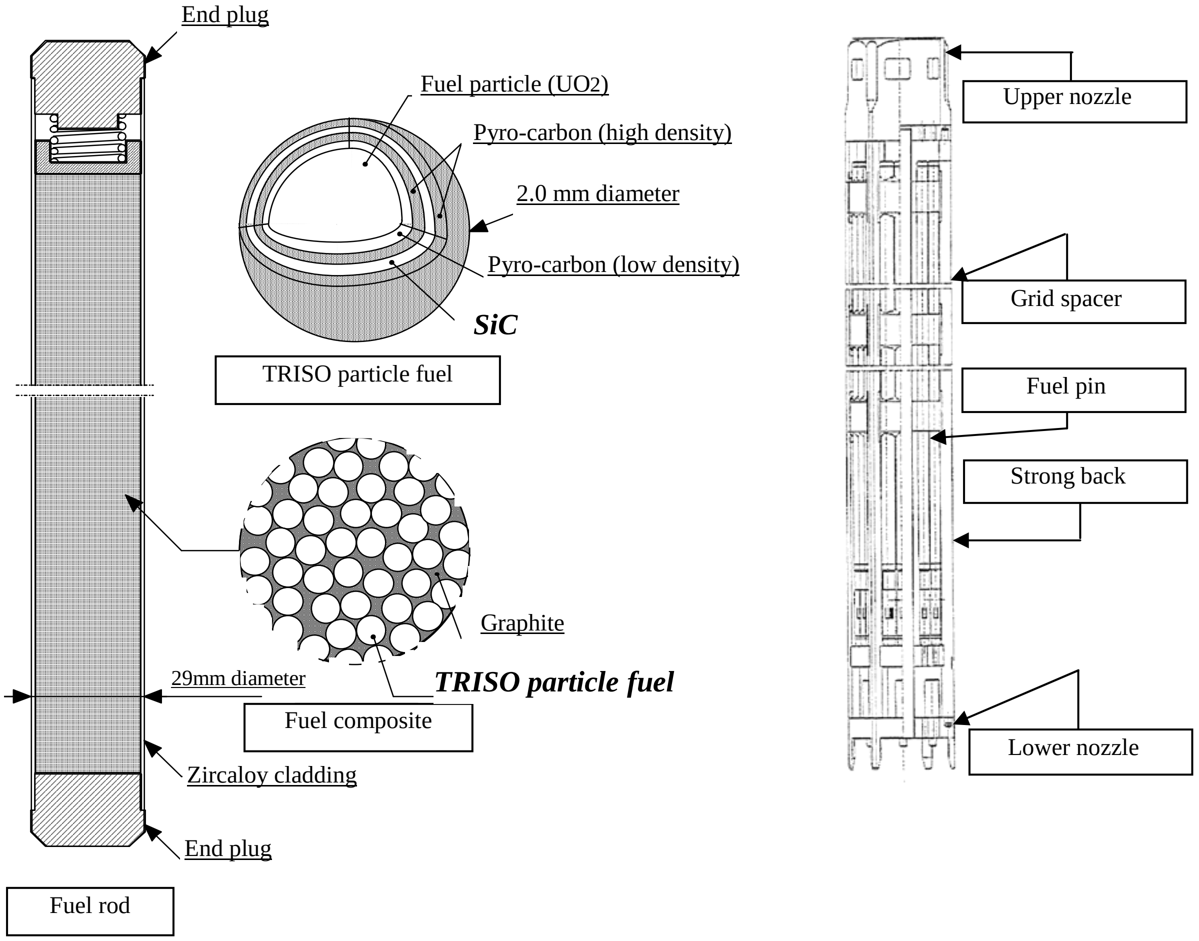


FIG. IX-3. PFPWR50 fuel and fuel assembly design.

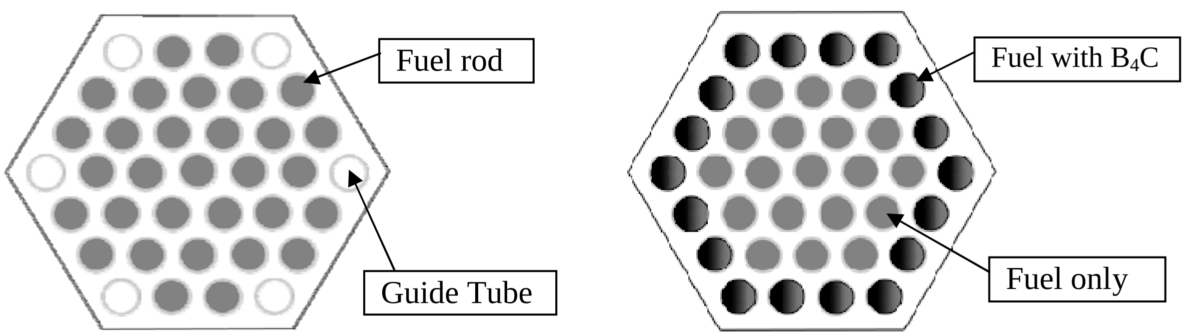
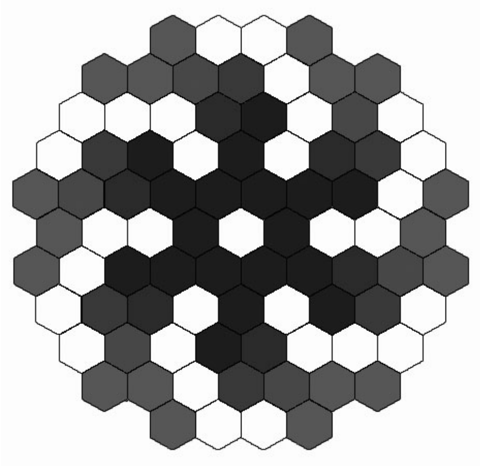


FIG. IX-4. Cross-sections of GT (left) and BP (right) fuel assemblies.



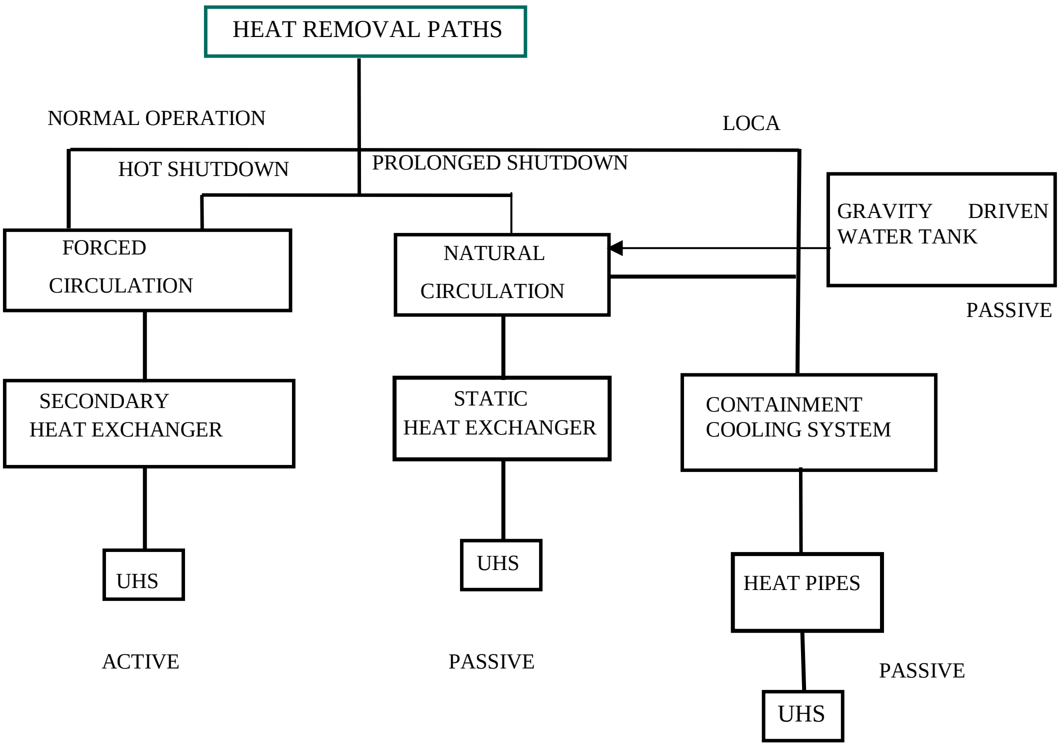
(BP, GT and standard fuel assemblies are in grey, black and white, respectively)

FIG. IX-5. Horizontal cross-sectional view of the PFPWR50 core.

Main heat transport system

The function of the main heat transport system is to remove nuclear heat from the reactor core in forced circulation mode under normal operation and in natural convection mode under shutdown conditions.

The heat removal paths of the PFPWR50 under various operating states and in LOCA are shown in Fig. IX-6.



UHS is for ultimate heat sink

FIG. IX-6 Heat removal paths of PFPWR50.

REFERENCES

[IX-1] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors, IAEA-TECDOC-978, Vienna (November 1997).

[IX-2] PONOMAREV-STEPNOI, N.N., et al., Prospects of VVER reactors with micro fuel elements, Atomnaya Energia, vol. 86 (6) (June 1999, in Russian).

[IX-3] SHIMAZU, Y., TOCHIHARA, H., AKIYAMA, Y., ITHO, K., PWRs using HTGR fuel concept with cladding for ultimate safety, ARWIF 2001 (Workshop Proceedings, Chester, UK) pp. 235–243 (Oct. 2001).

[IX-4] KAUCHI, M., SHIMAZU, Y., Optimization of burnable poison loading in a PWR with carbon coated particle fuel, ICAPP'02 (Proc. Int. Congress, Hollywood, FL, June 9–13, 2002).

[IX-5] KAUCHI, M., SHIMAZU, Y., Optimal burnable poison loading in a PWR with carbon coated particle fuel, Nuclear Science and Technology, Vol.40, No.1, pp. 22–29 (2003).

ANNEX X

BOILING WATER REACTOR WITH MICRO FUEL ELEMENTS (VKR-MT)

**VNIIAM, Russian Research Centre “Kurchatov Institute”,
Russian Federation**

X-1. General information, technical features and operating characteristics

X-1.1. Introduction

The VKR-MT is a vessel type reactor with the core based on micro fuel elements directly cooled by boiling water. VKR-MT is an English spelling of the Russian abbreviation for boiling water reactor with micro fuel elements.

VKR-MT is a direct successor of the VK-300 reactor developed for the renovation of reactor facilities previously used for weapon plutonium production [X-1]. The VKR-MT also makes use of the basic propositions of a concept of VVER reactor with micro fuel elements as developed by the Russian Research Centre “Kurchatov Institute” (RRC KI), the All-Russian Institute of Atomic Machinery (VNIIAM), and the Scientific and Production Association “Luch” (SPA “Luch”) [X-2].

The VKR-MT is a power reactor targeted at achieving a very high level of radiation safety through the elimination of significant releases of fission products from fuel in severe accidents, including the ones caused by reactor vessel rupture, by fall down of a heavy plane and by human actions of malevolent character. Being thoroughly based on the intrinsic safety features, light water reactors with micro fuel elements could create favourable conditions for a wide scale deployment of nuclear power, especially in developing countries.

The VKR-MT concept is based on:

- The experience in operation of VVER and BWR reactors;
- The experience in the design and operation of micro fuel elements (coated particles) in reactors of HTGR type;
- The results of the out-of-pile tests of micro fuel elements in water coolant performed by the RRC KI and VNIIAM;
- The results of the in-pile irradiation tests of micro fuel elements in water coolant performed in the IVV-2M research reactor at Zarechny (Russia);
- The design experience for fuel assemblies with micro fuel elements as accumulated in the VNIIAM;
- The results of calculation optimization of the design scheme of a micro fuel element in application to the VVER operating conditions;
- The experience in the design of a refuelling system that performs once-at-a-time reloading of micro fuel elements in an hourglass mode.

X-1.2. Applications

Small NPPs with the VKR-MT reactors are targeted for deployment in the developing countries with insufficient nuclear infrastructure, as well as in the remote areas disconnected from the main electricity grids. An NPP with VKR-MT could supply settlements and industrial facilities with electricity, heat, and steam, and could be also used for the production of potable water and district heating. The plants can be located in the immediate vicinity of populated areas and can operate in a basic load as well as in a load follow mode. The ratio between electricity and heat production is variable. The NPP could operate in a purely condensation mode with zero heat-extraction load, while at a heat-extraction load of 600 MW it could cogenerate 180 MW of the electric power. The VNIIAM, the RRC KI and the SPA "Luch" have performed a significant number of mock-up and in-pile tests and design studies for the VKR-MT; altogether, they prove the feasibility of this concept.

X-1.3. Special features

The VKR-MT is developed for a land-based NPP. Maximum use of the benefits provided by micro fuel elements in a pebble bed form makes it possible to develop a reactor design, in which the core operates for 10 years without reloading and shuffling of fuel, and the refuelling is performed once-at-a-time without opening the reactor vessel cover, through the use of special containers with fresh and spent micro fuel elements and a dedicated micro fuel element hydraulic transport system. From the standpoint of proliferation resistance, such technical feature is equivalent to an option of factory fabricated and fuelled reactor, but essentially simplifies the associated transport operations. The refuelling could be performed by a special team dispatched from a fuel factory and shall involve strict control and verification measures.

Because of a high level of radiation safety, the NPP could be located near big cities as well as small settlements.

X-1.4. Summary of major design and operating characteristics

A power unit with the VKR-MT reactor includes:

- A direct cycle boiling water reactor with forced circulation of coolant by jet pumps;
- The reactor cooling system;
- A saturated steam turbine plant with a two-stage separation, without the reheater;
- Heating unit;
- A process steam production unit (for example, for seawater desalination);
- The residual heat removal system based on passive operation with all media and having no active stop or control valves, except for the maintenance valves;
- The refuelling system for micro fuel elements;
- Safety systems.

Figure X-1 presents a schematic diagram of the VKR-MT power unit. Figure X-1 shows only heat exchangers of the systems of district heating and industrial steam production. Heat exchangers of the intermediate circuit of a district heating system are not shown. The steam turbine plant has no such potentially non-reliable components as high-pressure heaters or low pressure reheaters of steam, which contributes to its high reliability. The reduction of moisture in steam at turbine outlet down to an acceptable level is accomplished by the use of two-stage steam separation and by the separation devices located inside the turbine.

The scheme of coolant circulation inside the reactor vessel is shown in Fig. X-2. Jet pumps that are driven by the excess pressure drop created by an electric pump of the steam turbine circuit provide forced re-circulation of water inside the reactor vessel. The use of jet pumps contributes to an increased reliability of the VKR-MT power unit.

The major design and operating characteristics of the VKR-MT are summarized in Table X-1. The neutron physical characteristics are given in Table X-2.

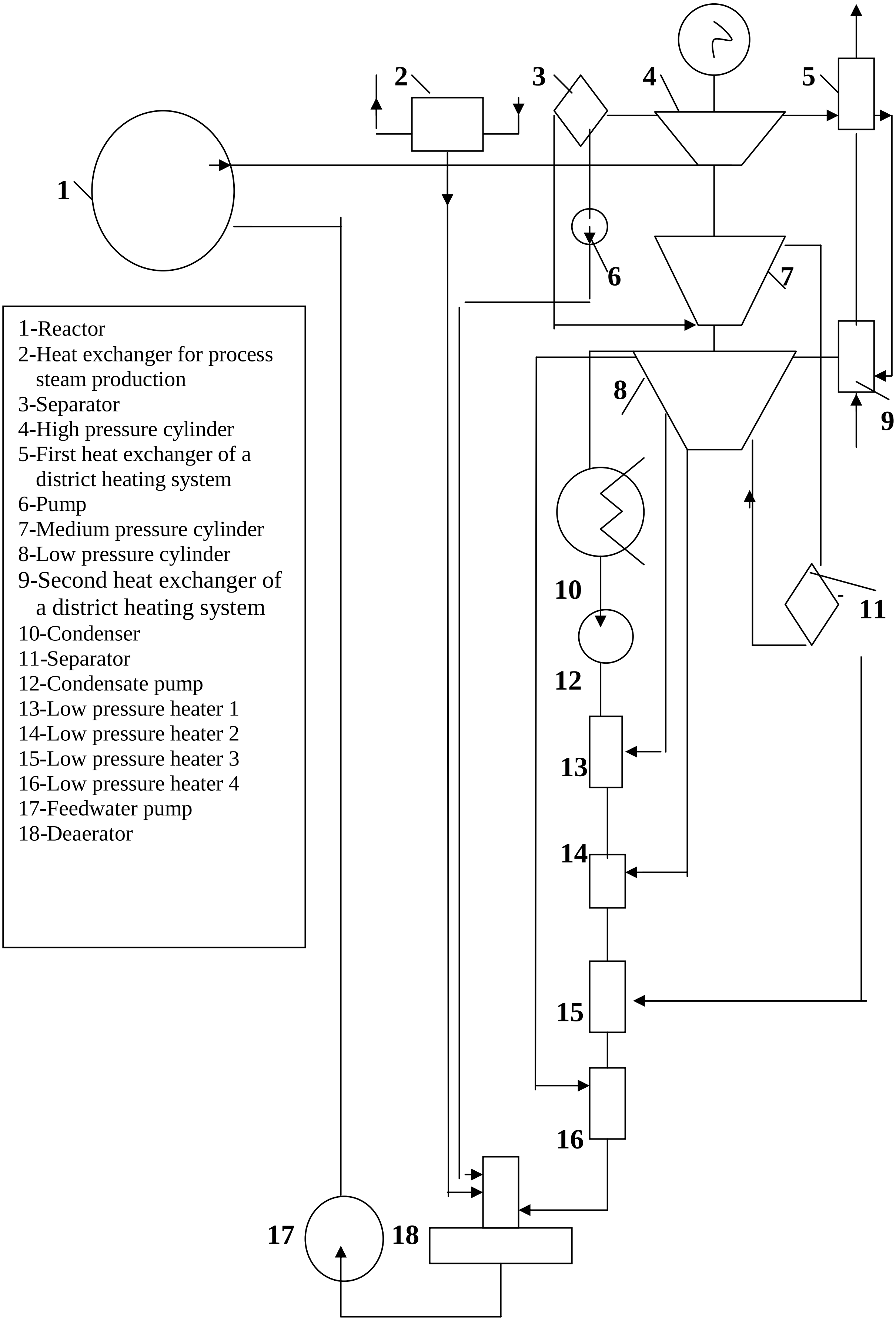


FIG. X-1. Schematic diagram of an NPP with the VKR-MT reactor.

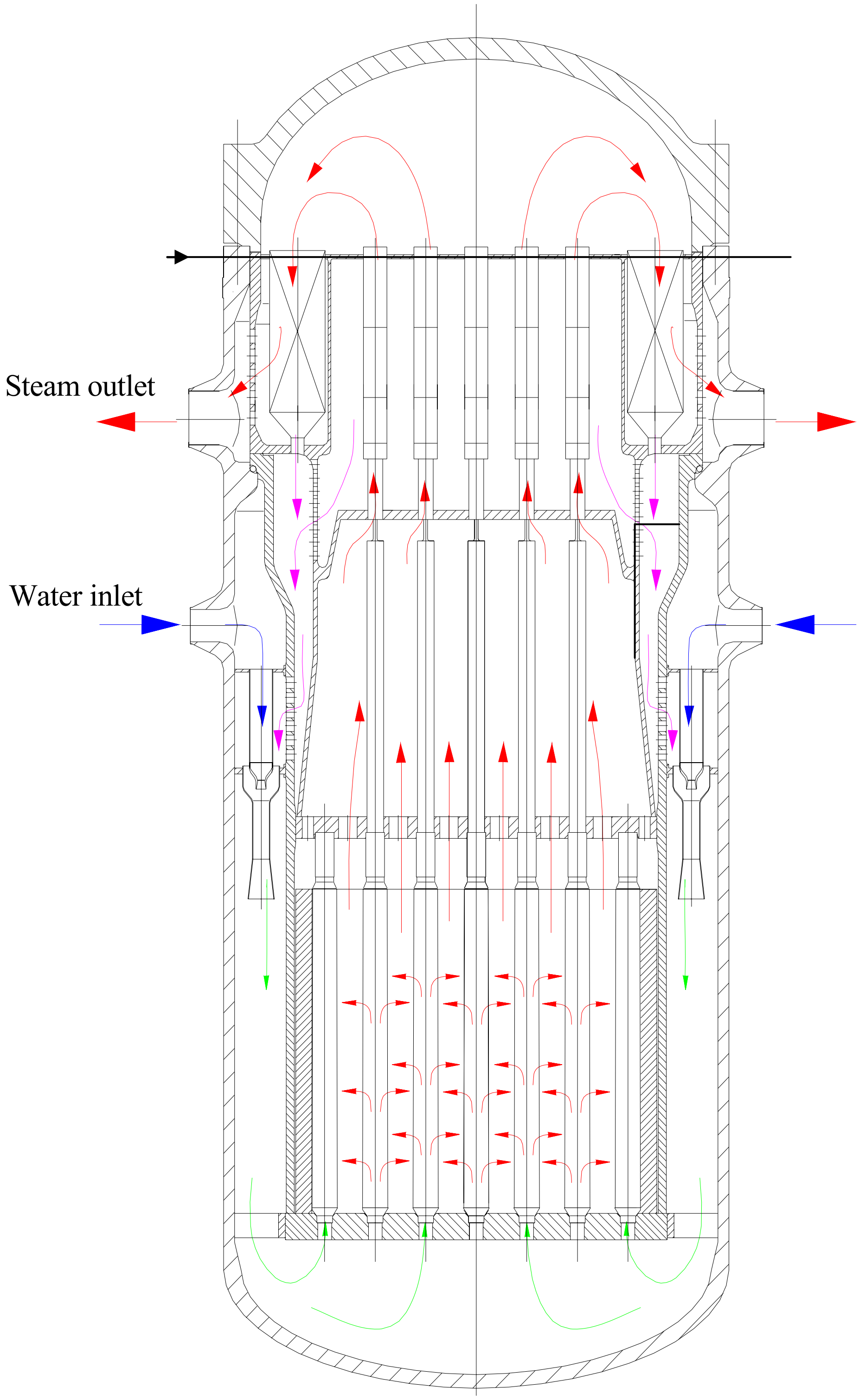


FIG. X-2. Scheme of coolant circulation inside the reactor vessel.

TABLE X-1. SUMMARY TABLE OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

CHARACTERISTIC	VALUE OR DESCRIPTION
Thermal power, MW Maximum electric power, MW Heat supply for industrial needs and/or for district heating, MW	890 300 600
Type of fuel	Uranium dioxide. The core is based on micro fuel elements of 1.8 mm outer diameter. The diameter of UO_2 kernel is 1.4 mm, the cumulative thickness of a multi-layer coating is 0.2 mm
Fuel enrichment	10% by ^{235}U
Coolant	Boiling water
Moderator	Boiling water
Structural materials	Vessel steel with welding deposition of stainless steel. The fuel assembly is made of austenitic steel.
Reactor core	Effective cylinder; contains 151 hexagonal fuel assemblies; the effective diameter is 3.0 m; the core height is 3.7 m
Reactor vessel	Vessel diameter: 5000 × 80 mm; vessel height: 11 000 mm,
Number of circuits; type of thermo-dynamic cycle	Single-circuit scheme typical of BWR; no reheating of steam; two separation stages in the turbine; extractions for district heating and process steam or potable water production
Plant type	BWR; mono-block
Mode of operation	Load follow and cogeneration
Plant efficiency, %	34
Load factor (target)	0.95
Plant lifetime, years	100

TABLE X-2. NEUTRON-PHYSICAL CHARACTERISTICS

CHARACTERISTIC	VALUE
UO_2 load, t	58.5
^{235}U load, kg	5156
Average/maximum fuel burn-up, % of fissile materials	6.1/9.8
Neutron fluence (E > 0.2 MeV) in micro fuel element coating, 10^{22} 1/cm^2	2.2
Lifetime of micro fuel element, effective days	3500
Doppler coefficient, 1/^{o}C	-5×10^{-5}
Void reactivity effect	<0
Reactivity margin for fuel burn-up, %	5.3
Radial power peaking factor	1.35
Volume power peaking factor	1.61

Figure X-3 presents the reactivity change over fuel burn-up cycle in a VKR-MT core with burnable poisons, calculated in the assumption that control rods are not inserted. It can be seen that the maximum burn-up reactivity to be compensated by control rods is around 5.5% Δk/k.

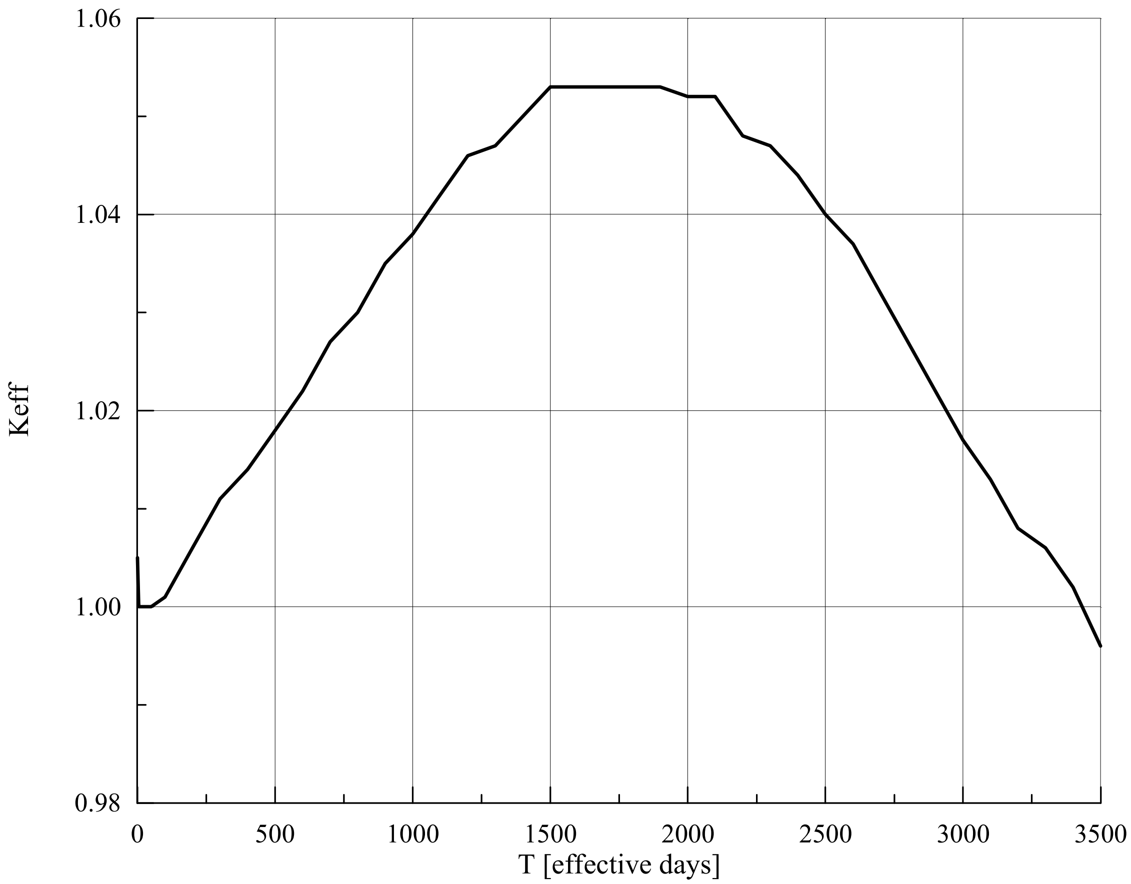


FIG. X-3. K_{eff} changes with fuel burn-up in a VKR-MT core.

The design provides for two independent and diverse reactivity control systems. The first system is based on control rods with electromagnetic drives. The second system is a liquid boron system providing for an injection of $Na_2B_4O_7$ solution to the coolant. The reactivity control mechanism is summarized in Table X-3.

Thermal-hydraulic characteristics of the VKR-MT are presented in Table X-4.

TABLE X-3. REACTIVITY CONTROL MECHANISM

TYPE OF REACTIVITY CONTROL	DESCRIPTION	PURPOSE
1. Mechanical system	104 electromagnetic drives with 12 boron carbide based control rods per each drive	Transfers core to a subcritical state without cooling down the reactor
2. Liquid boron system	System of liquid $Na_2B_4O_7$ injection	Transfers core to a subcritical state and cools down the reactor
3. Burnable poison	Borated steel	Compensates the margin for burn-up reactivity swing

TABLE X-4. THERMAL-HYDRAULIC CHARACTERISTICS

CHARACTERISTIC	VALUE
Circulation type	Forced
Core inlet pressure, MPa	7.5
Average/maximum fuel assembly power, MW	5.8/8.6
Average/maximum steam fraction at core outlet, %	12.5/25
Coolant flow rate through the core, kg/s	3650
Core inlet/outlet coolant temperature, °C	280/290
Average/maximum heat flow through the outer surface of a micro fuel element, MW/m^2	0.024/0.05
Fuel temperature, °C	285–295
Core pressure drop, bar	<0.1

A very large heat exchange surface in the core eliminates heat exchange crisis on the surface of micro fuel elements. Because of this, an NPP with the VKR-MT can operate in a load follow mode, which is important for energy systems of small capacity.

Table X-5 gives the material balances of the VKR-MT.

TABLE X-5. MATERIAL BALANCES

CHARACTERISTIC	VALUE
Fuel lifetime, effective full power days (EFPD) Interval between refuellings, EFPD Fraction of reloaded fuel, %	3500 3500 100
Annual consumption of ^{235}U without spent fuel reprocessing, kg/year	515
Specific consumption of ^{235}U without spent fuel reprocessing, kg/MW(e)/year	1.72
Specific consumption of ^{235}U with spent fuel reprocessing, g/(MW electric/thermal)	0.37/0.126
Specific consumption of natural uranium for an open fuel cycle, kg/MW(e)/year	287*
Specific consumption of natural uranium for a closed fuel cycle, kg/MW(e)/year	185*
Consumption of stainless steel	
Annual consumption of pyrolythic graphite, kg/year	300
Specific annual consumption of pyrolythic graphite, kg/(year MW electric/thermal)	1.0/0.34
Annual consumption of silicon carbide, kg/year	1200
Specific annual consumption of silicon carbide, kg/(year MW electric/thermal)	3.0/1.0

* The enrichment of depleted uranium is assumed to be 0.2%.

The use of micro fuel elements makes it possible to avoid using the expensive Zirconium alloys. The fuel assembly structures are made of stainless steel and remain in the core for the whole reactor lifetime (refuelling is performed without opening the reactor vessel cover).

The design limits adopted for the VKR-MT are presented in Table X-6.

TABLE X-6. DESIGN LIMITS

PARAMETER	VALUE
Outer surface temperature of micro fuel elements in normal operation (3500 EFPD), °C	~300
Outer surface temperature of micro fuel elements in accidents (2 hours), °C	700
Temperature of core structures in normal operation, °C	300
Temperature of core structures in accidents, °C	700
Maximum/average fuel burn-up, % fissile materials	10

For micro fuel elements, the limit temperature in steam-air mixture is 1600°C in the course of 6 hours. For stainless steel core structures, the limit temperature is 1200°C. Exceeding this limit results in a failure of the structures due to their chemical interaction with micro fuel elements.

Table X-7 presents the evaluated economy characteristics of the VKR-MT.

TABLE X-7. PLANT ECONOMY DATA, MILLION US$

ITEM \ REACTOR TYPE	VVER-1000	VKR-MT	COMMENT
1. Reactor compartment	270	160	Without steam generators, hydro-accumulators and pressurizer
2. Steam turbine plant including the generator	540	200	Balance of physical components
3. Service water system	95	30	The same
4. Transformer and electric plants	175	53	The same
5. Heat plant equipment	–	60	The same
6. Other	270	130	Evaluation
7. First load	240	330	
8. Capital costs	1350	633	
9. Annual operation and maintenance costs	150	120	
10. Specific capital costs, $US/kWe	1350	2110	
11. Specific capital costs including first load costs, $US/kWe	1530	2340	
12. Capital component of electricity cost, cent/(kWh)	3.50	5.35	
13. Fuel component of electricity cost, cent/(kWh)	0.87	1.60	
14. Operation and maintenance component of electricity cost, cent/(kWh)	0.22	0.51	
15. Electricity cost, cent/(kWh)	4.66	7.46	

Economy characteristics of the VKR-MT were evaluated under the following assumptions:

- A power unit with serial VVER-1000 reactor was selected as reference;
- Annual load for both reference unit and the VKR-MT is 7000 hours of full power operation;
- The discount rate for capital costs is 16%;
- A four-year cycle with annual refuelling was considered for VVER-1000 and, therefore, ¾ of the first load cost were attributed to capital costs at a 16% discount rate, while ¼ of the first load cost was attributed to annual fuel costs;
- Correspondingly, the VKR-MT first load cost was attributed to capital costs at a 16% discount rate;
- The electricity cost for VKR-MT was evaluated for a cycle with no heat extraction or process steam production.

X-1.5. Fuel cycle options

Possible scheme of a closed fuel cycle is presented in Fig. X-4.

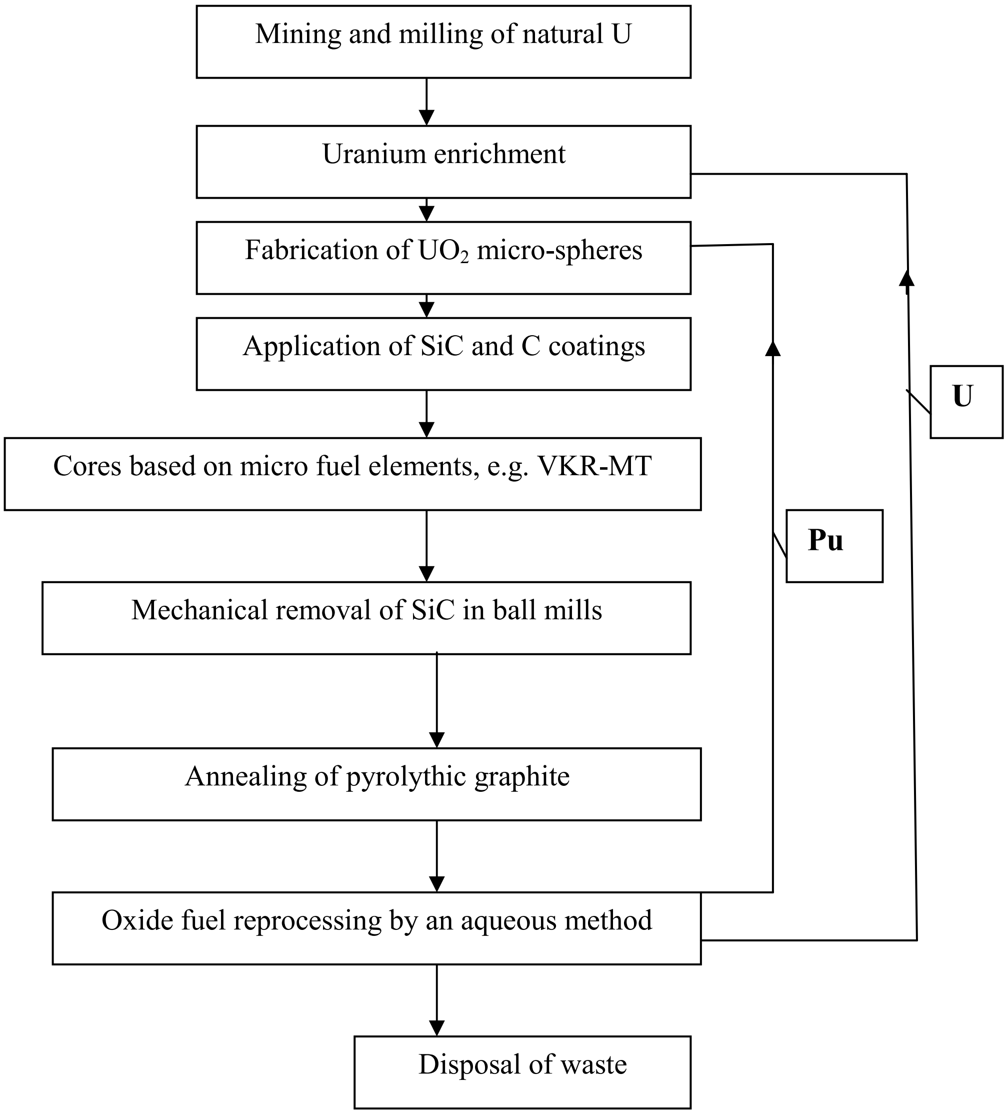


FIG. X-4. Closed fuel cycle scheme for VKR-MT.

Both open and closed fuel cycle options are possible for the VKR-MT. Open fuel cycle scheme may be essentially the same as for the VVER and VK-300 [X-1] reactors. The specific feature of a VKR-MT closed cycle is that the SiC outer coating of micro fuel elements is resistant to nitric and sulphuric acids. Therefore, the mechanical removal of these coatings in ball mills should be added to a process chart [X-3]. The remaining layers of pyrolythic graphite are removed through the heating of micro fuel elements in air at 800°C. After that, a conventional aqueous method could be applied to reprocess the uranium dioxide fuel.

X-1.6. Technical features and technological approaches that are definitive for VKR-MT performance in particular areas

X-1.6.1. Economics and maintainability

The economy of an NPP with the VKR-MT is based on the use of well-established boiling water reactor technology that ensures relatively low capital costs. Lower capital costs are also defined by the absence of steam generators, large pumps, accumulators, pressurizer, reheaters and high-pressure heaters.

The use of micro fuel elements makes it possible to simplify safety systems and to reduce operation and maintenance costs and radiation exposures for the plant personnel, particularly through the refuelling being performed with a closed reactor vessel cover.

The fabrication cost of a core based on pebble bed of micro fuel elements is ~40% lower than that of a traditional core based on rod-type fuel elements [X-4]. The reason behind this is that the fabrication of a pebble bed core eliminates costly welding and air-tightness control operations, as well as mechanical treatment of uranium dioxide kernels.

The use of micro fuel elements with radioactive release probability of $\sim 10^{-7}$ secures a low level of radioactivity in the turbine under normal operation modes. This contributes to the reduction of the turbine circuit cost through a lower cost of biological shielding and also reduces the operation and maintenance costs for the turbine circuit.

X-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

A nuclear installation with the VKR-MT eliminates significant releases of radioactivity in severe accidents, including those caused by a fall down of heavy plane or by malevolent human actions, which makes the reactor a good candidate for deployment in many developing countries. The micro fuel elements themselves provide a reliable structure for long-term storage or disposal of spent nuclear fuel. They are well protected from possible impacts of acids and high temperatures, as well as from mechanical impacts.

X-1.6.3. Safety and reliability

The use of micro fuel elements directly cooled by light water coolant makes it possible to develop a reactor with a high level of nuclear and radiation safety in many severe accidents. This quality is defined by a unique combination of the physical and chemical properties of micro fuel elements and the safety properties of boiling water reactors. The ceramic coatings of micro fuel elements retain their strength and air-tightness at very high temperatures, which provides a perfect confinement of fission products in accidents with a failure of the active systems of reactor cooling, such as pump trip, total NPP blackout with simultaneous failure of the active reactor shutdown system, and also in any depressurization of a reactor vessel, including the reactor vessel bottom rupture.

The direct cooling of micro fuel elements by light water coolant to a certain extent excludes their failure under a positive reactivity insertion. The reason behind this is that the average time of heat transfer from a micro fuel element to the coolant is about 0.03 seconds (for a micro fuel element of 1.8 mm diameter). Therefore, any positive reactivity inserted over more than 0.03 s will be effectively compensated by the evaporation of a water coolant-moderator. Different from that, in a PWR or a BWR with standard rod-type fuel elements the characteristic time of heat transfer from fuel to the coolant is 3–5 s, depending on fuel rod diameter. If positive reactivity is inserted faster, this will result in fuel melting, while the coolant will not evaporate before the cladding fails.

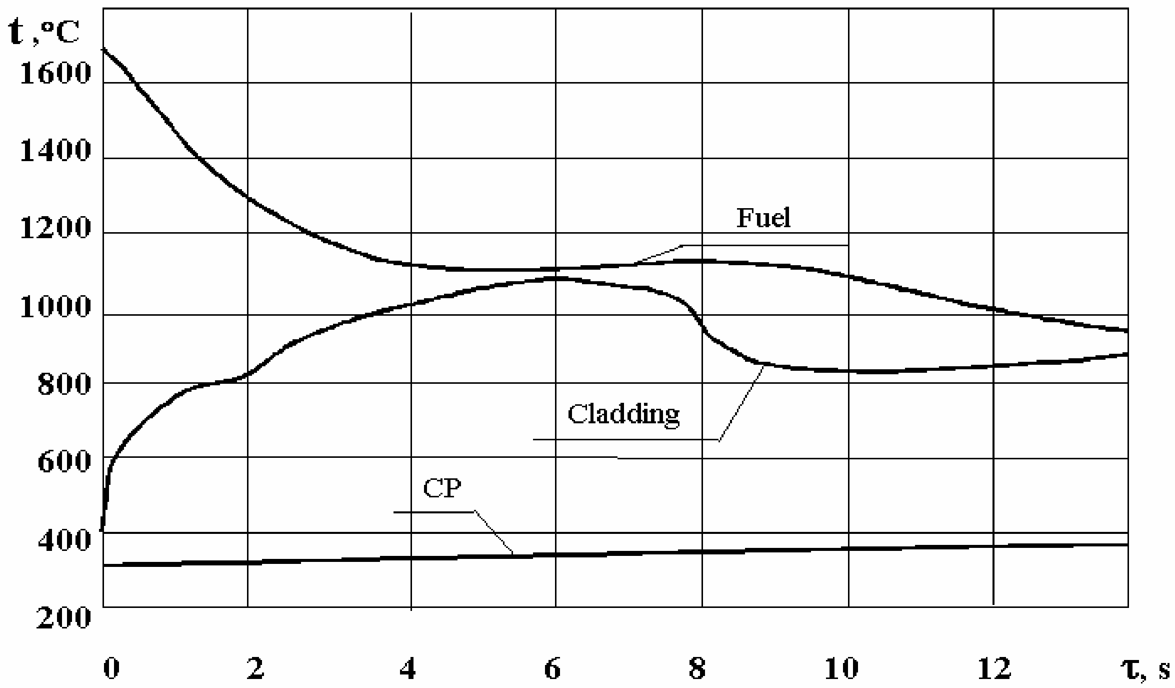
The calculation results illustrating safety performance of an NPP with the VKR-MT are presented for the three accident scenarios, including:

- Rupture of a feedwater pipeline;
- Total NPP blackout with simultaneous failure of the reactor shutdown system;
- The reactor vessel bottom rupture.

The first two scenarios are classified as design basis accidents, while the last one is considered as a beyond design basis accident.

Rupture of a feedwater pipeline

The accident considered is initiated by a rupture of the maximum diameter pipeline, which for the VKR-MT case is a pipeline of 200 mm diameter supplying feedwater to the reactor vessel. The calculation results are presented in Fig. X-5.



CP – temperature of micro fuel elements in the VKR-MT core; two curves at the top show the temperatures of fuel and cladding in fuel elements of a VVER-1000 reactor.

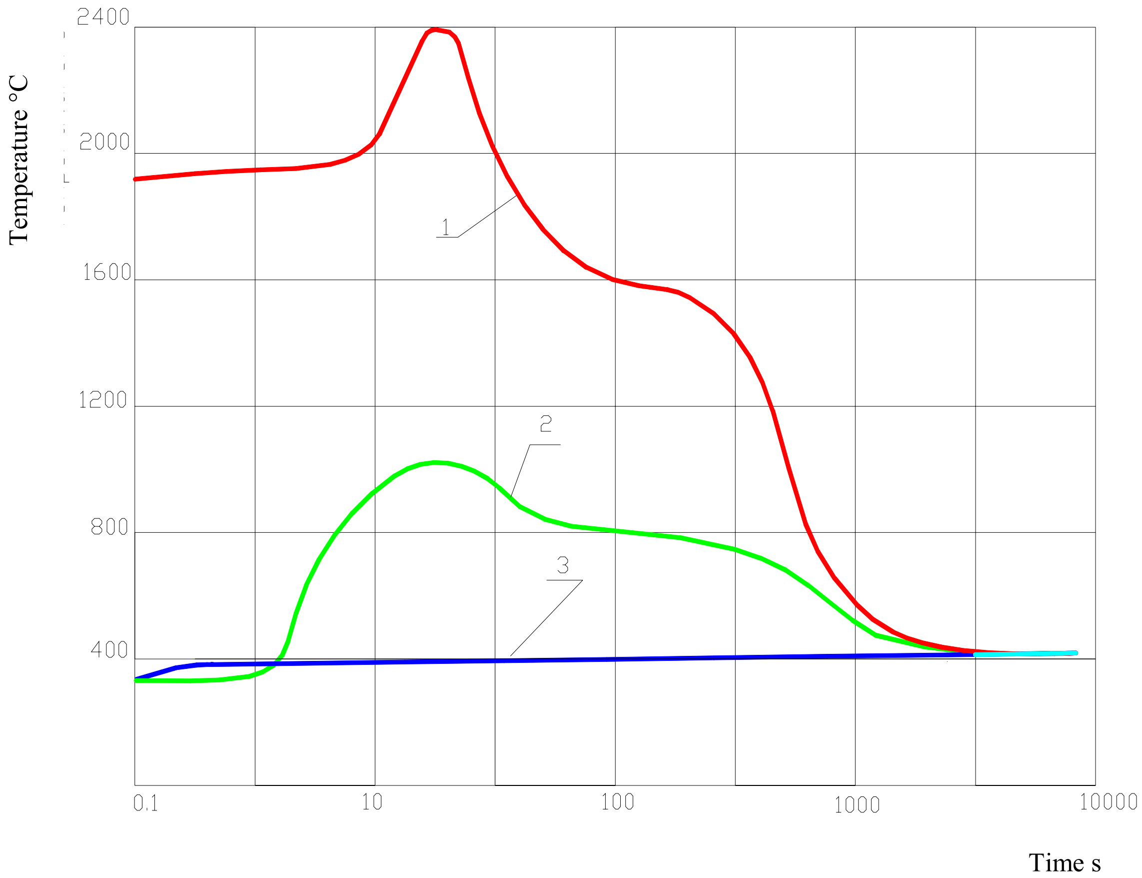
FIG. X-5. Accident with rupture of a maximum diameter pipeline.

For comparison, this figure also shows the results obtained for a scenario with the rupture of a pipeline of 850 mm diameter in a VVER-1000 reactor [X-2].

As it follows from Fig. X-5, the evolution of this accident for the VKR-MT and a VVER-1000 is essentially different. In VVER-1000, the temperature of zirconium claddings increases promptly due to high-temperature heat accumulated in the uranium dioxide pellets and due to heat removal deterioration. The VKR-MT core is practically not heated in the first seconds of the accident process, as the temperatures of micro fuel elements and the coolant in normal operation are different by a few degrees only. Later on, the temperature of micro fuel elements slightly increases due to residual heat up until the start-up of the emergency core cooling system (ECCS) operation. The accident is localized after the core is filled with the ECCS water. As the temperature of micro fuel elements is well below 1500°C, the release of radioactivity to the containment remains at the level of $\sim 10^{-7}$.

Total NPP blackout

The results of calculation for a scenario with total NPP blackout accompanied by simultaneous failure of the reactor shut down system are shown in Fig. X-6. Similar results obtained for an NPP with standard VVER-1000 reactor [X-2] are presented for reference.



Maximum temperature of fuel (1) and cladding (2) in a standard VVER-1000;
(3) temperature of micro fuel elements in the VKR-MT core.

FIG. X-6. Accident with total NPP blackout without scram.

It can be seen that the character of this scenario is also different for the two core types. For a standard VVER-1000 with rod-type fuel elements, the decrease of power takes place very slowly because of the positive Doppler reactivity being inserted when fuel, which is at 1000°C in normal operation, gets cooled. Fission reaction is stopped after ~1000 s, when nearly all primary-circuit water is released through the safety valves. In this, the temperature of zirconium claddings exceeds 1000°C after about 20 s after the accident start.

The scenario is quite different for the VKR-MT core incorporating micro fuel elements. As there is nearly no stored heat, the temperature of micro fuel elements remains at the level of about 300°C. Chain reaction is terminated at the expense of density and temperature reactivity effects on the coolant without the operation of the control rods. The release of radioactivity to the coolant remains at the level of $\sim 10^{-7}$.

Rupture of reactor vessel bottom

Such an accident is never considered for the reactors with cores based on rod-type fuel elements, as core cooling with an acceptable temperature of zirconium claddings cannot be provided in this case. For VKR-MT, the rupture of reactor vessel bottom is considered as a beyond design basis accident.

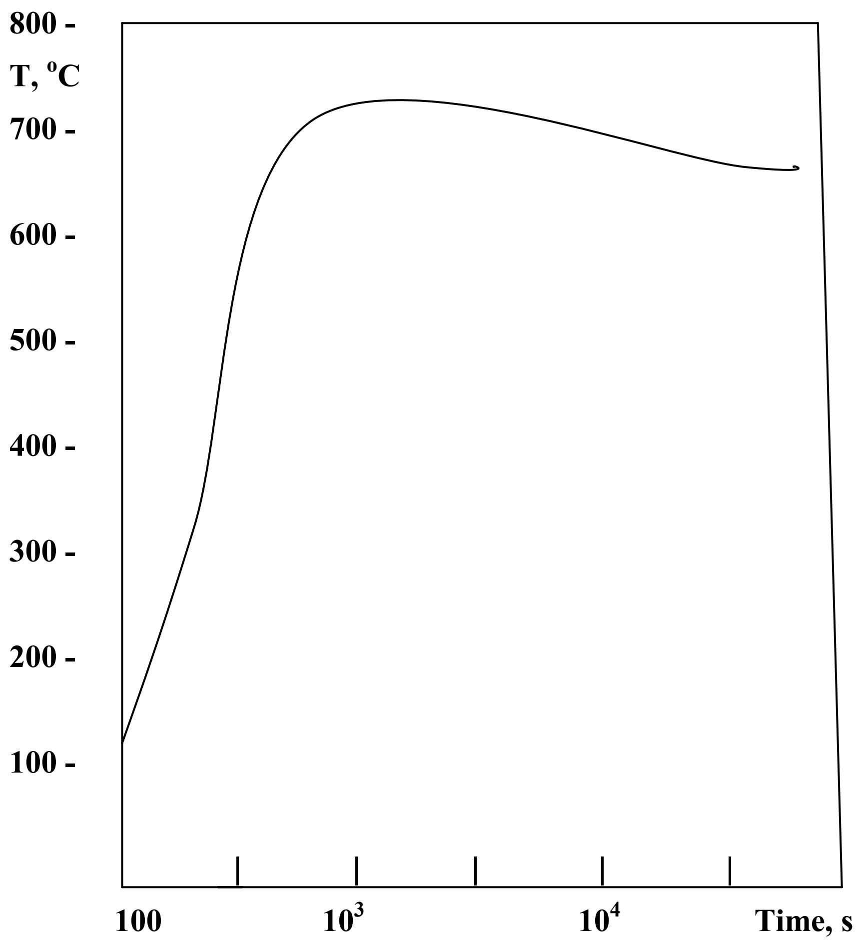


FIG. X-7. Hypothetical accident with rupture of reactor vessel bottom.

The calculation results are presented in Fig. X-7. After the reactor vessel bottom rupture, total NPP blackout and the prompt coolant loss and vessel decompression are observed. The pressure inside the primary containment increases up to 2 bar. After that, the ECCS operation starts, and cold water from the ECCS is released through the ruptured reactor vessel bottom. The core becomes voided in about 100 seconds, and during this period its temperature decreases from 300°C to about 120°C. After that, fast heating of the core starts due to the residual heat. Core cooling is performed by natural convection of steam between the core and the ECCS condenser at a pressure of ~2 bar. The natural convection flow rate increases from zero to 10 kg/s in about 10 minutes. In about 40 minutes, the core temperature reaches its maximum of 720°C, Fig. X-7. Further on, a slow decrease of the core temperature is observed. Within the range of temperatures observed in Fig. X-7, the release of radioactivity from micro fuel elements will be nearly the same as in normal operation, i.e. $\sim 10^{-7}$.

The maximum temperatures of the VKR-MT fuel and claddings in accidents and abnormal operation occurrences (AOO) without scram are evaluated in Table X-8.

TABLE X-8. MAXIMUM TEMPERATURES OF VKR-MT FUEL AND CLADDINGS IN UNPROTECTED AOO AND ACCIDENTS

Initiating event	Maximum temperature, °C	
	Fuel	Cladding
Positive reactivity insertion	310	305
Termination of feedwater supply	300	300
Total NPP blackout	295	295
Total NPP blackout without scram	300	300
Rupture of reactor vessel bottom	720	720

A very low heat energy stored in the VKR-MT core during normal operation defines low temperatures of the micro fuel elements in accidents. The temperatures of fuel and claddings in accidents with residual heat removal differ by fractions of a degree only.

High reliability of an NPP with the VKR-MT reactor is also secured by the absence of such potentially non-reliable components as steam generators, circulating pumps, high pressure hydro-accumulators, reheaters, as well as high-pressure circuits of the turbine circuit.

X-1.6.4. Proliferation resistance

The VKR-MT core is based on 10% enriched uranium fuel. The reactor refuelling is performed once in 10 years without opening the reactor vessel cover. The micro fuel elements with fresh fuel are supplied to the NPP in a sealed metallic reservoir, from which they are loaded to the core by the hydraulically driven ball transport pipelines of 20 mm diameter. Similar hydraulic pipelines and reservoirs are used to discharge and accommodate micro fuel elements with spent fuel. The whole VKR-MT core is refuelled once-at-a-time; during operation, there is no equipment for fuel discharge from the reactor vessel.

The above mentioned design features of the VKR-MT are a prerequisite for an effective implementation of safeguards, in a way similar to how it could be done for factory fabricated and fuelled reactors.

From the standpoint of proliferation resistance, it is also important that micro fuel elements cannot be reprocessed directly, with the use of well-known and established aqueous methods. To reprocess them, it would be necessary to apply mechanical removal of the corrosion resistant and mechanically strong silicon carbide coatings [X-3]. The annealing in air at 800°C is used to remove other coating layers made of pyrolythic carbon. Only then standard aqueous methods can be applied to reprocess the uranium dioxide kernels.

X-1.6.5. Technical features and technological approaches used to facilitate physical protection of VKR-MT

In the VKR-MT reactor, micro fuel elements are used in a pebble bed form and are directly cooled by the lateral flow of a light water coolant-moderator. This feature could make it possible to create a reactor with a high degree of inherent safety features that prevent large releases of radioactivity from fuel in many accidents, including those caused by malevolent human actions. The use of passive air-based residual heat removal system secures that the NPP equipment retains its integrity and operability under failures of the normal cooling systems.

Another feature that could facilitate physical protection of an NPP with the VKR-MT is the long refuelling interval achieved through the use of a relatively large core volume and a relatively low specific power, and through the extensive use of burnable poisons. The reactor is reloaded once in 10 years without opening the reactor vessel cover. In this, no fresh or spent nuclear fuel is stored at the site during reactor operation.

X-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of VKR-MT

The VKR-MT is a concept proposed in Russian Federation from a developing country perspective. This is a small reactor without on-site refuelling; it could be attractive for deployment in many developing countries with insufficient infrastructure and human resources. The design provides for a large component of local participation during construction and makes a provision for fuel leasing. Its deployment could be facilitated by licensing/design certification reciprocity arrangements between different countries.

X-1.8. List of enabling technologies relevant to VKR-MT and status of their development

A nuclear steam supply system with the VKR-MT is based on a synthesis of the technologies of boiling water reactors and high temperature gas cooled reactors. The enabling technologies for the VKR-MT are as follows:

- The use of micro fuel elements (coated particles with outer SiC layer) in a pebble bed directly cooled by light water coolant-moderator;
- The use of lateral-flow fuel assemblies appearing as tanks accommodating pebble beds of micro fuel elements with inner perforated tubes acting as inlet collectors and perforated assembly duct walls acting as outlet collectors;
- A long-life core providing for 10 years of operation without reloading or shuffling of fuel;
- A once-at-a-time core reloading performed with the use of the external reservoirs with fresh and spent micro fuel elements; from these reservoirs, elements are loaded to or discharged from the core by the hydraulically-driven ball transport pipelines, all without opening the reactor vessel cover;
- An original separator design, which combines the cyclone separator and the dryer within a single device;
- A steam turbine with the in-built separators.

In addition to this, the VKR-MT design makes an extensive use of many well established PWR and BWR technologies, such as the technologies of reactor pressure vessel, control rods, control rod drivelines, pipelines, ECCS, reactor compartment, primary and reinforced containment, etc.

The R&D necessary to develop and deploy a heat and power plant with the VKR-MT reactor is as follows:

- Design and technology development for micro fuel elements to be used in the conditions of boiling water cooled core (the operating temperature is relatively low for ceramic materials; the fast neutron fluence is by an order of magnitude higher than in high temperature gas cooled reactors; the coating layers are relatively thin; and the outer diameter of micro fuel elements is relatively large, etc.);
- Study of a chemical interaction between micro fuel elements and stainless steel at high temperatures;
- Study of a mechanical interaction between a pebble bed of micro fuel elements and steel structures at temperature changes;
- Technology development and demonstration for a reactor refuelling with closed reactor vessel cover;
- Development and demonstration of an option to perform the refuelling without depressurization and power reduction;
- Solution of certain thermal-hydraulic and design problems for fuel assemblies of new design for lateral flow of a coolant, e.g. the design with several inlet collectors - tubes with perforated walls, etc.;
- Design and technology development for new steam separation devices that combine the functions of a cyclone separator and a dryer;
- Design and technology development for a new turbine with the in-built separation devices, two external stages of steam separation, and without the reheaters.

X-1.9. Status of R&D and planned schedule

The out-of-pile corrosion tests of micro fuel elements have been completed for the following regimes:

- In water coolant, at 19 MPa and 350°C, in the course of 18 months;
- In superheated steam, at 10 MPa and 550°C, in the course of 15 months;
- In high temperature steam, at 800-900°C, in the course of 14 days (simulation of an accident with the failure of all cooling systems);
- In smoke gases, at temperatures up to 1600°C (simulation of the regimes of severe accidents);
- In contact with stainless steel, at temperatures up to 1200°C.

These tests have shown that micro fuel elements with the silicon carbide outer coating have very high corrosion resistance in water coolant. The loss of the outer coating mass was less than 0.1% in 18 months. When tested in smoke gases, the micro fuel elements preserved their air-tightness at 1600°C in the course of 6 hours. There was no chemical interaction between micro fuel elements and steel at temperatures up to 1200°C.

The in-pile tests of micro fuel elements in a water loop of the IVV-2M research reactor in Zarechny (Russian Federation) are ongoing currently. The micro fuel elements originally produced for the conditions of HTGR type cores, in which the fast neutron fluence is 20 times lower as compared to that in VVER-1000 and VKR-MT, are used as samples in these tests. The samples were a priori annealed in air at 800°C to remove the outer pyrolythic graphite layer. With this layer being removed, the fuel burn-up targeted in the tests appears to be higher than that matching the decreased coating thickness. After the targeted burn-up is reached, it is planned to perform post-irradiation examinations of the irradiated micro fuel elements in the atmosphere of smoke gases at temperatures above 1000°C, which corresponds to the conditions of a beyond design basis accident. The results of these tests will provide a basis for the designing of micro fuel elements for the VKR-MT and VVER-1000 reactors. A projection for the schedule of further R&D is outlined in Table X-9. This projection was made in the assumption of optimum financing conditions.

TABLE X-9. R&D SCHEDULE FOR VKR-MT

NAME	RESPONSIBLE RUSSIAN ORGANIZATION	TARGETED PERIOD
1. Completion of the in-pile tests of micro fuel elements	IVV-2M, RRC KI	2006
2. Fuel design for the conditions of VKR-MT	SPA “Luch”, RRC KI	2005–2007
3. Fuel assembly design for VKR-MT	Experimental Design Bureau OKBM, VNIIAM	2005–2006
4. Testing of a fuel assembly in the VK-50 reactor of the Research Institute of Atomic Reactors (RIAR)	RIAR, Experimental Design Bureau OKBM	2006–2007
5. Construction of a facility for the validation of refuelling performed with closed reactor vessel cover	VNIIAM	2005–2006
6. Detailed design for a prototype VKR-MT plant	Experimental Design Bureau OKBM	2005–2007
7. Construction of a prototype plant in the RIAR, Dimitrovgrad (Russia)		~2010

X-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Conceptual approaches used in the design of the VKR-MT fuel, fuel assembly, and separator are radically different from any existing practice and, therefore, the construction of a demonstration prototype will be required before licensing the VKR-MT into series. It is proposed to construct the prototype on a territory of the Research Institute of Atomic Reactors (RIAR) in Dimitrovgrad, Russian Federation.

X-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The use of micro fuel elements is being considered for VVER-1000 reactors [X-2] and for a direct-flow light water cooled reactor with superheated steam at core outlet, a concept developed by the Pacific Northwest National Laboratory (PNNL, USA). However, both these reactors are not SMRs. The Fixed Bed Nuclear Reactor (FBNR) described in this report incorporates certain design approaches that could be of relevance to the VKR-MT.

X-2. Design description and data for VKR-MT

X-2.1. Description of the nuclear systems

Fuel design

The micro fuel element, shown in Fig. X-8, was developed for the conditions of fuel operation in a light water cooled and moderated core. It appears as a sphere of 1.8 mm outer diameter and includes the uranium dioxide kernel and a three-layer coating made of high-temperature ceramic materials. The kernel has a diameter of 1.4 mm. The first coating layer is made of porous pyrolythic graphite (PyC); it has a density of ~1 g/cm^3. The thickness of this layer is ~100 mkm. The second layer is made of dense PyC (the density is about 1.8 g/cm^3) and is ~5 mkm thick. The third, outer layer is made of silicon carbide (SiC) and has the thickness of about 95 mkm.

The VKR-MT micro fuel elements are designed to confine fission products in normal operation with the probability of radioactivity release ~10^{-7}, and in accidents, at temperatures up to 1600°C in the course of several hours, with the probability of radioactivity release 10^{-5}. The maximum fuel burn-up should not exceed 10% of fissile materials.

The VKR-MT fuel assembly was designed with the use of a certain experience in the design of assemblies with pebble beds of micro fuel elements and lateral coolant flow, available for gas cooled fast reactors [X-4].

The design was selected as follows:

- The fuel assembly structures provide the accommodation of a pebble bed of micro fuel elements;
- The fuel assembly provides:
  - Lateral flow of coolant though the pebble bed of micro fuel elements with an acceptable hydraulic resistance;
  - An acceptable non-uniformity of the steam quality at pebble bed outlet;
  - The accommodation of the guiding tubes for control rods and burnable poison in the number necessary to meet safety requirements and the desired burn-up cycle characteristics;

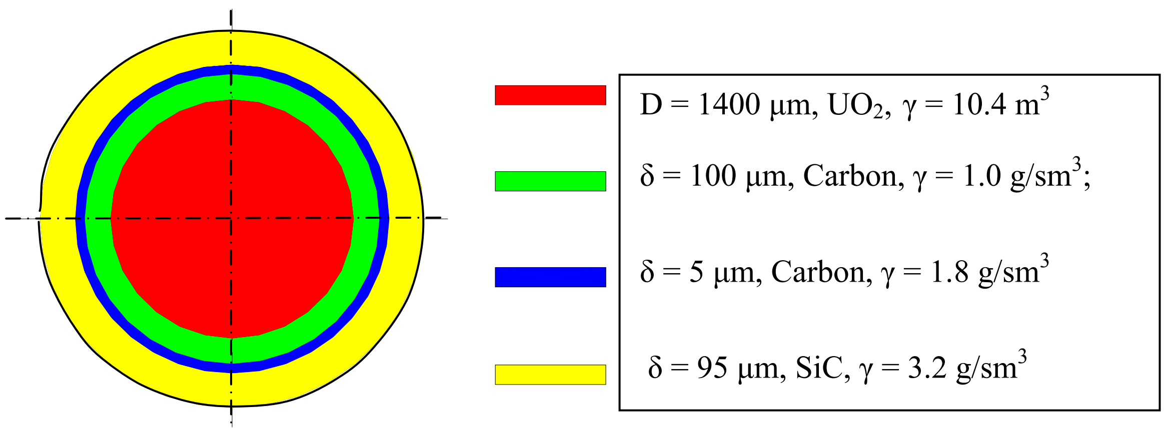


FIG. X-8. Micro fuel element (coated particle).

The immobility of micro fuel elements under the coolant flow;

- Elimination of the interaction between fuel assembly structural materials and outer coatings of the micro fuel elements in the course of 100 000 hours in normal operation, and in the course of several hours in accidents;
- Retaining of the assembly and core geometry in accidents;
- Exclusion of the chemical reactions with an intensive hydrogen release in accidents;
- Ensuring an acceptable deterioration of the properties of materials under neutron flux impact.

A fuel assembly of the design scheme shown in Fig. X-9 incorporates 7 inlet collectors that are cylindrical tubes of 28 × 2 mm diameter. The space between fuel assemblies acts as an outlet collector of steam-water mixture. The inlet collectors are selected cylindrical because they also increase the average density of boiling water coolant in the assemblies. The resulting axial non-uniformity of the flow rate is not so important in this case, because the velocity of water in a collector is relatively low. The calculation of thermal-hydraulic characteristics was performed using the methodology and code [X-5] that are based on the experimental data [X-6].

Each assembly accomodates12 guiding tubes of 16 × 0.8 mm diameter for control rods, located within the pebble bed. They are brought together by a crosspiece and connected to the electromagnetic drive by a bar. The periphery of each fuel assembly includes 18 tubes of 11.2 × 06 mm diameter filled with water, which is needed to increase the effective amount of moderator in the area adjacent to the outlet collector (duct wall). Else, the density of steam-water mixture in this area will be ~3 times lower than in the assembly on average. The inlet collectors and the duct are made of borated stainless steel and act as burnable poisons for the compensation of burn-up reactivity. They also flatten fuel burn-up in the areas adjacent to water cavities (collectors).

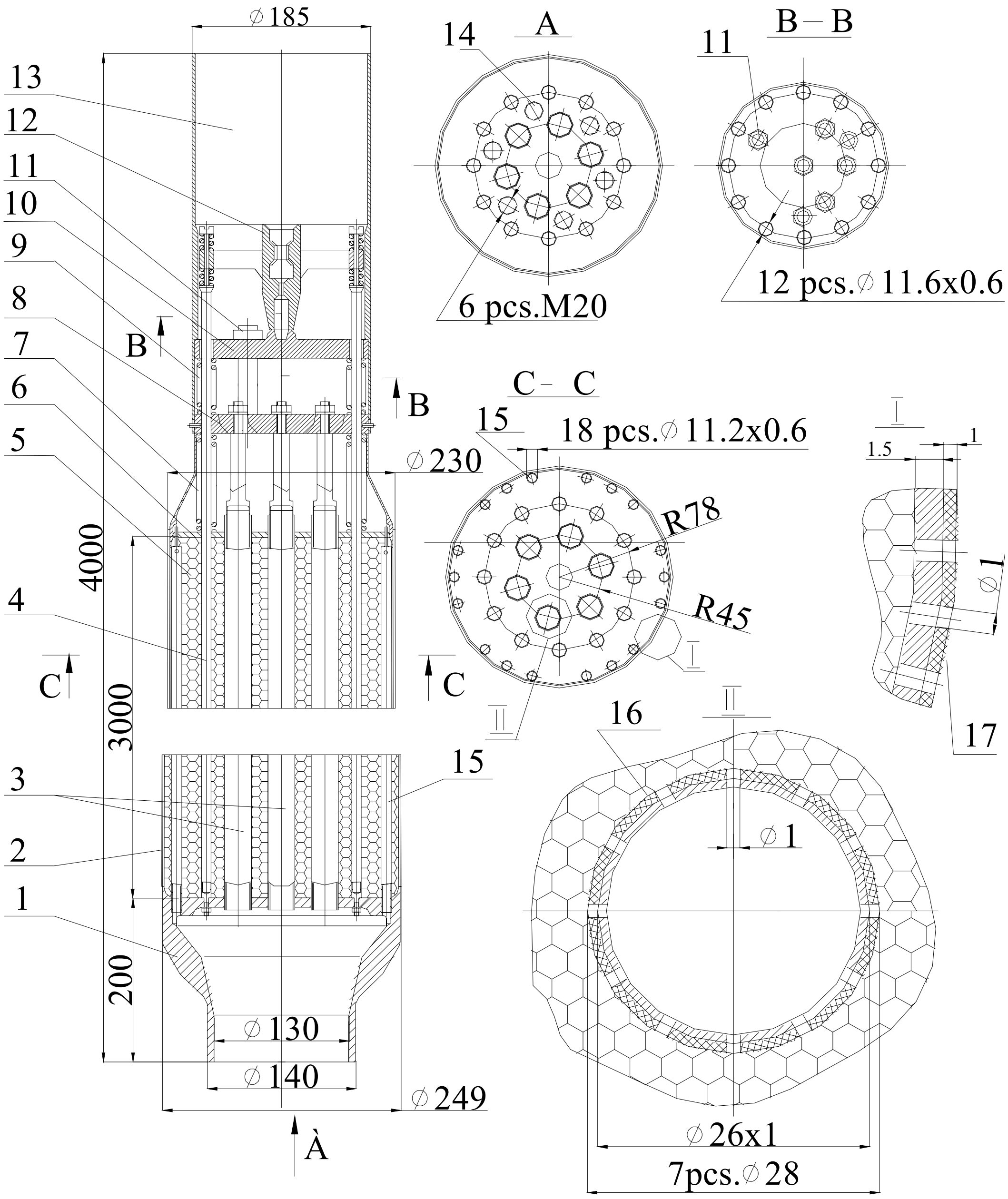
Characteristics of the VKR-MT fuel assembly are shown in Table X-10.

Core design

The design of the VKR-MT core was selected as follows:

- Micro fuel elements with multi-layer ceramic coatings capable to confine fission products effectively at temperatures up to 1600°C;
- The outer coating layer made of silicon carbide (SiC), which is corrosion resistant in water and steam under the targeted operating conditions;
- The use of zirconium alloys is not permitted;
- The core lifetime of 10 years without reloading or shuffling of fuel;
- Fuel burn-up not higher than 10% of fissile materials;
- Reactivity compensation by burnable poisons and active control rod clusters.

The core consists of 151 fuel assemblies accommodating the pebble beds of micro fuel elements that are directly cooled by lateral flow of boiling water coolant. Design data of the VKR-MT core are given in Table X-11.



1 – Tail part	2 – Canister	3 – Inlet collector
4 – Guiding tube of a control rod	5 – Pebble bed of micro fuel elements	6 – Compression bottom
7 – Spring	8 – Supporting bottom	9 – Spring
10 – Foot hub	11 – Foot bottom	12 – Cluster
13 – Head	14 – Plug	15 – Tube
16 – Assembly duct (outlet collector wall)		17 – Canister shell

FIG. X-9. Fuel assembly design scheme.

TABLE X-10. CHARACTERISTICS OF VKR-MT FUEL ASSEMBLY

CHARACTERISTIC	VALUE
1. Number of fuel assemblies in the core	151
2. Distance between fuel assembly centres, mm	250
3. Number of inlet collectors	7
4. Number of outlet collectors	1
5. Maximum diameter of inlet collectors, mm	28 × 2
6. Maximum diameter of the assembly duct, mm	249 × 1.5
7. Minimum diameter of the assembly duct, mm	235 × 1.5
8. Perforation density in inlet collectors, % 9. Perforation density in an outlet collector (duct wall), %	1–5 3
10.Number of guiding tubes for control rods, 16 × 0.6 mm	12
11. Number of tubes with water, 11.2 × 0.6 mm	18
12. Porosity of a pebble bed of micro fuel elements	0.37
13. Number of micro fuel elements	23×10^6
14. UO_2 load, kg	387
15. Pebble bed height, m	3.7
Structural material of the guiding tubes, the assembly duct, and the inlet collectors	Cr18Ni10Ti borated steel; boron: (1.5–3)% by weight

TABLE X-11. CORE DESIGN DATA

CHARACTERISTIC	VALUE
1. Height/effective diameter, m/m	3/3.7
2. Core volume, m^3	24.5
3. Number of fuel assemblies	151
4. Number of micro fuel elements	3.47×10^{9}
5. Compensation of reactivity 5.1. Number of drives 5.2. Number of control rods per drive 5.3. Absorber material 5.4. Burnable poison 5.5. Liquid boron shutdown system	104 12 B_4C Borated steel $Na_2B_4O_7$
6. Refuelling system 6.1. Number of branch pipes for fuel loading (in the reactor vessel cover) 6.2. Number of branch pipes for fuel discharge (in the reactor vessel bottom)	 5 1

Reactivity control system

In line with the regulatory requirements, there are two independent and diverse reactivity control systems. The first is a mechanical system with control rods. The second is a liquid boron system. Table X-12 presents the calculated worth of the reactivity control systems.

The mechanical system of reactivity control is based on conventional cylindrical control rods and electromagnetic drives. It includes 104 electromagnetic drives, of which 37 drives are used to compensate reactivity changes with fuel burn-up and to flatten power distribution in the core. Of them, 12 drives are used for the automatic control of power. The remaining 67 drives combine the functions of operation control and reactor shutdown. The mechanical system is capable to bring the VKR-MT to a cold shut down state at the beginning of life (BOL) only. Later on, the operation of liquid boron shutdown system should be added to achieve this state, while the mechanical system will be capable to bring the reactor to a shutdown state only at 250°C.

TABLE X-12. REACTIVITY CONTROL WORTH (K_{eff} AND $NA_2B_4O_7$ CONCENTRATION AT 20°C)

	BOL	MIDDLE OF LIFETIME	EOL
Control rods	0.88	0.99	0.96
Control rods + Liquid boron system $Na_2B_4O_7$, g/kg	0.88 0.0	0.94 2.7	0.94 0.8
Liquid boron system $Na_2B_4O_7$, g/kg	0.94 8.5	0.94 12.8	0.94 8.4

The design of control rods and the allocation of drives are similar to those in a VVER type reactor. The drives are mounted upon the reactor vessel cover. The rods appear as stainless steel tubes of 11.6 × 0.8 mm diameter filled by boron carbide pellets. The length of the absorbing part is 3500 mm. The positions of the control rods in a fuel assembly are shown in Fig. X-9.

Coolant and structural materials of the core

Boiling water acts as coolant and moderator, while separated steam is used to drive the turbine. Maintenance of the water chemistry is performed by the systems that are conventional for reactors of BWR type. The coolant circuit equipment is made of stainless steel, or has a welding deposition made of stainless steel. The fuel assemblies are also made of stainless steel. The outer coatings of micro fuel elements are made of silicon carbide (SiC).

Main systems of the coolant circuit

The schemes of the nuclear steam supply system and the coolant circulation inside the VKR-MT vessel are given in Fig. X-1 and X-2, respectively.

The reactor cooling system includes:

- The internal jet pumps;
- The internal separators;
- Systems of feedwater supply;

- Steam removal systems;
- Pulse pipelines of the control systems;
- Other auxiliary systems, conventional in their design and functions.

The water at 280°C enters the core, where it is heated and partially evaporates. The steam-water mixture from the core at 290°C and a mass steam fraction of 12.5% enters the inter-pipe space of the block of protective tubes, where an agitation takes place. From there, steam-water mixture enters the first stage of cyclone separators, which is to perform the preliminary separation of steam. The cyclone separators of the second stage have an original design providing for the separation down to a moisture fraction of 0.2%. Therefore, the chevron separators that are common to BWRs are not used in the VKR-MT design. The separated water goes to an inlet of the jet pumps, in which it is mixed with feedwater, compressed, and supplied to the core at 280°C.

Two pipelines are used to supply the feedwater to the reactor, and two other pipelines — to withdraw the saturated steam. The appropriate accessories are mounted on the water and steam lines in the area of their penetration to the primary containment, which is common to BWR reactors.

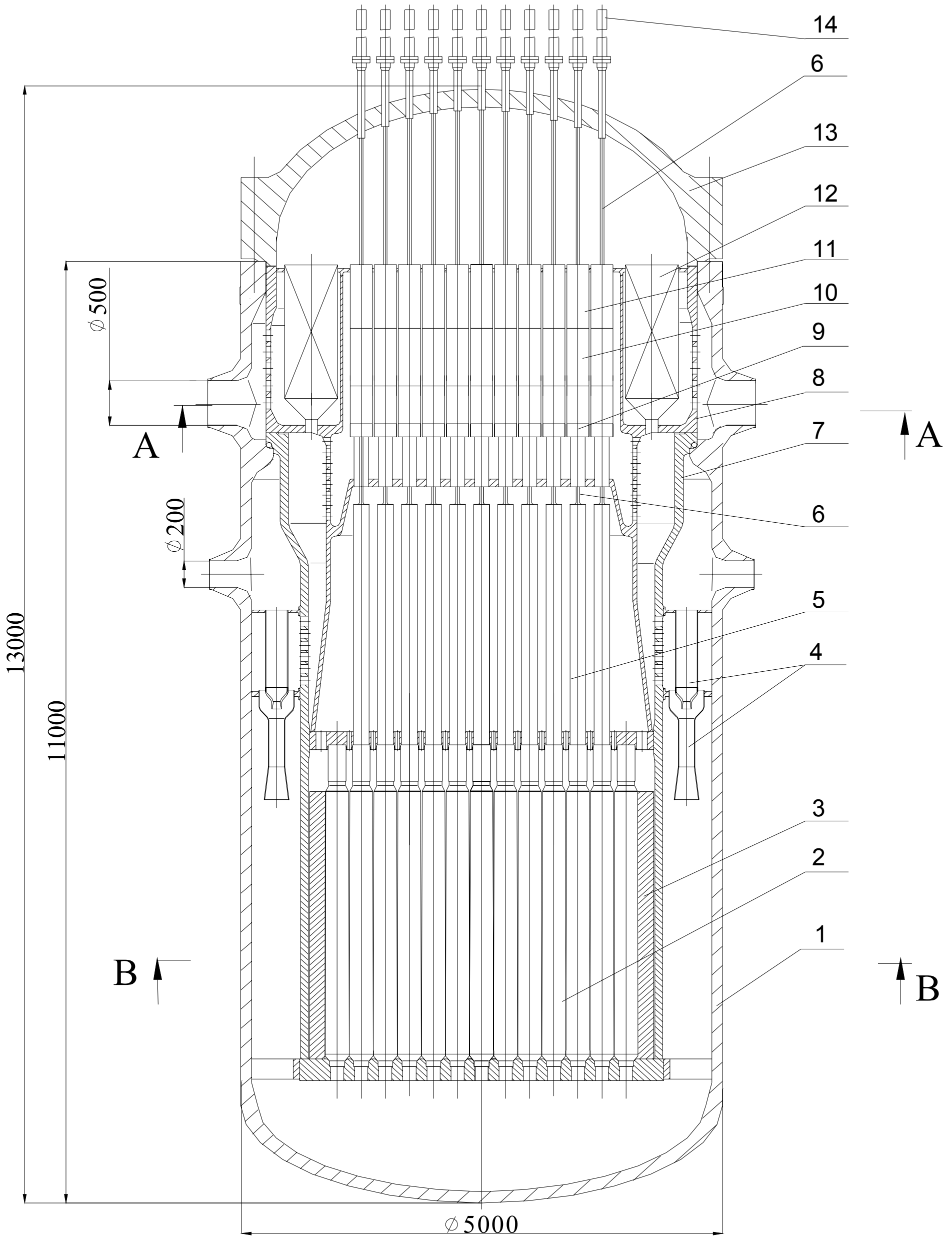
The coolant circuit data is summarized in Table X-13.

TABLE X-13. COOLANT CIRCUIT DATA

PARAMETER	VALUE
Rated thermal power, MW	890
Pressure in steam space (at water level mark), MPa	7.5
Feed water pressure at reactor vessel inlet, MPa	8.1
Feed water temperature, °C	212
Steam production rate (or feed water flow rate), kg/s	230

The reactor and the cooling system are located inside the primary containment, which is a part of the localizing system, traditional for BWRs.

The VKR-MT nuclear steam supply system uses a direct cycle vessel-type boiling water reactor, which includes (see Fig. X-10 and X-11): (a) The reactor vessel; (b) The reactor vessel cover; (c) An internal metallic shaft; (d) The core; (e) Two stages of the centrifugal separators; (f) The block of protective tubes; (g) Radiation and thermal shielding of the reactor vessel; (h) The supporting structure; (i) Jet pumps; (j) Control rod drives; (k) The reloading system for micro fuel elements.



1 – Vessel	2 – Fuel assembly
3 – Enclosure	4 – Jet pump
5 – Guiding tube of a cluster	6 – Protective tube of the bar of a control rod drive
7 – Internal metallic shaft	8 – Block of protective tubes
9 – Anti-holdup device	10 – First-stage separator
11 – Second-stage separator	12 – Re-dehydrator
13 – Vessel cover	14 – Control rod drive

FIG. X-10. Design scheme of VKR-MT.

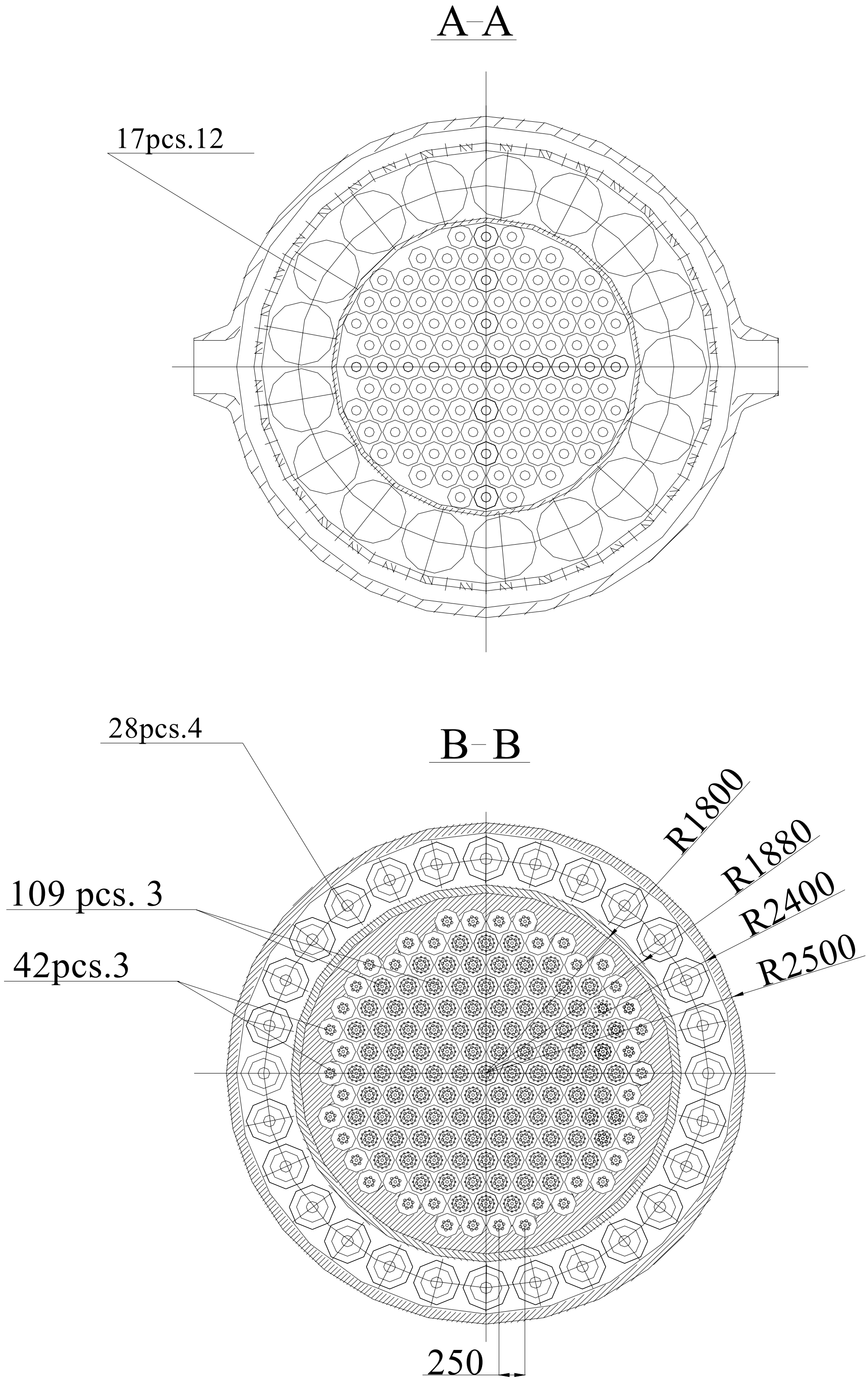
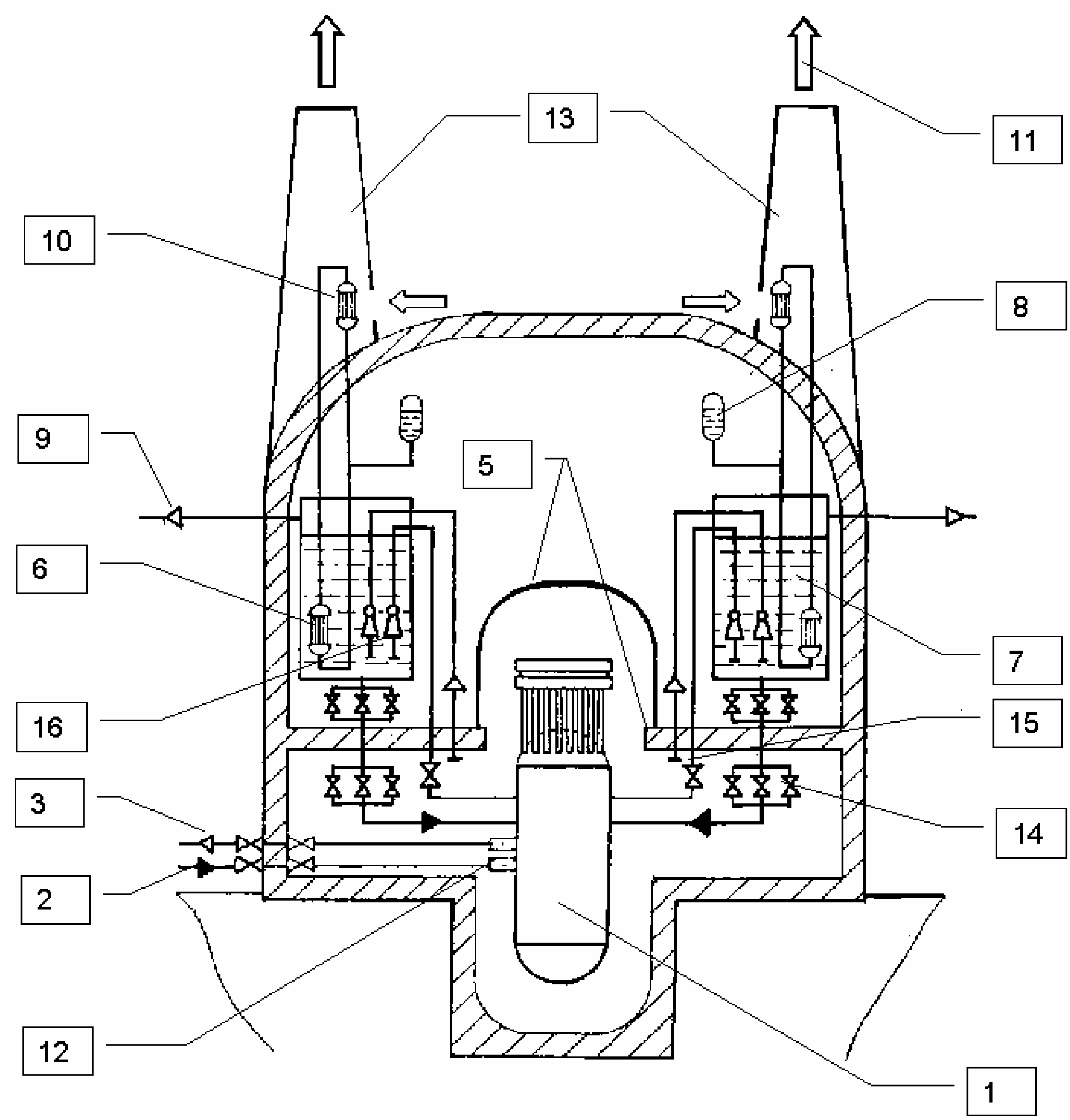


FIG. X-11. Cross-section of the VKR-MT core.

Reactor compartment

The VKR-MT reactor compartment is similar to that of many BWRs. The reactor and all major items of the primary circuit equipment are located within the primary containment, Fig. X-12. The primary containment and the auxiliary equipment are located inside the reinforced concrete containment.



1 – Reactor
2 – Feedwater supply
3 – Steam to turbine
4 – Condenser of the residual heat removal system
5 – Primary containment
6 – Heat exchanger – cooler
7 – Emergency water supply tank
8 – Pressurizer of the emergency heat removal system
9 – To a gasholder
10 – Water-air heat exchanger
11 – Air
12 – Leak limiter
13 – Draft tubes
14 – Check valves
15 – Valves of depressurization system
16 – Steam-jet injectors - condensers

FIG. X-12. Reactor compartment.

Refuelling system

A specific feature of the VKR-MT is the refuelling that is performed without opening the reactor vessel cover. Such refuelling is due to a pebble bed arrangement of micro fuel elements in the reactor core. A principal scheme of the refuelling system is shown in Fig. X-13.

This system consists of the three sub-systems:

- System of fresh fuel loading;
- System of spent fuel discharge from fuel assemblies;
- System of spent fuel discharge from the reactor.

System of fresh fuel loading

The system of fresh fuel loading includes 5 fresh fuel reservoirs, 5 spreaders, 5 pipe collectors, 163 pipelines for ball transport located within the block of protective tubes, and the inlet ball transport pipelines in 151 fuel assemblies.

The fresh fuel reservoir consists of a high-pressure vessel equipped with the loading and discharge ball transport pipelines and the pipelines for the supply and removal of the coolant. All pipelines are equipped with stop valves. The inlet ball transport pipeline is connected to a distributor and is equipped with the electromagnetic stop valve.

The distributors are to distribute micro fuel elements between fuel assemblies. They are located in 5 branch pipes of 180 mm diameter mounted upon the reactor vessel cover. The branch pipes are similar in design to those used for control rod drivelines. Each distributor includes a loading pipeline for ball transport, a silo of conical shape with the guiding pipeline for ball transport, the silo support, a stepping rotary electromagnetic drive for the rotation of the silo, the lock of a guiding pipeline for ball transport, and the detectors of guiding pipeline position. The guiding pipeline is equipped with a stop valve and connected to the fresh fuel reservoir.

The pipe collector appears as a block of 30–36 pipelines of 20 mm diameter for ball transport. The bottom ends of these pipelines are aligned with the pipelines laid within the block of the protective tubes, while the upper ends are fixed to the 20 mm branch pipes in the reactor vessel cover.

System of spent fuel discharge from fuel assemblies

The system of spent fuel discharge from fuel assemblies includes the inlet pipeline for ball transport and a stop valve in the tail part of a fuel assembly, the electromagnetic drive of the stop valve, and an internal repository of spent micro fuel elements.

The internal repository of spent micro fuel elements includes a silo mounted upon the reactor vessel bottom. The silo is equipped with an outlet ball pipeline aligned with the branch pipe (or with several branch pipes) located in the reactor vessel bottom. This branch pipe is used for the discharge of micro fuel elements and is connected to an electromagnetic drive. The silos are equipped with the guiding tubes, in which the bars of the stop valves of the ball transport pipelines move. The number of guiding tubes is equal to the number of fuel assemblies, i.e., 151. All structural elements of the silo are made of borated steel to ensure that the repository is subcritical.

The cooling of spent micro fuel elements is performed by natural convection of a pressure chamber coolant. The silos are perforated in their lower part to make this convection possible. The thermal power of the repository is about 100 kW. The capacity of the repository makes it possible to operate without discharge for several months. After being exposed in repository, the spent micro fuel elements have very low residual power.

System of spent fuel discharge from reactor

This system includes a pipeline of 20 mm diameter for ball transport located in the reactor vessel bottom and connected to the silo of an internal repository through a hydraulic lock, the outer high-pressure reservoir, and the atmospheric pressure containers for spent fuel storage.

The outer reservoir includes a high-pressure vessel, the inlet pipeline for ball transport with a stop valve, the outlet pipeline for ball transport with a stop valve, and the stop valves with electromagnetic drives. The reservoir is also equipped with the pipelines for the supply and removal of the coolant. These are also equipped with stop valves.

The atmospheric pressure container includes a tank with the devices for its transportation and the inlet pipeline for ball transport. The cooling of micro fuel elements in this container could be performed by natural convection of air, as the power of those micro fuel elements that have been exposed in the internal repository is very low.

Refuelling system operation

The refuelling is assumed to be performed in a shut down reactor at atmospheric pressure. However, the refuelling system makes it possible to discharge a specified amount of the micro fuel elements even when the reactor operates at low power level.

The refuelling system operates in an hourglass mode as follows:

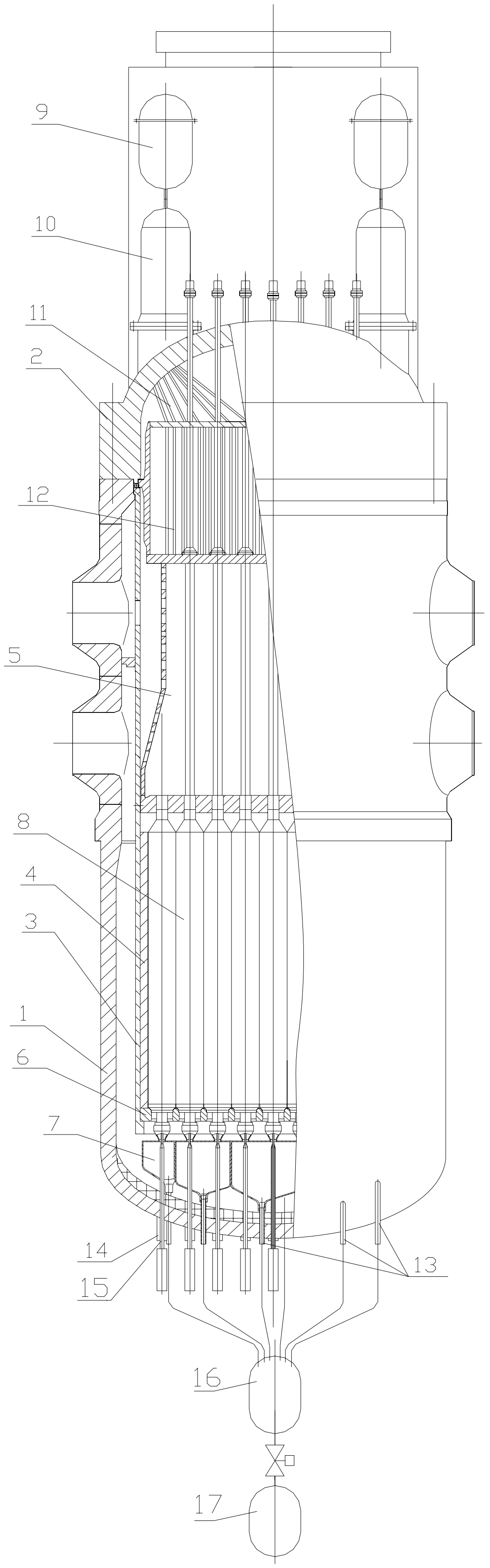
First, reservoir 9 is filled with the necessary amount of fresh micro fuel elements, Fig. X-13. With the use of a rotating electromagnetic drive, the distributor 10 turns the rotating pipeline for ball transport to a corresponding pipeline of the pipe collector 11. Then, the stop valve of reservoir 9 opens, and a batch of fresh micro fuel elements, driven by gravity, enters the silo of distributor 10 and through the rotating pipeline enters the corresponding pipeline of upper collector 11. After that, through pipeline 12 located in the block of protective tubes, a batch of the fresh micro fuel elements are supplied to the corresponding fuel assembly 8.

The discharge of spent micro fuel elements from fuel assembly 8 to the internal repository 7 is performed as follows. The electromagnetic drive opens the stop valve in the bottom part of fuel assembly 8. The spent micro fuel elements from fuel assembly 8 are poured out, driven by gravity, to the internal repository 7.

With the accumulation of spent micro fuel elements in the internal repository 7, they are discharged to external reservoir 16. For this, the electromagnetic drive opens the stop valve of the internal repository 7, and the spent micro fuel elements enter the external reservoir 16 through a branch pipe13 in the reactor vessel bottom 1.

With the accumulation of spent micro fuel elements in the external reservoir 16, they are being discharged as follows. The external reservoir 16 is cut off from the coolant circuit. Then, the stop valve of the external reservoir 16 is opened, and the spent micro fuel elements are sucked off from reservoir 16 by the hydraulic transport and enter the atmospheric pressure container 17.

As the atmospheric pressure container 17 gets filled, it is just replaced by another one. The spent micro fuel elements are either sent for reprocessing or transported to storage outside the NPP site.



1 – Vessel;
2 – Cover;
3 – Internal metallic shaft;
4 – Enclosure;
5 – Block of protective tubes;
6 – Support plate;
7 – Internal repository for spent micro fuel elements;
8 – Fuel assembly;
9 – Reservoir for fresh micro fuel elements;
10 – Distributor;
11 – Pipe collectors;
12 – Pipelines for ball transport within the block of protective tubes;
13 – Branch pipes for the discharge of micro fuel elements;
14 – Branch pipes in the reactor vessel bottom;
15 – Guiding branch pipes;
16 – High pressure reservoir for spent micro fuel elements;
17 – Atmospheric pressure container for spent micro fuel elements.

FIG. X-13. Principal scheme of the VKR-MT refuelling system.

The reservoirs marked '9' contain fresh micro fuel elements only during the period of reactor refuelling, which is to be performed under strict security measures. When the reactor operates, the reservoirs 9 are empty and it is impossible to discharge irradiated micro fuel elements from the reactor vessel, as the reactor has no equipment for hydraulic transport of micro fuel elements. Such equipment is delivered to a site along with the reservoirs for spent fuel just prior to a refuelling and moved away immediately after the refuelling is completed.

When necessary, the refuelling could be accomplished in a conventional mode, i.e., with the opening of the reactor vessel cover. Some inspection, repair and maintenance operations could be performed during this period. The VKR-MT design provides for it, since all circuits for the transport of micro fuel elements are made with the use of detachable joints and have no micro fuel elements permanently present in them. Stop valves in the tail parts of fuel assemblies 8 prevent the spilling of micro fuel elements in the operations of fuel assembly extraction.

Figure X-14 presents the scheme of main heat transport system for the VKR-MT with specification of heat removal path in normal operation and in accidents. It could be seen that in normal operation the heat is removed to the turbine plant and, after that, about 1/3 of the heat is converted to electricity and removed to the grid through a generator, while 2/3 of the heat is removed to process water through the condenser. When the plant is operated in a heat extraction mode, the electric power generation is reduced, while an essential part of the overall thermal power is removed from the turbine to a district heating circuit through the intermediate circuit heat exchangers. When heat production is set to a maximum, this part may reach 2/3 of the total plant power. Omit losses in heat transport, the air in living and industrial premises acts as an ultimate heat sink for the energy that is removed for district heating purposes. A very small fraction of power (about 1 MW thermal) is permanently removed to the air through the Residual Heat Removal System (RHRS), which contributes to keeping this system workable during the whole operation period of a plant.

In accidents, heat removal is accomplished through natural convention of all media in all systems. The residual heat is removed to the surrounding air by the RHRS.

X-2.2. Description of the turbine generator plant and systems

The turbine unit implements a cycle based on the use of two external stages for intermediate moisture separation without reheating. An enhanced system of moisture removal is incorporated into the turbine setting. It includes a special separator stage combined with some design features to enhance the peripheral and in-channel separation. The turbine includes the combined high and medium pressure cylinders of original design and a low pressure cylinder, in which a turbine rotor blade of the last stage has a diameter of 1200 mm. This blade is a modification of the turbine rotor blade of the serial Russian turbines for NPPs of 1000 MW(e).

The turbine is designed for a maximum heat-extraction load of 600 MW, at which the electric load would be reduced down to 180 MW(e).

Lateral section of the cogeneration turbine is shown in Fig. X-15.

The turbine setting, see Fig. X-15, incorporates the state-of-the-art in moisture separation systems for turbines [X-7 and X-8]. The low-pressure cylinder incorporates a dedicated next to last separation stage, which has moisture collection grooves on the convex surface of the peripheral part of a turbine rotor blade. It also incorporates a roof-type shroud and an ejector-type moisture trap. The next to last stages of the high- and medium-pressure turbine parts provide for the dehydration of steam, which is accomplished through the use of a jet and the turbine rotor blades of special design.

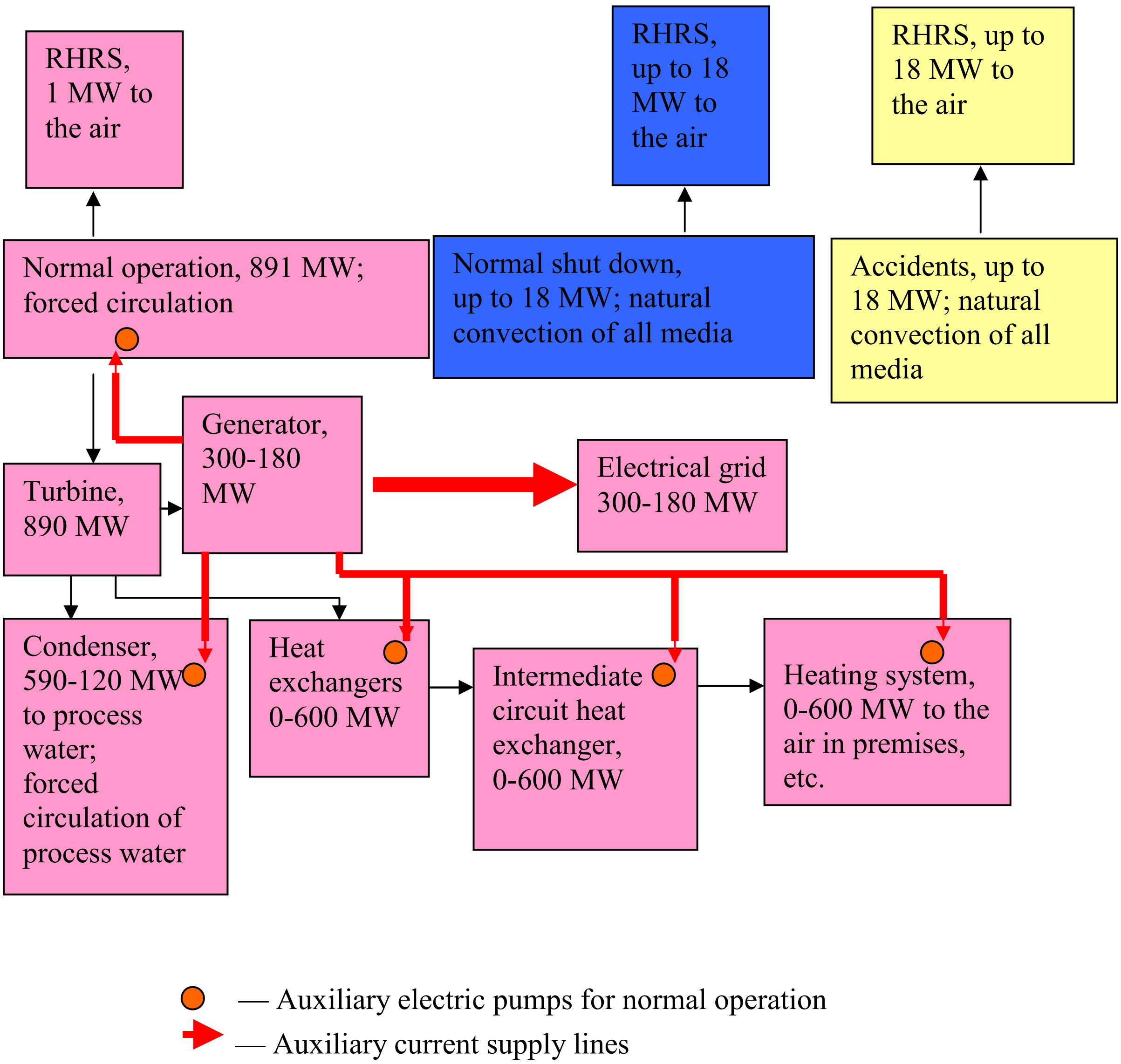


FIG. X-14. Heat removal paths in normal operation and in accidents.

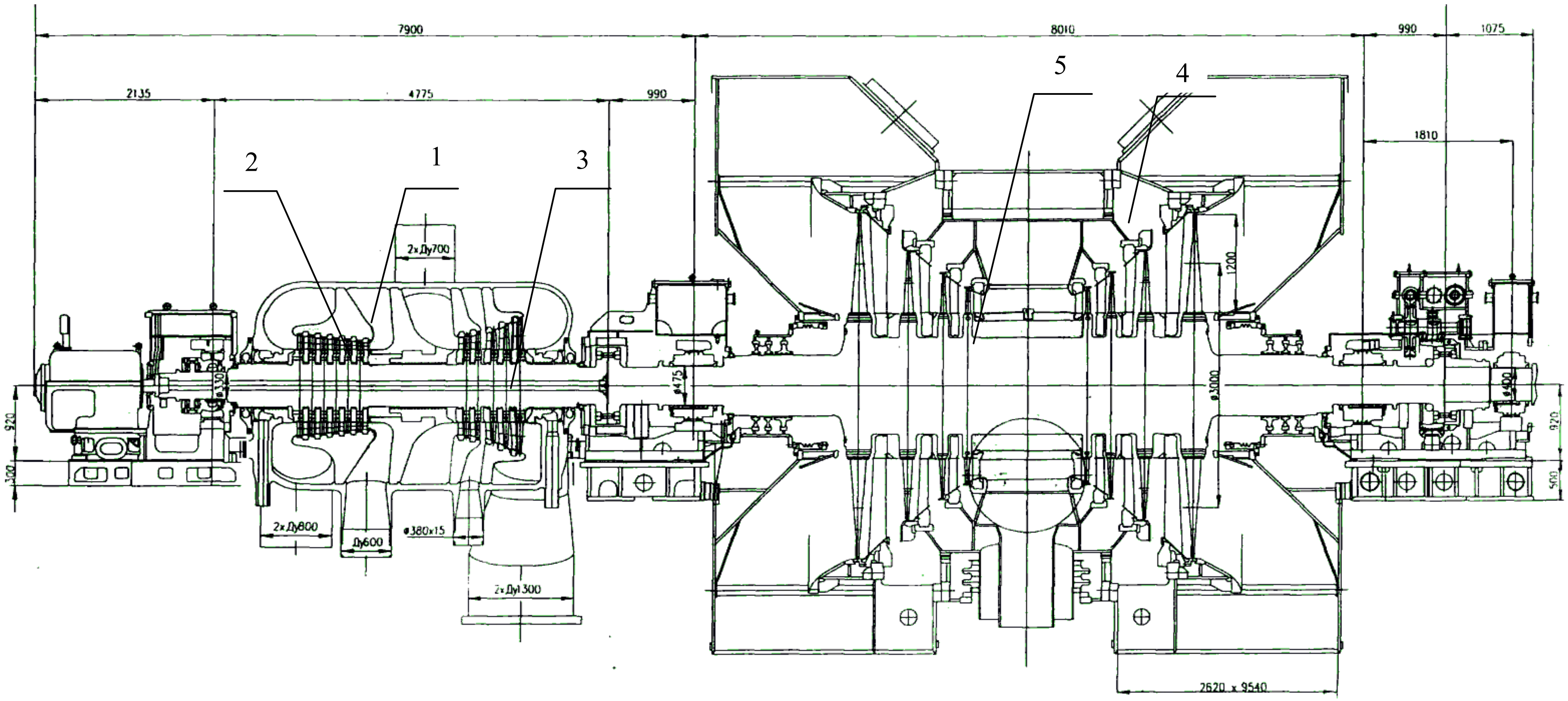
The diaphragms of the last two stages of high-pressure cylinder include an enhanced system for moisture removal. The pellicular moisture is removed to inner cavities of the turbine through the slots in the convex and concave blade surfaces. After this, the moisture is removed through the diaphragm rims to a discharge branch pipe.

X-2.3. Systems for non-electric applications

The system for district heating includes two stages of the primary circuit heat exchanger, an intermediate circuit, and heat exchangers of the network. The pressure of intermediate circuit coolant is higher than the primary circuit pressure and, therefore, the ingress of radioactive coolant to the heating network is excluded.

X-2.4. Plant layout

At the moment, the layout of the VKR-MT plant is assumed to be similar to that of an NPP with the VK-300 reactor [X-1].



1 – High and medium pressure cylinder;
2 – High pressure part;
3 – Medium pressure part;
4 – Low pressure cylinder;
5 – Separation stage.

FIG. X-15. Lateral section of the cogeneration turbine.

REFERENCES

[X-1] KUZNETSOV, Y.N., et al., NPP with VK-300 boiling water reactor for power and district heating grids: IAEA-SR-218/32, Small and Medium Sized Reactors: Status and Prospects (Proc. Int. Seminar, Cairo, Egypt, 27–31 May 2001) C&S Paper Series 14/P, Vienna (2002).

[X-2] PONOMAREV-STEPNOI, N.N., et al., Prospects of VVER reactors with micro fuel elements. Atomnaya Energia, Vol. 86 (6) (June 1999, in Russian).

[X-3] BYLKIN, B.K., INOZEMTSEV, YU.N., FEDOSEENKOV, A.N., KHRULEV, A.A., Methods and equipment for the reprocessing of HTGR fuel elements. Atomnaya Tekhnika za Rubezhom, No. 2, 3–8 (1991, in Russian).

[X-4] GRATTON C.P., Design and assessment studies performed by the GBR Association on gas cooled fast reactor. Journal of the BNES, p. 331–336, No.10 (1972).

[X-5] GRISHANIN, E.I., DENISOV, E.E., LIUBIN, A.YA., FALKOVSKY L.N.,. Mathematical model for calculation of coolant parameters in fuel assembly of light water cooled reactor with micro fuel elements. Tyazheloye Mashinostroyenie, No. 9, 11–20 (1995, in Russian).

[X-6] VINOGRADOV, O.S., SMIRNOV, I.P., TIGRAEV, I.P. Hydrodynamics of fuel assemblies with pebble beds. Annals of TsKTI, issue 145, Studies and Validation of Equipment for NPPs, 107–119, Leningrad (1977, in Russian).

[X-7] KIRYUKHIN, V.I., FILIPPOV, G.A., NAZAROV, O.I., Study and optimization of systems for moisture separation in turbines of NPPs, Teploenergetika, No. 6, 24–29 (1997, in Russian).

[X-8] KIRYUKHIN, V.I., et al. Operation experience of the turbine drives for feedwater pumps with a special moisture separation stage for NPPs. Teploenergetika, No. 5, 19–24 (1997, in Russian).

ANNEX XI

LONG-LIFE CORE SMALL NUCLEAR REACTOR WITH CLOSED-VESSEL REFUELLING — ATOMS FOR PEACE REACTOR (AFPR-100)

Pacific North-West National Laboratory (PNNL),
United States of America

XI-1. Basic summary

The Pacific North-West National Laboratory (PNNL) of the USA develops the concept of a small light water reactor with coated particle based pebble bed fuel directly cooled by lateral flow of light water coolant; this reactor, which has a long core lifetime and is refuelled without opening the reactor vessel cover, is tentatively named the Atoms for Peace Reactor (AFPR).

Reactor design

The AFPR core, see Fig. XI-1, appears as a pebble bed of the so-called micro fuel elements (MFEs), which are uranium dioxide based small particles (with the enrichment of about 10%) coated with SiC-PyC layers, like TRISO fuel for high temperature gas cooled reactors. These particles are in direct contact with the water coolant flowing laterally and leaving the pebble bed as steam through the perforated walls of the fuel assemblies, as shown in Fig. XI-1. To protect “bare” fuel design from corrosion-erosion in the high temperature and high pressure water and steam, the outer coating layer is assumed to be manufactured of very strong and resistant protective coating materials, such as the nano-layered nitride materials like TiN/NbN or AlN/CrN.

The AFPR is designed to store both the fresh fuel and the spent fuel generated during ~40 years of operation inside the reactor vessel. The design incorporates storage tanks for fresh and spent MFEs, see Fig. XI-1, and a valve system providing for a kind of on-line refuelling of the reactor core achieved via downward movement of the gravity-driven MFEs controlled by opening of the discharge valves. In this, neither fresh nor spent fuel can be accessed from the outside of the reactor vessel without using special equipment (a hydraulic transport system) that is not present in the reactor or on the site during the whole period of reactor operation — this equipment is brought to a site only once in ~40 years to accomplish the ‘external’ refuelling operations under strict safeguards and verification measures.

The AFPR incorporates a top-mounted control rod system; the control rods move inside guiding tubes laid vertically in the pebble bed of MFEs, Fig. XI-1.

Enhanced proliferation resistance

Consistent with proliferation-resistance objectives, there are no fresh or spent fuel storage facilities outside the AFPR vessel at the plant site.

It is also assumed that the reactor vessel lid would not be removed during the whole period of reactor operation. In these conditions and taking into account that the AFPR is essentially a reactor with thermal spectrum of neutrons, there would be no possibility for irradiation of undeclared fertile material within or around the reactor.

The refuelling is accomplished internally without opening the reactor vessel, except for the ‘external’ refuelling performed by a special team once in about 40 years.

At the time of this report, no practicable means existed to extract plutonium from the coated fuel particles. Uranium-thorium and uranium carbide fuel could also be used in the AFPR to enhance its proliferation resistance.

The fresh fuel (MFEs) for the AFPR could be manufactured in states with strong non-proliferation credentials. The spent fuel would then be shipped back to the original manufacturer.

Plant design

The reactor could be designed, built and operated as a boiling water reactor (BWR), a pressurized water reactor (PWR) or a direct-flow system with superheated steam at the core outlet. The latter offers the potential for having a steam cycle thermal efficiency of about 43%, which could decrease the capital costs per kW(e) by nearly one-third. The present short description is based on the data for a BWR version of the AFPR.

The power of a single AFPR unit operating for 40 years is limited to 100–50 MW(e), due to limitations arising from the weight of the transported fuel and the in-vessel fuel storage volume. For 100 MW(e), the anticipated fresh fuel weight is less than 100 tons; such weight could be transported by a cargo aircraft or by barge.

For a BWR version of the AFPR, the energy conversion system employs a Rankine cycle and relies on proven conventional steam turbine systems.

Safety design

The AFPR design strongly relies on inherent safety features.

The ceramic multi-layer coating effectively confines fission products at 1400°C for a long period and at 1600°C in the course of a few hours. At such temperatures the removal of residual heat can be performed by natural convection, conduction and radiation on a passive basis. Given the characteristic coated particle size (the diameter of ~2–4 mm), the heat from coated particle fuel is transferred to the coolant with a delay of only 0.1 s. Therefore, the core of a reactor with boiling water coolant that directly cools such coated fuel particles would provide a very rapid self-compensation of practically any positive reactivity if it is introduced not faster than in 0.1 s.

The core with coated particle fuel has low stored heat, since the temperature of fuel is only 10–15°C higher than that of the surrounding coolant, due to an extremely large heat exchange surface and small thermal resistance of the coated particles. For such a core, there are practically no limits related to critical heat flux or DNB.

The passive safety systems of the AFPR include: passive containment cooling system; reactor isolation condenser, core flood tanks and suppression chamber tanks.

XI-2. Major design and operating characteristics

Major characteristics of the reactor are summarized in Table XI-1. Major characteristics of an NPP with the AFPR (a BWR version) are given in Table XI-2. A vertical cross-section of the reactor vessel is shown in Fig. XI-1.

TABLE XI-1. REACTOR CHARACTERISTICS

ITEMS	SPECIFICATIONS
Fuel design	
Fuel type	Small spherical particles — micro fuel elements (MFE)
Diameter of MFE, mm	2–4
Diameter of UO_2 kernel, mm	1.5–3
Reactor core parameters	
Core outer diameter, m	3.1
Core height, m	3.0
Core volume,m^3	25.6
Core structure	Four annular fuel zones, see Fig. XI-1
Fuel bearing core volume, m^3	12.8
Pebble bed porosity	0.35
MFE density, g/cm^3	5.775
Mass of MFEs in the core, tones	48
Mass of UO_2 in the core, tones	33
Mass of UO_2 in the internal fresh fuel storage, tones	40
Mass of U^{235} (core + fresh fuel storage), tones	7.3
Enrichment by ^{235}U, weight %	8–13
Average core power density, MW/m^3	13.25
Coolant type	Boiling water
Coolant flow direction	Lateral (cross-flow)
Reactor internals	
Inlet headers	Three water inlet headers, see Fig. XI-1
Steam headers	Two steam headers
Fuel burn-up/operation cycle parameters	
Spent fuel burn-up, GW d/t (for equilibrium core)	60–100
Core fuel residence time, days/years	13 140/~40
Reactor vessel	
Vessel type	Cylindrical shell
Inner diameter, m	5
Vessel height, m	13

TABLE XI-2. PLANT CHARACTERISTICS

ITEMS	SPECIFICATIONS
Reactor type	Boling Water Reactor (BWR)
Electric power, MW(e)	100
Thermal power, MW(th)	300
Thermodynamic cycle type	Direct; Rankine cycle with steam turbine
Primary coolant system	
Primary coolant circulation	Forced
Feedwater pressure, MPa	7.5
Steam pressure, MPa	7.2
Core inlet temperature, °C	270
Core outlet temperature, °C	291

XI-3. List of enabling technologies and their development status

A list of the enabling technologies for the AFPR is presented in Table XI-3.

TABLE XI-3. LIST OF ENABLING TECHNOLOGIES FOR AFPR

ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Coated particle fuel technology for light water reactors	Further corrosion and irradiation tests are necessary, including studies of the advanced materials for outer coatings, such as TiN/NbN or ALN/CrN
Reactor internals and on-line refuelling system	Further R&D, mock-up tests and demonstration in a prototype reactor would be necessary
Long-life reactor operation without opening the reactor vessel lid	Step-by-step demonstration in a prototype reactor would be necessary; in-service inspection methods need to be developed and validated
Demountable equipment for once-at-a-time ‘external’ refuelling	Research and development (R&D) would be necessary

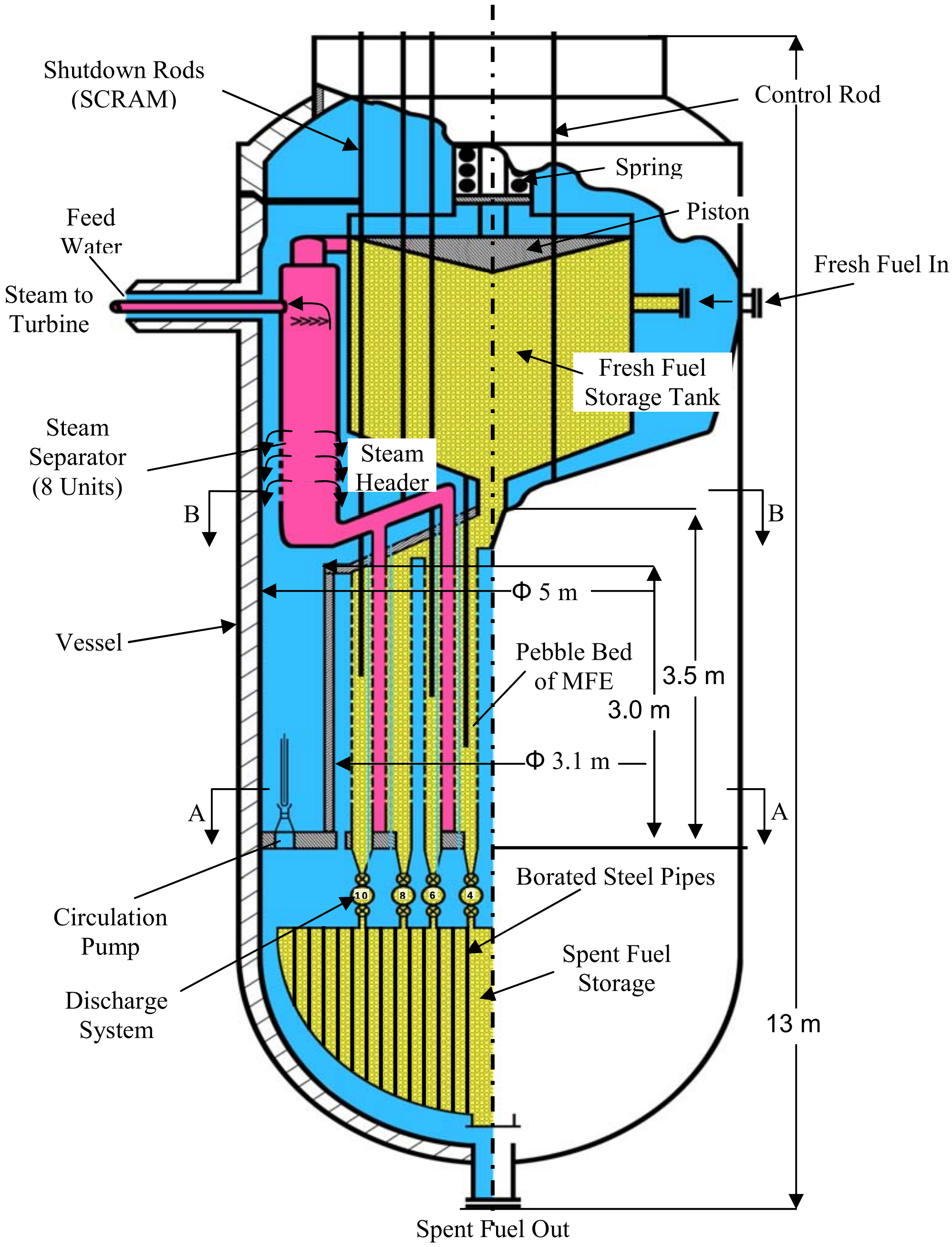


FIG. XI-1. Vertical cross-section of the AFPR-100 reactor vessel.

ANNEX XII

FIXED BED NUCLEAR REACTOR (FBNR)

Federal University of Rio Grande do Sul,
Brazil

XII-1. General information, technical features and operating characteristics

XII-1.1. Introduction

The Fixed Bed Nuclear Reactor (FBNR) concept assumes the use pressurized water reactor (PWR) technology, but incorporates high temperature gas cooled reactor (HTGR) type fuel and the concept of a suspended fixed bed core. Spherical fuel elements are fixed in the suspended core by the flow of water coolant. Any accident signal will cut off the power to the coolant pump causing a stop in the flow. This would make the fuel elements fall out of the reactor core, driven by gravity, and enter a passively cooled fuel chamber where they would reside in a subcritical condition. The Fixed Bed Nuclear Reactor (FBNR) is a simplified version of the fluidized bed nuclear reactor concept [XII-1 to XII-9]. In the FBNR, spherical fuel elements are in a fixed position in the core; therefore, there is no concern about the consequences of multiple collisions between them, an issue that may be raised about the fluidized bed concept. Relatively little work has been done for the fixed bed nuclear reactor so far, but the experiences gained from the development of a fluidized bed reactor can facilitate the development of the FBNR.

The FBNR concept is being developed in the Federal University of Rio Grande do Sul, (UFRGS - Brazil) in cooperation with several research groups in the institutes around the world such as the Imperial College of University of London (England), the Institute for Nuclear Science and Technology (Vietnam), the Gazi University (Turkey), and the Catholic University (Uruguay). More broad international cooperation for the development of FBNR is being sought for.

XII-1.2. Applications

The FBNR is designed to produce electricity alone or to operate as a cogeneration plant producing electricity and potable water or steam for industrial purposes. As an option, the FBNR may be designed for district heating.

XII-1.3. Special features

The FBNR is a land-based nuclear power plant for urban or remote locations. A lifetime core operation without on-site refuelling is envisaged.

XII-1.4. Summary of major design and operating characteristics

Some major design and operating characteristics of the FBNR are given in Table XII-1; the major design objectives are outlined in Table XII-2.

The reactor is modular in design, and each module is assumed to be fuelled at the factory. The fuelled modules in sealed form are then transported to and from the site. The FBNR has a long fuel cycle time and, therefore, there is no need for on-site refuelling. Else, the reactor makes an extensive use of PWR technology.

TABLE XII-1. MAJOR DESIGN CHARACTERISTICS OF FBNR

ATTRIBUTES	DESIGN PARTICULARS
Thermal power generated per module	134 MW
Electric power generated per module	40 MW
Targeted availability	95%
Core configuration	Suspended core, integrated primary circuit.
Fuel element	15 mm diameter spherical fuel elements incorporating TRISO type micro spherical fuel particles.
Fuel material	UO_2, (^{233}U-Th) O_2, or MOX
Fuel element cladding	SiC as an option
Moderator/Coolant	Pressurized light water
Coolant flow rate	668 kg/s
Module diameter	200 cm
Module pressure	160 bar
Core inner diameter	20 cm
Core outer diameter	160 cm
Active core height	200 cm
Specific power density in the core	33.7 MW(th)/m^3
Shutdown system	Pump turn-off initiated by reactor protection system
Slow reactivity control	Movement of a fuel limiter
Fast reactivity control	Fine-motion control rod

TABLE XII-2. DESIGN OBJECTIVES OF FBNR

OBJECTIVE	DESIGN APPROACH
High level of safety	Strong reliance on inherent and passive safety features and passive systems.
Enhanced safeguard ability	Fuel elements are confined in the fuel chamber that could be sealed by authorities for inspection at the end of the fuel life. The reactor vessel is cladded by neutron-absorbing materials to eliminate the possibility of neutron irradiation of any external fertile material.
Enhanced proliferation resistance	Use of thorium based TRISO type fuel.
Reduced nuclear waste	The spent fuel elements have the size and shape adequate to serve as a source of radiation for applications in industry and agriculture.
Reduced adverse environmental impacts	Underground containment in a garden like site.
Improved economy	Modular design to be produced in series. Design simplicity. Elimination of burnable poisons.
Technology transfer	The technology could be open to all nations of the world under the supervision and control of international authorities.

OBJECTIVE	DESIGN APPROACH
Long core lifetime	Insertion of fresh fuel into the core is performed continuously to compensate for fuel burn-up.
Enhanced security	Reactivity excursion accident cannot be provoked. The reactor core is filled with fuel only when all operational conditions are met.
Mitigation of steam generator leakage problem	The water heated in the reactor core passes through an integrated steam generator producing steam to drive the turbine.
Resistance to unforeseen accident scenarios.	Any probable accident, through cutting off the power to the pump, causes the fuel elements fall out of the core driven by the force of gravity. The normal state of control system is “switch off”. The pump is “on” only when all operating conditions are simultaneously met.

The FBNR is modular in design such that any size of reactor can be constructed from the basic modules. It is an integrated primary system design. The basic module has in its upper part the reactor core and a steam generator and in its lower part the fuel chamber, Fig. XII-1. The core consists of two concentric perforated Zircaloy tubes inside which, during the reactor operation, the spherical fuel elements are held together by the coolant flow in a fixed bed configuration, forming a suspended core. The coolant flows vertically up into the inner perforated tube and then, passing horizontally through the fuel elements and the outer perforated tube, enters the outer shell where it flows up vertically to the steam generator. The fuel chamber is a 25 cm diameter tube made of high neutron absorbing alloy, which is directly connected underneath the core tube. A steam generator of the shell-and-tube type is integrated in the upper part of the module. A control rod slides inside the centre of the core for fine reactivity adjustments. The reactor is provided with a pressurizer system to keep the coolant at a constant pressure. Each module has an independent pump. The pump circulates the coolant inside the module moving it up through the fuel chamber, the core, and the steam generator. Thereafter, the coolant flows back down to the pump through the concentric annular passage. At a certain pump velocity, the water coolant carries up the 15 mm diameter spherical fuel elements from the fuel chamber into the core. A fixed suspended core is formed in the module. In a shut down condition, the suspended core breaks down and the fuel elements leave the core and fall back into the fuel chamber.

Any signal from any detector due to any initiating event is assumed to cut-off power from the pump, causing the fuel elements to leave the core and fall back into the fuel chamber, where they remain in a highly subcritical and passively cooled condition. The fuel chamber is cooled by natural convection, transferring heat to the pool of water or to the air surrounding the fuel chamber.

A detailed heat transfer analysis of the fuel elements has shown that, due to a high convective heat transfer coefficient and a large heat transfer surface, the maximum power extracted from the reactor core is restricted by the mass flow of the coolant corresponding to a selected pumping power ratio, rather than by design limits of the materials.

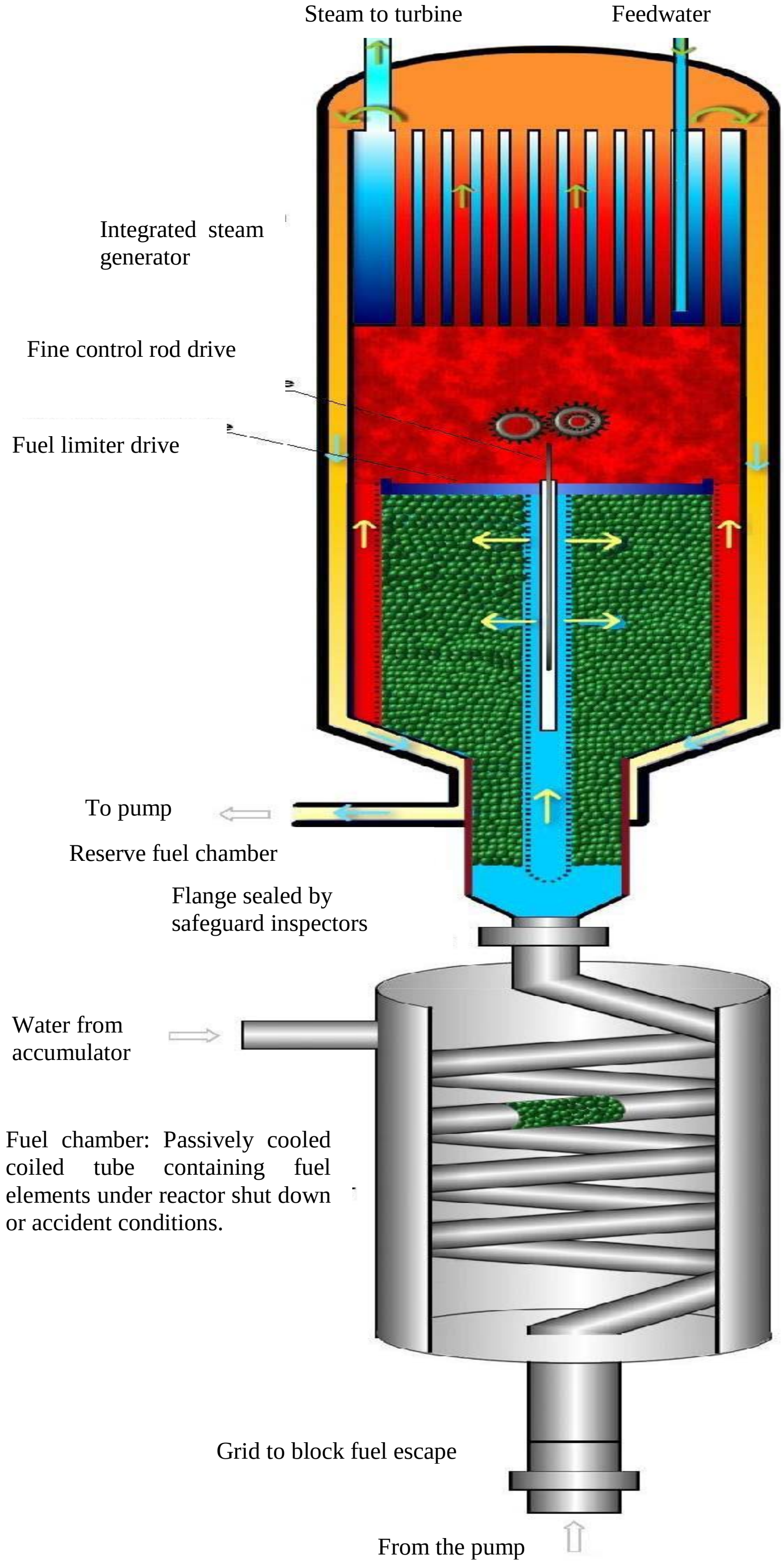


FIG. XII-1. Simplified schematic diagram of the FBNR module.

The proposed reactor concept is very flexible in its nature, which makes it possible to devise several alternative designs:

(1) *Fixed bed with supercritical steam as coolant.* The concept of a direct cycle reactor operating at supercritical pressure is attractive for radically improving the thermal efficiency. Such reactor could combine the fixed bed concept with the idea of using a direct cycle reactor operating at supercritical pressure, for example, as proposed in [XII-8]. Supercritical steam is used as the reactor coolant. The critical pressure of water is 221 bar. When the reactor operates at 250 bar, the supercritical water does not exhibit a change in phase, and the phenomenon of boiling does not exist. The water density decreases continuously with temperature.

    The coolant inlet temperature in the lower part of the bed may be 290ºC, and the outlet temperature in the upper part of the bed is then ~416ºC. Therefore, the water density decreases continuously from 0.744 to 0.137 g/cm^3 along the bed. The recommended pressure of 250 bar is due to the smooth and mild variation of density with pressure in this region, resulting in a stability of flow in the core. The power production is much higher in this option as the difference in inlet and outlet enthalpy is much higher than in a conventional pressured or even boiling water reactor. The plant thermal efficiency is estimated to exceed 40%. By adopting supercritical steam as a coolant, the turbines could be made smaller compared with those used in existing light water reactors (LWRs). The superheated steam is fed directly into the turbine. The steam water separation is not needed for direct cycle reactor. Other advantages of a supercritical option include the absence of steam generators, and the reduced waste heat. However, a new design of spherical fuel elements will be required, since SiC is not corrosion resistant in water at supercritical parameters [XII-9].

(2) *Fixed bed with helium gas as coolant.* In this option, the fixed bed is cooled by helium, bringing in all advantages of a gas cooled reactor, including the use of a direct cycle gas turbine and the resulting high efficiency. In this case, the reactor has fast neutron spectrum.

Neutron-physical characteristics

The neutron-physical characteristics of the FBNR are not determined as yet. They are expected to be similar to those of the conventional pressurized water reactors.

Reactivity control mechanism

The FBNR does not use burnable poisons, and slow reactivity control to compensate for fuel burn-up is achieved by introducing fresh fuel to the core by raising the fuel limiter and allowing fresh fuel elements enter the module from the reserve fuel chamber, Fig. XII-1. Fast reactivity control is provided by a fine-motion control rod with internal drive mechanism. This rod moves in a guide tube located in the centre of the core. This guide tube also acts as an inlet collector for the core, Fig. XII-1.

Thermal-hydraulic characteristics

A PWR type reactor operating at 160 bar with the selected inlet/outlet temperatures will have about 33% efficiency with an indirect cycle.

The high surface-to-volume ratio of spherical fuel elements results in excellent heat transfer characteristics yielding a low maximum-to-average fuel temperature ratio. The core is cooled by forced convection, but the residual heat produced in the fuel chamber is removed by natural convection. No heat exchange crisis is anticipated.

The water coolant flows into the core at a rate of 668 kg/s with the inlet temperature of 290°C and leaves the core at 326°C, being directed to the steam generator.

Fuel lifetime/period between refuellings

The fuel lifetime is targeted to be more than 10 years, depending on considerations of plant economy and energy security requirements. It could be easily achieved by adequate dimensioning of the reserve fuel chamber. Within the fuel lifetime, the reactor is assumed to operate with a weld sealed vessel.

Design basis lifetime for reactor core, vessel and structures

The module, the fuel chamber and other parts are relatively small pieces and can simply be replaced as needed.

Design and operating characteristics of systems for non-electric applications

The FBNR can operate within a cogeneration plant producing both electricity and desalinated water. A Multi-Effect Distillation (MED) plant may be used for water desalination. An estimated 1000 m^3/day of potable water could be produced at 1 MW(e) reduction of the electric power.

Economics

The total electricity production cost from the FBNR is estimated at 2.1 cents/KWh, with reference to US$. In this, the capital cost is about 1.6, the fuel cost is 0.3, and the O&M cost is 0.2 cents/KWh. Costs at this level may compete well with alternative energy sources. The construction cost is estimated as ~1000 US$/kW(e), the construction period is about 2 years.

XII-1.5. Outline of fuel cycle options

A standard fuel cycle of high temperature gas cooled reactors could be used as basic option for the FBNR. A variety of alternative fuel cycle options could be used according to the demand. These include a plutonium burner mode using plutonium-thorium oxide fuel and a closed fuel cycle based on ^{233}U-Th.

XII-1.6. Technical features and technological approaches that are definitive for FBNR performance in particular areas

XII-1.6.1. Economics and maintainability

The simplicity of design, short construction period, and an option of incremental capacity increase through modular approach are expected to result in a much smaller capital investment for the FBNR as compared to conventional PWRs.

Research & development (R&D) and licensing for the FBNR could be essentially simplified, since it may be enough to develop, validate and license by test only a single module.

The elimination of on-site refuelling and long core lifetime could reduce the operation and maintenance costs of the reactor. The elimination of burnable poison contributes to the improved neutron economy and results in essentially lower fuel enrichment, contributing to reduced fuel enrichment costs. There are no fuel assemblies in the FBNR core, which would also contribute to a reduction in fuel fabrication costs.

The total investments required to design, construct, and commission the FBNR, including the investments during construction, are evaluated to be very low and easy to raise. A single module FBNR plant of 40 MW(e) plant could cost about US$ 40 million. Therefore, the risk of investments in the FBNR could be sufficiently low too.

XII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The elimination of burnable poison and high fuel burn-up contribute to a more efficient use of uranium resources. Various fuel options available for the FBNR broaden the available resource base, which could also include plutonium from dismantled nuclear weapons and the abundant thorium available in countries like Brazil and India.

The increased fuel burn-up and a fuel form that is capable of perfect confinement of fission products at high temperatures are factors that contribute to the minimization of waste.

The inherent safety feature — each fuel particle having its own containment — reduces the probability of a large release of radioactivity to the environment. Also, each reactor module contains a relatively small quantity of thermal energy, due to low operating temperature of fuel. The envisaged simple underground containment, an additional defence-in-depth feature, also contributes to reduced adverse environmental impacts.

Spent fuel from the FBNR can be treated or reprocessed in a way similar to that for HTGR fuel. Should reprocessing not be allowed, the FBNR spent fuel elements could easily be vitrified within the modules and the whole module could then be deposited directly in a waste repository.

XII-1.6.3. Safety and reliability

Safety concept and design philosophy

The safety philosophy behind the FBNR design is strong reliance on inherent and passive safety features. To the extent possible, laws of nature and physics should govern safety of the reactor.

Provisions for simplicity and robustness of the design

Modularity, strong reliance on passive safety design options, and low power density in the core are the factors contributing to simplicity and robustness of the FBNR design.

Active and passive systems and inherent safety features

A "fail-safe" passive control system is assumed to govern the reactor system. In this system, numerous signals from various redundant detectors of different origin enter the circuit. When all signals are within the pre-defined ranges of values, then the pump power will be in the "on" position. In any other situation, the coolant pump is in its normal "switched off" position, and the fuel elements leave the reactor core, driven by gravity, and become deposited in a passively cooled fuel chamber.

The use of HTGR type fuel capable of confining fission products at very high temperatures adds to this a large margin to fuel failure, which is an important inherent safety feature.

The active safety systems include a control rod, which is used only for fine control of reactivity during normal operation, and a slow-movement fuel limiter that allows fresh fuel elements from the reserve chamber enter the core to compensate for the bulk of reactivity change due to fuel burn-up. The design of fuel limiter drive could be made similar to that of contemporary control rod drives, i.e., preventing its inadvertent upper movement to the extent possible.

Structure of the defence-in-depth

In the FBNR, fission products are confined inside the fuel elements of a type designed to resist temperatures of about 1600°C. At the same time, these fuel elements are at temperatures less than 350°C under normal operating conditions. In other words, each fuel particle has a small containment - SiC coating layer that effectively prevents the release of radioactivity up to very high temperatures, and the margin to fuel failure is around 1250°C. The fuel elements are in the reactor core only when all reactor components operate within the design ranges of parameters. Otherwise, they leave the reactor core and reside in a passively cooled subcritical state. In addition to this, the reactor is located inside an underground containment building.

Design basis accidents and beyond design basis accidents

The safety system of the FBNR could take care of any conceivable design basis or beyond design basis accidents by relying on inherent safety features and passive systems only. Any abnormality in operation is expected to result in a passive shut down of the reactor. Should it for whatever reason fail, a large margin to fuel failure would simplify accident management.

Probability of unacceptable radioactivity release beyond the plant boundaries

The target is 10^{-7}.

Measures planned in response to severe accidents

The target is to eliminate off-site emergency planning.

XII-1.6.4. Proliferation resistance

Adopting a thorium fuel cycle is an intrinsic measure that could hinder the possibility of misuse of nuclear materials for nuclear weapons. Within such cycle, ^{233}U is produced with a noticeable admixture of a highly radioactive ^{232}U, which essentially complicates reprocessing and assembly operations for nuclear weapons. The mixing of thorium with low enriched uranium or plutonium results in the production of ^{233}U that is additionally diluted with ^{235}U or ^{238}Pu. The access to pure ^{233}U will only be possible through isotope separation techniques. The high ^{238}Pu to ^{239}Pu ratio and the production of gamma emitting ^{208}Tl in the thorium cycle are hindrances to nuclear proliferation. ^{238}Pu has a spontaneous fission that contributes to increased residual heat of spent fuel that will complicate the production of nuclear weapons.

An additional barrier is provided by SiC coating layers which need to be mechanically removed before conventional aqueous methods are applied for reprocessing of fuel kernels.

The fuel elements of the FBNR are confined in the fuel chamber, which could be sealed by the authorities and inspected at the end of fuel lifetime.

The FBNR has a very long lifetime (more than 10 years) and will not be refuelled on the site. The fuel is located in the sealed fuel chamber outside the pressure vessel; and the refuelling is performed just by taking the sealed fuel chamber to a factory, which could be performed by an authorized team under strict security measures.

XII-1.6.5. Technical features and technological approaches used to facilitate physical protection of FBNR

The fuel is contained in a sealed module inaccessible to outsiders for a long period of operation without on-site refuelling. It can only be manipulated at the factory. Only under design operating conditions the fuel remains in the core and the reactor becomes critical; under any other situations, the fuel leaves the core and is stored in a subcritical state.

XII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of FBNR

With its design simplicity and strong reliance on inherent and passive safety features, the FBNR could be a good choice for many developing countries. A long-life core operation without on-site refuelling could provide certain guarantees of sovereignty for those countries that would prefer to lease fuel or even a nuclear power plant rather than master an indigenous fuel cycle. The design and technology development for FBNR could benefit from cooperation of the researchers and designers in many developing and industrialized countries around the world.

XII-1.8. List of enabling technologies relevant to FBNR and status of their development:

The list of enabling technologies for the FBNR is given in Table XII-3.

TABLE XII-3. ENABLING TECHNOLOGIES AND THEIR DEVELOPMENT STATUS

DESIGN AREA	ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Fabrication of fuel elements based on coated particles.	Technology for application of SiC coatings to spherical fuel elements of 15 mm diameter. Fabrication technology for coated particles.	Fabrication technology for coated particles is available in several countries. Technology for application of SiC coatings to spherical fuel elements of 15 mm diameter is being developed at UFRGS-Brazil. After the pilot fuel elements are fabricated, irradiation tests and post-irradiation examinations would be required. Since the irradiation can be performed for small batches of spherical fuel elements, it could be performed in various facilities already available around the world.
Long term reactivity control.	Method of securing reserve reactivity by fresh fuel insertion without the use of burnable poison.	It is planned to use the existing control rod drive technology to design fuel limiter drive.
Pump control.	The normal state is “switched off”. The pump is “on” when all signals from all detectors governing the operating conditions are simultaneously within the design ranges of values.	R&D planned.
Neutron-physical calculations.	Equivalence models to relate cylindrical and spherical geometry.	Equivalence models need to be developed, and then standard codes developed for PWRs could be used.

DESIGN AREA	ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Thermal-hydraulic calculations.	Thermal-hydraulic modelling of a suspended core.	Reliable codes for PWRs exist; their applicability to calculation of suspended cores needs to be examined.
Study of FBNR hydraulic performance.	A full size experimental hydraulic module made of transparent materials using stainless steel balls to simulate fuel elements is required to perform testing. The module is to be provided with instrumentations to measure the basic hydraulic parameters such as pressure drop as a function of coolant flow velocity under different core configurations. Videotape is to be made of the operation in order to analyse the core behaviour under various simulated operating and accidental conditions.	R&D and construction of test facility are planned.
Passive cooling of fuel chamber.	Passive cooling of fuel chamber by natural convection of water with heat transfer to air and water through the chamber wall.	Calculations are being performed.
Reliability of materials under long-life core operation.	Relevant experience in validation, testing and demonstration of fuel and structural materials from other designs of small reactors without on-site refuelling around the world could be used to develop the R&D programme.	Not started yet.

XII-1.9. Status of R&D and planned schedule

The current design stage is very preliminary, just a start-up of conceptual design. The programme of R&D for FBNR visualizes the following steps:

- Conceptual design development;
- Construction of a full size non-nuclear hydraulic module to verify the hydraulic performance and determine the basic parameters of a suspended core;
- Performance of neutron physical, thermal-hydraulic, fuel behaviour and structural calculations;
- Fabrication and testing of pilot batches of fuel;
- Engineering design of a prototype reactor;
- Performance of a zero power experiment with one module in a nuclear experimental facility;
- Construction of a single module prototype.

The institutions that so far have shown interest in participating in this project include Imperial College of the University of London, Institute of Theoretical and Experimental Physics (ITEP) and the Institute of Physics and Power Engineering (IPPE) in the Russian Federation and some individual scientists in Uruguay, Vietnam, Turkey, Finland, Switzerland, and the USA. Increased international cooperation would be helpful for the promotion of the FBNR project.

Estimate of an overall time frame within which the design could be implemented under favourable financing conditions is ~10 years. It is estimated that about one million US$ dollars is needed to build a zero power prototype of the FBNR and demonstrate the concept feasibility

XII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The fixed bed suspended core reactor concept incorporates radical conceptual changes in design approaches and system configurations in comparison with existing practice and would, therefore, require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented before launching the FBNR into series.

XII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

There are similar activities ongoing in the All-Russian Institute of Atomic Machinery (VNIIAM) in Moscow (Russian Federation) and in the USA, in the Pacific North-west National Laboratory (PNNL). These activities are related to the development of a pebble bed boiling water reactor concept with superheated steam [XII-9]. The reactor supplies energy to a 1500 MW(e) plant, and an option of a smaller 300 MW(e) plant has been considered.

XII-2. Design description and data for FBNR

XII-2.1. Description of the nuclear systems

Reactor core and fuel design

The FBNR fuel is a 15 mm diameter spherical fuel element made of compacted micro fuel elements (MFEs) with the fuel density of 5.9 g/cm^3, cladded by silicon carbide. The matrix surrounding coated particles to form a fuel element is pyrolythic graphite, Fig. XII-2.

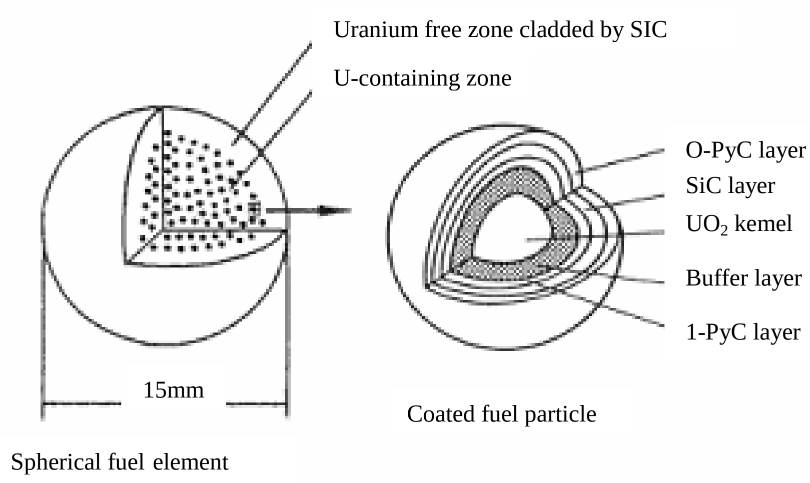


FIG. XII-2. Fuel element of FBNR.

The MFEs are coated particles similar to TRISO fuel with the outer diameters of about 2 mm. They consist of 1.5–1.64 mm diameter uranium dioxide spherical kernels coated with 3 ceramic layers. The inner layer, called a buffer layer, is made of 0.09 mm thick porous pyrolythic graphite (PyC) with the density of 1 g/cm^3, providing space for gaseous fission products. The second layer is made of 0.02 mm thick dense (1.8 g/cm^3) PyC, and the outer layer is 0.07–0.1 mm thick corrosion resistant silicon carbide (SiC). The fourth, outer PyC layer is assumed to be absent. SiC protection layers, manufactured by chemical vapour deposition (CVD) method, create resistance of graphite components against water and steam at high temperatures. Small fuel elements are able to confine fission products indefinitely at temperatures below 1600°C.

Main heat transport system

A scheme of the FBNR main heat transport system with indication of heat removal path in normal operation and in accidents is given in Fig. XII-3.

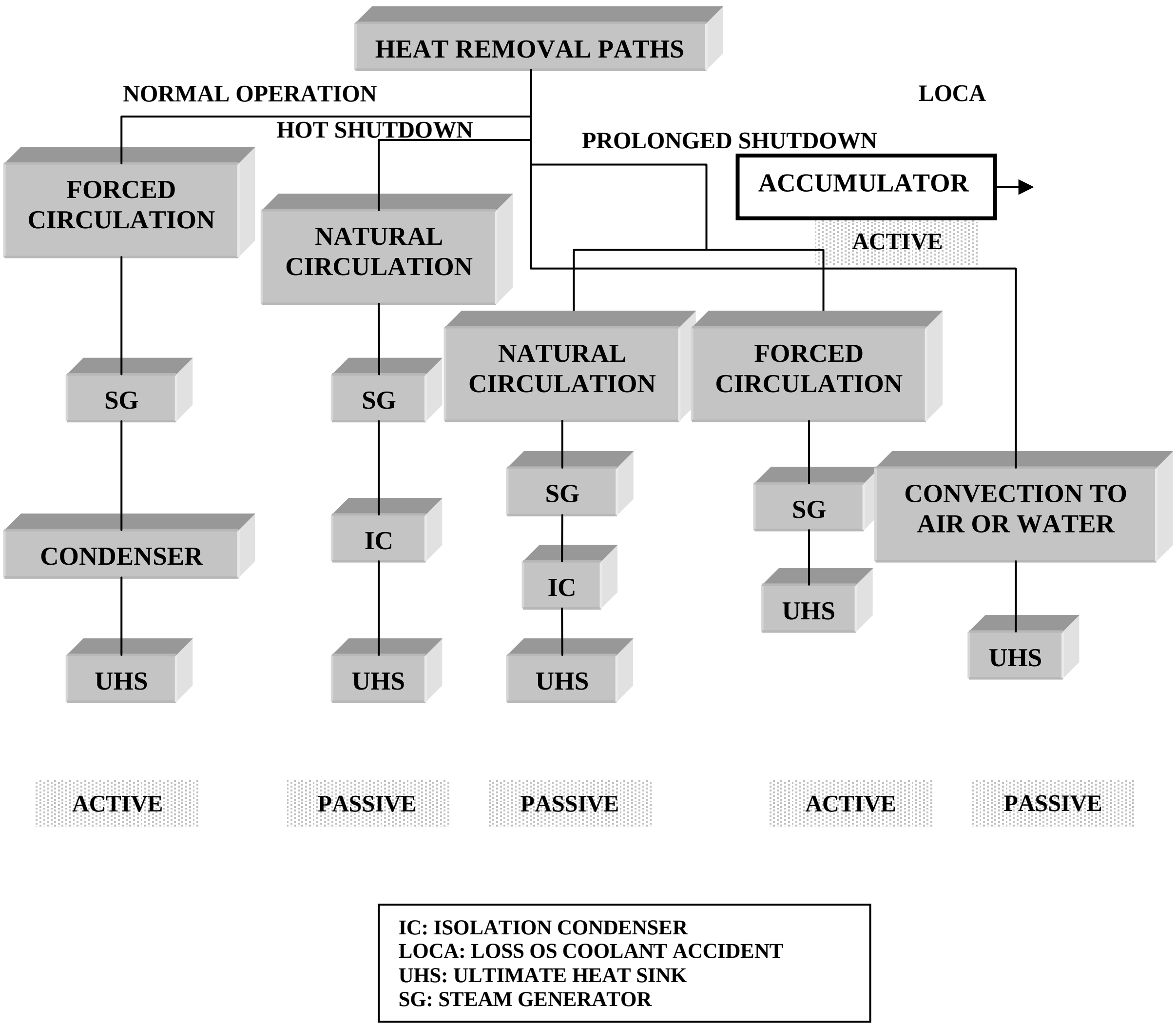


FIG. XII-3. Heat removal paths of FBNR.

XII-2.2. Description of the turbine generator plant and systems

A conventional turbine generator plant could be used.

XII-2.3. Systems for non-electric applications

No information was provided

XII-2.4. Plant layout

The plant is assumed to be located underground to avoid any negative visual impact. The nuclear power plant site is envisaged to incorporate garden like surroundings. The observed part could be the administration building and the chimney for air exhaustion. In this building, the reactor control room could be located. The swimming pool above ground serves as the accumulator to supply water to cool the fuel chamber, and eventually it could be used as a heat sink for the residual heat removal through an isolation condenser (IC).

A vehicle is assumed to transport the reactor module and fuel chamber to the underground building through a double door with an isolation area.

A general view of the FBNR plant is shown in Fig. XII-4.

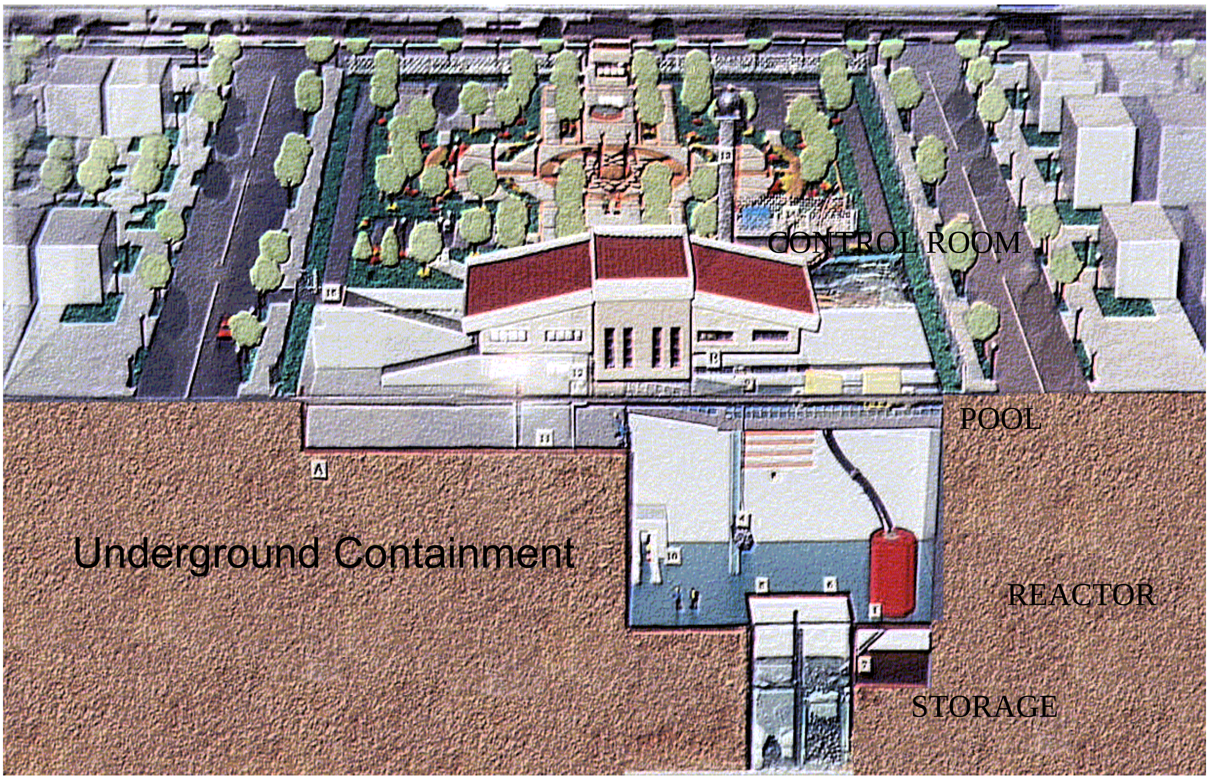


FIG. XII-4. General view of the FBNR nuclear power plant.

REFERENCES

[XII-1] SEFIDVASH, F. Preliminary evaluation of the fixed and fluidized bed nuclear reactor concept using the IAEA-INPRO methodology, Kerntechnik, 69, 3 (February 2004).

[XII-2] SEFIDVASH, F. Fixed bed suspended core nuclear reactor concept, Kerntechnik, 68 56–59 (February 2003).

[XII-3] SEFIDVASH, F., Status of the small modular fluidized bed light water nuclear reactor concept, Nuclear Engineering and Design, 167 pp. 203–214 (1996).

[XII-4] SEFIDVASH, F., A preliminary thermal-hydraulic study of the fluidized bed nuclear reactor concept, Kerntechnik, 60, 1, pp.48–51 (1995).

[XII-5] SEFIDVASH, F., A fluidized bed nuclear reactor concept, Nuclear Technology, Vol. 71, No. 3, pp. 527–534 (1985).

[XII-6] SEFIDVASH, F., Loss of coolant accident in the Fluidized Bed Nuclear Power Reactor, Atomkernenergie/Kerntechnik, 42, 2, pp. 125–126 (1983).

[XII-7] SEFIDVASH, F., Haroon, M.R., Preliminary reactor physics calculations of a Fluidized Bed Nuclear Reactor Concept, Atomkernenergie/Kerntechnik, 35, 3, pp.191–195 (1980).

[XII-8] OKA, Y., et al., Systems design of direct-cycle supercritical water cooled fast reactors, Nuclear Technology, .109, pp. 1–10 (1995).

[XII-9] TSIKLAURI, G., et al., Pebble bed boiling water reactor concept with superheated steam. ICONE-2 (Proc. 10th Int. Conf. on Nuclear Engineering, Arlington, Virginia, USA, April 14–18, 2002). ICONE2–22058.

[XII-10] For up to date information visit www.rcgg.ufrgs.br/fbnr.htm .

GAS COOLED SMALL REACTORS

ANNEX XIII

FAST NEUTRON BATTERY-TYPE GAS COOLED POWER REACTOR OF 300 MW THERMAL (BGR-300)

Russian Research Centre “Kurchatov Institute”,
Russian Federation

XIII-1. Basic summary

The concept of a fast neutron battery-type gas cooled power reactor of 300 MW(th) (BGR-300) is being developed by the Russian Research Centre “Kurchatov Institute”.

Reactor design

The BGR-300 is a small tank-type reactor with the secondary vessel acting as a safety system. The BGR-300 has a three-region core profiled by the fissile material content and effective density. A molten salt reflector plays the role of an in-vessel radiation shield. The reactor core uses a porous matrix fuel in the form of quasi-homogeneous heat generating blocks with cross-circulated coolant.

Plant design

The BGR-300 is a modular-type power plant with a single reactor unit. The BGR-300 operates in an indirect cycle with air in the secondary circuit and an air turbine used for the electricity generation. There is no intermediate heat transport system. Through implementing a very high temperature heat exchanger in the primary circuit before the primary-to-secondary heat exchangers, the BGR-300 provides an option of hydrogen production, which could be accomplished through thermo-chemical processing of natural gas.

Safety design

The safety design philosophy is to exclude core damage in all conceivable accidents through strong reliance on inherent safety features and passive systems. For example, natural convection of molten salt from side reflector is assumed to remove heat from the core in loss of coolant accidents.

XIII-2. Major design and operating characteristics

Main characteristics of the BGR-300 core are summarized in Table XIII-1. Major characteristics of an NPP with the BGR-300 are given in Table XIII-2. A simplified schematic diagram of the BGR-300 plant is shown in Fig. XIII-1.

TABLE XIII-1. CORE CHARACTERISTICS

ITEM	SPECIFICATION
Fuel type	Fuel based on porous carbon or carbide matrix
Fuel composition	(U-Pu)C, (U-Pu)N
Fuel enrichment	(14–15.5)% of fissile Pu isotopes
Structural materials	Mono-crystalline nickel based alloy
Average power density in fuel, W/cm^3	~900

ITEM	SPECIFICATION
Fuel assembly type	Cylindrical; with lateral coolant flow; using a single collector with perforated walls
Fuel assembly material	Stainless steel or nickel alloy
Active core height, cm	310
Effective diameter of active core, cm	260
Power flattening	3-zones of different material content and density.
Reflector	Molten salt side (radial) reflector
Reflector thickness, cm	52
In-core breeding ratio	~1.03÷1.05
Burn-up reactivity swing, ΔK/K	0.3%
Temperature reactivity effect, cold-to-hot state, ΔK/K	–(0.15–0.45)%
Adiabatic temperature reactivity coefficient at BOL in: Hot active core, pcm/K Cold active core, pcm/K	 –1 –0.31
Adiabatic temperature reactivity coefficient at EOL in: Hot active core, pcm/K Cold active core, pcm/K	 –0.29 –0.3
Period of operation without on-site refuelling, years	12

TABLE XIII-2. PLANT CHARACTERISTICS

ITEM	SPECIFICATION
Reactor type	Pressure vessel; guard vessel
Maximum thermal output, MW(th)	300
Maximum electric output, MW(e)	130
Cycle type/thermodynamic efficiency, %	Indirect/44
Number of loops	4
Main circulation pipelines	4 pipelines, tube-in-tube type
Material of pipes	High-temperature resistant stainless steel.
Coolant of primary circuit	He or He-Xe (as an option)
Core inlet/outlet temperature	350^{o}C/850^{o}C
Primary circuit pressure	16 MPa
Primary coolant flow rate	88.9 kg/s
Coolant of the secondary circuit	Air
Residual heat removal in accidents	Heat is removed from the reactor vessel by natural convection of air

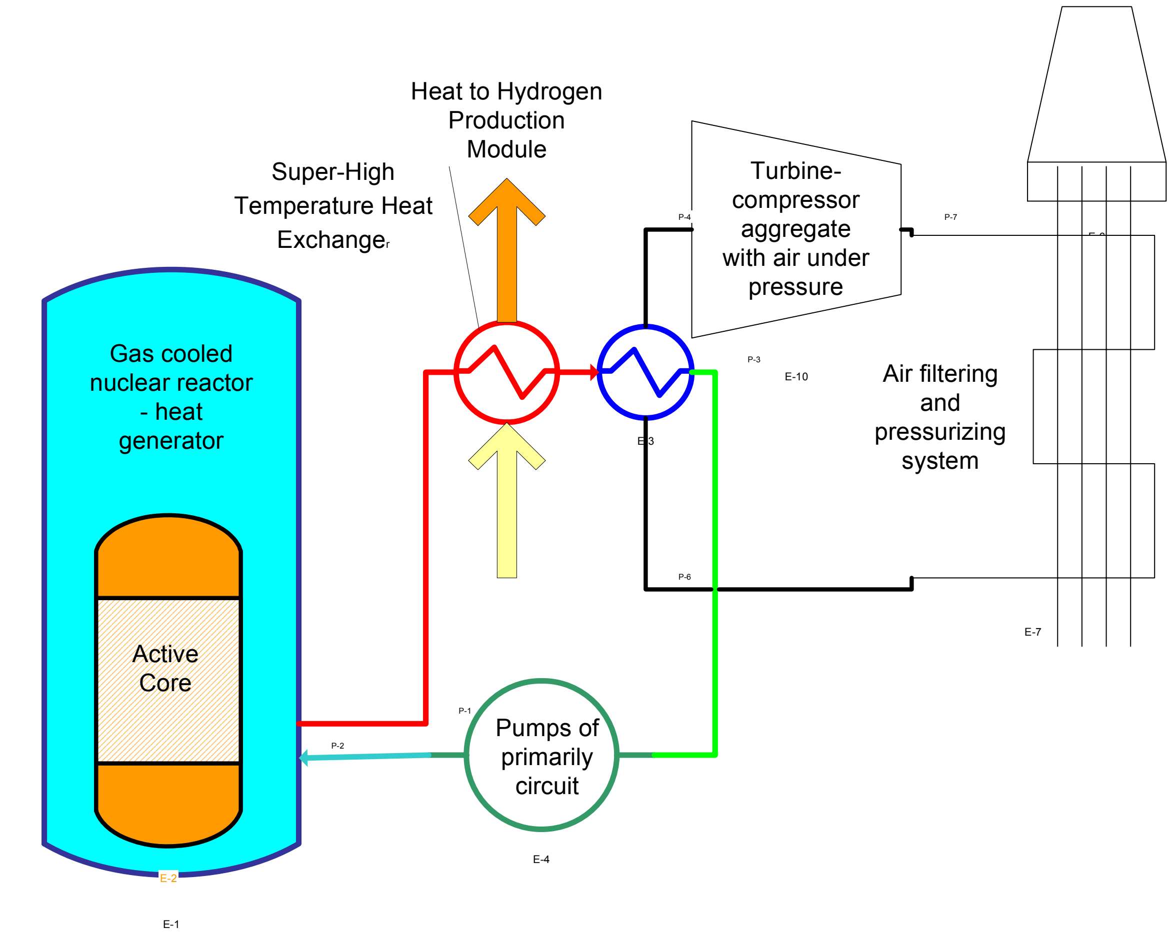


FIG. XIII-1. Schematic diagram of the BGR-300 power plant.

XIII-3. Enabling technologies and their development status

The enabling technologies for the BGR-300 are specified in Table XIII-3.

TABLE XIII-3. ENABLING TECHNOLOGIES FOR BGR-300

ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Use of matrix type porous nuclear fuel	Conceptual design
Molten salt technology	Conceptual and engineering design
High temperature structural materials	Already tried-out in other projects

SODIUM COOLED SMALL REACTORS

ANNEX XIV

SUPER-SAFE, SMALL AND SIMPLE REACTOR (4S, TOSHIBA DESIGN)

Toshiba Corporation and Central Research Institute of Electric Power Industry (CRIEPI), Japan

XIV-1. General information, technical features and operating characteristics

XIV-1.1. Introduction

The 4S (super-safe, small and simple) is a nuclear power plant with a sodium cooled small reactor without on-site refuelling. Being developed as distributed energy source for multi purpose applications, the 4S offers two outputs of 30 MW(th) and 135 MW(th), respectively. These energy outputs have been selected from demand analyses [XIV-1].

Japan has a long-term national plan to introduce sodium-cooled fast breeder reactors (FBRs) for effective utilization of natural uranium; to provide their initial fuel load, plutonium will be extracted from the spent fuel of existing light water reactors (LWRs).

To accomplish this plan, the sodium cooled experimental reactor JOYO is now under operation in the Japan Atomic Energy Agency (JAEA), a new organization combining the former Japan Nuclear Cycle Development Institute (JNC) and Japan Atomic Energy Research Institute (JAERI). The JOYO reactor is being operated to develop and validate sodium, fuel, and material technologies, etc.

The prototype FBR MONJU was constructed by former JNC to demonstrate electricity generation by FBRs and build sufficient experience with sodium cooled power plants, aiming at their commercialization in Japan in the future. The technologies gained through these experiences support the base of the 4S design as a sodium cooled reactor.

Apart from the prototype FBR MONJU, much research and development (R&D) has already been performed to complete the design of the Demonstration FBR, sponsored by nine Japanese utilities, Electric Power Development Co., Ltd., and the Japan Atomic Power Company (JAPC). The R&D included the development of new types of equipment for sodium cooled reactors such as highly reliable electromagnetic pumps and double-walled tube steam generators with leak detection systems for both sodium and water/steam. This new equipment is considered to become more important for the commercialization of sodium cooled reactors, and the 4S is adopting these technologies in its design.

Since 2002, CRIEPI, JAERI, Osaka University, and the University of Tokyo are performing the R&D focussed on the technologies of the 4S reactor core, fuel and reflectors, sponsored by the Japan Ministry of Education, Culture, Sports, Science and Technology (MEXT). Critical experiments for the 4S have been performed at the Fast Critical Assembly (FCA) in Tokai-mura (former JAERI). The Argonne National Laboratory (ANL) and the Idaho National Laboratory (INL, former ANL-West) have developed the metal fuel technology, which is a keystone to achieving the desired features the 4S, and much experience with the metal fuel has been gained through the operation of the EBR-II reactor in the USA.

The 4S is being designed and developed mainly by Toshiba Corporation and the Central Research Institute of Electric Power Industry (CRIEPI) in Japan.

XIV-1.2. Applications

As it was already mentioned, the 4S concept offers two different thermal outputs, which are 30 MW(th) and 135 MW(th). When all thermal energy produced is converted into electric power, the 4S will generate 10 MW(e) and 50 MW(e) respectively.

The plant can be configured to deliver not only electricity but also hydrogen and oxygen using the process of high temperate electrolysis (HTE). The HTE is a technology that can produce both hydrogen and oxygen from steam and electricity; the latter are produced by the 4S without environmentally disadvantageous by-products, such as carbon dioxide. The production ratio of electricity to hydrogen/oxygen could be adjusted in the balance of plant (BOP) design according to the demand at each specific site. The 4S units of 30 MW(th) and 135 MW(th) are capable of producing hydrogen/oxygen at a rate of 3000 Nm^3/hour and 14 000 Nm^3/hour respectively when all generated electricity is utilized for hydrogen/oxygen production.

The plant can also be configured to produce potable water using a two-stage reverse osmosis system for seawater desalination. The amount of potable water produced could also be selected in response to demand at each site, but the maximum capabilities of two types of the 4S to produce potable water are 34 000 m^3/day and 170 000 m^3/day respectively, when all generated energy is utilized for desalination.

XIV-1.3. Special features

The 4S is a land-based nuclear power station with the reactor building basically embedded underground for security reasons, to minimize unauthorized access and enhance inherent protection against externally generated missiles. The BOP including a steam turbine system and the HTE units or desalination system is located at ground level.

To assure high quality of the reactor building and reactor components, they are shop-fabricated and transported to a site. Taking the advantage of small-size and lightweight design, the reactor building with major components like steam generators can be transported by barge. The transportability offers the advantage of a short on-site construction period.

Finally, the 4S is a reactor without on-site refuelling designed to operate for 30 years without reloading or shuffling of fuel in the core.

XIV-1.4. Summary of major design and operating characteristics

Some major design and operating characteristics of the 4S are given in Table XIV-1.

The 4S is a sodium-cooled reactor; therefore, its neutron spectrum is fast. However, the 4S is not a breeder reactor since blanket fuel, usually consisting of depleted uranium located around the core to absorb leakage neutrons from the core to achieve breeding of fissile materials, is not provided in its basic design.

The 4S is a reactor without on-site refuelling in which the core has a lifetime of approximately thirty years. The movable reflector surrounding the core gradually moves, compensating the burn-up reactivity loss over the thirty-year lifetime.

The reactor power can be controlled by the water/steam system without affecting the core operation directly. The capability of power self-adjustment makes the reactor applicable for a load follow operation mode.

TABLE XIV-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF 4S

ATTRIBUTES	DESIGN PARTICULARS	
Thermal rating	30 MW(th)	135 MW(th)
Electric output	10 MW(e)*1	50 MW(e)*1
Mode of operation	Base load or load follow	
Load factor/ availability (targets)	> 95 %	
Reactor type	Pool type (integral type)	
Fuel material	Metal fuel (U-Zr alloy) based on enriched uranium	
Coolant	Sodium	
Neutron energy spectrum	Fast	
Core and fuel lifetime	30 years (no refuelling during the whole lifetime)	
Reactivity control system	Axially movable reflectors / Fixed absorber	
Reflector type	Cylindrical type; divided into 6 sectors	
Primary shutdown system	Axially movable reflectors of 6 sectors	
Back-up shutdown system	A single ultimate shutdown rod	
Inherent shutdown system	Inherent characteristics based on reactivity feedbacks	
Type of primary pump	Two electromagnetic (EM) pumps in series	
Reactor vessel diameter	Approximately 2.8 m	Approximately 3.6m
Shutdown heat removal system (1)	Reactor vessel auxiliary cooling system (RVACS)	
Shutdown heat removal system (2)	Intermediate reactor auxiliary cooling system (IRACS)	Primary reactor auxiliary cooling system (PRACS)
Boundary for primary sodium	Double boundary: reactor vessel (RV) and guard vessel (GV)	
Containment system	GV and top dome	
Secondary cooling system	One sodium loop: heat transport from intermediate heat exchanger (IHX) to steam generator (SG)	
Type of secondary pump	EM pump	
Number of steam generators (SGs)	1	
Type of SG	Helical type	
Type of tubes in SG	Double wall tubes with leak detection system	

* In the case when all thermal output is used for electricity generation in the balance of plant (BOP).

A vertical cross-section of the 4S is shown in Fig. XIV-1; a simplified schematic diagram of the 4S based electric power plant is given in Fig. XIV-2. Although the 4S has two designs, those of 10 MW(e) and 50 MW(e), both of these figures show the 10 MW(e) design.

The reactor is a pool type (integral type) as all primary components are installed inside the reactor vessel (RV). Major primary components are the IHX, primary EM pumps, moveable reflectors which form a primary reactivity control system, the ultimate shutdown rod which is a back-up shutdown system, radial shielding assemblies, core support plate, coolant inlet modules and fuel subassemblies.

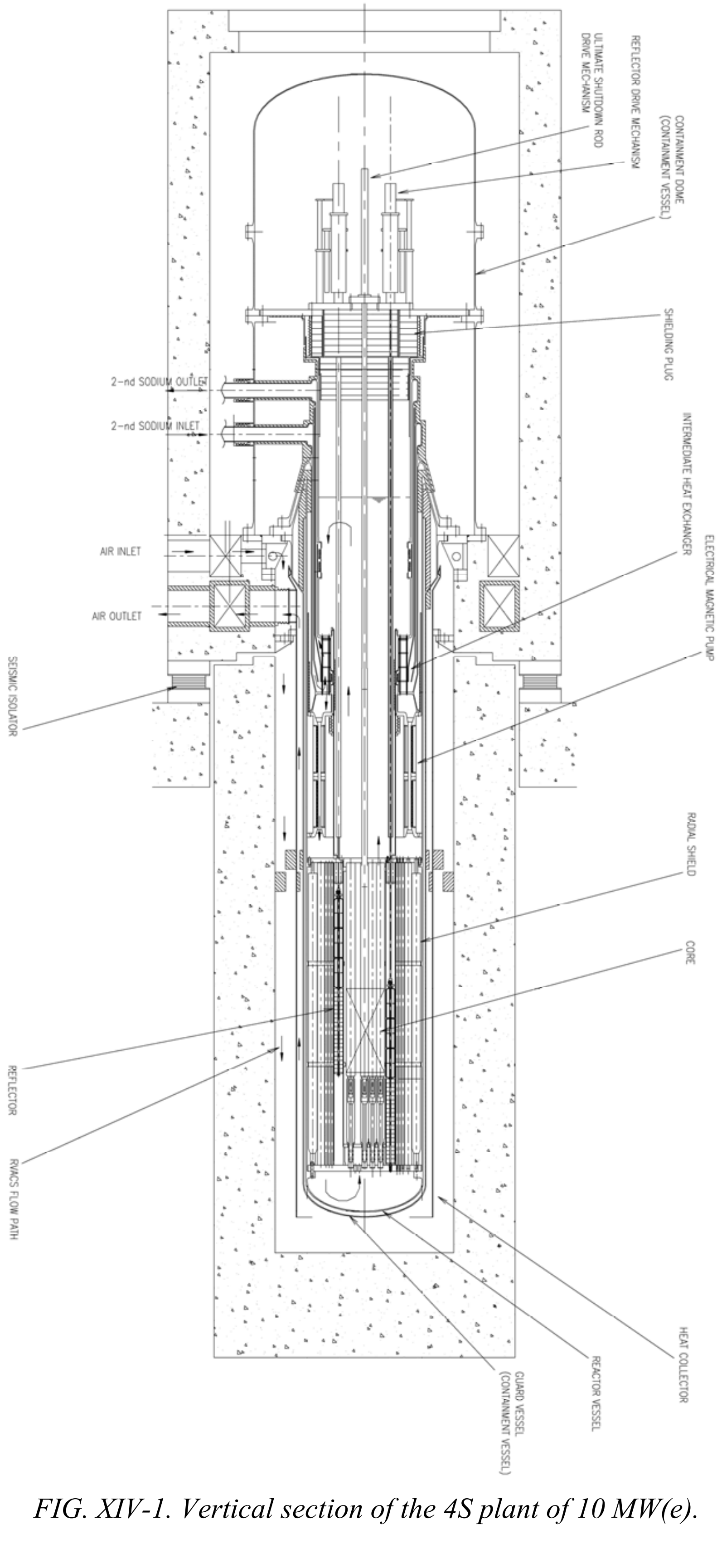


FIG. XIV-1. Vertical section of the 4S plant of 10 MW(e).

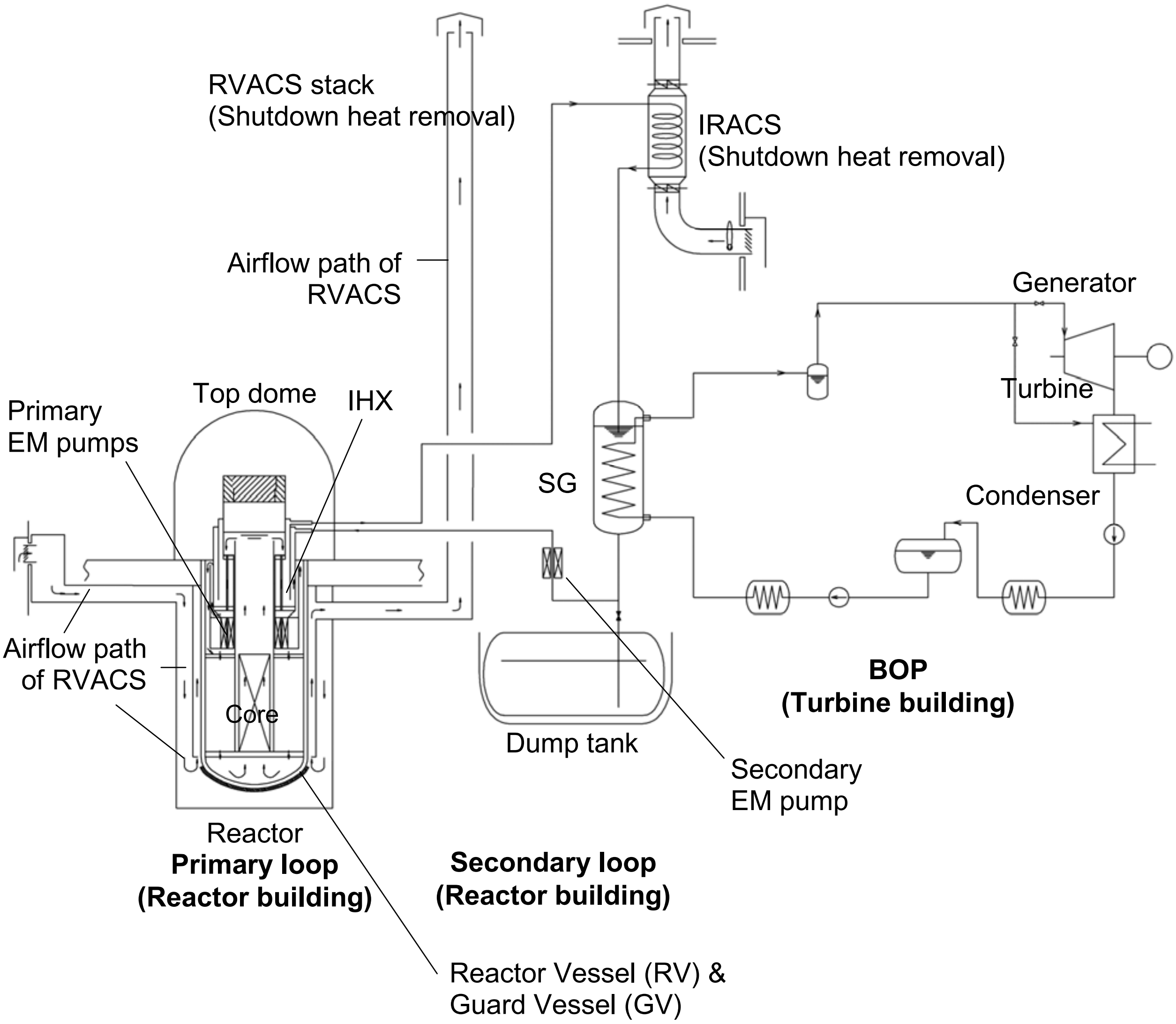


FIG. XIV-2. Simplified schematic diagram of the 4S plant of 10 MW(e).

The IHX is located at the upper position inside the RV. Heat produced in the core is transferred from the primary sodium to the secondary sodium at the IHX. The primary EM pump system, located beneath the IHX, consists of two units arranged in series to insure redundancy for the circulation capability of primary sodium in case of one pump failure. Each EM pump unit produces a half of the head needed to circulate sodium in the reactor primary coolant system. A shielding plug seals the RV at the top. The cover gas (argon) fills the region between the surface of the primary sodium and the bottom of the shielding plug. The guard vessel (GV) provides a second boundary for the primary sodium at the outer side of the RV. The containment system consists of the GV and the top dome, which covers the upper region of the RV, a shielding plug and the equipment located on the shielding plug. Horizontal seismic isolators are adopted for the reactor building.

The primary sodium circulates from the EM pumps downward, driven by pump pressure, and flows through radial shielding assemblies located in the region between the RV and the cylindrical dividing wall. The coolant flow changes its direction at the bottom of the RV and then goes upward, mainly into the fuel subassemblies and partly into the movable reflectors.

The coolant flow is distributed appropriately to fuel subassemblies of each type and to the movable reflectors. Here, the guide wall separates the core and reflector regions. Heat produced in the core is transferred to the coolant while it flows through the fuel pin bundles. Reflectors are also cooled so that the temperature becomes sufficiently low and the temperature distribution is flattened to maintain integrity through 30 years. The coolant gathers at the hot plenum after flowing through the fuel subassemblies and the reflectors. The heated primary sodium then goes into the IHX to transfer heat to the secondary sodium.

The secondary sodium loop acts as an intermediate heat transport system and consists of the IHX, piping, dump tank, EM pump, and SG. Secondary sodium coolant heated in the IHX flows inside the piping to the SG where heat is transferred to water/steam of the power circuit to be supplied to the steam turbine generator.

The heat transfer tubes of the SG are double wall tubes. Between the inner and outer tube, wire meshes are provided, which are filled with helium and act as a detection system for a one side tube failure.

For heat removal from a shutdown reactor, two independent passive systems are provided, which are the reactor vessel auxiliary cooling system (RVACS) and the intermediate reactor auxiliary cooling system (IRACS). The RVACS is completely passive and removes shutdown heat from the surfaces of the guard vessel using natural circulation of air. There is no valve, vane, or damper in the flow path of the air; therefore, the RVACS is always in operation, even when the reactor operates at rated power. Two stacks are provided to obtain a sufficient draft.

The IRACS removes shutdown heat via the secondary sodium. In normal shutdown, heat is removed by forced sodium circulation and natural air convection with normal electric power supply; the IRACS can also remove the required amount of heat solely through natural circulation of both air and sodium in case of postulated accidents.

Figure XIV-3 shows a general view of the 4S reactor for a 50 MW(e) plant; although the size and dimensions differ from those of the reactor for a 10 MW(e) plant, nearly all basic concepts are the same, except that the primary reactor cooling system (PRACS) is used in the 50 MW(e) design instead of the IRACS in the 10 MW(e) design.

Figure XIV-4 gives a general view of the 4S core.

The neutronic design of the 4S has been optimized to achieve the following design targets:

- Improvement of the public acceptance, regulations, policies and safety: all reactivity coefficients by temperature and sodium void reactivity of the core are negative;
- Minimization of fuel cost and operation and maintenance (O&M) cost; ensuring enhanced proliferation resistance (fuel costs are affected by the burden of fuel transport and storage problems in rural areas): no refuelling incurred during the whole 30-year core lifetime,
- Ensuring public acceptance; taking into account certain political circumstances such as non-proliferation regime and early deployment option: use of uranium fuel with the enrichment by ^{235}U less than 20% (by weight) ;
- Minimization of fuel cost and securing fuel integrity under long-life operation of the core: adequate fuel burn-up;
- Minimization of construction costs: reduction of core size.

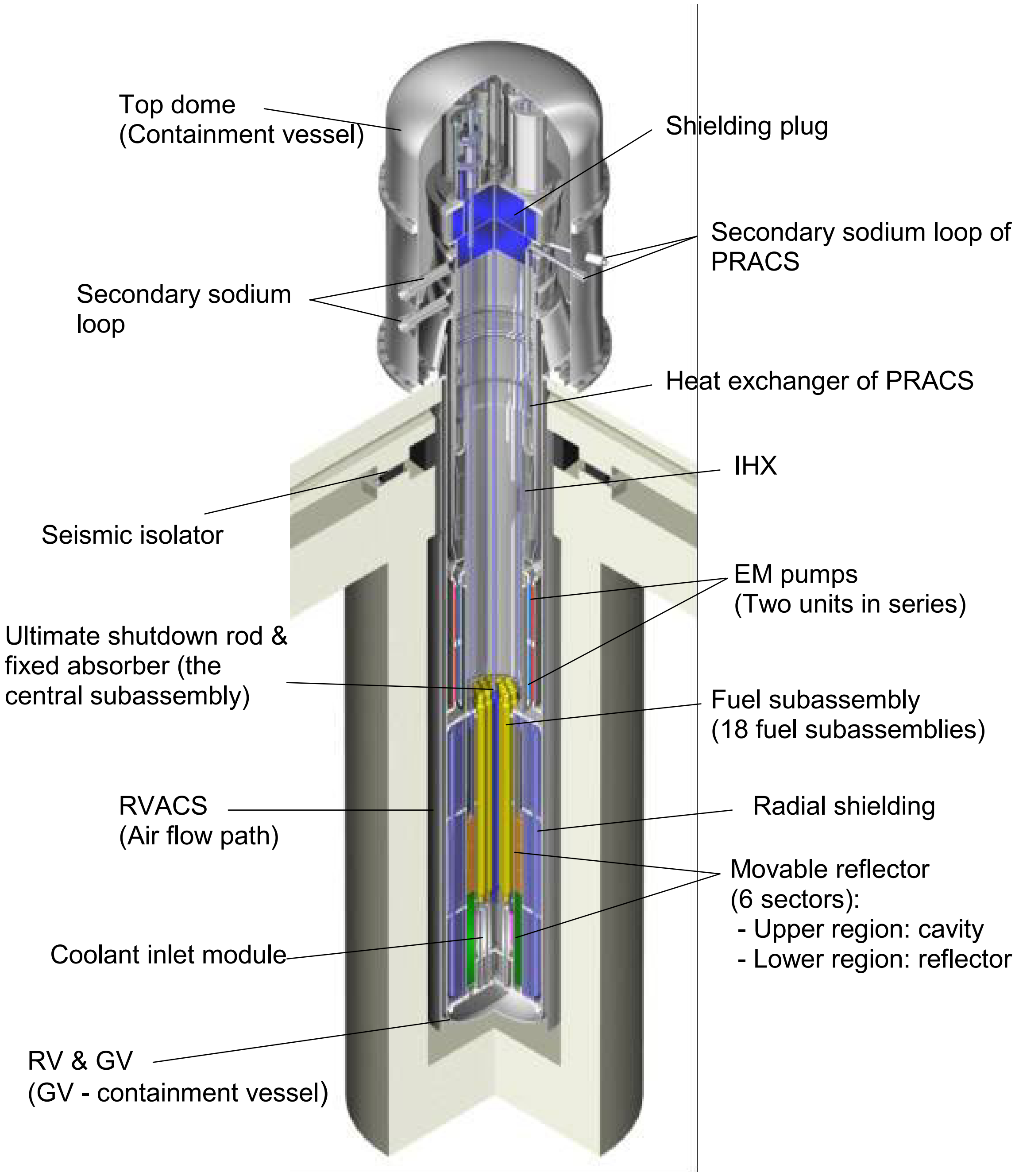


FIG. XIV-3. General view of the 4S reactor for a 50 MW(e) plant.

The above mentioned design targets were defined after deliberations regarding the actual needs or demands at each site in rural areas and taking into account the factors of acceptability to the public, early deployment option, regulation policies, and (international) political circumstances including non-proliferation, cost competitiveness, etc.

A summary of the neutron-physical characteristics of the 4S reactor is provided in Table XIV-2.

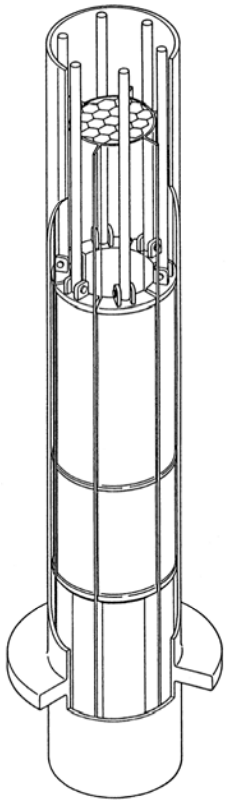


FIG. XIV-4. General view of the 4S core.

TABLE XIV-2. NEUTRON-PHYSICAL CHARACTERISTICS OF THE 4S

Electric output, MW(e)	10	50
Number of uranium enrichment zones	2 (inner / outer core)	2 (inner / outer core)
Uranium enrichment (% by weight)	17.0 / 19.0	12.0 / 18.0
Average linear heat rate (W/cm)	39	110
Conversion ratio	0.45	0.53
Average burn-up (GW·day/t)	34	90
Burn-up reactivity swing (% dk/kk’)	5.5	10
Coolant void reactivity (% dk/kk’)	-0.4	0

The burn-up reactivity swing is compensated by axially movable reflectors; electromagnetic impulsive force (EMI) is applied in the driving mechanism of the reflectors. In an EMI system, the inertia of the reflector is the force behind the mechanics; an EMI unit is provided for each of the six reflectors [XIV-2 to XIV-4]. While an EMI technology will be developed for 4S, a combined system of ball screw and hydraulic mechanism as a developed system might be adopted for a reflector drive system in an initial phase of 4S deployments.

The drive mechanism of the reflectors carries them upward to conform to the predicted or pre-adjusted curve to give the core a constant reactivity-worth (Fig. XIV-5).

A mismatch between reactivity added by the reflectors and the reactivity lost via fuel burn-up is adjusted by the feedwater control of the water/steam system. Therefore, the reactivity control is unnecessary at a reactor side and this is an important factor to simplify the reactor operation.

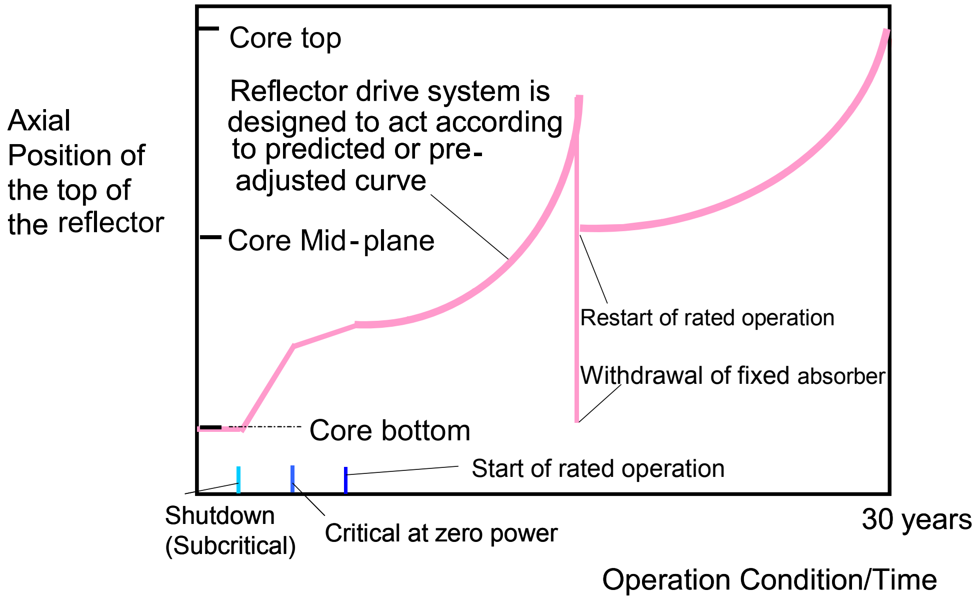


FIG. XIV-5. Axial position of the top of the reflector versus operation time.

In addition to the inherent safety features, there are two independent systems for reactor shutdown. The primary shutdown system provides for a drop of several sectors of the reflector, and the back-up shutdown system provides for insertion of the ultimate shutdown rod, located as a central subassembly on a stand-by in a fully “out” condition.

The 4S is sodium-cooled reactor; therefore, an intermediate heat transport system is employed to avoid a reaction between the primary (radioactive) sodium and water/steam of the power circuit. The 4S has three heat transport systems: the primary sodium system located inside the RV, the secondary sodium system in which sodium is sufficiently non-radioactive to define it as an “uncontrolled area”, and the water/steam turbine system.

The thermodynamic efficiency is approximately 33% for the 30 MW(th) plant and 37% for the 135 MW(th) plant.

The main thermal-hydraulic characteristics of the 4S are shown in Table XIV-3.

The 4S operation without on-site refuelling is one of the keystones for the reactor application in rural areas, for a variety of reasons. The core and fuel lifetime as well as the plant lifetime would be approximately 30 years; the fuel in the 4S does not need to be reloaded or shuffled during the plant lifetime. The fuel is just installed when the 4S is constructed at a site. Therefore, the concept of “annual flow of fuel and non-fuel materials” is of somewhat limited meaning for the 4S.

The material balances for the 4S are given in Table XIV-4. The major part (more than 95%) of the discharged minor actinides (MA) is neptunium.

The design lifetime of the core and fuel as well as the reactor vessel and components is 30 years. The reactor building including the concrete silo can be used for more than 60 years.

TABLE XIV-3. THERMAL-HYDRAULIC CHARACTERISTICS OF THE 4S

Electric output, MW(e)	10	50
Primary circulation: - Normal operation - Unprotected loss of flow (ULOF)	Forced circulation (two EM pumps in series) Flow coastdown with synchronous motor systems, then natural circulation	The same as for the 10 MW(e) plant
Primary coolant system: - Coolant temperature - Pressure - Pressure loss in the fuel subassembly	355 / 510°C (inlet / outlet). Non-pressurized Less than 0.1 MPa	The same as for the 10 MW(e) plant Less than 0.2 MPa
Maximum temperature of fuel cladding	650°C (hot spot)	The same as for the 10 MW(e) plant
Secondary cooling system: - Coolant temperature - Pressure	310 / 485°C Non-pressurized (slightly higher pressure than in the primary system)	The same as for the 10 MW(e) plant
Steam/water system: - Coolant temperature - Pressure	210 / 453°C 10.5 MPa	The same as for the 10 MW(e) plant

TABLE XIV-4. MASS BALANCES OF FUEL MATERIALS FOR THE 4S

Electric output, MW(e)	10	50
Heavy metal (U) inventory, tons	9.23	16.2
Fissile (^{235}U) inventory, tons	1.69	2.58
Average annual flow* of: - Heavy metals (U), kg/year - Fissile materials (^{235}U), kg/year	 308 56	 539 86
Average annual flow per MW(e)* of: - Heavy metals (U), kg/year/MW(e) - Fissile materials (^{235}U), kg/year/MW(e)	 31 6	 11 2
Average burn-up of discharged fuel, GW day/t	34	90
Inventory of materials discharged after 30 years, tons: - Heavy metals total - U - ^{235}U - Pu - MA, kg	 8.90 8.75 1.36 0.15 2	 14.7 14.1 1.36 0.65 17
Natural uranium requirements**: - For fabrication of fresh enriched uranium fuel load, tons - Average specific flow of natural uranium, kg/year/MW(e)	 320–400 1070–1320	 500–620 330–410

* Total inventory is divided by 30-years.
** It is assumed that ^{235}U content in depleted uranium is in the range between 0.2% and 0.3% (by weight); reprocessing/recycle of fissile materials remaining in the fuel is not taken into consideration in this calculation.

Two kinds of systems for non-electric applications have been incorporated in the 4S; they are:

- Seawater desalination system; and
- Hydrogen and oxygen production system.

Combinations of these systems and the turbine generator system as balance of plant (BOP), including the capacity of each system, would be determined to meet the actual needs at each site.

To be a viable option for power generation in remote areas, the 4S must provide competitive cost of electric power determined as busbar cost. "Busbar cost" is that required to generate a kilowatt-hour of electricity as measured at the plant busbar, i.e., the conducting boundary in the plant where the generated electricity is transferred to the external grid.

A preliminary effort to estimate the 4S busbar cost has been conducted under the following assumptions:

- A levelling period of 30-years;
- An assumed construction period of 12 months under normal site conditions;
- An assumed house load factor of 8% for the 4S plant operation;
- A mass production phase, i.e., N^{th}-of-a-kind plant.

Preliminary cost estimations for the 4S show its competitiveness compared to the SMR costs estimation devised by the US DOE, which are in the range of US$5.4 cents per kW-h and US$10.7 cents per kW-h at the 50 MW(e) size and in the range of US$10.4 cents per kW-h and US$24.3 cents per kW-h at the 10 MW(e) size [XIV-5].

XIV-1.5. Outline of fuel cycle options

A metal uranium fuel is used for the 4S. Viewed from the current situation regarding the capacity of actual reprocessing facilities for metal fuel, in the first phase of the 4S spent fuel would be stored/cooled and then preserved geologically in medium or long-term storage. In other words, a once-through fuel cycle is assumed for the first phase of the 4S.

In the next phase, spent fuel from the 4S or other reactors including LWRs could be reprocessed using pyro-process technology developed at the Argonne National Laboratory (ANL, USA) and/or CRIEPI (Japan). In this phase, plutonium and MA recovered from spent fuel could be used as fresh fuel for the 4S. Here, a centralized reprocessing plant would be preferable for the 4S because each 4S plant is a small distributed power station and a collocated reprocessing like in the IFR seems inappropriate for this type of power stations. To put it short, in the next phase, the 4S would be operated in a closed nuclear fuel cycle.

The 4S can be configured for a variety of alternative fuel cycle options to meet actual demands of its users. These include a plutonium or TRU burner option using a metal fuel such as a U-Pu-Zr alloy or using inert materials to avoid further production of plutonium from the installed 238U [XIV-6, XIV-7].

XIV-1.6. Technical features and technological approaches that are definitive for 4S performance in particular areas

XIV-1.6.1. Economics and maintainability

The inherent and passive safety features, the operation without on-site refuelling, lower projected maintenance and operating requirements, plant transportability in construction scheme and lower busbar costs of the 4S could facilitate its deployment in developing countries with limited technological resources. However, as the first step, the 4S design should be approved and certified for production in series by a reliable regulatory body in a developed country.

The main design features of the 4S supporting a reduction of its capital cost is as follows:

- Reduced volume and weight of materials achieved by the use of simple systems and structures (the concepts of simple operation, simple inspection and strong reliance on inherent safety features are supported by the use of simple systems and structures);
- Passive principles of reactor operation; the operation of almost all systems of the 4S is based on natural phenomena, taking the advantage of small reactor size;
- Shop fabrication and transportability of the reactor building including the SG and the reactor, resulting in a reduced site construction load and a shorter construction period of approximately 12 months.

For example, the absence of a necessity of fuel reloading and shuffling (for a period of 30 years) eliminates the need of a permanent fuel handling system, which could be substituted by a demountable temporary system, which would be shared among several 4S plants. In the 4S, the control of feedwater at the BOP side can ensure the control of reactor power through changes in coolant temperature and the associated reactivity feedbacks in the core. Therefore, the 4S reactor has no control rods, drive mechanisms or upper internal structure (UIS). The RVACS is a completely passive system using air naturally circulated around the guard vessel and is a final heat sink in one of the shutdown heat removal systems. In heating, ventilating, and air conditioning (HVAC), there is no need to use systems with seawater as an ultimate heat sink; a system of heat release to the air will be sufficient for this purpose because of the small thermal output of the plant.

The 4S is being designed to operate safely without active involvement of the plant operators. The design features to support such operation are as follows:

- Burn-up reactivity swing is automatically compensated by the fine motion reflectors;
- Reactor power can be controlled automatically by the feedwater flow rate,
- There is no need in reloading and shuffling of fuel in the course of 30 years;
- A reduction in maintenance requirements achieved by adopting static devices such as EM pumps or static devices continuously monitored by simple systems;
- Reduction of in-service inspections (ISI) achieved by taking advantage of the non-pressurized systems of a sodium cooled reactor and by applying a “continuous monitoring” process based on “leak before break (LBB)” detection to ensure safety of the 4S.

During the 4S operation, the operation personnel are required only for monitoring or checking. There might be a possibility of reducing a security effort because of the earth-sheltered embedded plant (Fig. XIV-6). If this would be authorized through discussion with the regulatory side, the O&M costs could be further reduced from the current estimation.

Burn-up of the discharged fuel directly influences fuel cost. In the 4S, fuel costs account for approximately 27% of the busbar cost for a 50 MW(e) plant and 15% of the busbar cost for a 10 MW(e) plant. The burn-up of the discharged fuel is higher than in typical LWRs (see Table XIV-4).

XIV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

In the next phase of the 4S, when recovered plutonium and minor actinides (MA) would become politically and commercially available because of the shortage of natural fissile materials, fresh fuel consisting of the reprocessed fissile materials and depleted or natural uranium could be installed in the 4S. A fast neutron spectrum of the 4S avoids the degradation of fissile materials through burn-up; therefore, the recovery process for the spent fuel of the 4S could be repeated many more times than for LWRs, resulting in a higher degree of natural uranium utilization.

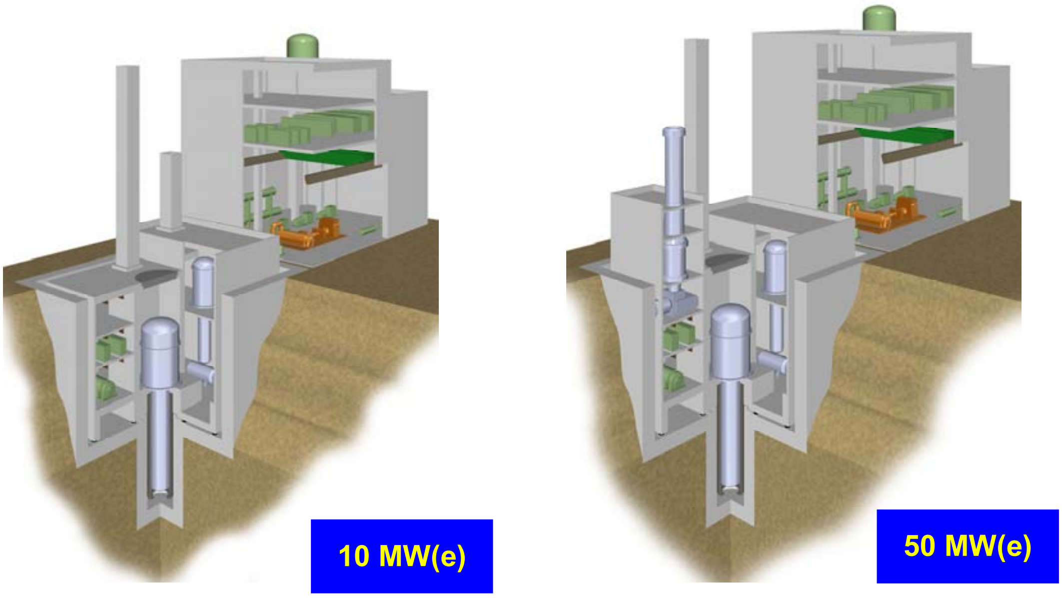


FIG. XIV-6. Earth-sheltered reactors of the 4S plants.

Radioactive waste is mainly generated through the cleanup of equipment or those reactor internals that are used in the primary sodium (with such cleanup being performed for maintenance, repair or exchange). The absence of refuelling during 30 years and the resulting reduced maintenance requirements for a sealed reactor would facilitate a considerable reduction in the radioactive gas, liquid and solid wastes.

XIV-1.6.3. Safety and reliability

Safety concept and design philosophy

The philosophy behind the 4S safety concepts is to put an emphasis on simplicity achieved by strong reliance on passive and inherent safety features as a major part of the defence in depth strategy. The ultimate objective of the 4S safety concept is to eliminate the requirement of evacuation as an emergency response measure. The 4S safety concept provides for three functions to be shouldered by the defence in depth in each phase of the abnormal operation or an accident; these three functions are the following:

- Prevention;
- Mitigation;
- Confinement of radioactive material.

Provisions for simplicity and robustness of the design

Incorporation of several passive and inherent safety features, such as low power density in the core, good thermal characteristics of the metal fuel bonded by sodium, negative reactivity coefficients by temperature, passive shutdown heat removal by both natural circulation of the coolant and natural air draft, and a large coolant inventory are some important provisions for simplicity and robustness of the 4S design.

Active and passive systems and inherent safety features

The active and passive systems and inherent safety features of the 4S are applied with the following main objectives:

- To reduce the probability of component failure; the inherent features of the 4S design supporting such a reduction are the following:
  - By-design elimination of active systems and feedback control systems from the reactor side;
  - By-design elimination of components with rotating parts (use of static devices such as EM pumps);
  - By-design limitation of the radioactivity confinement area (no refuelling during the whole reactor lifetime and no systems relevant for fuel reloading or shuffling);
- To prevent core damage in accidents; the active and passive systems and inherent safety features of the 4S supporting such prevention are the following:
  - Two independent active shutdown systems, including:
    - The actively-initiated drop of several sectors of the reflector;
    - Active insertion of the ultimate shutdown rod;
  - Enhancement of inherent safety features via the use of metal fuel in the core (lower accumulated enthalpy of fuel);
  - All-negative reactivity coefficients by temperature (an inherent safety feature);
  - A higher capability for natural circulation of sodium after a pump trip enabled by low pressure loss in the fuel subassemblies and a simple flow path inside the reactor (an inherent safety feature);
  - Two fully passive shutdown heat removal systems, including:
    - RVACS, based on natural circulation of primary sodium and natural air draft around the guard vessel; and
    - IRACS, based on natural circulation of secondary sodium and natural air draft through the air heat exchanger;
  - A large inventory of primary sodium (an inherent safety feature);
- To confine the radioactive materials; the design features of the 4S supporting this objective are as follows:
  - Multiple barriers against fission product release, including:
    - The fuel cladding;
    - The reactor vessel, upper plug and the IHX tubes;
    - The top dome and the guard vessel as a containment;
  - The small radioactive inventory typical of a small sized power reactor;

- To prevent sodium leakage and to mitigate the associated impact if it occurs; the design features of the 4S supporting this objective are the following:
  - A by-design double boundary for sodium in the primary system and in the important parts of the secondary system with a detection system for small leakages occurring via a one-boundary failure, including:
    - The reactor vessel and guard vessel boundary for primary sodium;
    - The heat transfer tubes have double walls in both the SG and the air cooler of IRACS;
  - A passive sodium drain system from the SG to the dump tank; if a sodium-water reaction occurs, an increase in cover gas pressure in the SG would cause disk rupture and make secondary sodium to drain rapidly to the dump tank located beneath the SG.

Structure of the defence in depth

Some major highlights of the 4S design and systems, structures and components corresponding to various levels of the defence in depth are brought out as follows:

Level 1: Prevention of abnormal operation and failure:

(A) Prevention of loss of coolant:

- Double boundaries for primary and secondary sodium in SG tubes and leak detection systems of continuous operation;

(B) Prevention of loss of flow:

- Primary EM pumps are arranged in two units connected in series where each single unit takes on one half of the pump head;
- A combined system of the EM pumps and the synchronous motor systems (SM) ensures a sufficient flow coastdown characteristics;

(C) Prevention of transient overpower:

- Elimination of feedback control of the movable reflectors,
  - A pre-programmed reflector-drive system, which drives the reflector without feedback signals;
  - The moving speed of the reflector is approximately 1mm/week;
- The limitation of high-speed reactivity insertion by adopting the electromagnetic impulsive force (EMI) as a reflector driving system;
- The limitation of reactivity insertion at the start-up of reactor operation;

(D) Prevention of sodium-water reaction:

- A leak detection system in the heat transfer tubes of the SG using wire meshes and helium gas, capable of detecting both:
  - An inner tube failure (water/steam side of the boundary); and
  - An outer tube failure (secondary sodium side of the boundary).

Level 2: Control of accidents within the design basis.

The design features of the 4S supporting Level 2 of the defence in depth are as follows:

- Increased reliability of the reactor shutdown systems achieved by the use of two independent systems with each of them having enough reactivity for a shutdown, including:
  - The drop of several sectors of the reflector;
  - Insertion of the ultimate shutdown rod;
- Increased reliability of the shutdown heat removal systems achieved by the use of two passive systems based on natural convection;
- Increased reliability of the sodium-leakage prevention systems achieved by the use of double-wall SG tubes with detection systems for both inner and outer tubes.

Level 3: Control of severe plant conditions, including prevention of accident progress and mitigation of the consequences.

The design features of the 4S supporting Level 3 of the defence in depth are as follows:

- Inherent safety features of a metal fuelled core, such as excellent thermal conductivity and low accumulated enthalpy;
- All-negative reactivity coefficients by temperature;
- The fully passive shutdown heat removal system (RVACS) based on natural air draft and natural circulation of sodium;
- Large inventory of primary sodium to meet the requirements for increased grace periods;
- The rapid system of sodium drain from the SG to the dump tank as a mitigation system for sodium-water reaction.

Level 4: Mitigation of radiological consequence of significant release of radioactive materials.

The inherent and passive safety features of the 4S are capable to eliminate an occurrence of fuel melting in any accident without scram (AWS) or anticipated transient without scram (ATWS).

A preliminary evaluation has been conducted where failure of all fuel element claddings (approximately 5 000 fuel pins) was hypothetically assumed to calculate site suitability source term (SSST). The status of major nuclides defining the source term and their behaviour are as follows:

- Plutonium (Pu) is retained in the metal fuel slug because fuel melting never occurs;
- Caesium (Cs) could be solidified and retained in a lower temperature area using a leakage path from the coolant to the reactor vessel, including the upper plug and the IHX, and then to the containment;
- Iodine (I) is retained in the sodium coolant within NaI compound because fuel melting never occurs; therefore, iodine migration does not occur also.

It was assumed that 100% of the noble gases including krypton and xenon are released from the sodium coolant to the cover gas. Further migration of noble gases was considered as follows:

- At a leak rate of 0.02%/day from cover gas through the reactor vessel, upper plug and IHX and then to the top dome, during 30 days;

- At a leak rate of 1%/day from the top dome to the reactor building;
- Noble gases in the reactor building were assumed to be released off-site.

The analytical results obtained show that the dose equivalent in this case is 0.01 Sv at a distance of 20 m from the reactor. It means that only 20 meters are required as a site boundary for the 4S.

Design basis accidents and beyond design basis accidents

A major objective of the 4S design is to ensure the capability of withstanding a wide range of postulated events without exceeding the specified temperatures of fuel, cladding, and coolant boundaries, thereby maintaining the fuel pin and coolant boundary integrity. For the safety analysis of the 4S, design basis events (DBEs) have been selected and identified systematically with consideration of the 4S operation cycle and the events postulated for MONJU, DFBR (Japan), and LWRs. A broad variety of events have been considered in the following categories:

- Power transients;
- Loss of flow;
- Local fault;
- Sodium leakage;
- Balance of plant (BOP) failure and loss of off-site power;
- Multiple systems failure.

For the safety analysis of the 4S, beyond design basis events (BDBEs) have been selected and identified in a similar manner. The criteria for anticipated transients without scram (ATWS) and accidents without scram (AWS) are as follows:

- ATWS events:
  - Maximum CDF (Cumulative Damage Fraction) less than 0.5;
  - Maximum fuel temperature lower than the melting point;
  - The coolant boundary limit does not exceed the service level D in ASME
- AWS events:
  - Maximum coolant temperature lower than the boiling point;
  - Maximum fuel temperature lower than the melting point;
  - The coolant boundary limit does not exceed the service level D in ASME standards.

Some analytical results for dominant and severe events in the 4S are summarized below.

(A) LOSS OF SITE POWER WITHOUT SCRAM (ATWS EVENT)

This scenario is typically called an unprotected loss of flow (ULOF).

EM pumps of both primary units trip with the flow coastdown facilitated by the synchronous motor (SM) system; then the fuel and coolant temperatures rise because of the coolant flow decrease. However, the reactor power is decreased via the negative feedbacks resulting from Doppler, fuel expansion, and steel and coolant reactivity coefficients. After the flow coastdown by the SM systems, the primary coolant circulates within the reactor vessel, driven by natural convection, with a flow rate of approximately 20% of the nominal. Then, the inherently decreased power and the convection flow rate are balanced to a steady state.

As a consequence, peak temperature of the nominal hottest fuel element cladding reaches 740°C and then settles at lower than 580°C. The CDF is less than 0.5 and the claddings do not fail, Fig. XIV-7.

(B) SUDDEN LOSS OF HEAD IN ALL PRIMARY PUMPS WITHOUT SCRAM (AWS EVENT)

This scenario is more severe than ULOF described in (A) and, therefore, it is categorized as an AWS. Primary flow is suddenly lost because of the dielectric breakdown in the EM pumps. Even though there are two EM pump units arranged in series, both of them are assumed to suddenly fail in this scenario. A flow rate of approximately 20% of the nominal is assured by natural circulation, and a temperature rise in the fuel, coolant and steel produces negative feedbacks; then, the power decreases. The rate of temperature rise is lower than in typical fast breeder reactors (FBRs) because the power density is lower in the 4S. The inherently decreased power and the convection flow rate are balanced into a steady state. As a consequence, neither coolant boiling nor fuel melting occurs.

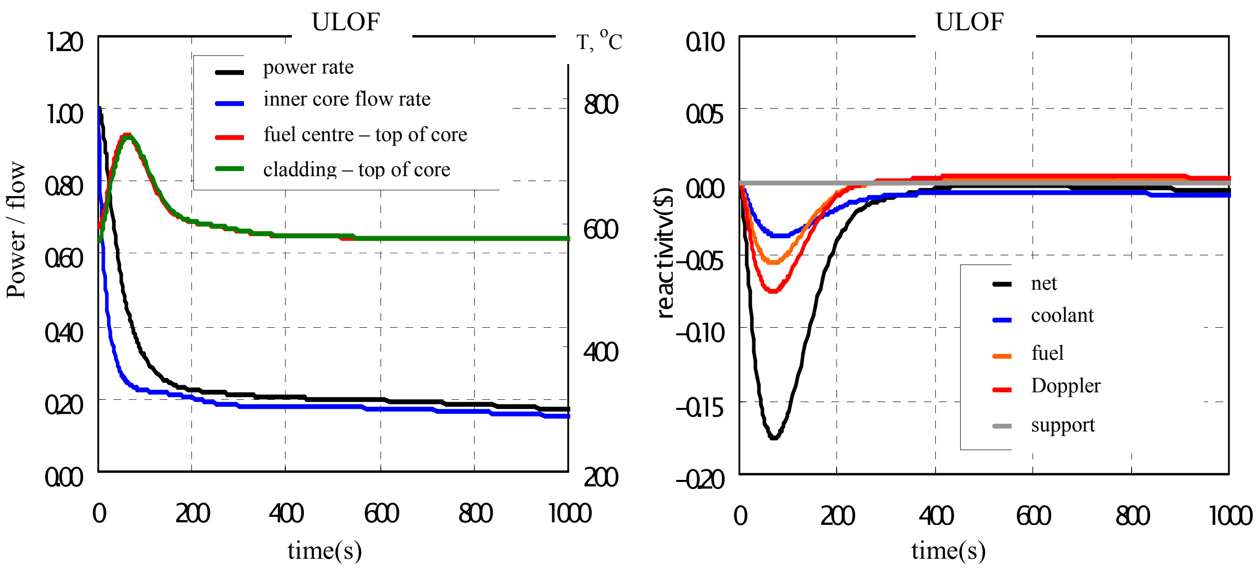


FIG. XIV-7. ULOF event (loss of site power without scram).

(C) FAILURE OF THE REFLECTOR DRIVE SYSTEM IN RATED POWER OPERATION WITHOUT SCRAM (ATWS EVENT)

This scenario is typically called an unprotected transient overpower (UTOP).

The functions of the reflector drive system in rated power operation are limited to compensation of the burn-up reactivity swing, i.e. the requested reactivity insertion speed is very slow. The electromagnetic impulsive force (EMI) mechanism of the reflector drive is designed to provide a reactivity insertion rate of 0.00035 cents/s at maximum, even if the system fails. Even though the reflectors are divided into six sectors and each sector has its own drive system, it was assumed that all of the sectors fail and move upward altogether.

The reactor power increases very slowly and the coolant temperature follows it and rises also slowly. Even after 12 hours, the maximum cladding temperature does not exceed 700°C and the fuel pins do not fail. Enough time can be provided to shutdown the core manually before the cladding failure.

(D) FAILURE OF THE REFLECTOR DRIVE SYSTEM IN A START-UP WITHOUT SCRAM (AWS EVENT)

The functions of the reflector drive system during the 4S start-up are limited to compensation of a reactivity resulting from the temperature rise between cold stand-by (critical) and rated power reactor states. Even though the reflector operations occur as only one sector is manually lifted up using separate drive systems, it was assumed that all of the sectors fail and move upward simultaneously (a more severe condition). Therefore, this scenario was categorized as an AWS, because of a more hypothetical assumption.

The analytical conditions were 0.01 cents/s as an insertion rate and a total of 75 cents as the upper limit of reactivity insertion, in this calculation defined by the total worth of all reflectors for power reactivity compensation from 25% to full power; the actual value could be even smaller than 75 cents. The progression of this scenario is illustrated in Fig. XIV-8.

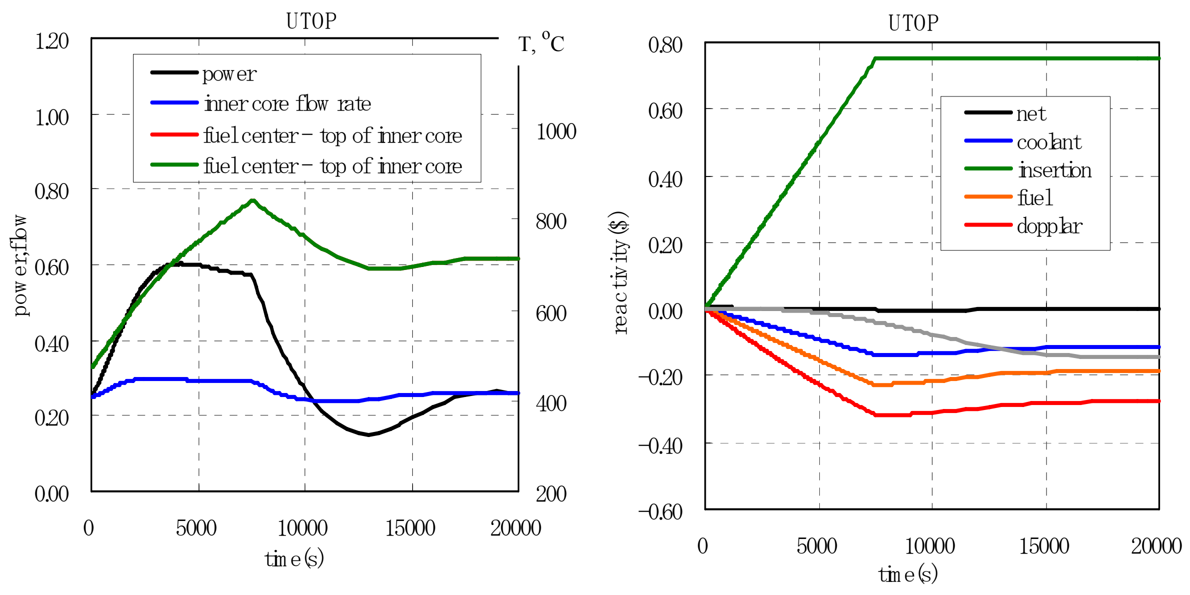


FIG. XIV-8. Failure of reflector drive system in a start-up operation without scram.

After the power increase due to the inserted reactivity, the system temperature starts rising, resulting in negative feedbacks. The negative feedbacks remain in place as core inlet temperatures begin to rise; the power decreases to the level of the initial value and the coolant does not boil.

(E) FAILURE OF IRACS AND RVACS WITH THE COLLAPSE OF BOTH OF THE TWO STACKS

This postulated event is considered to be a more severe one than the AWS, although the reactor shutdown can be successfully achieved.

In the consideration performed, two more failures were assumed in addition to one transient as an initiator; therefore, this event is more severe than a typical protected loss of heat sink (PLOHS). The analytical conditions were as follows:

- Complete failure of the IRACS;
- In the RVACS, both of the two stacks collapse, and half-blockage of the air flow path occurs in the RVACS but it continues operation with a reduced air draft.

After the reactor shutdown, primary pumps trip with the flow coastdown facilitated by the SM system; then natural circulation begins within the reactor vessel. Coolant temperature in the hot plenum (upper core region) decreases because of the shutdown. Only the RVACS with reduced draft removes decay heat by air convection. Temperatures in the primary boundary (structure) rise very slowly due to the large inventory of primary and secondary sodium and become stable. The maximum temperature of the reactor vessel is less than 520°C, the claddings do not fail, and the structure of the primary boundary is not damaged, Fig. XIV-9.

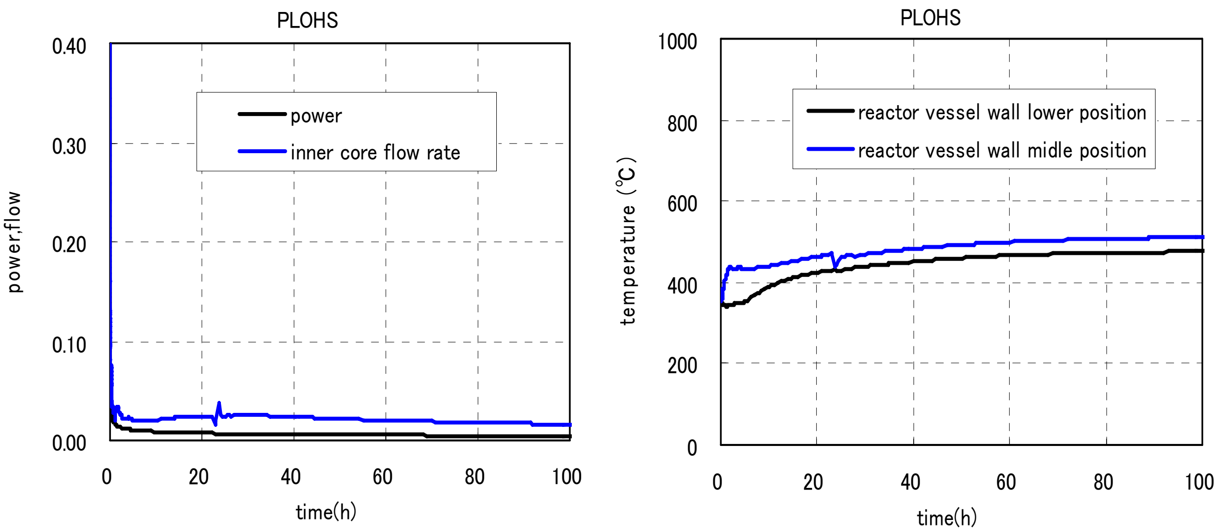


FIG. XIV-9. Failure of IRACS and RVACS with the collapse of both stacks (PLOHS).

In the 4S design, the reactor building is isolated horizontally by seismic isolators. The design standard already exists for such isolators for nuclear power plants (NPPs) in Japan [XIV-8]. The thin reactor shape results in a higher characteristic frequency; therefore, the 4S reactor could be rigid against vertical shock.

For the 4S it has been shown that fuel never melts under hypothetically postulated conditions like ATWS and AWS (both - BDBE). Some fuel pins with maximum cladding temperature might fail in more severe AWS events. However, from the source term calculations it was found that, even if all fuel element claddings fail, the required distance for a site boundary is only 20 meters. This meets one of the most important design objectives of the 4S, which is to enhance the level of safety so as to eliminate the need for public intervention beyond the plant boundaries as a consequence of any postulated accident.

XIV-1.6.4. Proliferation resistance

Technical features of the 4S contributing to a high level of proliferation resistance are:

- Uranium based fresh fuel with the ^{235}U enrichment less than 20% by weight;
- Low plutonium content in the spent fuel, less than 5% by weight;
- The reprocessing technology available for metal (alloy) fuel, such as U-Zr or U-Pu-Zr, ensures that plutonium is always recovered with the accompanying minor actinides, which include highly radioactive and radiotoxic nuclides.

Absence of refuelling during the whole core lifetime and low maintenance requirements resulting from continuous operation of a sealed reactor in the course of 30 years provide a substantial physical protection of nuclear material. There is no opportunity for fuel to come out of the thick reactor vessel except for the period of loading at the beginning of the 30-year lifetime and discharge at the end of the lifetime. The number of fuel subassemblies is small (only 18 subassemblies), which makes it easy to monitor and scrutinize all of the subassemblies.

The 4S is designed in a way that there are no facilities or equipment to discharge the fuel subassemblies, or to disassemble the fuel subassemblies into fuel pins and extract nuclear material from the metal fuel slugs. The fuel-handling machine is temporarily provided to discharge spent fuel subassemblies after 30-year operation and only following adequate cooling inside the reactor. The spent fuel subassemblies are then encased in a cask and transported to a geological storage site (in the first phase) or to a recycle centre (in the next phase). Therefore, it would be difficult to perform an undeclared production of fissile material in the 4S just because there is no facility or apparatus available to enable such production.

As for unauthorized use of the fuel-handling machine, this kind of machine is a temporary system for the 4S and would be shared among several 4S sites. There would be no available machine for fuel assembly handling during the operation.

XIV-1.6.5. Technical features and technological approaches used to facilitate physical protection of 4S

The designers of the 4S consider embedding the whole reactor underground as one of the most natural and substantial methods of physical protection against unauthorized access and external missiles. Other features of the 4S contributing to an enhanced physical protection are as follows:

- No refuelling during the whole reactor lifetime of 30 years;
- The reactor operates completely sealed;
- The operation is automatic without the need of operator actions.

The fundamental concept of the 4S is that of "continuous monitoring" rather than "active operation". The reactor operates using a system of pre-programmed movable reflectors and the power control is executed from the outside, through feedwater flow rate changes in the power circuit. The plant and component conditions and/or unauthorized access could be continuously monitored from outside the site, e.g. by satellite systems.

XIV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of 4S

The operation without on-site refuelling could meet well the concept of fuel leasing and full scope fuel cycle service agreements using a reprocessing centre common for all 4S plants.

Such features of the 4S as operation without on-site refuelling, plant transportability for construction, reduced maintenance requirements and less need in active operator involvement might meet the concept of a NPP leasing.

XIV-1.8. List of enabling technologies relevant to 4S and status of their development

A list of enabling technologies relevant to the 4S and status of their development are given in Table XIV-5.

TABLE XIV-5. ENABLING TECHNOLOGIES RELEVANT TO THE 4S

ITEM/AREA	TECHNOLOGY	STATUS OF DEVELOPMENT
Core	Control of neutron leakage by the reflector	Basic test (critical) has been conducted at Toshiba Critical Assembly (TCA) [XIV-9, XIV-10]. Critical experiments have been conducted at JAERI*/ FCA since 2004.

ITEM/AREA	TECHNOLOGY	STATUS OF DEVELOPMENT
Core	Negative temperature coefficient	Critical experiments have been conducted at JAERI*/ FCA since 2004.
Fuel	Integrity for a long lifetime of 30 years	Sufficient data for the design and licensing of fuel element claddings based on established materials is available. Development of a higher-creep strengthened steel including irradiation tests is in progress at JNC* [XIV-11, XIV-12]
Fuel	Tight lattice of fuel pins with low pressure loss	Hydraulic tests are being conducted at CRIEPI since 2003.
Reflector drive system	Highly reliable and fine motion drive system using electromagnetic impulsive force (EMI)	Mock-up tests including hydraulic tests have been conducted since 2003.
EM Pumps	Annular type EM pumps; two units in series (the 4S-type pumps)	Demonstration of a large-capacity single EM pump unit has been completed [XIV-13]. Demonstration of the 4S-type EM pumps would be beneficial.
SG	Double wall tube SG with leak detection system	Basic characteristics of double wall tube were confirmed in sodium tests at ETEC (USA). Demonstration of leak detection system would be beneficial.
RVACS	Natural circulation of air. Enhancement of heat radiation from steel to the air.	In the USA, ANL has conducted the tests and GE has designed the system for the PRISM reactor. The experiments were also conducted by CRIEPI in Japan.
Seismic isolation	Horizontal seismic isolator	A standard has been established for NPPs in Japan [XIV-8].
Modified 9Cr-1Mo steel	Structural material with higher resistance against irradiation	A standard has been established by ASME, but it needs further elaboration as comes to irradiation characteristics (or new standards need to be established)
Safety	Demonstration of shutdown systems	Design-by-analysis approach is being used based on the data and R&D results of previous sodium-cooled reactors. Demonstration tests would be of benefit.
Safety	Demonstration of inherent safety features	To be performed in the 4S demonstration reactor.
Production of oxygen and hydrogen	High temperature steam electrolysis (HTE) technology	R&D in progress.
Seawater desalination	Two-stage reverse osmosis system.	The technology has been demonstrated.

* Japan Atomic Energy Agency (JAEA) at present.

XIV-1.9. Status of R&D and planned schedule

The 4S design is being developed at TOSHIBA and CRIEPI in Japan. Chubu Electric Power Company supported the initial phase of some R&D relevant to the 4S.

The current R&D focussed on core, fuel and reflector technologies are conducted under the sponsorship of MEXT in Japan.

The design development of the 4S is mainly done by TOSHIBA and CRIEPI.

Members involved in the current R&D sponsored by MEXT are CRIEPI, JAERI, Osaka University, and the University of Tokyo.

The conceptual design and major parts of the system design have been completed, and their results are rated as sufficient for the initial safety review by a regulatory body. Subject to the availability of funds, the targeted schedule of the 4S development and design standardization could be as follows:

- Preliminary safety review by a regulatory body — the end of 2006;
- Construction of the 4S "demonstration reactor" and safety tests — in the early 2010s;
- Approval and/or certification of the 4S "standard design" for commercial deployment — in the early 2010s.

XIV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The reasons why demonstrations including safety tests at a "demonstration reactor" are required for the standardization or commercialization of the 4S are as follows:

- The 4S is a sodium cooled reactor, not a LWR;
- A 30-year lifetime of the 4S core and fuel; the use of metal fuel; and reactivity control with the movable reflector;
- High reliability requirements to the reflector driving system, which is to operate continuously for 30 years;
- A targeted significant reduction of maintenance requirements and ISI supported by the use of the non-pressurized sodium systems and static devices without rotating parts;
- The projected coupling with the HTE system for oxygen and hydrogen production.

XIV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

ENHS and STAR-LM are SMR concepts similar to the 4S [XIV-14, XIV-15].

XIV-2. Design description and data for 4S

XIV-2.1. Description of the nuclear systems

Reactor core and fuel design

Figure XIV-10 shows a cross section of the 4S core.

The core and fuel are designed to eliminate the need for refuelling during approximately thirty years and to make all reactivity temperature coefficients negative. Metal fuel, which has an excellent thermal conductivity, is applied. The core is shaped as a cylinder; its main dimensions are given in Table XIV-6. The core can be operated during thirty years by axially moving reflectors installed at the outside of the core, upward from the bottom. No reloading or shuffling of fuel is required during the whole core lifetime.

Figure XIV-11 shows the fuel subassembly of the 4S. The fuel element (fuel pin) consists of fuel slugs of U-Zr alloy, bonding sodium, cladding tube, and plugs at both ends. A gas plenum of an adequate length is located at the upper region of the fuel slugs.

In the fuel subassembly, fuel pins are assembled with grid spacers and a top shield is installed to prevent activation of the EM pumps and the secondary sodium in the IHX. Coolant inlet modules located beneath the fuel subassembly provide a lower shielding for the reactor internal structures including the core support plate and air in the RVACS.

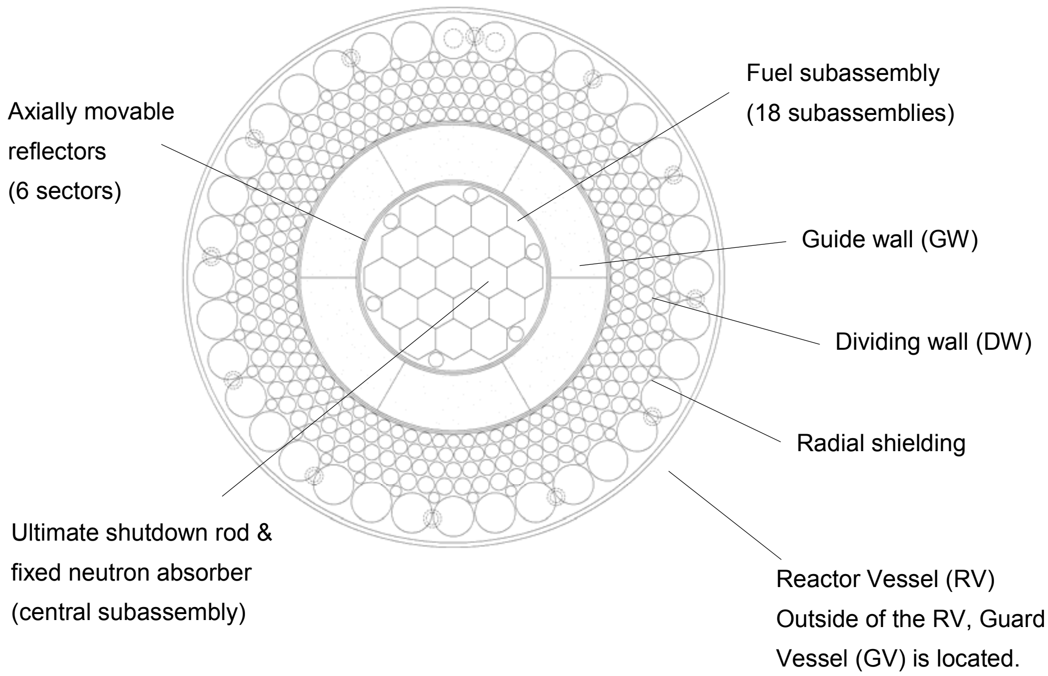


FIG. XIV-10. Cross-section of the 4S core (10 MW(e) plant).

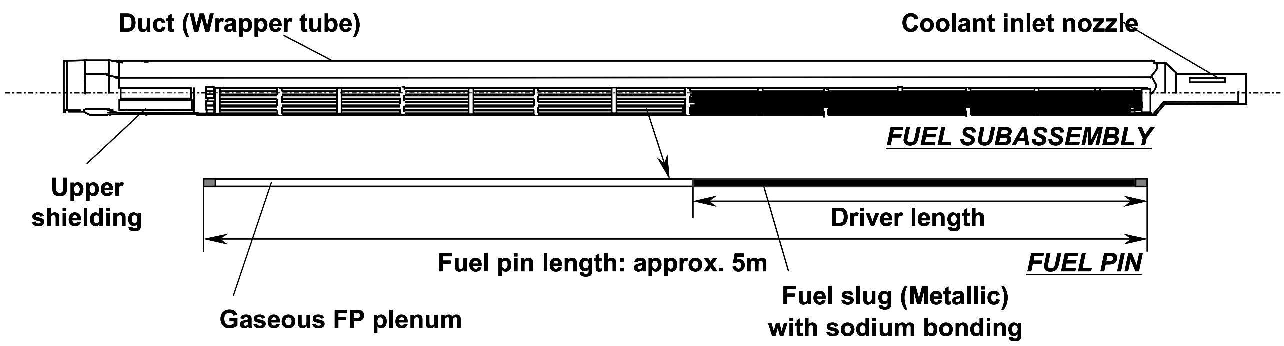


FIG. XIV-11. 4S fuel subassembly (10 MW(e) plant).

TABLE XIV-6. MAIN DESIGN PARAMETERS OF THE 4S CORE AND FUEL

ATTRIBUTES	DESIGN PARTICULARS	
Thermal rating, MW(th)	30	135
Active core height, m	2.5	2.5
Core equivalent diameter, m	0.9	1.2
Core configuration	Cylindrical shape	
Number of fuel subassemblies	18	
Type of fuel subassembly	Triangular fuel pin arrangement (Hexagonal cross section)	
Number of fuel pins per subassembly	169	271
Fuel assembly arrangement pitch, mm	206	259

Main heat transport system

A schematic of the 4S main heat transport system with specification of heat removal path in normal operation and in accidents is given in Fig. XIV-12; a brief explanation of this scheme is provided below.

Normal operation

The primary system is enclosed inside the reactor vessel (RV); sodium coolant is circulated by two EM pump units arranged in series. The heat generated in the reactor is transferred to the coolant of secondary sodium via the IHX located at the upper region in the RV. The secondary sodium is circulated by one EM pump unit. The heat is transferred to the water/steam system via heat transfer tubes in the SG. Water/steam is circulated by the feedwater pump.

The reactor vessel auxiliary cooling system (RVACS) is a system for shutdown heat removal; however, to keep the fully passive features, it is continuously operating even at normal operation of the reactor. The intermediate reactor auxiliary cooling system (IRACS) is a sodium loop with an air cooler for shutdown heat removal, arranged in series with the secondary sodium loop (Fig. XIV-2).

Shutdown heat removal

The RVACS removes shutdown heat with natural circulation of air. In the IRACS operation for shutdown heat removal, dampers are adjusted for the required capacity of heat removal. In case of a long-term operation for decay heat removal, IRACS is directed into a natural circulation mode via the adjustment of the dampers.

The water/steam system is also available for normal shutdown heat removal.

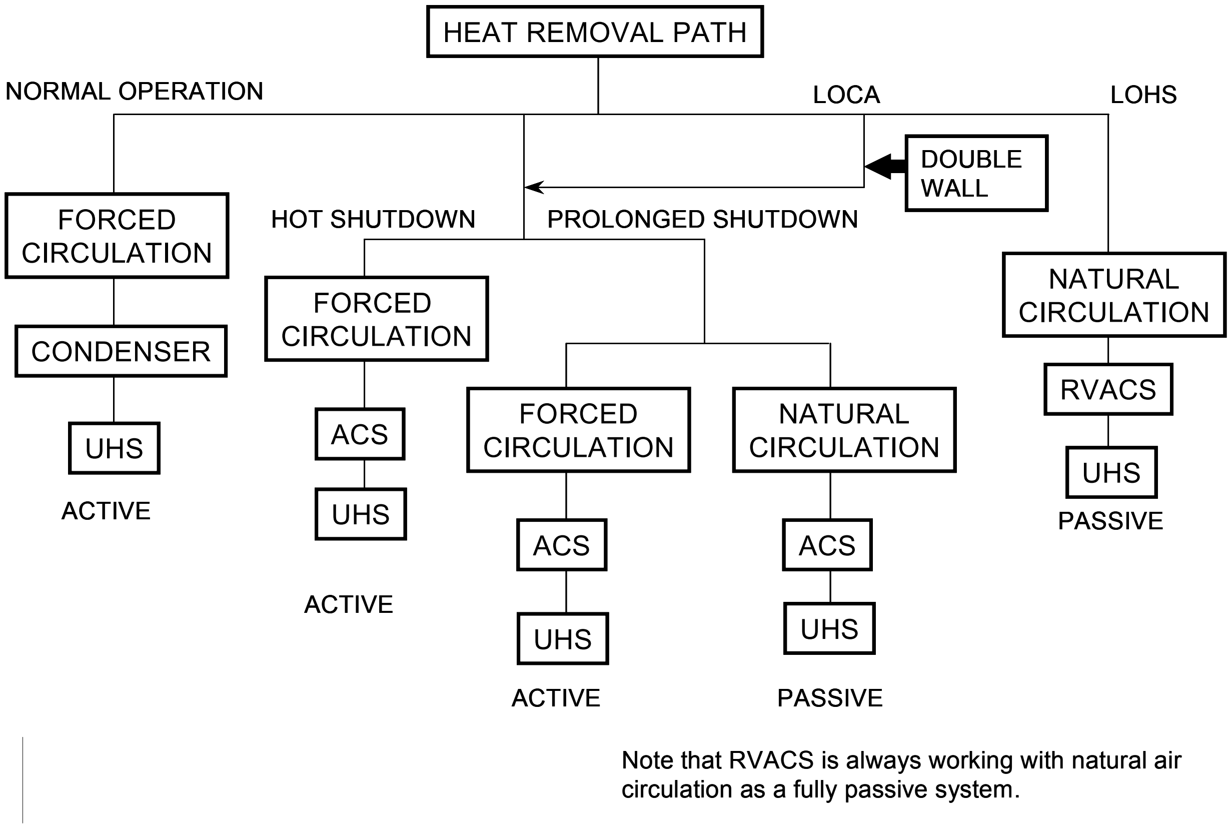
Loss of coolant (LOCA)

The 4S is a sodium cooled reactor; therefore, its primary system is “non-pressurized”. Hence, if sodium leakage occurs, the leak rate is quite small and the leaked sodium is retained by the second boundary, i.e. by the guard vessel, in all cases provided by the design; therefore, the core is always immersed in sodium. In case of a failure of the first boundary, both shutdown and normal shutdown heat removal systems operate.

Loss of heat sink (LOHS)

In case of a failure of the IRACS start-up, which is a failure of dampers, the dampers should be opened manually to provide for the removal of heat. If opened manually, IRACS would act as a heat sink, in a natural circulation mode.

If the IRACS fails completely, the RVACS is able to remove shutdown heat as a fully passive system of air convection.



ACS – auxiliary cooling system: RVACS, or RVACS + IRACS

LOHS – Loss of heat sink UHS – Ultimate heat sink

FIG. XIV-12. Heat removal paths of the 4S.

XIV-2.2. Description of the turbine generator plant and systems

Turbine, generator, and condenser of the 4S are commercially available systems.

XIV-2.3. Systems for non-electric applications

In remote areas, there is a demand for a power supply technology free from the burden of fuel transportation. Also, there is an underlying request for robust energy systems and a flexible energy supply to secure the energy independence of these areas. The 4S, a fast reactor without on-site refuelling, is a concept suiting the first request; it could also suit the second one if the energy is used diversely, such as for hydrogen production.

The high temperature steam electrolysis (HTE) is an appropriate method to produce hydrogen when coupled with the 4S, because HTE operates under a wide range of temperatures without emitting carbon dioxide due to the use of water as a feedstock.

The electrolysis of water is performed by introducing energy (ΔH) to a solid oxide electrolysis cell (SOEC) with high temperature steam, as shown in the equations below:

$$H_2O \dashrightarrow H_2 + 1/2O_2 - \Delta H \quad (1)$$

$$\Delta H = \Delta G + T\Delta S$$

In these equations, ΔG is provided by electricity and TΔS is provided by heat.

Figure XIV-13 shows a schematic drawing of hydrogen production by the HTE coupled with the 4S. The nuclear reactor of the 4S generates heat, a turbine-generator converts part of this heat to electricity, and the residual heat is transported to the HTE system. The electricity is used as power supply for the SOEC (via the rectifier) and is also delivered to the grid.

The maximum hydrogen production rate is estimated at around 14 000 Nm^3/hour with a reactor of 130 MW(th) and around 3000 Nm^3/hour with a reactor of 30 MW(th).

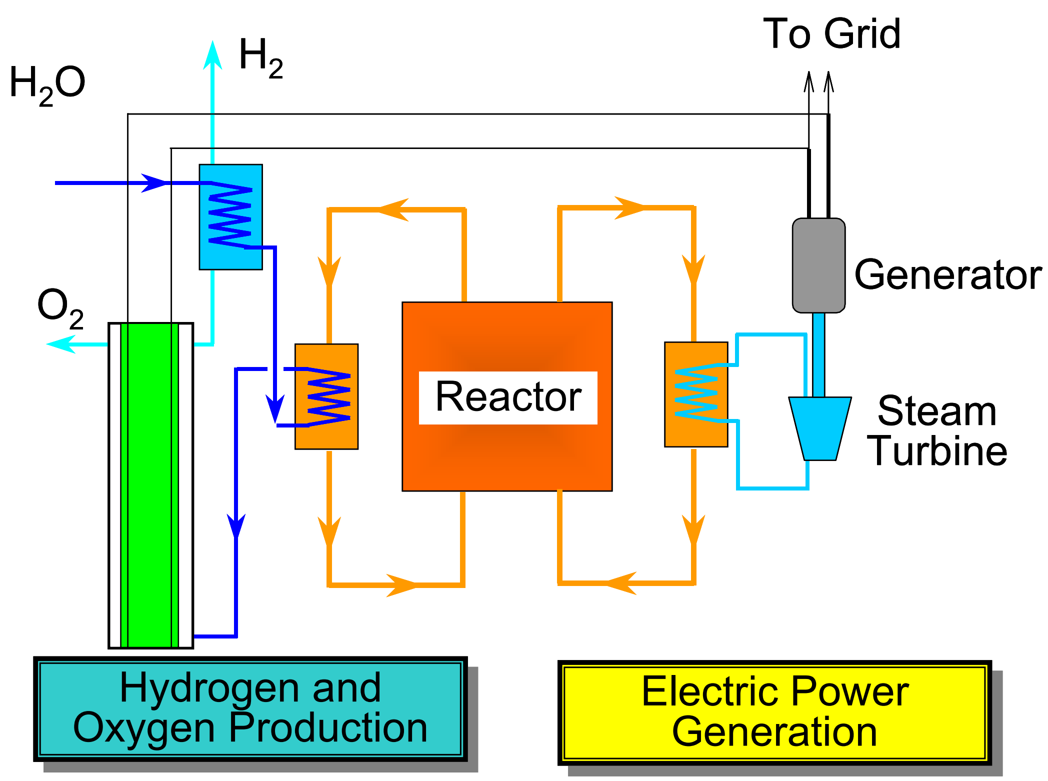


FIG. XIV-13: Schematic of hydrogen and oxygen production by the 4S with HTE.

In the above mentioned way, the 4S can produce hydrogen and, at the same time, supply electricity to the grid. The amount of electricity supplied to the grid and the volume of hydrogen production can be easily changed to meet the demand. When electricity demand is low, more hydrogen could be produced and stored as a reserve energy source.

By using this system, the independence of energy sources in remote areas becomes possible. Also, because oxygen is produced simultaneously by the HTE, the industries making use of the oxygen could be developed in the vicinity of the 4S plants.

The selected system for seawater desalination is described in detail in [XIV-16] and [XIV-17].

XIV-2.4. Plant layout

The plant layout of the 4S is optimized to meet various functional needs; the requirements for safety, radiation zoning, and piping and cabling; construction requirements; and access and security considerations. The general philosophy of the 4S plant layout is as follows:

- Efficient space utilization and minimization of volume of the buildings;
- Horizontal seismic isolation for the reactor building;
- An embedded reactor building, securing that the reactor is earth-sheltered;
- Lightweight buildings to assure a high degree of transportability in construction;
- The secondary sodium loop area is categorized as a “non-controlled area”; to achieve this, a sufficient shielding of the IHX is to be provided.

Vertical and horizontal cross-sectional views of the 4S plant are given in Fig. XIV-14 and Fig. XIV-15 respectively.

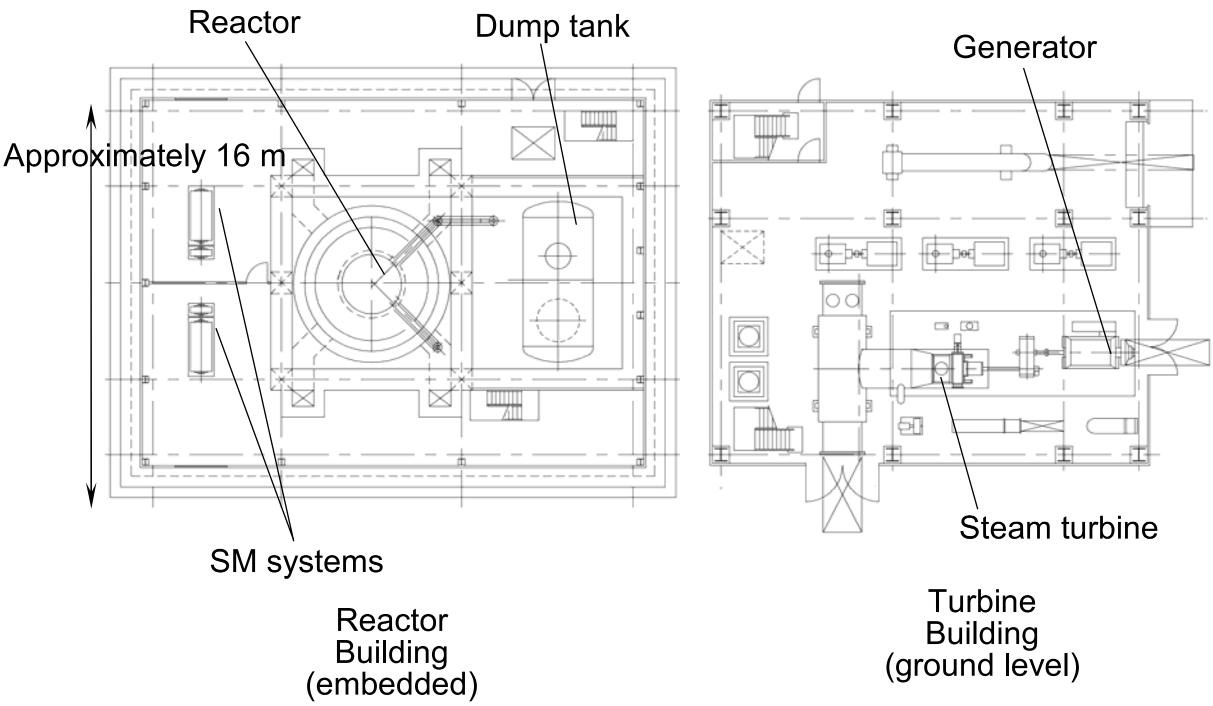


FIG. XIV-14. Horizontal cross-section of the 4S plant of 10MW(e).

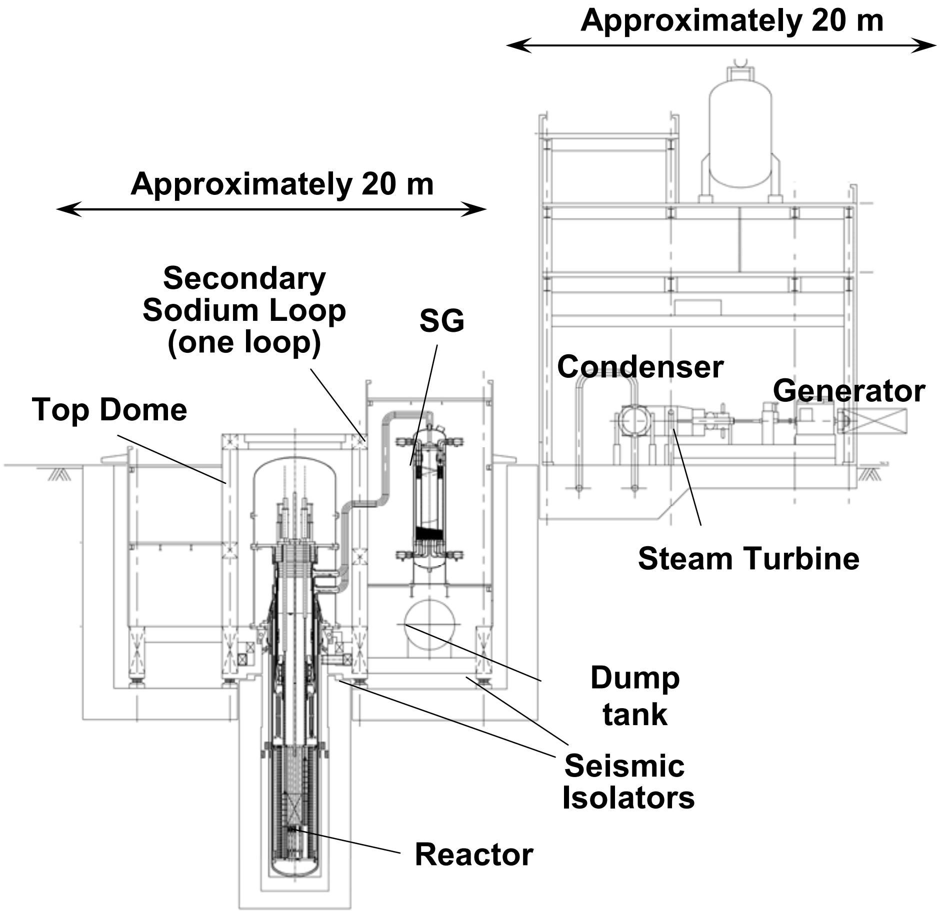


FIG. XIV-15. Vertical cross-section of the 4S plant of 10MW(e).

REFERENCES:

[XIV-1] HATTORI, S., HANDA, N., Use of super-safe, small and simple LMRs to create green belts in desertification area, Transactions of the American Nuclear Society (ANS), vol. 60, pp. 437 (1989).

[XIV-2] MATSUMIYA, H., et al., Development of a reflector drive mechanism using magnetic repulsion force for 4S Reactor, ANS Winter Meeting (Paper presented at meeting in Washington, D.C., Nov. 12–16, 2000).

[XIV-3] TSUJI, K., et al., Development of an innovative reflector drive mechanism using magnetic repulsion force for 4S reactor, ICONE-9 (Proc. 9th Int. Conf. on Nuclear Engineering, Nice Acropolis, France, April, 2001), ASME.

[XIV-4] KOBAYASHI, Y., et al., A study of magnetic impact type micro-stepping actuator to drive large payload reflector for 4S reactor, IMECE2004-59372, 2004 ASME International Mechanical Engineering Congress and RD&D Expo (Anaheim CA, USA, Nov. 13–19, 2004).

[XIV-5] Report to Congress on Small modular nuclear reactors, USDOE (May, 2001).

[XIV-6] AOKI, K., et al., A feasibility study on TRU burning in a small fast reactor with negative sodium void worth, ICENES'93 (Paper presented at the 7th Int. Conf. on Emerging Nuclear Energy Systems, Makuhari, Chiba, Japan, September 20–24, 1993).

[XIV-7] AOKI, K., Hattori, S., Long term Pu storage and MA burning in a small fast reactor", GLOBAL'95, France (1995).

[XIV-8] Japanese Electric Association Guide (Nuclear), JEAG 4614-2000.

[XIV-9] WATANABE, M. et al., Toshiba nuclear critical assembly physics benchmark experiments for reflector-control-type small fast reactors, ANS Winter Meeting (Paper presented at meeting in Washington, D.C., November 12–16, 2000).

[XIV-10] KAWASHIMA, M. et al., Physics benchmark experiments and analysis for reflector-control-type small fast reactors at Toshiba Nuclear Critical Assembly, Nuclear Data for Science and Technology, ND2001 (Paper presented at conf. in Tsukuba, Japan, October, 2001).

[XIV-11] UKAI, S., et al., Characterization of high temperature creep properties in recrystallized 12Cr-ODS ferritic steel claddings, Journal of Nuclear Science and Technology, Vol. 39, No. 8, pp. 872 (2002).

[XIV-12] UKAI, S., et al., Development of 9Cr-ODS martensitic steel cladding for fuel pins by means of ferrite to austenite phase transformation, Journal of Nuclear Science and Technology, Vol. 39, No. 7 (2002).

[XIV-13] OTA, H., et al., Development of 160m^3/min large capacity sodium-immersed self-cooled electromagnetic pump, Journal of Nuclear Science and Technology, Vol. 41, No. 4, pp.511–523. (2004).

[XIV-14] WADE, D.C., et al., ENHS: the encapsulated nuclear heat source - a nuclear energy concept for emerging worldwide energy markets, ICONE-10 (Proc. of 10th Int. Conf. on Nuclear Engineering, Arlington VA, US, 2002), ICONE-10-22202, ASME.

[XIV-15] BROWN, N. W., et al., Liquid metal cooled reactors and fuel cycles for international security, ICONE-11 (Proc. 11th Int. Conf. on Nuclear Engineering, Tokyo, Japan, 2003), ASME.

[XIV-16] HATTORI, S., MINATO, A., Application of super safe small and simple reactor for nuclear desalination in the desertification area, Tech. Responses to Global Environmental Challenges (Paper presented at Conf. in Kyoto, Japan, Nov., 1991).

[XIV-17] HATTORI, S., MINATO, A., Human welfare by nuclear desalination using super-safe, small and simple reactors (4S), Desalination, 99, pp. 345–365 (1994).

SUPER-SAFE, SMALL AND SIMPLE LIQUID METAL COOLED REACTOR (4S-LMR, CRIEPI DESIGN)

Central Research Institute of Electric Power Industry (CRIEPI), Japan

XV-1. General information, technical features and operating characteristics

XV-1.1. Introduction

The Central Research Institute of Electric Power Industry (CRIEPI, Japan) is developing a small-sized Super-Safe, Small and Simple liquid metal cooled reactor (4S-LMR), which is a sodium cooled nuclear reactor of 50 MW(e) being designed for use as a distributed energy source with multi-purpose applications. Design development for the 4S-LMR is performed jointly with the Toshiba Corporation (Japan).

Another nuclear power plant with a small reactor of the 4S name, developed by the Toshiba Corporation in cooperation with the CRIEPI and other Japanese organizations, is described in ANNEX XIV of this report.

Both designs rely on certain experience in development and operation of fast sodium cooled reactors in Japan, described in Annex XIV.

The organization presently responsible of liquid metal cooled fast reactor technologies, including sodium cooled fast reactor technologies in Japan, is the Japan Atomic Energy Agency (JAEA), a new organization consolidating the former JAERI and JNC.

Japan has a national plan to introduce fast breeder reactors (FBRs) to utilize Pu extracted from spent fuel of existing light water reactors (LWRs). CRIEPI is supporting this national plan as a utility laboratory, and has proposed several innovative FBR concepts for future deployment. The 4S-LMR is one of the innovative reactor concepts proposed by CRIEPI.

XV-1.2. Applications

The 4S-LMR is designed as an energy source to supply electricity and potable water, see Fig. XV-1.

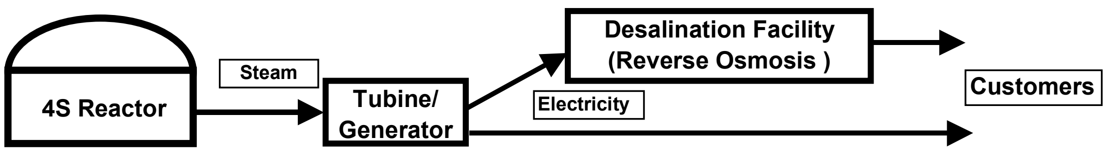


FIG. XV-1. 4S-LMR applications.

XV-1.3. Special features

The 4S-LMR [XV-1 to XV-3] is a concept of small sodium cooled fast reactor in which the design efforts are concentrated toward meeting the global power market needs. To achieve this, the 4S-LMR has been designed to incorporate the principles of simple operation and simplified maintenance, including the refuelling, a high level of safety and improved economic performance. Special features of the 4S-LMR are as follows:

- Operation for 10 years or more without reloading or shuffling of fuel;
- Quality assurance and a short construction period based on factory fabrication of certain parts of a NPP;
- Load following capability achieved without the operation of reactor control systems;
- Strong reliance on inherent safety features and passive safety systems, targeting the elimination of core damage in any conceivable initiating event without reactor scram;
- Minimum maintenance and inspection of the reactor components.

One of the important tasks of nuclear technology for the future sustainable growth of mankind is the stable and reliable energy supply to remote populated areas. These areas may be remote islands or deep interior regions where sophisticated technological infrastructure is not expected and where power demands are generally modest. The benefits of nuclear energy could best be brought to these communities by small and simple power generation.

XV-1.4 Summary of major design and operating characteristics

Design of the 4S-LMR was modified in 2001 to select a metal fuel core as basic option for further development. The efforts of the 4S-LMR designers are focussed on optimizing the core configuration because safety requirements strongly depend on core performance. Major design and operating characteristics of the 4S-LMR are summarized in Tables XV-1 and XV-2.

TABLE XV-1. MAJOR DESIGN SPECIFICATIONS OF THE 4S-LMR CORE

ITEMS	SPECIFICATIONS
Thermal output, MW	135
Electric output, MW	50
Mode of operation	Basic load or load following
Load factor (target)	90 %
Core height, m	1.0/1.5
Core diameter, m	1.2
Number of subassemblies	12
Number of movable reflector units	6
Fuel type	Metal fuel
Fuel composition	U/Pu/Zr
Smear density, % of theoretical density	75
Pu enrichment, weight %	17.5/20.0
Fuel pin diameter, mm	10.0
Number of fuel pins	469
Maximum linear heat rate, kW/m	25
Maximum temperature of fuel, °C	620
Cladding thickness, mm	0.59
Bundle pitch, mm	258
Pitch/diameter ratio	1.15

ITEMS	SPECIFICATIONS
Duct thickness, mm	2
Duct gap, mm	2
Movable reflector material	Graphite
Reflector thickness, m	0.3
Reflector height, m	2.1
Conversion ratio (middle of core life)	0.71
Coolant void reactivity (end of core life)	~ 0
Burn-up reactivity swing, % ρ	~ 9
Number of circuits	Three, see Table XV-2.
Thermodynamic cycle efficiency, %	37%
Average fuel burn-up, GW·day/t	70
Core lifetime, years	10
Plant lifetime, years	30

TABLE XV-2. SPECIFICATIONS OF PRIMARY COOLANT, SECONDARY COOLANT AND STEAM CONDITIONS

ITEM	SPECIFICATION
Primary coolant system	
Primary coolant	Sodium
Primary circulation	Forced, electromagnetic pump (EMP)
Primary coolant conditions:	
Temperature (outlet/inlet), °C	510/355
Flow rate, kg/s	152
Pressure drop, MPa	~ 0.1
Intermediate heat transport system	
Secondary coolant	Sodium
Secondary circulation	Forced, electromagnetic pump
Secondary coolant conditions:	
Temperature (outlet/inlet), °C	475/310
Flow rate, kg/s	142
Steam turbine circuit	
Steam conditions:	
Temperature/ pressure at steam generator inlet, °C/MPa	210/10.8
Temperature/ pressure at steam generator outlet, °C/MPa	453/12.4
Flow rate, kg/s	12.8

A schematic of the 4S-LM reactor is given in Fig. XV-2. A simplified schematic diagram of the 4S-LMR plant is given in Fig. XV-3.

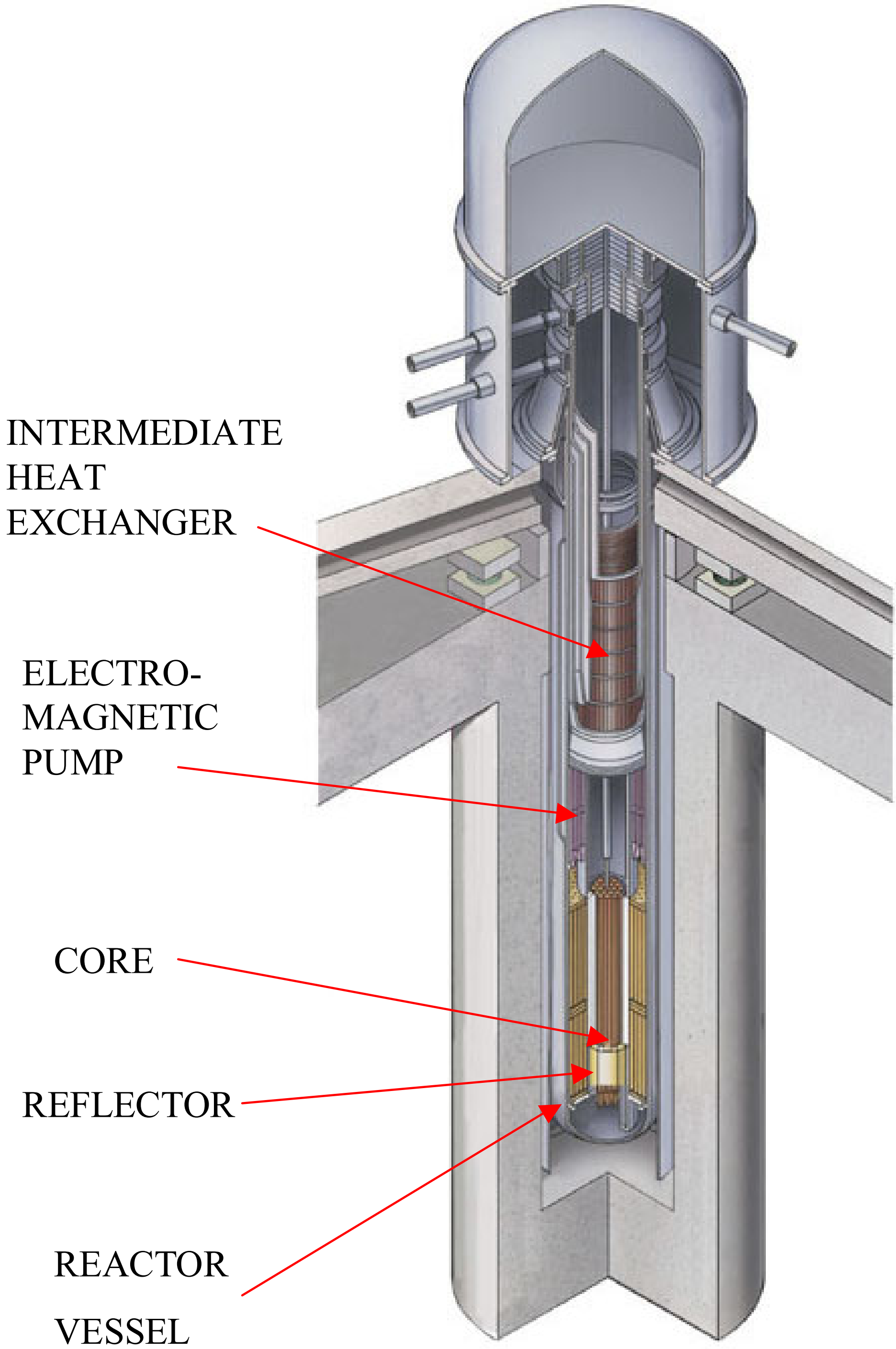


FIG. XV-2. 4S-LMR reactor.

The 4S-LMR incorporates neutron reflectors to control the core reactivity without neutron absorber rods. The reflectors are driven from outside the reactor vessel and move very slowly; the movement speed is below 1 mm/day. Electromagnetic pumps are used for primary coolant circulation. Incorporation of these design features eliminates fast moving or rotating components, contributing to a decreased component failure and reduced maintenance.

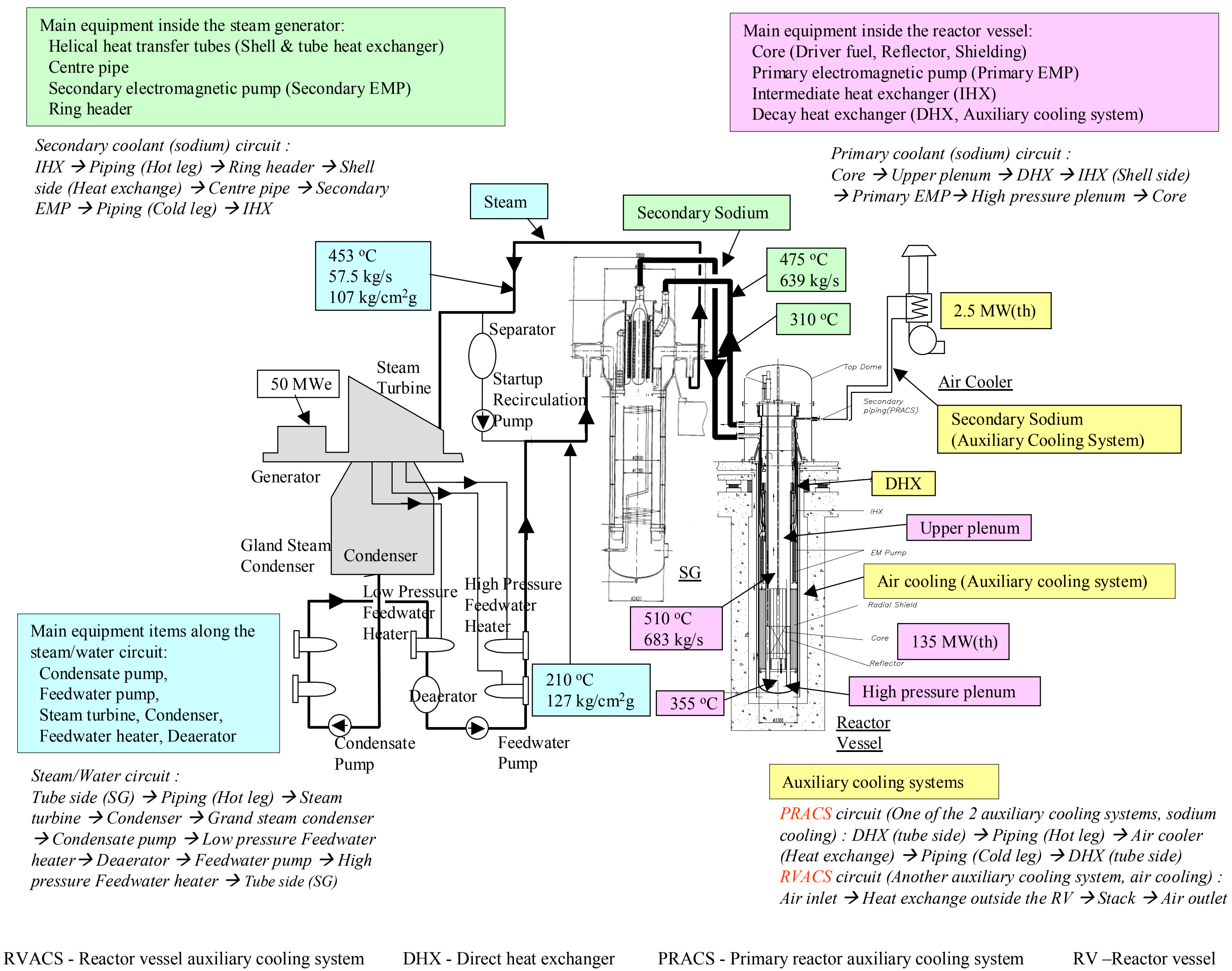


FIG. XV-3. Simplified schematic diagram of the 4S-LM plant.

Table XV-3 shows the temperature feedback coefficients integrated over the core region. The graphite reflector helps increase the Doppler coefficient, which is slightly smaller than that of a large MOX core. With graphite reflector, the neutron energy spectrum in the 4S outer core becomes relatively soft compared to a previously considered 4S-LMR core with the steel reflector.

TABLE XV-3. REACTIVITY FEEDBACKS ON TEMPERATURE

Doppler	$\left(T\frac{dk}{dT}\right)$	-7.07×10^{-3}
Fuel	$\left(\frac{\Delta k/kk'}{^{o}C}\right)$	-2.68×10^{-6}
Coolant	$\left(\frac{\Delta k/kk'}{^{o}C}\right)$	~0
Structures	$\left(\frac{\Delta k/kk'}{^{o}C}\right)$	-8.94×10^{-8}

Since the output can be controlled by the turbine-water system alone using a negative coolant reactivity coefficient, the reactivity control system to adjust the output can be removed, which contributes to improved economic characteristics of the 4S-LMR.

XV-1.5. Outline of fuel cycle options

The reactor is assumed to operate in a closed fuel cycle with reprocessing of fuel. The reactor core operates for 10 years without reloading and shuffling of fuel. Following a whole-core refuelling, the spent fuel will be sent to regional or national centres for reprocessing.

A metallic fuel is used in the 4S core in consideration of pyro-reprocessing. A fast reactor technology using a metal fuel based cycle (pyro-processing of spent fuel) appears to be a promising approach [XV-4]. The technology is of value because it has the potential to simplify reprocessing and fuel fabrication processes and nuclear waste disposal, and also could help reduce fuel cycle costs.

XV-1.6. Technical features and technological approaches that are definitive for 4S-LMR performance in particular areas

XV-1.6.1. Economics and maintainability

The following design features contribute to improved economic performance and simplified maintenance of the 4S-LMR:

(1) All core fuel is replaced in one batch recognizing that the volume of maintenance work, including fuel exchange, should be largely reduced; the core is designed to operate for a long period without refuelling;

(2) The design incorporates no rotating plug and the mechanism of fuel exchange is designed so that fuel can be removed and loaded using a simple and inexpensive fuel-handling machine;

(3) The design makes no use of control rods for power and reactivity control during reactor operation and, therefore, eliminates the complicated upper-core structure typical of many other reactors; the reactor is designed so that the annular reflector

surrounding the core performs all start-up, shutdown and burn-up control functions;

(4) The design provides for shop fabrication of certain plant components and their easy installation at the site; the performed design studies have shown that the specific (per output) construction cost of the 4S-LMR could be maintained at a level matching that of a plant with a large water cooled reactor.

The abovementioned design features of the 4S-LMR are also viewed as necessary conditions for plant installation at remote locations in developing or developed countries. Specifically, these features contribute to achieving a simplified reactor design, with the per-output weight of structural materials in the 4S reactor being lower than in a typical large reactor.

The estimates performed in 1990 for the 4S-LMR of previous design pointed that the construction cost per output could be 20% lower than in a large light water reactor if the factor of mass production is taken into account with the capacity of 10 units per year. This is an approach alternative to the economy of scale used to improve the economic characteristics of large reactors.

The mechanism of cost reduction through production in series is the same as in conventional, non-nuclear industries. Costs of the design, production facilities, plant construction and operation for nuclear reactors are levelled by the number of units produced. In mass production, the largest cost components are those of the inspection and materials. If the automation of the inspection is advanced, it will be the reduced volume of bulk materials that would directly govern the economic competitiveness of a serially produced reactor. As a result, the construction cost could be reduced to about 30% of that corresponding to the construction of a single reactor, if the 4S-LMR is manufactured at a rate of 10 units per year continuously for 10 years. The additional merit of a small reactor is that the total development cost up to commercialization is dramatically smaller than that for a large reactor.

For the 4S-LMR, the target is to have construction costs below 2000 US$/kW(e) under specific conditions, such as factory fabrication, mass production, etc. The targeted construction period is 2 years per unit.

XV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The cumulative wastes generated by the 4S-LMR would be reduced compared with those in conventional reactors that do not have a long-life core, because of the infrequent refuellings. As a fast reactor, the 4S-LM can also effectively incinerate minor actinides.

If the 4S-LM is used to produce desalinated water for a plantation in the desert or a similar place, it would indirectly contribute to absorbing some carbon dioxide.

Under its present design, the 4S-LMR is not a breeder reactor; however, it has a relatively high conversion ratio and its spent fuel could be reprocessed and the Pu could be extracted for future use, contributing to a more efficient use of natural uranium resources.

XV-1.6.3. Safety and reliability

Safety concept and design philosophy

The design target for the 4S-LMR is to ensure that there is no core damage in any conceivable initiating events without scram and to secure complete confinement of radioactivity in all operating conditions and under decommissioning. The design features adopted to facilitate achieving this goal are as follows:

(1) Negative temperature reactivity coefficients and negative coolant void reactivity effect, provided by an appropriate selection of the design parameters; the incorporation of negative temperature reactivity coefficients facilitates realization of passive safety features [XV-2] and also simplifies the power control system so that only feed water control in the power circuit can regulate reactor power;
(2) High thermal conductivity of metal fuel, which contributes to an enhanced protection against core disruption, is several postulated events such as unprotected loss of flow (ULOF);
(3) All passive systems incorporated in the design are independent of the emergency power;
(4) All decay heat removal systems are passive.

Provisions for simplicity and robustness of the design

Provisions for simplicity and robustness of the 4S-LMR design are the following:

(1) The core is designed so that the annular reflector surrounding the core performs all start-up, shutdown and burn-up control functions, which ensures simple core burn-up control without control rods and control rod driving mechanisms, and also makes unnecessary the complicated upper-core structure;
(2) Heat is released from the reactor vessel by natural circulation of sodium and air to enable effective heat discharge in the period of reactor shutdown after an accident;
(3) The output can be varied by control of the turbine-water system alone using a negative coolant temperature reactivity coefficient of the core, which makes it possible to abandon the traditional reactor control system;
(4) The sodium-water reaction product release system is designed so that the integrity of the primary boundary is maintained under assumed damage to all of the heat transfer tubes in the steam generator and loss of the protective function of the steam system;
(5) The 4S-LMR design provides for minimum maintenance and inspection of reactor components.

Active and passive systems and inherent safety features

The active safety systems of the 4S-LMR are:

- System of dropping down a reflector;

The passive safety features and systems are:

- Radial expansion mechanism of the core;
- Reactor vessel auxiliary cooling system (RVACS) and primary reactor auxiliary cooling system (PRACS), see Fig. XV-3.

The inherent safety features are:

- Negative temperature reactivity coefficients and negative void reactivity coefficients;
- High heat conductivity of the metal fuel.

The safety related systems and features of the 4S-LMR are summarized in Table XV-4.

Structure of the defence-in-depth

The defence in depth concept of the 4S-LMR is similar to the one used in sodium cooled fast reactors of previous designs, such as the PRISM.

TABLE XV-4. SAFETY RELATED SYSTEMS AND FEATURES OF THE 4S-LMR

ITEM	SPECIFICATION
Core burn-up reactivity compensation system	Annular reflector upward movement with a very low speed, below 1 mm/day
Primary pump	Electromagnetic pump, no movable parts
Primary flow after shutdown	Natural circulation
Cavity cooling	Natural circulation
Containment cooling	Natural circulation
Secondary pump	Electromagnetic pump, no movable parts
Maintaining integrity of the reactor core during and after an accident	High thermal conductivity of metal fuel; high heat capacity of the primary and secondary coolant
Shutdown heat removal	Natural circulation

Design basis accidents and beyond design basis accidents

The design basis events (DBE) considered in the 4S-LMR are similar to those analyzed for the previous designs of fast reactors; however, the beyond design basis events (BDBE) considered in the 4S-LMR are those in which the reflector does not drop down for any reason, categorized as anticipated transients without scram (ATWS).

Typical hypothetical accidents were analyzed to demonstrate the passive safety capability of the 4S-LMR. The accidents analyzed were chosen to demonstrate whether or not the passive heat removal capability and the passive reactor shutdown capability play a significant role [XV-5]. The term “passive reactor shutdown” does not mean a true shutdown but a function of reducing the reactor power to a level where heat removal can be accomplished passively with the use of the RVACS and with no core damage.

All components, such as intermediate heat exchanger (IXH), EM pumps, and steam generator (SG), were modelled in one dimension. Figure XV-4 shows the schematic network model of the 4S-LMR used in the analyses; it also illustrates a unique flow path configuration provided to enhance the performance of the RVACS. Exhausted coolant from the primary EM pumps flows in two directions, as shown in the right part of Fig. XV-4.

One is in the main direction leading to the core inlet through the shielding. The other is upward along the inside of the reactor vessel, and then returning to the pump inlet as a bypass flow. Without the pump head, coolant flow in this region reverses its direction after a reactor shutdown, as shown in the left part of Fig. XV-5, contributing to an increase in the effective surface for heat radiation. The bypass flow is 10 % of the rated flow.

The design criteria are to have no coolant boiling and no fuel melting and to ensure that temperature does nor exceed 650°C for the primary boundary structure. Temperatures are evaluated for the nominal hottest pin, which is assumed to have a nominal hot channel factor of 1.53 without the engineered safety factor. The outlet coolant temperature is 593°C in normal operation.

The protected loss of heat sink (PLOHS) event was simulated to predict the heat removal capability of the RVACS. PLOHS was assumed to be initiated by a loss of the external alternate current (AC) power, resulting in a total loss of AC power, because the 4S-LMR has no emergency AC power on-site. The steam/water system cannot remove the decay heat in this event. The primary coolant flow shifts to natural convection mode. The design heat removal capabilities of the PRACS and the RVACS are 2.5 MW(th) and 1 MW(th), respectively.

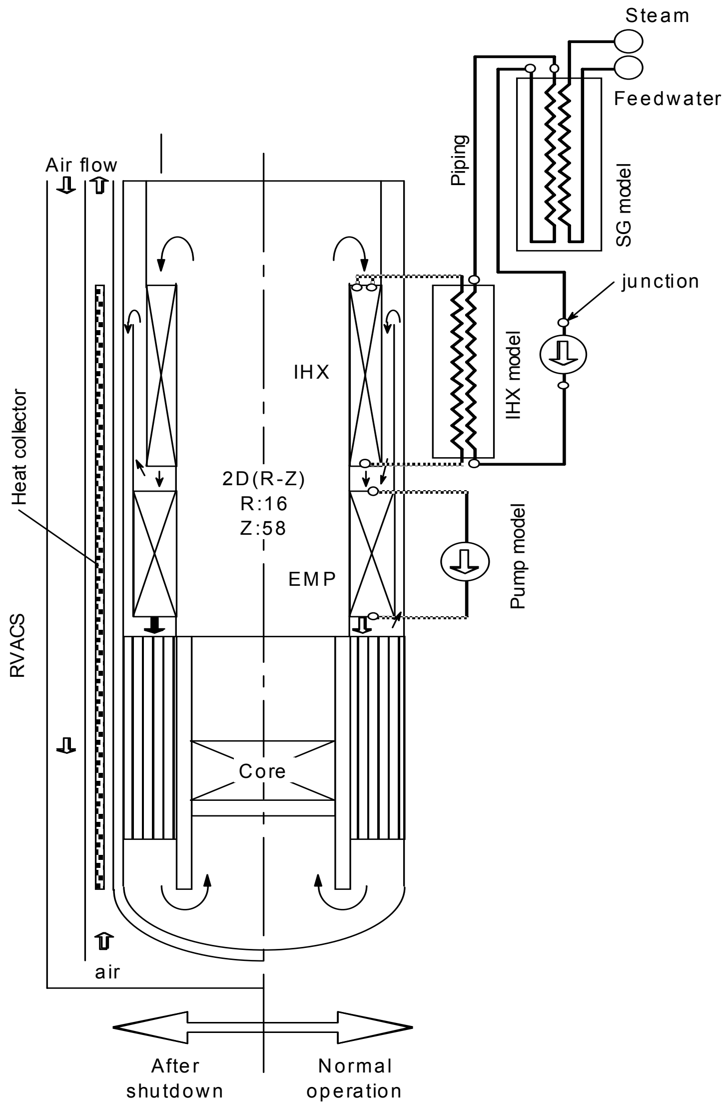


FIG.XV-4. Schematic network model of the 4S-LMR.

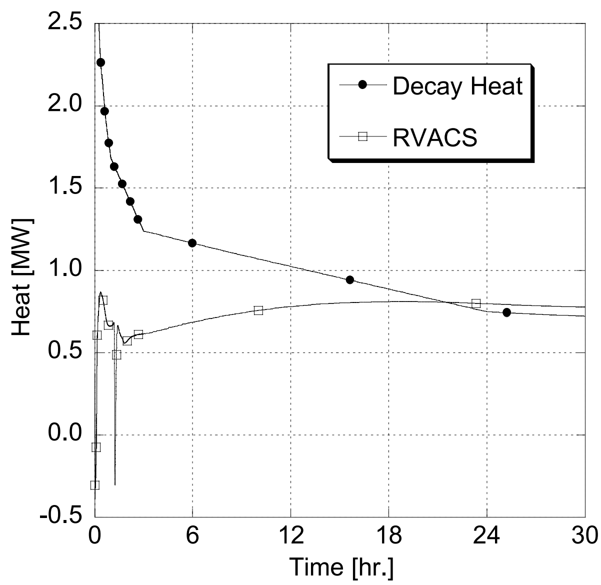


FIG. XV-5. Predicted heat removal capability of RVACS in PLOHS.

Although the PRACS can remove the decay heat under a natural convection mode, it was assumed to be out of work and only the RVACS was available — this conservative analysis was performed to evaluate the heat removal capability of the RVACS. The movable reflector was assumed to be moved down in this event.

The accident starts with a primary pump trip after 1 second following the reactor shutdown. Feed water stops at 6 seconds and the steam/water blow valve is opened for 34 seconds.

Figure XV-5 shows the predicted heat removal capability of the RVACS. Since the temperature distribution in the primary hot plenum is in transition and the secondary flow is unstable for about 1 hour after the shutdown, the heat and flow are observed to move up and down. The RVACS removes about 0.8 MW(th) after stable conditions are reached, which is higher than the core decay heat. The difference from the design value of 1.0 MW(th) results from the difference in the temperature at the primary hot plenum -the design value was defined for 650°C. The heat transfer area of the RVACS is 130 m^2 at the outer surface of the reactor vessel with an effective height of 13.8 m. The maximum averaged heat flux is 6.2 kW/m^2. The primary coolant flows at a 3.4% of the nominal rate. The coolant flow in the bypass region is 3.3 kg/s, which is about 5% of the core flow rate in stable conditions. Figure XV-6 shows the temperature variation during the event. The “HP Na (RV Top)” is used to denote the coolant temperature at the top of the down comer, which is representative of the primary boundary temperature. The maximum temperature of the coolant is low enough to meet the criterion. The maximum primary boundary temperature is also lower than 650°C. The results of thermal-hydraulics analysis, performed with the use of the CERES code, indicate that neither stagnant areas nor local vortex flow are observed in the flow pattern due to a simple flow path configuration. With relation to the flow pattern, it is also predicted that neither a hot spot nor a cold spot will be observed in the temperature distribution [XV-5].

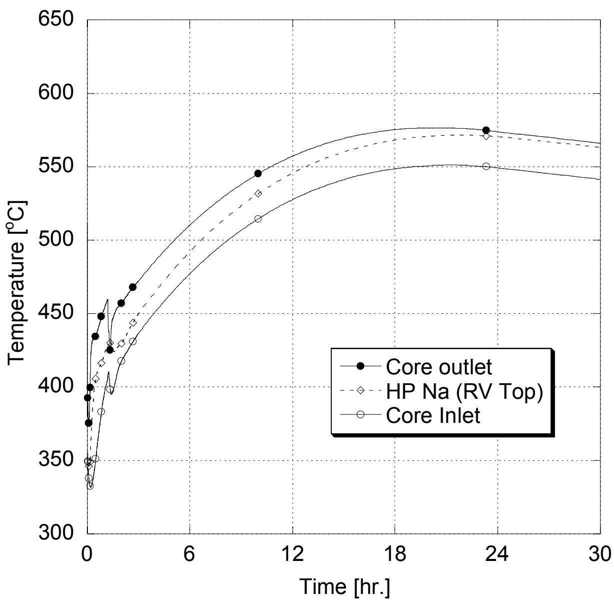


FIG. XV-6. Predicted temperatures in PLOHS.

In a protected loss of flow, moving the reflector to a lower position helps accomplish the reactor shutdown and the decay heat is then removed passively. As it was already mentioned, an anticipated transient without scram (ATWS) for the 4S-LMR was defined as an accident in which the active reactor shutdown system does not work. Therefore, the unprotected loss of flow (ULOF) and transient overpower (UTOP) are categorized as ATWS.

A ULOF event is initiated by loss of the external AC power of the primary pumps without reactor scram. As a result, the core temperature rises due to a power-to-flow mismatch. The core flow decreases faster than the core power is reduced by the negative reactivity feedbacks. Because the typical fast reactor has a positive coolant void worth (coolant density feedback is also positive), the transient may result in catastrophic core damage after the onset of coolant boiling. A non-positive void worth enhances the negative feedback and prevents the insertion of a large positive reactivity. Flow halving time and core kinetic characteristics, especially, the Doppler and the sodium density feedback coefficients, mainly govern the consequences of ULOF [XV-6, XV-7].

Figure XV-7 shows the predicted temperature changes of the fuel and the coolant at the nominal hottest pin, which has the highest temperature of all the pins. The maximum temperatures are lower than the fuel melting point of 1180°C and the sodium boiling point of 960°C, respectively.

In addition, a more severe ULOF was analyzed, which is an instantaneous loss of circulation head of one of the two EM pumps connected in series. Such accident is initiated by the loss of one EM pump, and the duration time during which the flow rate decreases from 100 to 50% of the rated value is shorter than in the ULOF considered previously. Besides, the flow coast down shape maintained by only one pump is also less favourable. Altogether, the flow decrease characteristics indicate that this ULOF case may be more severe than the one considered previously.

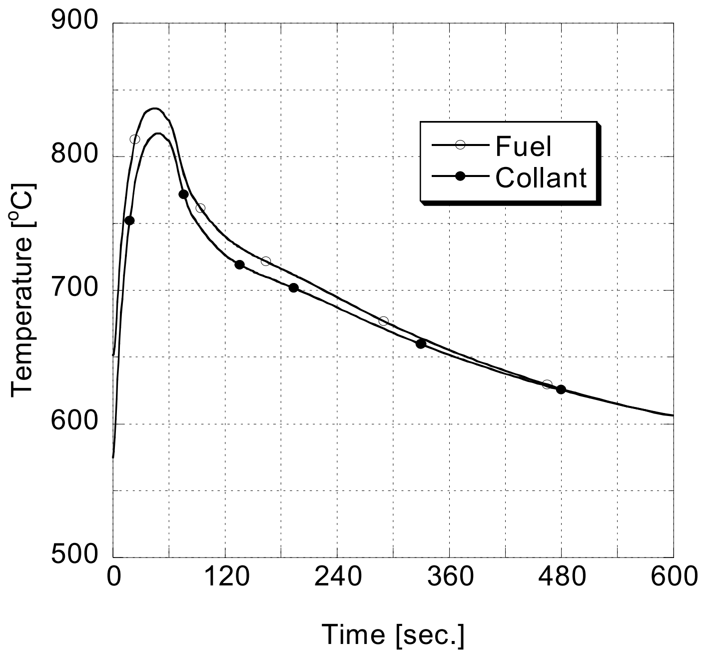


FIG. XV-7. Predicted temperatures in ULOF.

When pumps are connected in parallel, more than three of them would be required in such ULOF because there may be some reverse flow through the pump at fault. In a serially arranged two-pump system of the 4S-LMR, the core flow rate can be sustained at more than 50% of the rated flow, because there is no reverse flow and the functioning pump could be able to sustain a higher flow due to its flow-head curve. The analytical results obtained for the 4S-LMR indicate that this more severe ULOF pushes the temperatures up by about 10°C only, which is acceptable. They also point to certain margins in core pressure drop and reactivity feedback characteristics, which could be effectively used to improve the 4S-LMR performance in ULOF, e.g. by applying a tighter lattice pitch in the core.

The unprotected transient overpower (UTOP) events were analyzed to estimate the allowable external reactivity insertion. In the 4S-LMR, slow upward movement of the reflector is used to compensate the burn-up reactivity loss. An UTOP is then the event initiated by an inadvertent reflector lifting without scram. The analytical assumptions used for the reference UTOP case are summarized below:

- No radial core expansion except that in the core support grid;
- Constant coolant flow rate in the primary and secondary circuits;
- Constant rate of heat removal via SG (135MW(th));
- PRACS and RVACS were not considered.

The external reactivity is inserted at a ramp rate of 0.1 ¢/s, which is ten times larger than the ramp rate required to change the reactor power by 1%/min. The peak coolant temperature at the hottest nominal fuel element reaches 970°C up to 1 $ insertion, while the hottest fuel temperature is below the melting point. The inserted reactivity is almost cancelled by the Doppler reactivity. The calculated reactor power rises to 1.31 of the rated value.

However, some previously performed mock-up tests indicate that the cladding of the irradiated fuel element may breach and fuel liquefaction may occur within the range of temperatures reached in a UTOP in the 4S-LMR. Specifically, a furnace test [XV-8] was conducted to evaluate the behaviour of an irradiated EBR-II Mk-V-type fuel element during a loss of flow event; the fuel element appeared as a combination of U-19Pu-10Zr fuel and HT-9 cladding. The fuel element was kept at about 820°C for 112 minutes. The cladding breached due to cladding thinning by the fuel/cladding metallurgical interaction. The fuel/cladding interaction also caused fuel foaming, because the iron atom diffused into the fuel matrix to form a low melting alloy.

In another UTOP with 70 ¢ insertion, the 4S-LMR power rises and becomes stable at 1.30 of rated value. The tendency is the same as in a 1$ insertion case; the coolant temperature rises to 860°C and comes down to 800°C after the reactivity insertion is completed; the peak fuel temperature is 940°C. If the transient lasts for hours, the fuel elements may be damaged resulting in molten fuel dispersion due to the iron atom diffusion and the liquefaction.

The abovementioned results indicate that that an insertion of the total reactivity slightly below 1$ may be acceptable to avoid fuel melting and coolant boiling in a short term and that the external reactivity should be limited by about 70 ¢ to ensure acceptable long-term behaviour of the 4S-LMR. It is noted that acceptable reactivity depends on analytical assumptions, which include the conditions of heat removal and the passive reactivity mechanism, especially, the core radial expansion (bowing).

The 4S-LMR of the latest design has about a 24 $ reactivity loss during a core lifetime of 10 years. This value, compensated by movement of the reflectors, is large enough to initiate severe core damage in case the reactivity of all reflectors is inserted. The reflector reactivity control system must include a stopper system to prevent unacceptable reactivity insertion; the design of this system can be figured out through various UTOP analyses.

Provisions for safety under seismic conditions

The 4S-LMR has a standard seismic design with horizontal seismic isolation systems, acceptable for a variety of siting conditions.

Probability of unacceptable radioactivity release beyond the plant boundaries

The probability of radioactivity release beyond the plant boundary is estimated at 10^{-7} 1/year.

Measures planned in response to severe accidents

There might be no necessity to plan off-site responses to severe accidents because a high core damage accident (HCDA) is essentially eliminated in the 4S-LMR through several specific design features, including those that prevent a primary coolant leak.

XV-1.6.4. Proliferation resistance

The main feature contributing to an enhanced proliferation resistance of the 4S-LMR is a lifetime core with infrequent refuelling. Because of a long period between whole-core refuellings (more than 10 years), it may be possible to eliminate the need for on-site refuelling equipment and a long period of on-site storage of the fuel. Refuelling of the eighteen fuel assemblies can be accomplished in a very short time and with special shipping casks, removed from the site to the recycling facilities. Access to the nuclear system is unnecessary during normal operation, which means that access to the fuel and the source of neutrons could be effectively restricted and easily monitored.

XV-1.6.5. Technical features and technological approaches used to facilitate physical protection of 4S-LMR

The fact that planned inspection and maintenance could be very infrequent and the footprint of the nuclear island is very small, permits maximizing access restrictions. It should be possible to install remote monitors of the entire nuclear island so that unauthorized access can be identified through satellite monitoring. Planned maintenance and inspection may be conducted by the reactor supplier without the need for a user to invest in nuclear expertise.

XV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of 4S-LMR

The realization and use of the 4S-LMR, or any small reactor, faces economic challenges arising from the reverse economies of scale. A new approach to safety design and design approval and nuclear plant assembly appears necessary to reduce costs. Shop assembly of major components is needed that would support site installation times similar to those for combined gas-turbine cycle plants. Also, the safety design and site approval time must be considerably reduced from what now occurs with large plants. The safety design of the 4S is such that it reduces sensitivity to site conditions and with the appropriate approach to regulatory review, a standard design can be approved, which would reduce licensing costs.

There is also a need to recognize that while many countries may benefit from a more reliable supply of electric and thermal power with the 4S-LMR, it would also be necessary to reduce the incentive of these countries to develop a domestic nuclear fuel cycle infrastructure. This could be possible with the 4S-LMR but requires a change from the traditional approach, which was to establish of a full set of nuclear engineering technologies in a country that intends to buy and operate an NPP.

XV-1.8. List of enabling technologies relevant to 4S-LMR and status of their development

Some of the enabling technologies of the 4S-LMR that require further research and development (R&D) are outlined below:

Sodium-water reaction free SG

The problems remain in a sodium-water accident. In a near term design, a double-wall tube is introduced into the steam generator (SG); a plate type heat exchanger is proposed instead of the ordinary SG for an advanced design. This new type SG is similar to the double-wall-tube SG as comes to layer configuration; both SGs have three layers: sodium, gas and steam/water. Sodium and steam/water are separated by double boundaries; helium gas is generally used because of its good heat conductivity.

A plate type or a plate fin type heat exchanger is used in many industrial fields. The primary merit of this SG is its compactness compared to a tube-in-tube type SG. This design may also permit inclusion of the SG into the reactor vessel and elimination of the secondary sodium. Such a change would make the system more compact and reduce costs.

A heat exchange unit is fabricated by hot isostatic pressing (HIP) process; HIP is one of the diffusion bonding technologies and a process that uniquely combines pressure and temperature to produce materials and parts with substantially better properties than achievable by other methods. Rectangular tubes are bent into a plate-like case formed by the outer plates; the inner plates envelop the side of tubes. Hundreds of units of plate type heat exchangers may be azimuthally installed instead of an intermediate heat exchanger (IHX). Further R&D is needed to define how to assemble the parts for the HIP and how to perform pre-service inspection.

Core with a lifetime of 30 years

The present 4S-LMR core has a 10-year lifetime; therefore, whole core refuellings would be required during the plant lifetime, which is more than 30 years. A 30-year core lifetime would eliminate the need for refuelling during the plant lifetime and will better meet the requirements of an enhanced proliferation resistance. A larger size and better neutron economy are required to gain a 30-year lifetime while keeping the negative coolant void reactivity. An attempt is being made to develop a 30-year core design but it might be necessary to reduce the maximum power level from the 50 MW(e) target.

XV-1.9. Status of R&D and planned schedule

At the time of this report, the 4S-LMR design is being developed by CRIEPI through technical discussion with Toshiba (Japan) and with an information support from the Lawrence Livermore National Laboratory (LLNL) and Argonne National Laboratory (ANL) in the USA. Chubu Electric Power Company (Japan) has supported the design study for the 4S-LMR. The preliminary conceptual design has been completed. Development of a new, improved design with respect to core configuration and safety and development of some key technologies, such as the driving mechanism for the reflectors, are being conducted under a contract with the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan.

Several problems were identified to be resolved to enhance the feasibility of an earlier realization of the 4S-LMR project, related to the reliability of a reactor shutdown system including a reflector control system, and chemical activity of the secondary sodium systems. The total R&D costs, including the construction of a prototype reactor needed to obtain data for licensing of the commercial reactor is estimated at under US$1 billion, under the present design conditions. These costs may depend on the future design conditions and other factors.

The targeted timeframe for deployment is 2015.

XV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The 4S has a long-life core and an innovative core burn-up control system based on upward movement of the reflectors surrounding the core; these are essentially innovative features never applied in commercial power reactors before. A demonstration of these features in the operating prototype reactor would be required before the 4S-LMR can be licensed for commercial operation.

XV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The Encapsulated Nuclear Heat Source (ENHS) concept [XV-9] developed by the University of California at Berkeley (the USA) is a similar design concept.

XV-2. Design description and data for 4S-LMR

XV-2.1. Description of the nuclear systems

Reactor core and fuel design

The original 4S design of 1991 has undergone certain modifications; these modifications are summarized in Table XV-5 and graphically illustrated in Fig. XV-8.

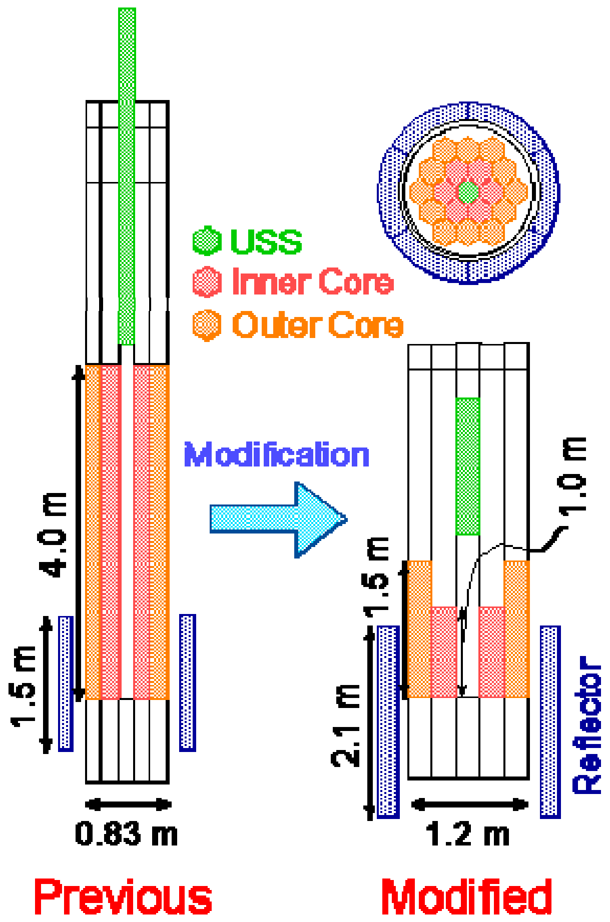
TABLE XV-5. EVOLUTION OF MAIN DESIGN SPECIFICATIONS OF THE 4S CORE

DESIGN	1991	2001
Thermal output, MW	125	135
Electric output, MW	50	50
Core diameter, m	0.84	1.2
Core height, m	4.0	1.0/1.5
Number of fuel assemblies	12	12
Number of reflector units	6	6
Reflector thickness, m	0.15	0.30
Number of fuel pins	217	469
Fuel pin diameter, mm	10.0	10.0
Cladding thickness, mm	0.50	0.59
Duct thickness, mm	3	2
Duct gap, mm	4	2
Bundle pitch, mm	182	258
Average burn-up, GW day/t	45	70
Pu enrichment, weight %	18.5/20.0	17.5/20.0
Conversion ratio (middle of cycle)	0.65	0.71
Coolant void reactivity (end of cycle), %ρ	-0.3	~0
Burn-up reactivity swing, %ρ	~8	~9
Core pressure drop, MPa	~0.2	~0.1

In the previous design, a specific problem was encountered related to the core height of 4 m causing difficulties in irradiation testing in consideration of the existing facilities. In 2001, the 4S-LMR design has been modified to include a metal fuel core with the height of less than 4 m.

Positive reactivity coefficient on coolant density is a characteristic strongly undesirable for sodium cooled fast reactors with a relatively low boiling point of the coolant. In general, the negative coolant density reactivity coefficient requires a large neutron leakage; on the other hand, a long core lifetime requires a high internal conversion, i.e., good neutron economy. These two requirements to a certain degree contradict each other.

To resolve this contradiction, metal fuel with its superior neutron economy and thermal conductivity characteristics was adopted for the new 4S-LMR design. Also, a new core configuration was selected with the core height different in the inner and outer core regions, Fig. XV-8. Such a configuration was shown to be optimal from the standpoint of providing a long core lifetime and ensuring negative sodium void reactivity coefficient.



(Previous: design of 1991; Modified: design of 2001)

FIG. XV-8. Core configurations of the 4S-LMR.

Reactivity control

The 4S reactor is designed to apply a reactivity control system with a movable annular reflector replacing the control rods and driving mechanisms, which traditionally require frequent maintenance. If applied, control rods would have to be replaced a number of times during the long core lifetime.

Vertical movement of the annular reflector during plant operation, including the start-up and shutdown, is the only mechanism for reactivity control provided for in the 4S-LMR. The reflector is installed inside the reactor vessel and heat generated in the reflector is removed by sodium.

The reflector is gradually lifted up to compensate for reactivity change due to fuel burn-up. Regular power operation is attained at a constant speed that is controlled according to the reflector differential reactivity worth. With no other reactivity control systems being used, the reactor thermal output drifts only by several percents during operation.

Figure XV-9 shows the averaged axial power profiles at several moments during the core lifetime. At the beginning of core life (BOC), a bare sub-critical core becomes critical by inserting reflectors to reduce the neutron leakage. The peak power is at a lower part of the core. As the core burns, the reflector is gradually lifted up to cover fresher fuel parts at the middle of core life (MOC). At the end of core life (EOC), the reflector is almost at the top of the core. Otherwise negative, the coolant density reactivity coefficient and the coolant void reactivity effect are approaching zero at the EOC.

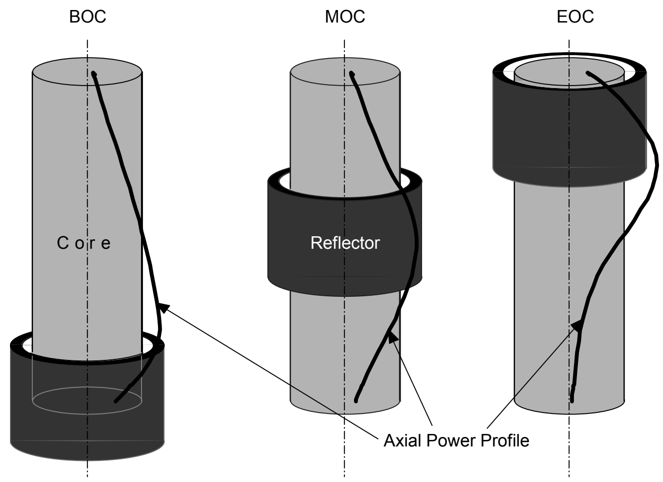


FIG. XV-9. Pattern of burn-up reactivity control by movable reflector.

The reflector drive mechanism consists of a hydraulic system that operates at the reactor start-up and shutdown and a ball screw system that connects a reflector to the motor that is actuated during normal operation (Figure XV-10). The mechanism has six driving systems corresponding to the six azimuthally separated reflectors. The six ball screw systems are fixed on a platform supported by the hydraulic system. For a reactor shutdown, the hydraulic pressure is released by opening scram valves to move the reflectors downward. The mechanical part of the reactor shutdown system has redundancy in that the platform is divided and the scram valves are set in parallel.

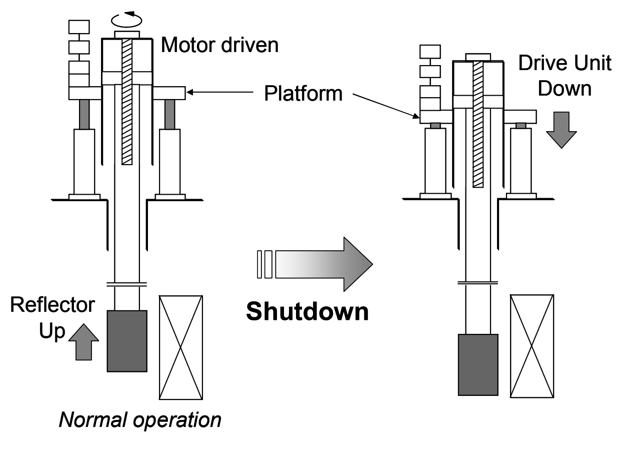


FIG. XV-10. Schematic of the reflector drive mechanisms.

Main heat transport system

A scheme of the 4S-LMR main heat transport system with the indication of heat removal path in normal operation and in accidents is given in Fig. XV-11. The reactor incorporates redundant passive decay heat removal systems. Specifically, a reactor vessel auxiliary cooling system (RVACS) is adopted in which the natural convection airflow removes the decay heat radiated through the guard vessel. The heat removal capability depends on the thermal radiation area. A specific (per thermal power) heat radiation area of small reactors is larger than that of medium sized or large reactors. It is expected that about 1% of the nominal power could be removed with the RVACS.

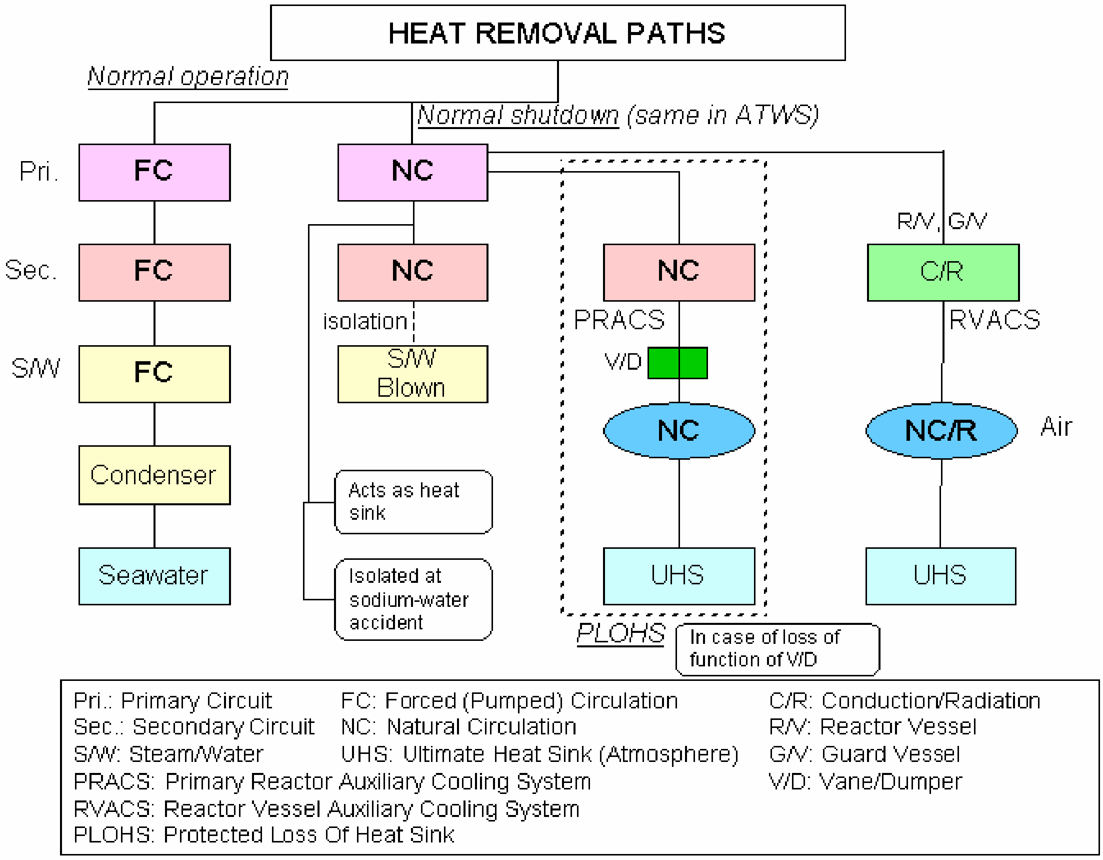


FIG. XV-11. Heat removal paths of the 4S-LMR.

XV-2.2. Description of the turbine generator plant and systems

In its present design, the 4S-LMR provides for the use of a standard turbine generator system; innovative turbine generator designs may be incorporated once developed.

XV-2.3. Systems for non-electric applications

There are no plans to use steam generated by the 4S-LMR for process applications; however, some or all of the electricity produced may be used to power a reverse osmosis plant for potable water production.

XV-2.4. Plant layout

The 4S-LMR plant layout is illustrated by Fig. XV-12. The area required to construct a reactor building including the turbine generator plant has been estimated as 25 m×50 m.

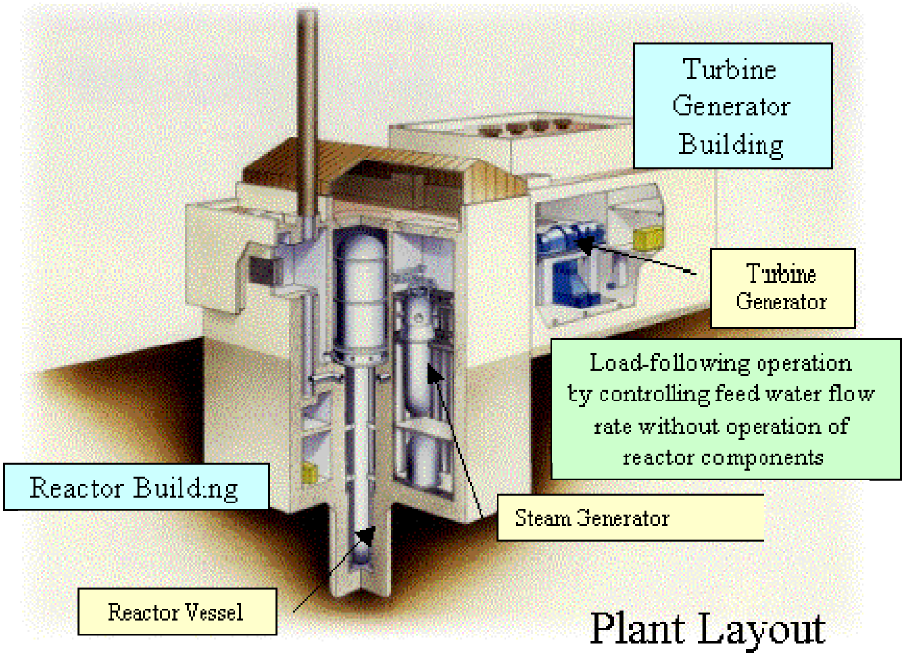


FIG. XV-12. Reactor building of the 4S-LMR (1991 design).

REFERENCES

[XV-1] UEDA, N., MINATO, A., HANDA, N., HATTORI, S., Super-safe, small and simple reactors for the global energy demand, FR'91 (Proc. of Int. Conf. on Fast Reactors and Related Cycles, Kyoto, Japan, Oct. 28–Nov. 1, 1991), Vol. IV, P5-7.

[XV-2] HATTORI, S., MINATO, A., Passive safety feature in 4S plant, ICONE-2 (Proc. of 2nd Int. Conf. on Nuclear Engineering, California, March 21-24, 1993), pp. 631–637, ASME.

[XV-3] HATTORI, S., MINATO, A., A large modular LMR power station which meets current requirements, ICONE-3 (Proc. of 3rd Int. Conf. on Nuclear Engineering, Kyoto, April. 23–27, 1995), S207-4, ASME-JSME.

[XV-4] TILL, C.E., CHANG, Y.I., Progress and status of the integral fast reactor (IFR): Fuel cycle development (Proc. of Int. Conf. on Fast Reactors and Related Fuel Cycles, Kyoto, Japan, Oct. 28-Nov. 1, 1991), Vol. I, P1-6.

[XV-5] UEDA, N., et al., Passive safety features in sodium cooled super-safe, small and simple reactor, ICONE-10 (Proc. of 10th Int. Conf. on Nuclear Engineering, Arlington, Virginia, US, April 14–18, 2002), ICONE-10-22354, ASME-JSME.

[XV-6] ENDO, H. et al., Safety features of fast reactors with metallic-fuelled core, Int. Fast Reactor Safety Meeting (Paper presented at Int. Meeting in Snowbird, Utah, USA, August 12–16, 1990).

[XV-7] WADE, D.C., Recent innovations in IFR safety research, ARS '94 (Paper presented at Int. Topical Meeting on Advanced Reactor Safety, Pittsburgh, PA, US, April 17–21, 1994).

[XV-8] LIU, L.L., et al., Behaviour of EBR-II Mk-V-type fuel elements in simulated loss-of-flow tests, Journal of Nuclear Materials, 204, pp.194–202 (1993).

[XV-9] Wade, D.C., et al., ENHS: the encapsulated nuclear heat source - a nuclear energy concept for emerging worldwide energy markets, ICONE-10 (Proc. of 10th Int. Conf. on Nuclear Engineering, Arlington, Virginia, US, April 14–18, 2002), ICONE-10-22202, ASME-JSME.

MODULAR PLANT WITH SODIUM COOLED FAST REACTOR (MBRU-12)

Experimental Design Bureau of Machine Building (OKBM),
Russian Federation

XVI-1. General information, technical features and operating characteristics

XVI-1.1. Introduction

The MBRU-12 is a modular nuclear power plant (NPP) with sodium cooled fast reactor. The name reflects the main engineering bases of the concept: a fast spectrum of neutrons, metallic (sodium) coolant, NPP assembly from factory-built equipment modules and a nominal value of electric power.

The MBRU-12 technical proposal was developed based on national experience in the design and operation of sodium cooled fast reactors with consideration of the world trends in nuclear power industry development.

The MBRU-12 plant of 12 MW(e) was developed in the 1990s by the Russian design and research organizations: Experimental Design Bureau, OKB Machine Building (OKBM), Nizhny Novgorod; Sankt-Peterburg Atomenergoproekt (SPb AEP); and State Scientific Centre Institute of Physics and Power Engineering (IPPE), Obninsk. It provides for cogeneration as a source of electricity and district heating or process steam or seawater desalination and could operate autonomously, as an energy source for house loads.

The main objective was to design a multi-purpose NPP based on a modular reactor installation with a sodium cooled fast reactor in support of economic, reliable and safe solutions to the issues of power and heat supply to consumers independent of large power systems.

The MBRU-12 design is based upon the following technical solutions verified in practice in the NPPs with fast sodium cooled reactors: BOR-60, BN-350, and BN-600, as well as during the design development of the BN-800 reactor [XVI-1 to XVI-4]:

- Coolant technology and structural materials for fast sodium cooled reactors;
- An integral design of the primary circuit;
- A three-circuit NPP layout with intermediate heat transport system (IHTS);
- The design of fuel elements, fuel assemblies and control rods;
- A passive system for emergency heat removal from the shutdown reactor;
- Fuel handling technology.

At this stage of preliminary design development for the MBRU-12, several options of core arrangement and reactor design were considered to enhance safety and economic effectiveness and to search for optimum solutions. One of key objectives in developing modifications was to facilitate transport of the reactor vessel and main NPP equipment by rail.

To enhance safety, the void reactivity effect was analyzed, and the selected small sized core ensured its non-positive value.

XVI-1.2. Applications

It is assumed that NPPs with the MBRU-12 fast reactors, along with electricity generation in the base load mode, can be used for the production of process heat, potable water or for district heating [XVI-5].

XVI-1.3. Special features

A NPP with the MBRU-12 reactor is characterized by the following innovative solutions aimed at improving the plant economy and providing a high safety level:

- Modular design of main plant equipment;
- The possibility of reactor operation without refuelling for the entire service life of the plant.

XVI-1.4. Summary of major design and operating characteristics

A NPP with the MBRU-12 includes one integral modular reactor (Fig. XVI-1), a two-loop intermediate circuit, and a steam-water circuit. Each loop of the intermediate circuit includes an intermediate heat exchanger, a steam generator, a circulation pump and the pipelines.

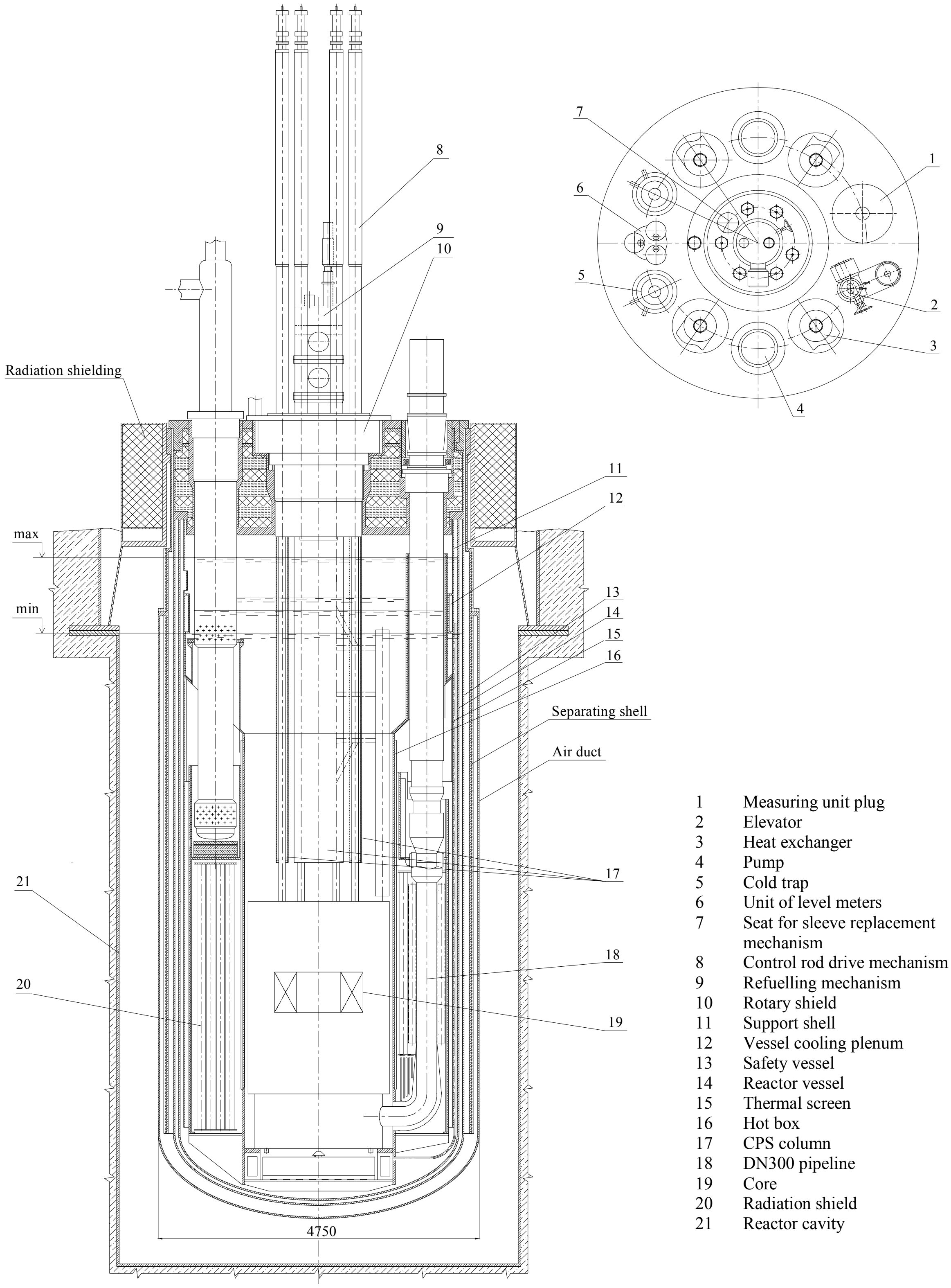
The reactor is located in a cavity with concrete walls and is covered with a leak-tight metallic liner. The liner above the reactor cover acts a leak-tight shell under which the control rod drive mechanisms (CRDMs), drives of the primary circuit pumps and the in-reactor refuelling mechanisms are located. Secondary circuit steam generators and the pipelines of each loop are arranged in individual rooms.

The plant emergency cooldown system is designed to remove residual heat to the ultimate sink (air) in case of initiating events such as NPP blackout or loss of feedwater supply to the steam generators. The emergency cooldown (aftercooling) system is a safety system and consists of passive elements, which do not require intervention of the operating personnel or actuation of automatic machinery to put them into operation.

Residual heat is removed from the reactor through the reactor vessel to atmospheric air at natural draught.

An integral arrangement of the reactor (Fig. XVI-2) was selected where primary circuit equipment is located in a single vertical vessel tank. The reactor vessel is enclosed in a full-strength safety vessel and has an upper support unit. The reactor cover is in the form of a flat ceiling and comprises a support for primary circuit equipment. The safety vessel has no thermal insulation to improve residual heat removal to the ultimate sink (air) at natural draught. The reactor vessel accommodates four intermediate heat exchangers and two main circulation pumps of the primary circuit. The core with a pressure chamber and a device to isolate core fragments and remove heat in case of beyond design basis accidents is installed on the vessel bottom. A multi-layer "hot box” installed on the pressure chamber is intended to divide areas of hot and cold coolant in the reactor.

The loading and unloading equipment of the plant provides for initial reactor loading with fresh fuel assemblies, unloading of spent fuel assemblies at the end of core life and shuffling of assemblies inside the core during the core life without opening the leak-tight shell.



1 Measuring unit plug
2 Elevator
3 Heat exchanger
4 Pump
5 Cold trap
6 Unit of level meters
7 Seat for sleeve replacement mechanism
8 Control rod drive mechanism
9 Refuelling mechanism
10 Rotary shield
11 Support shell
12 Vessel cooling plenum
13 Safety vessel
14 Reactor vessel
15 Thermal screen
16 Hot box
17 CPS column
18 DN300 pipeline
19 Core
20 Radiation shield
21 Reactor cavity

FIG. XVI-1. MBRU-12 module with equipment specification.

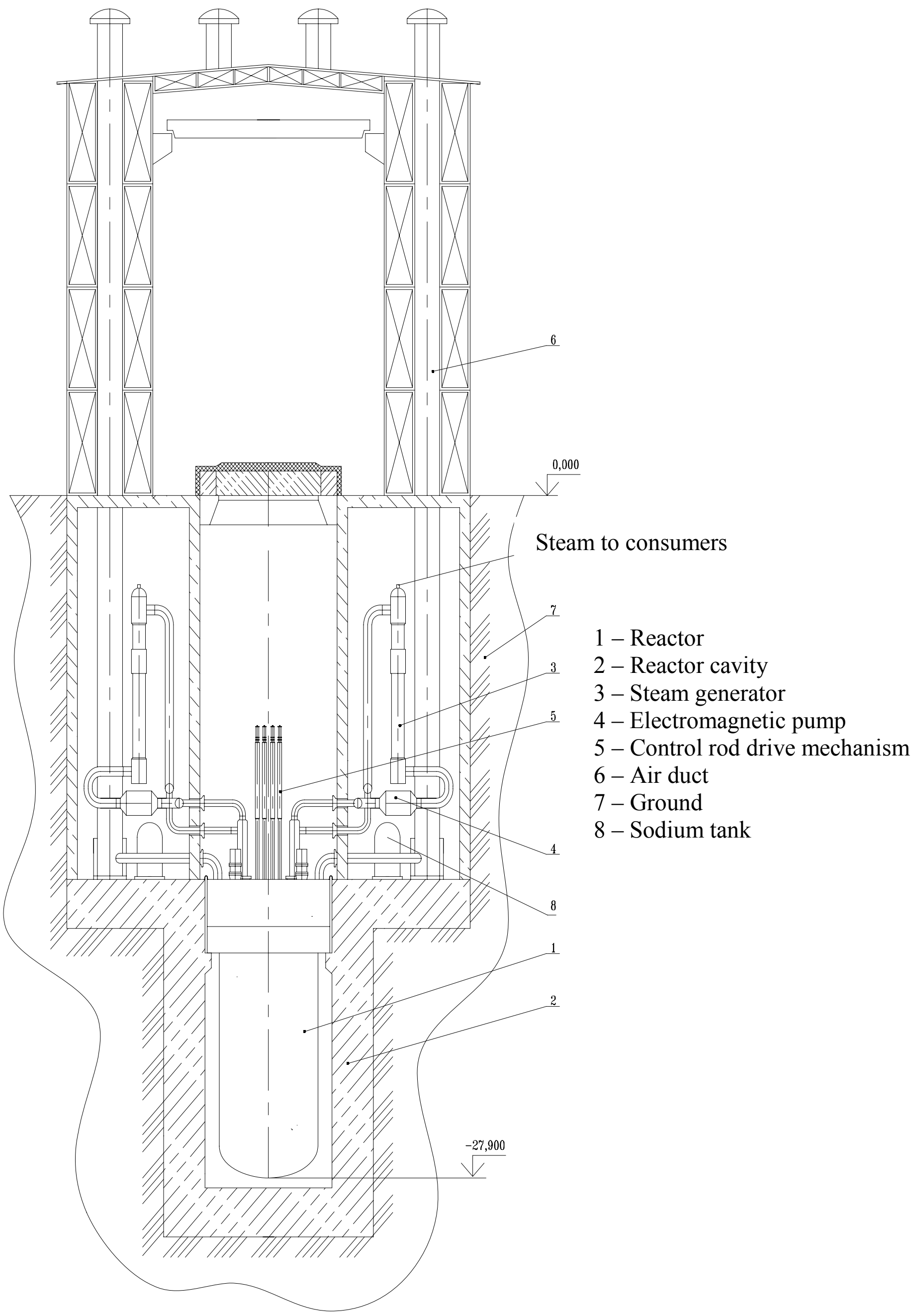


FIG.XVI- 2. General view of the MBRU-12 plant.

Major design and operating characteristics of the MBRU-12 are summarized in Table XVI-1.

TABLE XVI-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF MBRU-12

CHARACTERISTIC	VALUE/ DESCRIPTION
Installed capacity Thermal Electric	 48 MW 12 MW without heat take-offs
Fuel type	Cylindrical fuel elements with pellets of plutonium-uranium dioxide fuel in steel claddings
Enrichment	Plutonium content in oxide fuel: 28%
Coolant	Sodium
Moderator	None
Structural materials	Cladding of fuel elements: 1Cr13Mo2NbVB (EP-450) steel. Reactor structures: Cr18Ni9 austenitic stainless steel.
Reactor vessel	Cylindrical vessel with elliptical bottom and flat cover of the safety vessel. Dimensions of the main vessel (diameter × wall thickness × height, m): 4.5×0.03×18.
Number of circuits, type of thermal cycle	Three-circuit arrangement with sodium in the primary and secondary circuits and water-steam in the third circuit. Primary circuit has no loops. The secondary and circuit has two loops of equal power.
NPP style	Modular, integral primary circuit, loops in secondary and third circuit
Plant efficiency	25%
Capacity factor*	0.8
Mode of operation	Basic load
Design service life, years	30

* Capacity factor (CF)

The neutron-physical and operation cycle characteristics of the MBRU-12 are summarized in Table XVI-2; the reactivity control mechanism is described in Table XVI-3.

TABLE XVI-2. NEUTRON-PHYSICAL AND OPERATION CYCLE CHARACTERISTICS

CHARACTERISTIC	VALUE
Average specific core power, MW/m^3	47.4
Average fuel burn-up, MW·d/t	10^5
Maximum linear power in fuel assemblies, MW/m BOL EOL	 17.2 20.3
Sodium void effect	≤0
Reactivity variation with burn-up, %ΔK/K	0.44
Peaking factors (BOL/EOL) Axial Radial	 1.33/1.22 1.16/1.60
Interval between fuel shuffling, years	1
Core diameter, mm: Internal, at BOL; External, at BOL; Internal, at EOL; External, at EOL.	 790 1350 500 1190
Specified operation lifetime, years: Fuel assemblies of core and blanket; Control and protection system (CPS) rods.	 30 10

TABLE XVI-3. REACTIVITY CONTROL SYSTEM

ITEM	DESCRIPTION	FUNCTION/ VALUE
Types of reactivity control systems	Two independent protection systems with mechanical drives of reactivity control rods	Each system can shut down the reactor and maintain it in a subcritical state
Effectiveness of systems (BOL/EOL), % ΔK/K	First system (3 SR*) Second system (3 RR*)	3.1/2.5 2.3/3.4

* Regulating rod (RR)
* Safety rod (SR)

The thermal-hydraulic characteristics of the MBRU-12 are shown in Table XVI-4.

TABLE XVI-4. THERMAL-HYDRAULIC CHARACTERISTICS OF MBRU-12

PARAMETER GROUP	PARAMETER	VALUE
Circulation type	Forced circulation at power level; natural circulation (NC) for residual heat removal	
Circulation system parameters	*Primary circuit*	
	Sodium temperature at core inlet, C	330
	Sodium temperature at the inlet of intermediate heat exchanger, C	480
	Sodium flow rate, t/h	1044
	Sodium pressure at core inlet, MPa	0.2
	Pressure in gas cavity, MPa	0.05
	Secondary circuit	
	Sodium temperature at the inlet of intermediate heat exchanger, °C	280
	Sodium temperature at the outlet of intermediate heat exchanger, °C	460
	Sodium flow rate, t/h	1860
	Sodium pressure at pump discharge nozzle, MPa	0.5
	Third circuit	
	Feedwater temperature, °C	160
	Live steam temperature, °C	435
	Live steam pressure, MPa	9
Design limits	Temperature of fuel element cladding with account of uncertainty of parameters, °C	650
	Fuel temperature with account of uncertainty of parameters, °C	2500
	Vessel temperature, °C	450
	Temperature of in-reactor metallic structures, °C	600

Mass balances and flows of fuel materials for the MBRU-12 are given in Table XVI-5. Evaluations of the MBRU-12 economic characteristics are presented in Table XVI-6.

TABLE XVI-5. MASS BALANCES AND FLOWS OF FUEL MATERIALS

CHARACTERISTIC	VALUE	COMMENT
Fuel mass in the core, kg	4100	
Mass of fertile material, kg	11 784	Depleted UO_2 in axial, internal and radial blankets is also used to reduce burn-up reactivity swing
Total duration of fuel life, years	30	Based on fuel operation experience under irradiation
Interval between fuel shuffling, years	1	Selected to support long fuel lifetime and low peaking factors
Portion of fuel unloaded from the reactor	None	
Flow of fissile materials, kg/MW(th) per effective year	0.877	Related to one year of reactor operation at nominal power
Flow of natural uranium, kg/MW(th) per effective year	9.0	

TABLE XVI-6. ECONOMIC EVALUATIONS

METHOD/ CHARACTERISTIC	VALUE / DESCRIPTION	COMMENT
Method for evaluation of a specific capital outlay for construction	Through specific steel intensity of plant equipment, t/MW(e)	
Specific steel intensity of MBRU-12	56 t/MW(e)	Better than similar index for a NPP with light water reactor of 38 MW(th)
Other costs	Not evaluated at this design stage	

XVI-1.5. Outline of fuel cycle options

A specific feature of the MBRU-12 is long fuel residence time in the core coinciding with the plant service life.

Also, the MBRU-12 targets the possibility of being used as an element of the multi-component nuclear power system with optimized nuclide flows between the elements. Specifically, the MBRU-12 targets to use mixed uranium-plutonium fuel for core loading.

No detailed analysis of fuel cycle options for the MBRU-12 has been performed so far.

XVI-1.6. Technical features and technological approaches that are definitive for MBRU-12 performance in particular areas

XVI-1.6.1. Economics and maintainability

The specifics of the MBRU-12 is conditioned by the requirements of maximum autonomy and minimum maintenance. These requirements could be met by ensuring enhanced operating reliability in the base load mode, a high level of NPP safety and a high degree of physical protection. Considerable attention is also paid to decreasing the lump-sum capital investments.

Improvement of the MBRU-12 economic characteristics could be achieved due to the following concept-specific features:

(1) Through the reduction of dimensions and simplification of the equipment design in the MBRU-12 as compared with the BN-600 plant, a reduction of repair and maintenance and a considerable reduction in the number and time of NPP shutdowns to perform required maintenance is achieved, which could result in the increase of the capacity factor from 0.8 adopted for design analyses, to 0.95;
(2) The reactor and building design offer the possibility of removing the reactor module for disposal at the end of operation lifetime and installing a new module in its place. This allows the use of buildings and structures for a longer period than usual, with the further possibility of operation or replacement of modules of other thermal equipment;
(3) A considerable reduction in cost for the NPP could be achieved through maximum prefabrication and delivery of modules to a site in an assembled form, to reduce the scope and cost of mounting activities at the NPP site;
(4) Cost reduction would be facilitated by mass production;
(5) A reduction in cost for emergency power supply systems is achieved, since emergency cooldown of the reactor module is performed only by passive systems, relying on self-controlled natural circulation of air to which decay heat is removed through the reactor vessel.

XVI-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Fast reactors including the MBRU-12 could solve the problem of providing a sustainable power supply for a long time due to the possibility of transmuting uranium-238 to plutonium-239. The MBRU-12 is characterized by a high level of fuel utilization achieved by the introduction of the internal, radial and axial blankets for breeding of fissile materials.

The MBRU-12 design targets to meet requirements for the integrity of fuel elements in normal operating modes and in accidents (which is facilitated by moderate temperature levels in the core), ensuring a negligible level of irradiation dosage on the environment. The underground location of a NPP with the MBRU-12 and a small outage period for maintenance also contribute to achieving this target.

Evaluations of the operating experience of fast sodium cooled reactors show that for the MBRU-12, radiation impact on the environment could be as follows.

Activity release into the atmosphere through ventilation is conditioned by a primary circuit gas leak and activation of reactor cavity cooling air. During normal plant operation, the gas activity released into the atmosphere does not exceed 2.5×10^{10} Bq/day, which is much less than the allowable level of 1.9×10^{12} Bq/day [XVI-6].

More detailed evaluations of the radiological consequences of radioactivity release from the MBRU-12 were not performed.

XVI-1.6.3. Safety and reliability

Safety concept and design philosophy

The approach to the MBRU-12 safety design is based on the concept of retaining radionuclides in the fuel during normal operation and in emergency modes, so that radiation impact on personnel and in the NPP area is within the allowable limits. In this, the dose limits for the MBRU-12 design are set well below the current regulations.

The NPP arrangement in the cavity is also targeted at the enhancement of safety.

Active and passive systems and inherent safety features

The MBRU-12 design targets finding an optimum combination between the reliance on by-design and inherent safety features and the application of engineering (active and passive) systems to ensure a high level of plant safety.

Specifically, it has been shown that the passive emergency cooldown system allows the complete removal of residual heat in case of SG isolation at temperatures in the reactor not exceeding nominal values.

Structure of the defence-in-depth

The defence-in-depth protection provides for multiple barriers preventing radioactivity release from the fuel and measures to maintain the integrity of the barriers. This structure of barriers relies mainly upon the known properties of fuel to retain large amounts of radionuclides in the active fuel part and to prevent radionuclide release to the coolant. Barriers for fission product release are mainly claddings of the fuel elements. Additional barriers preventing radioactivity release to the environment are the reactor vessel, safety vessel, secondary circuit and the reactor cavity.

Design basis accidents and beyond design basis accidents

At this stage of design development, the list of accidents for the MBRU-12 was adopted based on the operating experience of existing NPPs and is not yet final. Probabilistic analysis of the entire spectrum of potential initiating events has not been performed so far.

The MBRU-12 safety level was analyzed using the example of two beyond design basis accidents:

- A NPP blackout with failure of all CPS rods;
- Leaks in the main and safety vessels.

The first accident progresses according to the scenario shown in Fig.XVI-3.

NPP operation at nominal power

↓

NPP blackout

↓

Failure of generation of the emergency protection signal

↓

Non-actuation of CPS rods

↓

Coastdown of primary and secondary circuit pumps
with shutdown in ~100 s

↓

Heat removal from the core is accomplished only through the reactor vessel surface to atmospheric air

FIG. XVI-3. Accident with NPP blackout and failure of all CPS rods.

The analysis of such an accident explicitly illustrates the self-protection features of the MBRU-12. Due to the implementation of negative reactivity feedbacks, the reactor power is reduced to the level of residual heat with sodium temperature at the core outlet increasing only up to 750°C.

Residual heat is removed only through the reactor vessel surface to the atmospheric air. In this, the reactor is cooled down without sodium boiling and the temperature of the main reactor vessel does not exceed 700°C.

The effective individual dose of irradiation does not exceed 3.7 mSv per accident beyond the radius of 1 km. Aerosol activity release does not exceed the allowable average monthly release for normal operation. Therefore, radiation consequences do not exceed the limit of 5 mSv for the first year after the accident, as prescribed by [XVI-7]. This excludes measures for population protection beyond the control area.

The second accident progresses according to the scenario shown in Fig. XVI-4.

NPP operation at nominal power

↓

Depressurization in the lower part
of the main reactor vessel

↓ *0.2 hours*

Failure of leak detectors

↓ *9 hours*

FEP* actuation upon sodium level reduction signal

↓ *5 hours*

Maximum pressure is achieved in the
safety vessel and depressurization occurs in its lower part

↓ *2 hours*

If there is a sodium circulation disruption in the primary circuit, sodium is heated in the reactor to 550°C. Core power is removed through casings of the intermediate heat exchangers to the circulating sodium of the secondary circuit, as well as through surfaces of the main and safety vessels until the cavity concrete equals them in temperature.

↓ *150 hours*

Leakage stops after pressures in the reactor, safety vessel and reactor cavity equalize.

FIG. XVI-4. Accident with leaks in main and safety vessels.

* Fast emergency protection

The analysis of radiation consequences has taken into account full release of gases from the sodium exiting the main vessel, release of gaseous fission products to the atmosphere through the ventilation, and release of sodium combustion products from the cavity through leaks of the isolating valves of the air-based emergency cooling system and then through the exhaust tube to the atmosphere.

The effective irradiation dose per person per accident is not more than 13 μSv, at a distance of more than 1 km from the source of radioactive release. This is much lower than in the first accident.

The abovementioned examples of beyond design basis accident scenarios, which were supported by tests, demonstrate a high level of the MBRU-12 self-protection being achieved through the inherent safety features of the core, thermo-physical properties of the sodium coolant, the use of natural circulation to organize emergency core aftercooling, and the use of passive systems in combination with traditional active systems.

Provisions for safety under seismic conditions

For the MBRU-12, safety under seismic impacts is to be ensured by following standard procedures prescribed by the regulatory documents.

It is noteworthy to mention that strength analysis performed for the BN-800, a reactor in many ways similar to the MBRU-12 but having the mass and size that considerably exceed those considered for the MBRU-12, has confirmed plant seismic stability at a maximum design earthquake of 7 points per the MSK scale.

Probability of unacceptable radioactivity release beyond the plant boundaries

The targeted probability of unacceptable radioactivity release beyond the plant boundary is less than 10^{-7} per reactor per year.

Measures planned in response to severe accidents

For the MBRU-12, the following main safety criteria are adopted, which could facilitate reduction or elimination of measures for population protection in accidents:

- Radioactive releases during normal operation and the respective effective population irradiation doses shall be at least by an order of magnitude less than the currently enforced dose limit [XVI-7], which amounts to 1 mSv/year per 5 years on average;
- In case of design basis accidents, the effective irradiation dose for population on the boundary of the exclusion area and beyond it shall not exceed 1 mSv per the first year since the accident. This corresponds to the limit of the dose established for normal operation;
- In case of beyond design basis accidents, the effective irradiation dose for population on the boundary of the exclusion area and beyond it shall not exceed 5 mSv per the first year since the accident;
- The assessed value of the probability of beyond design basis accidents with the above dose limits shall not exceed 10^{-7} for the reactor per year;
- The assessed value of the probability of severe core damage and melt shall not exceed 10^{-7} for the reactor per year. This is 10 times less than the limit defined in [XVI-8].

In addition to the abovementioned safety criteria, a number of additional measures to enhance safety are considered in the MBRU-12 design, among them:

- The provisions for effective management of the beyond design basis accidents, including long grace periods and appropriate technical means of control;

– A hypothetical accident with core destruction and melting would be considered deterministically, with no reference to its very low probability. The radiation consequences of this accident with an account of isolation safety systems shall prevent the need to evacuate population beyond the NPP boundaries;
– A core catcher would be provided in the lower part of the reactor vessel, ensuring the isolation and cooling of destructed core fragments in a hypothetical situation of core melting, to prevent the possibility of forming critical configurations from fallen fuel.

XVI-1.6.4. Proliferation resistance

The problems related to a potential proliferation of fissile materials from fast reactors (including the MBRU-12) are conditioned by the capability of surplus fuel breeding using fuel assemblies with fertile material for this purpose. Besides, fuel assemblies of fast reactors themselves can include breeding inserts, which do not noticeably worsen the working characteristics of the reactor.

The components of proliferation resistance are structured in two basic groups:

- Intrinsic features - integral properties of nuclear power systems (including plants, materials, applied technologies) where the reactor is operated;
- Extrinsic measures -organizational and supporting technical measures.

During conceptual design development of the MBRU-12, attention was paid to the issues relevant for both abovementioned groups.

The following factors were identified as potentially being among the intrinsic proliferation resistance features of the MBRU-12:

- The fuel loaded in the reactor prior to operation is not removed from the reactor vessel because core life coincides with the specified service life of the reactor right up to the NPP decommissioning;
- The NPP arrangement in the cavity eases the task of physical protection;
- The design of the fuel assemblies allows the following:
  — Measurement of nuclear materials prior to loading them into the reactor;
  — Fuel assembly cutting is performed after the fuel assemblies are withdrawn from the reactor, which could enhance the effectiveness of the IAEA control and verification, since in this way it is possible to measure all fuel elements and exclude the possibility of an undeclared replacement of intermediate rows of fuel elements for fuel elements with fertile material. With this approach, the accounting unit is an individual fuel element and not a fuel assembly;
- It is proposed to use non-aqueous methods of fuel reprocessing. Non-aqueous technologies developed in Russia and related to the processing of fuel from fast reactors with incomplete purification from fission products (about 1% of them remain in refabricated fuel) and with the release of only curium from the fuel (neptunium and americium and 1% of curium remain) allow the production of such fresh fuel for fast reactors that cannot be directly used to create nuclear weapons [XVI-9].

It should be noted that the abovementioned consideration is preliminary and that further confirmation of the MBRU-12 intrinsic proliferation resistance features should be pursued through analysis of the reactor performance within a selected nuclear energy system.

Obviously, the least intrinsically protected nuclear fuel cycle (NFC) stages involving the MBRU-12 are factories for fuel assembly fabrication and the operations of transport of both fresh and spent fuel assemblies. These would require intensified extrinsic measures, such as accounting, monitoring, physical protection and national and international inspections.

XVI-1.6.5. Technical features and technological approaches used to facilitate physical protection of MBRU-12

To enhance physical protection of the MBRU-12, the plan is to arrange the NPP in an underground cavity.

Transients started by initiating events during the NPP operation run rather slowly at a considerable heat capacity of the primary circuit and passive heat removal from the reactor vessel. This offers the chance to take timely measures for accident management.

Operation of passive systems is based on physical laws, substantially reducing the possibility of failure by premeditated personnel actions.

XVI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of MBRU-12

At this stage of the MBRU-12 design, specific requirements and market needs (especially with respect to developing countries) were not taken into consideration. It is anticipated that full-scope fuel cycle service agreements could be offered within an appropriate structure of nuclear energy systems with the MBRU-12.

XVI-1.8. List of enabling technologies relevant to MBRU-12 and status of their development

The MBRU-12 makes an extensive use of many technical solutions already confirmed by the operating experience of fast sodium cooled reactors.

The technologies that require further R&D and validation are:

- Passive cooldown (aftercooling) concept:
  - Currently there is no experimental validation of the concept of residual heat removal through the reactor vessel to the atmospheric air;
  - Tests are necessary to investigate the processes of core residual heat transfer to the reactor vessel and then to atmospheric air for normal and emergency operation conditions of the MBRU-12;
- Core operation without refuelling during the entire reactor service life:
  - Reliable experimental data on operability of the fuel elements with a specified service life in reactor conditions are needed.

The economic viability of the concept considerably depends on the capability of the fuel composition to provide a relatively high burn-up. By now, moderate parameters have been achieved in tests and operation; therefore, additional tests and post-irradiation examinations are required to validate higher fuel burn-up.

XVI-1.9. Status of R&D and planned schedule

The factors that could support development and implementation of the MBRU-12 in reasonably short terms are related to the fact that this reactor makes an extensive use of many technical solutions already confirmed by the operating experience of fast sodium cooled reactors. In addition to this:

- The qualified personnel for the design development of the MBRU-12 is available;
- The industrial base for fabrication of the entire complex of the MBRU-12 equipment and systems is established;
- The infrastructure to perform R&D in support of the innovative solutions of the MBRU-12 is in place.

As of 2004, conceptual studies of the MBRU-12 project were performed on the initiative of specialists of the OKBM (Nizhny Novgorod, Russia).

During recent years (1997-2003), activities for the MBRU-12 were also stimulated by the exchange of scientific and technical information with companies in the Russian Federation and abroad currently developing concepts of sodium cooled fast reactors. Among them, mentioned should be the Ministry of Atomic Industry of Kazakhstan.

The MBRU-12 development is at a conceptual design stage and does not yet allow for the specific definition of companies and institutions that could be involved in further R&D. The time frames for future activities are not defined also.

An optimum approach to further development of the MBRU-12 concept could be to join the efforts of Russian enterprises such as SPb AEP, OKBM and IPPE. Previously such cooperation produced detailed designs of power units based on the fast sodium cooled reactors BN-600 and BN-800.

XVI-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The MBRU-12 concept incorporates many known technical solutions used in its predecessors (BOR-60, BN-350, BN-600 and BN-800).

The MBRU-12 concept is rated innovative because it incorporates several design approaches differing considerably from those used in the present-day sodium cooled reactors.

The design features of the MBRU-12 that require demonstration in a prototype plant are the following:

- Concept of emergency residual heat removal based on passive principles only;
- Core operation without refuelling during the entire service life of the reactor.

It is clear that additional experimental studies of individual phenomena and processes are necessary to validate solutions for a small sized MBRU-12. However, from the history of fast sodium cooled reactors it is known that problems related to fuel and coolant were often solved through large-scale experimental studies and by taking into account the operation experience of a number of consequently built plants.

Individual innovative features and their combinations could be validated and demonstrated to a full extent on mock-ups and prototypes. Construction of a pilot full-scale plant may be required to make final decisions on the selection of operating parameters and modes.

XVI-1.10. List of other similar or relevant SMRs for which the design activities are ongoing

No information was provided.

XVI-2. Design description and data for MBRU-12

XVI-2.1. Description of the nuclear systems

Reactor core and fuel design

Direct prototypes of the MBRU-12 core are core designs of the fast sodium cooled reactors: BOR-60, BN-350, BN-600, and BN-800.

A large scope of the performed R&D and experience obtained during the operation of other fast sodium cooled reactors facilitated adoption of the following solutions for the MBRU-12 core:

- Use of the mixture of uranium and plutonium oxides as fuel;
- Use of the core arrangement with an inner blanket, containing depleted uranium dioxide.

Core operation without refuelling for the entire service life of the NPP (30 years) is adopted. Fuel assemblies are shuffled in the core at 1-year intervals. In case of fuel element depressurization, there is a possibility of replacing a defective fuel assembly by one of several ‘reserve’ fuel assemblies with fresh fuel, which can be installed behind the radial blanket.

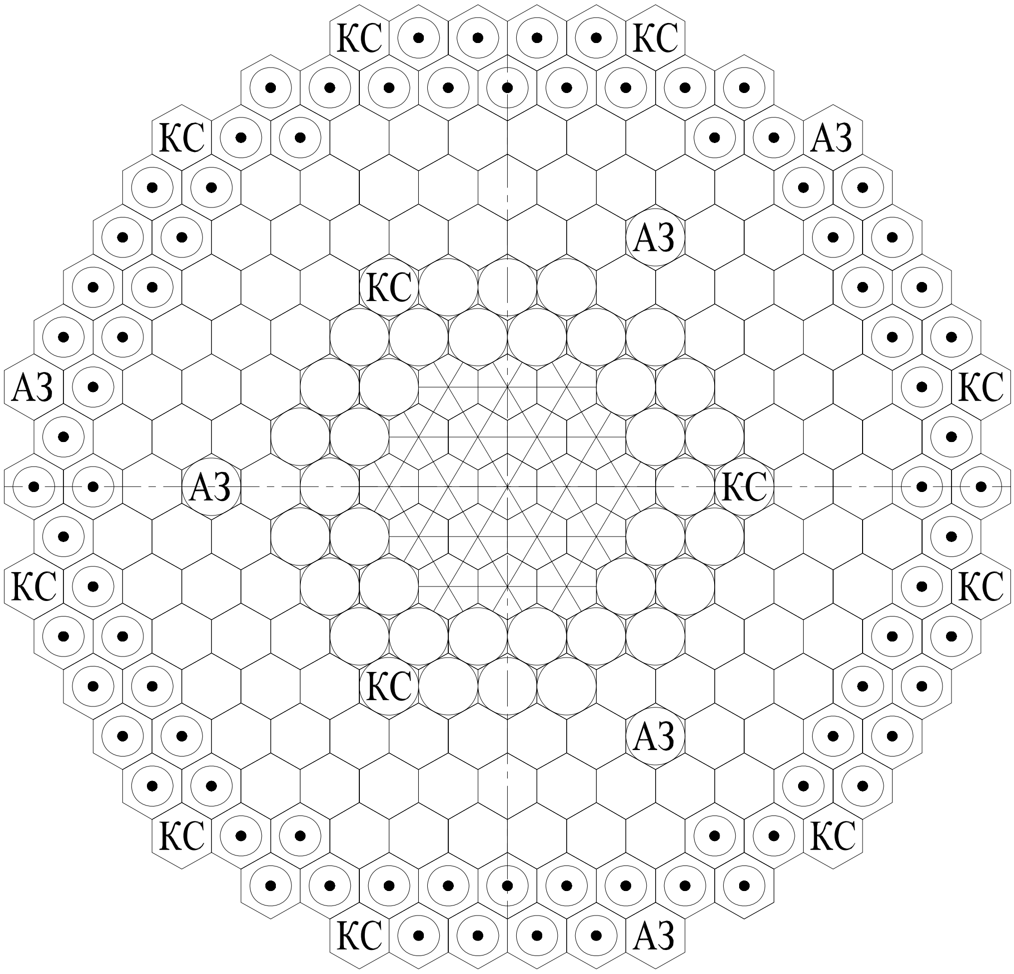
The core is annular and grouped concentrically; its central part is composed of steel assemblies and fertile fuel blanket assemblies (Fig. XVI-5). At the end of life, the core configuration is changed, with its external diameter being reduced; only steel assemblies remain in the centre, and the thickness of the external fertile blanket is increased (Fig. XVI-6). At shuffling, the reactivity margin is made up for the next interval of 1 year.

The design of a core fuel assembly is shown in Fig. XVI-7. The fuel assembly design data are given in Table XVI-7.

TABLE XVI-7. FUEL ASSEMBLY DESIGN DATA

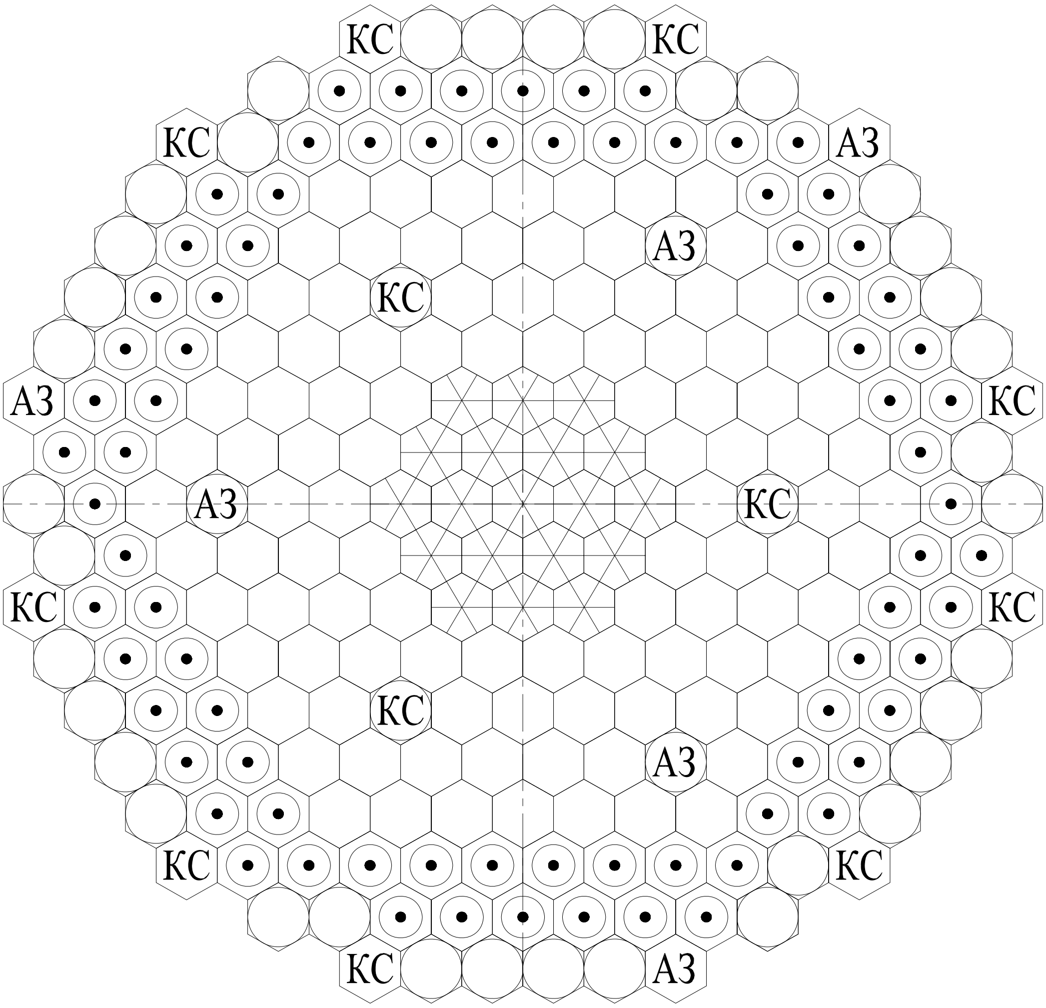
CHARACTERISTIC	VALUE
Width across flats of the fuel assembly wrapper, mm	96
Number of fuel elements in fuel assembly	37
Fuel element diameter, mm	14
Active part height, mm	1000
Height of upper and lower axial blankets, mm	2×300
Fuel assembly length, mm	3500

The fuel assembly of a blanket has the configuration similar to that of a core fuel assembly. The height of the fertile material column is 1600 mm.



Symbol	Description	Quantity, pcs.
	Core FA	102
	FA of internal breeding blanket	36
	FA of external breeding blanket	78
	Steel assembly	19
А3	Safety rod	3
КС	Ragulating rod	3
А3	Safety rod (reserve)	3
КС	Regulating rod (reserve)	9

FIG. XVI-5. Core configuration at BOL.



Symbol	Description	Quantity, pcs.
	Core FA	102
	FA of internal breeding blanket (after refueling)	36
	FA of external breeding blanket	78
	Steel assembly	19
АЗ	Safety rod	3
КС	Ragulating rod	3
АЗ	Safety rod (used)	3
КС	Regulating rod (used)	9

FIG. XVI-6 Core configuration at EOL.

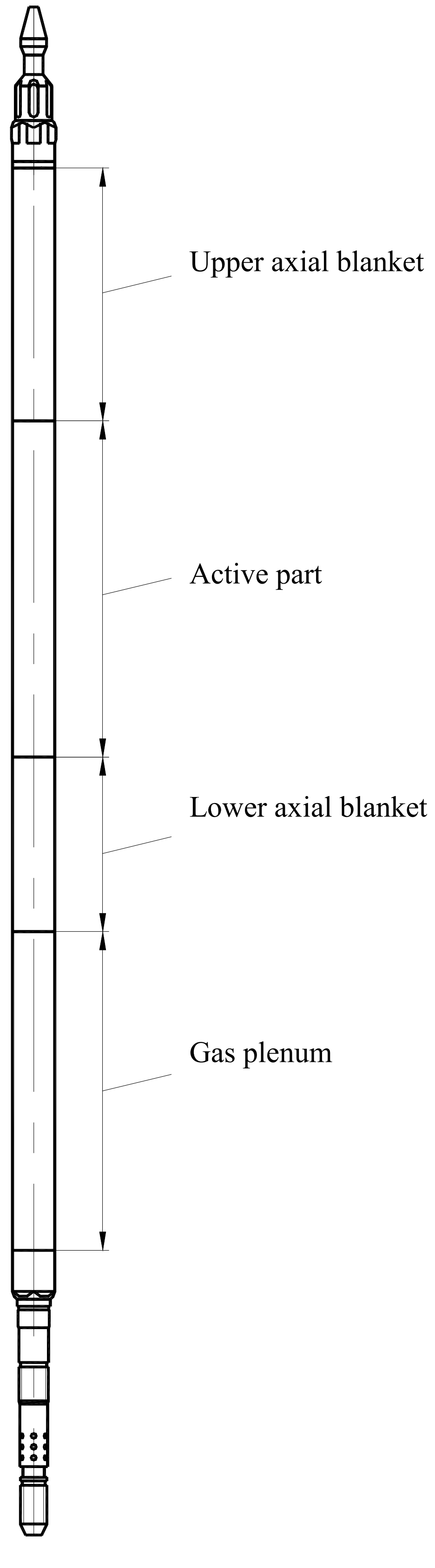


FIG. XVI-7. Core fuel assembly.

Reactor control and protection system (CPS)

According to Russian regulations, two independent and diverse protection systems shall be provided. The reactor uses two groups of reactivity control rods with mechanical drives of different types. Each of these systems can shut down the reactor and maintain it in a sub-critical state from any nominal or emergency state, provided that the most effective rod does not actuate.

The design of absorber rods is similar to those developed for the BN-800 reactor.

Three rods are intended for reactor emergency shutdown; other three rods compensate for reactivity effects and provide power control.

To ensure a 30-year period of core operation without refuelling, the plan is to replace CPS rods. as their 10-year lifetime expires, for redundant rods located in cells of the radial blanket. The reactivity balance in refuelling of the core and blanket fuel assemblies is given in Table XVI-8.

TABLE XVI-8. REACTIVITY BALANCE DURING REACTOR REFUELLING

BALANCE COMPONENT	VALUE, %ΔK/K (BOL/EOL)
Maximum reactivity margin	1.2/2.2
Total worth of CPS rods	5.4/5.9
Sub-criticality level of shutdown reactor	4.2/3.7

Systems for core power control by neutron flux measurement are traditional for BN type reactors.

Reactor module and primary circuit systems

The reactor module has an integral arrangement (Fig. XVI-2). All primary circuit systems, including the core, intermediate heat exchanger, circulation pumps of the primary circuit and filters for coolant purification from oxides are arranged in a double cylindrical vessel. Dimensions of the main and safety vessels are ∅4100×25 mm and ∅4250×25 mm, respectively; vessel height is about 18 m. The gap between the main and safety vessels is filled with pressure- regulated gas. The gap is also intended to heat up the reactor module prior to a start-up by the forced circulation of hot gas. In addition, the safety vessel provides isolation of the radioactive coolant in case of main vessel depressurization.

The safety vessel is surrounded by an additional vessel of the emergency cooldown system. A shell is installed in the gap between these vessels. This shell separates it into riser and downcomer channels where natural circulation of atmospheric air takes place, see Fig. XVI-2. The shell separating the gap into channels has very high thermal resistance. The channels are connected with the exhaust and input tubes. Nominal characteristics of the emergency cooldown system are given in Table XVI-9.

TABLE XVI-9. CHARACTERISTICS OF EMERGENCY COOLDOWN SYSTEM

PARAMETER	VALUE
Removed power, MW(th)	0.347
Air flow rate, kg/s	3.4
Air temperature, °C - At the inlet; - At the outlet	 From "minus" 50 to 50 143
Nominal hydraulic resistance, Pa	140
Service life, years	30

The reactor module does not contain external pipelines with primary coolant.

Main heat transport system

Figure XVI-8 shows the heat transport path from the core to the ultimate sink during normal operation and under emergency conditions.

The normal heat removal system is based on a three-circuit design and includes a loop-less (pool type) primary circuit in the reactor module, two equivalent loops of the intermediate sodium circuit, two loops of the steam-water circuit and the turbo-generator facility. During normal operation, heat released in the core, including residual heat release of the shutdown reactor, is transferred to the steam-water circuit. Steam can be taken off from the third (steam-water) circuit for process applications and/ or district heating.

Intermediate circuit

Apart from the reactor module, the MBRU-12 NPP includes:

- An intermediate sodium circuit with steam generators,
- A control and monitoring system,
- An in-reactor refuelling system,
- A system of gas heating for the reactor vessel,
- A primary coolant filling and drainage system,
- A primary coolant purification system and other systems supporting the NPP operation.

The option of arranging the reactor module and main equipment of the intermediate sodium circuit in separate cavity compartments located below the ground surface was considered.

Some design characteristics of the intermediate and third circuit are given in Table XVI-10.

XVI-2.1. Description of the turbine generator plant and other systems

No detailed information was provided, except for the data given in Tables XVI-4 and XVI-10.

TABLE XVI-10. DESIGN DATA FOR INTERMEDIATE AND THIRD CIRCUITS

CHARACTERISTIC	VALUE
Secondary (intermediate) circuit	
Number of loops	2
Sodium flow rate, kg/s	207
Temperature at the steam generator inlet, °C	460
Temperature at the steam generator outlet, °C	280
Third (steam-water) circuit	
Reheater type	Steam
Number of loops	1

XVI-2.3. Systems for non-electric applications

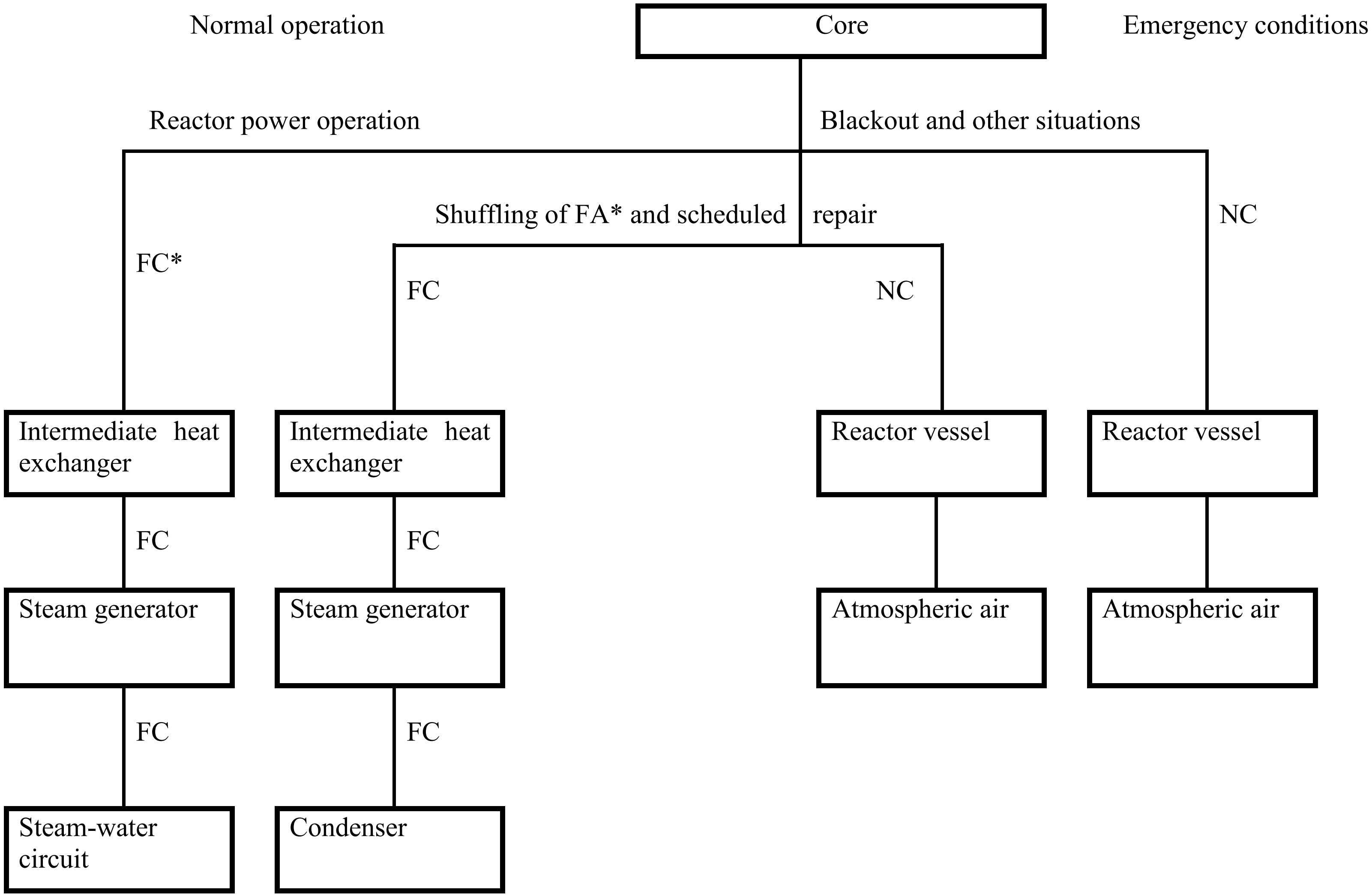
At this design stage, the systems for non-electric applications of the MBRU-12 were not considered in detail. Experience in the design of relevant systems for close parameters in the steam-water circuit (e.g., for the BN-350) makes it possible to conclude that the MBRU-12 could be adjusted for a cogeneration with seawater desalination and/ or district heating without any design-specific problems [XVI-3].

XVI-2.4. Plant layout

At this design stage, the general plan of a NPP with the MBRU-12 was not drawn up. Some considerations for the arrangement of the NPP systems and buildings are outlined in Table XVI-11.

TABLE XVI-11. DESIGN AND ARRANGEMENT OF MBRU-12 PLANT

NPP COMPONENTS	DESIGN AND ARRANGEMENT
Primary circuit	Modular integral reactor in the underground cavity
Secondary (intermediate) circuit, including steam generator (SG)	Modular SG in the underground cavity
Third (steam-water) circuit with turbo-generator	Ground based building
Automatic process control system	Ground based building
Turbo-generator compartment	Ground based building



* Fuel assembly (FA)
* Forced circulation (FC)

FIG. XVI-8. Heat removal path during normal operation and under emergency conditions.

REFERENCES

[XVI-1] BUKSHA, YU.K. et al., Operation experience of the BN-800 fast reactor, Nuclear Engineering and Design 173, pp. 67–79 (1997).

[XVI-2] KIRYUSHIN, A.I., et al., BN-800 - next generation of Russian sodium fast reactors, ICONE-10 (Proc. 10th Int. Conf. on Nuclear Engineering Arlington, VA, USA, April 14–18, 2002) ASME.

[XVI-3] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Liquid Metal Cooled Fast Reactor Technology, IAEA-TECDOC-1083, Vienna (1999).

[XVI-4] Safety of Beloyarsk NPP operation (collection of scientific proceedings), Ekaterinburg, Russian Federation: UrO RAN, ISBN 5-7691-0469-4 (1994).

[XVI-5] MITENKOV, F.M., et al., Use of missile shafts for NPP arrangement (Paper presented at Int. Conf. on Safety of Sodium-Cooled Fast Reactors, Russia, Obninsk, October 3–7, 1994).

[XVI-6] Sanitary rules for design and operation of NPP (SP AS-03), SP 2.6.1.24-03, Moscow, Minzdrav of Russia (2003).

[XVI-7] Radiation Safety Codes (NRB-99), SP 2.6.1.758-99, Moscow, Minzdrav of Russia, ISBN 5-7508-0040-7 (1999).

[XVI-8] General Principles of Safety Provision for NPPs, OPB-88/97. NP-001-97 (PNAE G-01-011-97). Moscow, Gosatomnadzor RF (1997).

[XVI-9] RUMYANTSEV, A., et al., Comparison of the wide-scale nuclear power development concepts with expanded fuel breeding and without it from the viewpoint of minimizing the proliferation risk (Paper presented at Int. Workshop on Methodologies of Quantitative Assessment of Nuclear Fuel Cycle Proliferation Resistance, June 2003, Obninsk, Russia).

WHOLE-CORE REFUELLED INTEGRAL DESIGN SMALL SODIUM COOLED FAST REACTOR (RAPID)

**Central Research Institute of Electric Power Industry (CRIEPI),
Japan**

XVII-1. General information, technical features, and operating characteristics

XVII-1.1. Introduction

RAPID (Refuelling by All Pins, Integral Design) is the abbreviation for a small sodium cooled reactor of 10 000 kW(th) (1000 kW(e)) with U-Pu-Zr metal fuel and fast neutron spectrum [XVII-1 and XVII-2]. It is one of the successors of the RAPID-L [XVII-3 to XVII-7] - the operator-free fast reactor concept designed for a lunar based power system.

The technical basis for the RAPID includes general experience with sodium cooled fast reactors. Specifically, the RAPID concept includes no control rods but incorporates the passive lithium expansion modules, lithium injection modules and lithium release modules to enable an operator-free operation mode. These systems utilize ^{6}Li as a liquid poison instead of B_4C rods. To verify the reactivity worth of ^{6}Li, the criticality test [XVII-5] using the fast critical assembly (FCA) of the Japan Atomic Research Institute (JAERI)* has been conducted. Also, the manufacturing technology of the lithium modules was mastered, and the performance and neutron radiography tests of the lithium expansion and lithium injection module pilots were conducted.

Central Research Institute of Electric Power Industry (CRIEPI), Tokyo, Japan, is developing the RAPID concept.

XVII-1.2. Applications

The RAPID is designed to supply electricity and potable water in remote areas, disconnected from electricity grids.

XVII-1.3. Special features

The RAPID is an operator-free factory fabricated and fuelled reactor with an infrequent refuelling interval of 10 years.

XVII-1.4. Summary of major design and operating characteristics

An essential feature of the RAPID concept [XVII-8] is that the reactor core consists of an integral fuel assembly rather than conventional subassemblies. In this small sized reactor core, 14 000 fuel pins are integrated and encased in a fuel cartridge. Refuelling is accomplished by replacing the whole fuel cartridge. The reactor can be operated without refuelling for up to 10 years.

Unique challenges in the design of reactivity control systems have been addressed in the RAPID concept. The reactor has no control rods but involves the following innovative reactivity control systems [XVII-1 to XVII-6, XVII-9 to XVII-11]: passive lithium expansion

* Currently Japan Atomic Energy Agency (JAEA)

modules (LEMs) for reactivity feedback improvement, lithium injection modules (LIMs) for passive reactor shutdown, and lithium release modules (LRMs) for automated reactor start-up. These systems adopt ^6Li as a liquid poison for reactor control instead of B_4C based rods. In combination with the LEMs, LIMs and LRMs, the RAPID could be operated in an operator-free mode.

The primary circuit of the RAPID is connected to a thermoelectric power conversion system. Some major design and operating characteristics of the RAPID are given in Table XVII-1.

TABLE XVII-1. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

CHARACTERISTIC	VALUE/ DESCRIPTION
Installed capacity, MW: - Thermal - Electric (gross/ net) - Waste heat Mode of operation	 10 1.2/ 1.0 8.8 Base load; operator-free
Load factor/ availability	95% (target); periodic inspection is performed during the refuellings, once in 10 years; no shutdown is planned except that for refuelling
Major design characteristics	
Type of fuel	U-Pu-Zr metal fuel
Fuel enrichment (by Pu)	14% for the inner core and 19% for the outer core
Coolant	Sodium
Structural materials	Austenitic steel (SUS316)
Core type and characteristic dimensions	Homogeneous, with two zones of different enrichment. The active core region is 1150 mm in diameter and 1000 mm high with a central channel of 220 mm diameter. The core consists of approximately 14 000 fuel elements, combined by a core support grid and several spacer grids, and assembled into a fuel cartridge.
Vessel type and characteristic dimensions	The reactor is essentially a loop type configuration with a reactor vessel 3.0 m in diameter and 6.8 m in height. The distinction from conventional sodium cooled reactors is the integral fuel assembly enclosed in a fuel cartridge.
Cycle type	Direct thermo-electric energy conversion system [XVII-12 to XVII-15]
Energy conversion efficiency	12%. A thermoelectric (TE) energy conversion system was adopted because of reliability, absence of maintenance and simple operation; these are definitive advantages over steam turbine generators in spite of the inferior energy conversion efficiency, which is not rated as very important for a small power reactor such as the RAPID.
Number of loops	2

Simplified schematic diagram

Electromagnetic pumps drive the primary coolant circulation. The reactor is coupled to four thermo-electric power conversion segments placed around the reactor, Fig. XVII-1. Each segment has a pumped sodium heat rejection loop.

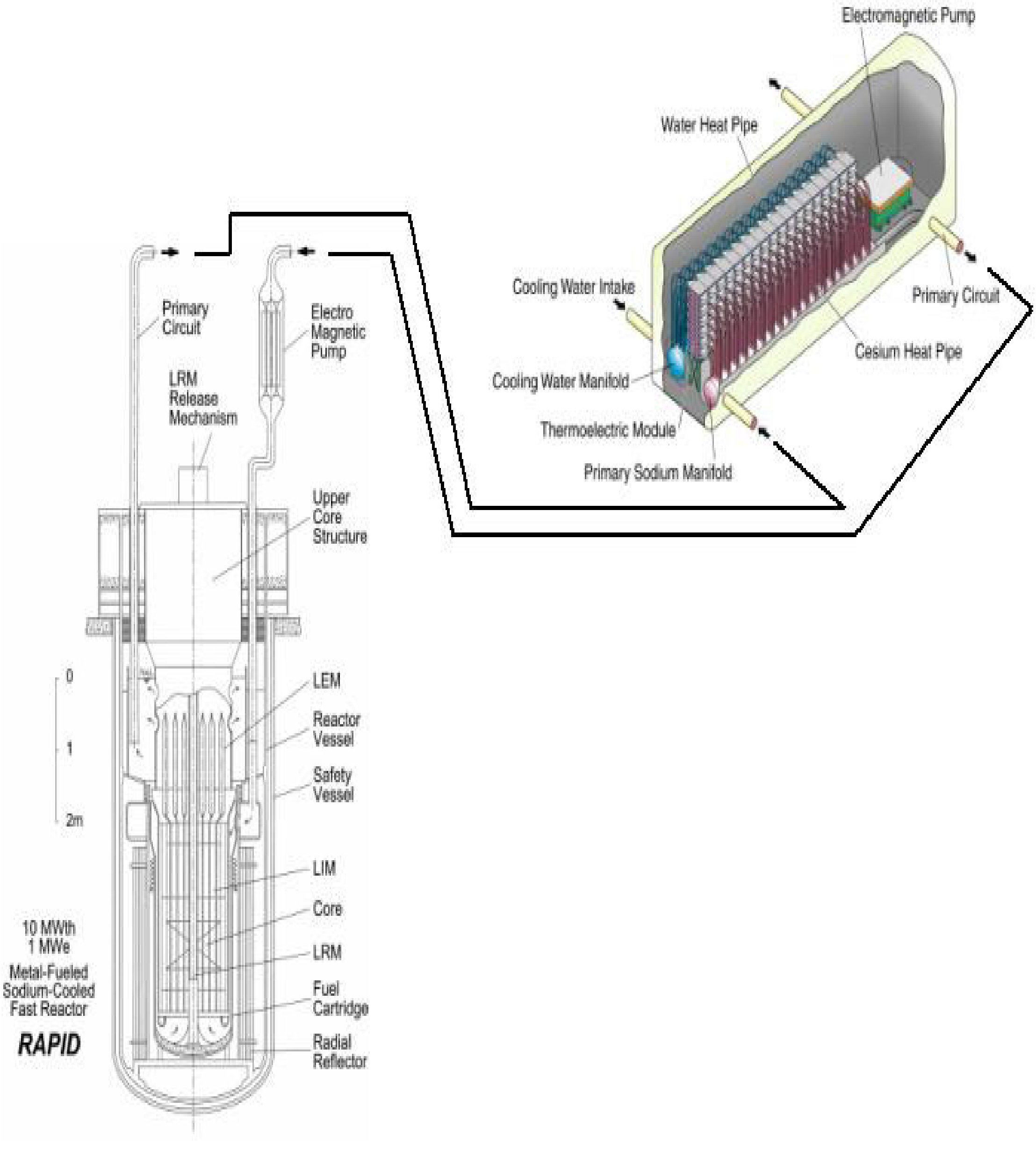


FIG. XVII-1. Schematic diagram of the RAPID power plant.

The fuel cartridge incorporates LEMs, LIMs, LRMs and radial reflectors. In the RAPID, the reactor vessel itself has neither a diagrid† nor a core support structure because they both are integrated into the fuel cartridge.

Neutron-physical characteristics

The neutron-physical characteristics are summarized in Table XVII-2.

TABLE XVII-2. NEUTRON-PHYSICAL CHARACTERISTICS

CHARACTERISTIC	VALUE/ DESCRIPTION
Delayed neutron fraction	$3.78\ 10^{-3}$
Power effect of reactivity	$2.2\ 10^{-3}$ (from 200°C to 380°C) and $1.02\ 10^{-3}$ (from 380ºC to nominal 530°C)
Void reactivity effect	+3.5$
Burn-up reactivity swing	-2.85$ over 10 years
Breeding ratio	1.0016

Reactivity control mechanism

As it was already mentioned, the RAPID control and protection system includes the ^{6}Li-based passively operated LEMs, LIMs and LRMs.

Lithium expansion module (LEM)

The lithium expansion module (LEM) is a passive device used to improve reactivity feedbacks. The concept of LEM is illustrated in Fig. XVII-2. The LEM is composed of an envelope of refractory metal in which a liquid poison of 95% enriched ^{6}Li is enclosed. Lithium-6 is suspended in the upper part of the envelope by surface tension exerted on the gas-liquid interface. The LEM is actuated by the volume expansion of the ^{6}Li itself; if the core exit temperature increases, the gas-liquid interface goes down and negative reactivity insertion can be achieved. The inside diameter of the LEM envelope is 12 mm; it was determined such that the gas-liquid interface in the LEM envelope can never be broken, even in an 1.3g earthquake‡.

A quick LEM is characterized by a rapid response; it only provides a negative reactivity insertion. Five out of 6 quick LEMs ensure 50 cents of negative reactivity insertion, Fig XVII-3. In nominal operation the gas-liquid interface is placed at the active core top. In case the core outlet temperature decreases, the gas-liquid interface goes up and no positive

† In reactors of traditional design, fuel and blanket subassemblies stand on a diagrid, which distributes primary coolant into the subassemblies; usually the diagrid consists of a high pressure plenum and a low pressure plenum, with the former providing a higher coolant flow rate for those assemblies that are in a higher neutron flux and, therefore, need a higher coolant flow rate.

‡ The maximum envelope inner diameter is defined by the following equation [XVII-9]:

$D = (12\ \sigma/n\gamma)^{1/2}$,

where:

D is the envelope inner diameter (m);

σ is surface tension per unit length (N/m);

n is anticipated acceleration over the earth’s gravity; in this case, n = 1.3 g = 1.0 g (gravity force) + 0.3 g (seismically isolated response);

γ is the specific weight of lithium (N/m^3).

reactivity insertion is expected. Quick LEMs have double-enveloped reservoirs, with the gap between the envelopes providing a vacuum insulation needed to adjust the response in the reactor transients. In the absence of double envelopes, the extra-sensitive response of the quick LEMs would result in oscillations and divergence of the reactor power.

The maximum inner diameter of the envelope in the above equation is 25 mm. A LEM envelope with larger inner diameter would result in the liquid lithium falling downward and gas rising in the envelope, under seismic vibration.

Slow LEMs can provide both negative and positive reactivity insertion with a moderate thermal response. Thirty-five slow LEMs provide the variation of reactivity between -2.79$ and +2.88$. The slow LEMs are used for automated burn-up reactivity compensation. In addition, slow LEMs partially realize the function of power control in accordance with the primary coolant flow rate. In nominal operation, the gas-liquid interface is placed in the active core region as shown in Fig. XVII-4. In case the core outlet temperature decreases, the gas-liquid interface goes up and positive reactivity is added, and vice-versa. To avoid quick positive reactivity addition, slow LEMs also have double-enveloped reservoirs (shells) with vacuum insulation. Therefore, only moderate thermal transients resulting from burn-up reactivity swing and primary flow rate variations affect slow LEMs. The design parameters of the LEMs are given in Table XVII-3.

Slow LEMs together with quick LEMs ensure that the reactor power can be controlled within a wide range (at minimum, 40% of the nominal power) by adjusting the primary coolant flow rate.

TABLE XVII-3. DESIGN PARAMETERS OF QUICK AND SLOW LEMS

PARAMETER	QUICK LEM	SLOW LEM
Envelope		
Inner diameter (mm)	12	12
Full stroke (mm)	1000	1000
Material	MoRe	MoRe
Reservoir		
Inner diameter (mm)	140	140
Length (mm)	1300	1350
Wall thickness (mm)	2	2
Gap of double envelope (mm)	1	5
Total LEM sensitivity (¢/K)	2.2	19.9
Single LEM sensitivity (¢/K)	0.43	0.57

Lithium injection module (LIM)

The LIM is another passive device installed in the RAPID-L. The concept of LIM is illustrated in Fig. XVII-4. LIM is also composed of an envelope (shell) enclosing a 95% enriched ^{6}Li. In case the core outlet temperature exceeds the melting point of the freeze seal, ^{6}Li is injected by a pneumatic mechanism from the upper to the lower region to achieve negative reactivity insertion. In this way the reactor is automatically brought into a permanently subcritical state with the temperatures being kept well below the boiling point of sodium (960ºC, considering hydraulic static pressure exerted on the core). Twelve LIMs with 20 mm-diameter envelopes are sufficient to perform passive reactor shutdown function.

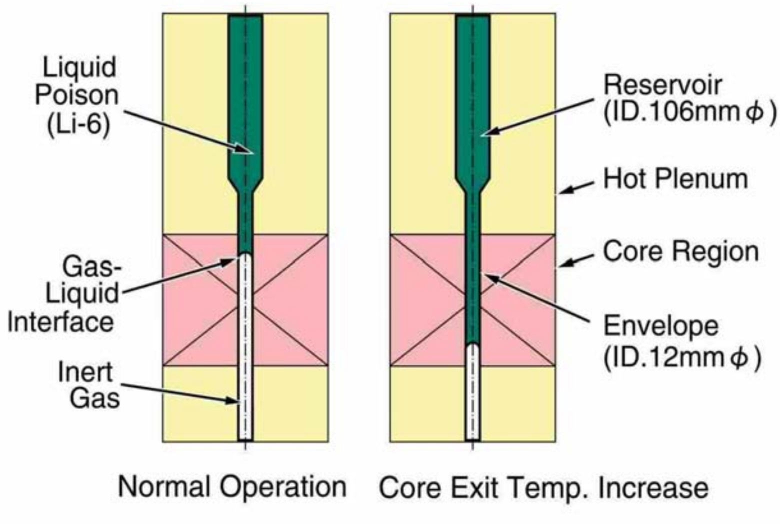


FIG. XVII-2. LEM concept.

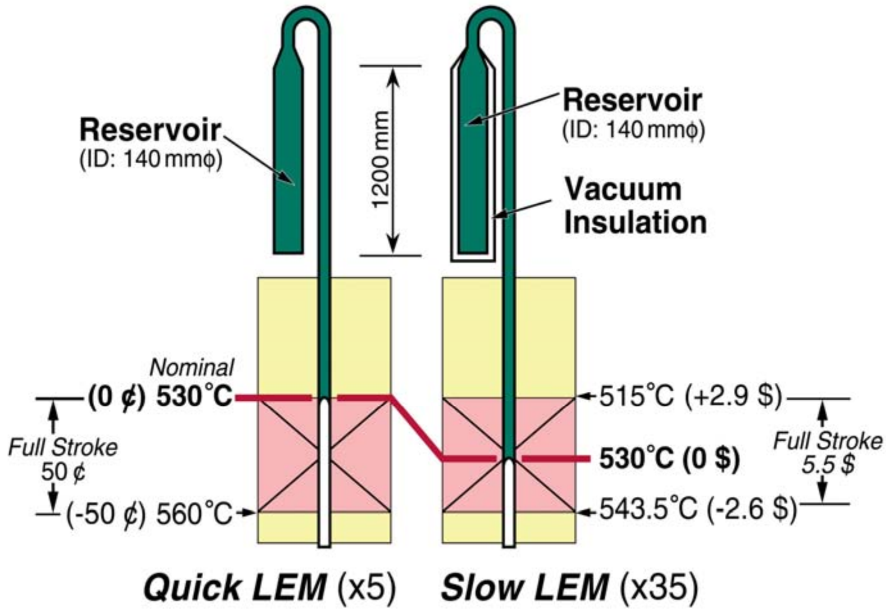


FIG. XVII-3. Elevation of the LEM gas-liquid interface.

Similarly to LEMs, LIMs ensure sufficient negative reactivity feedback in unprotected transients. The role of LIMs is to provide a diversity and redundancy of performing this function in transients. Either LEMs or LIMs can terminate such transients independently. The difference between LEMs and LIMs is that the former can achieve both negative and positive reactivity feedbacks reversibly, while the latter provide only permanently negative feedbacks.

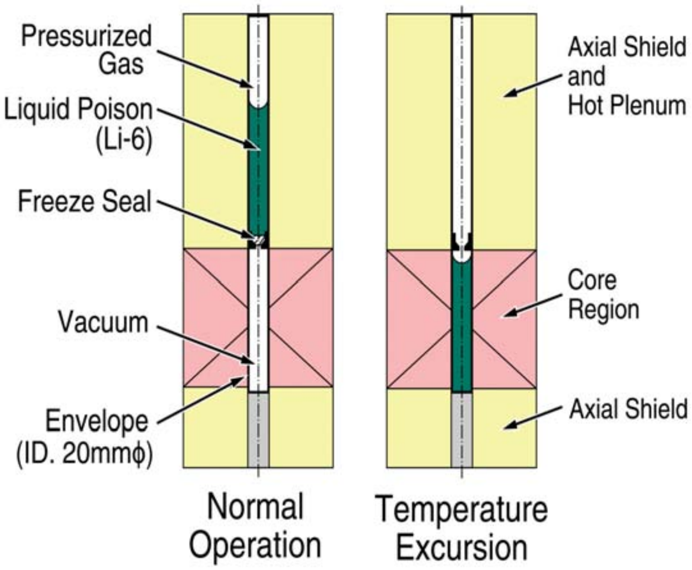


FIG. XVII-4. LIM concept.

Lithium release module (LRM)

Fully automated reactor start-up can be achieved by the LRM, yet another passive device incorporated in the RAPID concept. Figure XVII-5 shows the LRM basic concept. LRM is similar to LIM; however, ^{6}Li is reserved in the active core part prior to reactor start-up. The LRM is placed in the active core region where the local coolant void worth is positive, as is also the case with LEMs and LIMs. The RAPID is equipped with an LRM bundle in which 9 LRMs and an additional B_4C rod are assembled. The reactivity worth of the LRM bundle is +3.45 $, once each LRM includes a 95% enriched ^{6}Li enclosed in a 20mm-diameter envelope. A B_4C rod is used to ensure the shutdown margin (-0.5 $). An automated reactor start-up can be achieved by gradually increasing the primary coolant temperature with the primary pump circulation. The freeze seals of LRMs melt at the hot standby temperature (380ºC), and ^{6}Li is released from the lower level (active core level) to the upper level to achieve positive reactivity addition. An almost constant reactivity insertion rate is ensured by the LRMs because the liquid poison, driven by the gas pressure in the bottom chamber, flows through a very small orifice. It would take almost 14 hours for the liquid poison to move into the top chamber completely. A Sn-Bi-Pb alloy is used as the freeze seal material to ensure the reactor start-up at 380ºC.

Plant dynamic analyses were undertaken to demonstrate the fully automated reactor start-up. The boundary conditions were as follows: start-up duration 50 000 sec (13.9 hr); reactivity insertion rate 0.0069 cents/s.

The transient characteristics are shown in Fig. XVII-6. An automated reactor start-up was initiated from a subcritical state by the insertion of reactivity at a constant rate (0.0069 cents/s). About 2 hours after the start-up, the reactor power comes to a peak (12% overpower). Quick LEMs are actuated by this overpower to counterbalance the LRMs positive reactivity addition. Then, the net reactivity is kept slightly positive and the reactor power gradually approaches the nominal value.

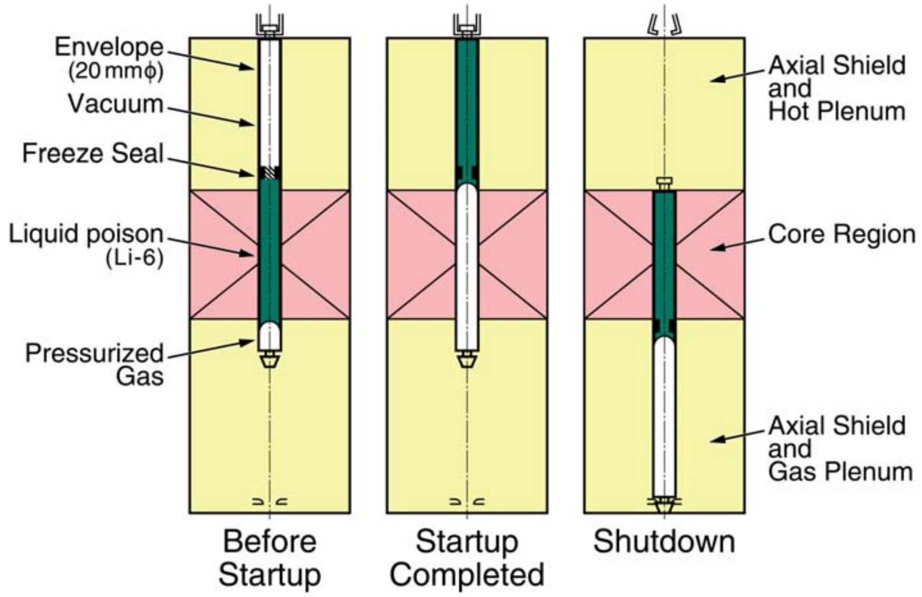


FIG. XVII-5. LRM concept.

The selected reactivity insertion rate is as moderate as that achieved by actively inserted control rods in conventional fast reactors. The maximum thermal transient of the primary circuit in reactor start-up (10 K/min) is similar to that observed in a cold shock transient with scram of the conventional land-based reactors (typically 5 to 10 K/min). The role of the quick LEMs in this case is to restrict the power overshoot by 2 hours after the reactor start-up.

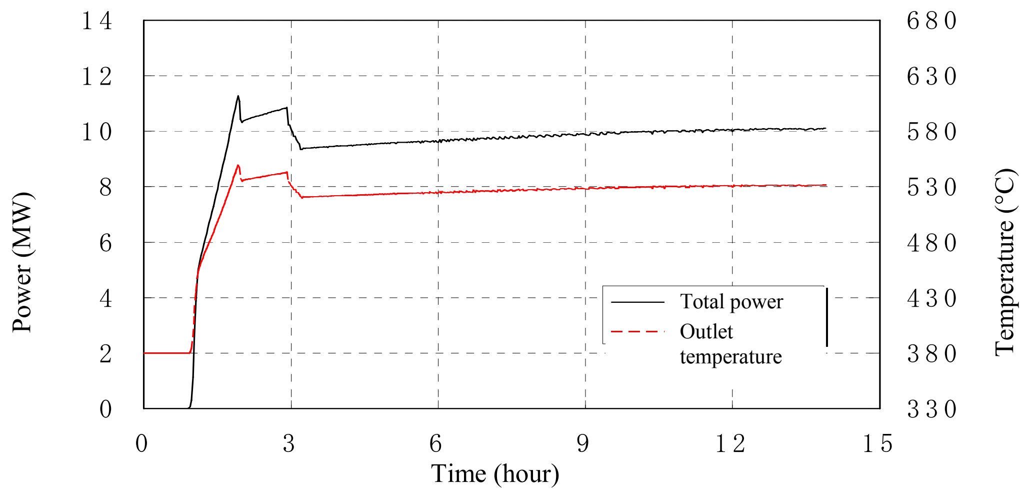


FIG. XVII-6. Transient response in the reactor start-up.

The reactivity worth data of the RAPID control and protection systems are summarized in Table XVII-4.

The reasons for abandoning conventional control rods are discussed in section XVII-1.6.3.

TABLE XVII-4. REACTIVITY CONTROL DATA

REACTIVITY CONTROL SYSTEM	ENVELOPE INNER DIAMETER (mm)	NUMBER	REACTIVITY WORTH RANGE ($)
Quick LEM	12	5+ (1)	-0.65 to 0
Slow LEM	12	35	-2.79 to +2.88
LIM	12	12	-5.36
LRM	12	9	+3.45
B_4C rod		1	-0.5

Thermal-hydraulic characteristics

The thermal-hydraulic characteristics are summarized in Table XVII-5, with some basic data being provided in Table XVII-1.

TABLE XVII-5. THERMAL-HYDRAULIC CHARACTERISTICS

CHARACTERISTIC	VALUE/ DESCRIPTION
Sodium temperature in the core (inlet/outlet), °C	380/ 530
Primary coolant flow rate (kg/s)	52.5
Average coolant velocity in the core (m/s)	0.22
Peak linear power at BOL (W/cm)	43
Excess pressure in the primary circuit	0.1 atmosphere
Maximum/average temperature of fuel in normal operation (°C)	580/535
Maximum/average temperature of structural materials in normal operation (°C)	575/530
Temperature limit for fuel (°C)	1100 (to avoid melting)/750 (to avoid eutectic reaction)
Temperature limit for claddings (°C)	610

Maximum/average discharge burn-up of fuel

The maximum fuel burn-up is 6600 MW·day/t; the average one is 3400 MW·day/t.

Fuel lifetime/period between refuellings

The fuel lifetime is 10 years, coinciding with the period of reactor operation without on-site refuelling.

Mass balances/flows of fuel materials

The consumption of natural uranium is 55 kg/MW(e) per year.

Design basis lifetime for reactor core, vessel and structures

The design lifetime of the RAPID is 40 years.

Design and operating characteristics of systems for non-electric applications

No information was provided.

Economics

The estimated capital cost for NPP construction is US$ 8000/kW(e). This estimate is based on a target of the electricity cost of 0.05 US$/kW(e)/hour, with a plant availability of 0.95 and a design lifetime of 20 years. However, it is the cost of a first-of-a-kind plant, which is expected to decrease with increased production. The modular approach is also effective for cost reduction.

XVII-1.5. Outline of fuel cycle options

Once-through fuel cycle could be the initial fuel cycle option for the RAPID. Later on, dry reprocessing of metal fuel could be applied to close the fuel cycle. The reprocessing is assumed centralized, with the discharged integral fuel assembly of the RAPID being transported to a centralized reprocessing plant.

The RAPID is designed to operate for 10 years without a shutdown. Prior to refuelling, the LRM bundle should be released into the lower part of the core so that liquid poison is again located in the active core region. A B_4C rod included in the LRM bundle also moves into the active core region to ensure a shutdown margin (-0.5 $). In this case, the LRM bundle acts as a poison rod; once released, it is clumped at the bottom and impossible to pull out again. Such design conforms to the US space reactor safety criteria. The reactor start-up is only possible by installing a new fuel cartridge.

The refuelling procedure, conducted every 10 years, is illustrated in Fig. XVII-7. The RAPID concept [XVII-1] enables fast and simplified refuelling after two weeks of reactor shutdown by which time the decay heat of the core is 10 kW. During refuelling, a lithium-filled fuel cartridge is removed from the reactor and loaded into a lithium-filled on-site storage cask (OSSC) located beside the reactor. After receiving the spent fuel, the OSSC is equipped with a heat pipe radiator for decay heat removal. It is stored in an excavated cylindrical hole to minimize the dose rate of the personnel involved. The dissipation of decay heat will solidify lithium in the OSSC one year after refuelling, then, the spent fuel together with the OSSC could be transported to the reprocessing plant.

XVII-1.6. Technical features and technological approaches that are definitive for RAPID performance in particular areas

XVII-1.6.1. Economics and maintainability

The RAPID is a factory-fabricated and fuelled small reactor, which does not benefit from the economies of scale but could benefit from the economies of mass production.

The RAPID could be useful for power systems on islands where diesel generators are currently employed. Other expected applications are power systems for ore deposit mining and seawater desalination plants in remote areas. Many ore deposits are uneconomic because they are located in very remote places, too distant from the usual sources of energy. Energy supply from the RAPID could improve the economics for many projects, which would otherwise be abandoned.

The operation without on-site refuelling and infrequent periodic inspections are factors contributing to the reduction of the operation and maintenance costs. The increase of a maintenance interval up to 10 years could be reasonable for operator-free reactors such as the RAPID because no moving mechanical parts are involved in their safety systems.

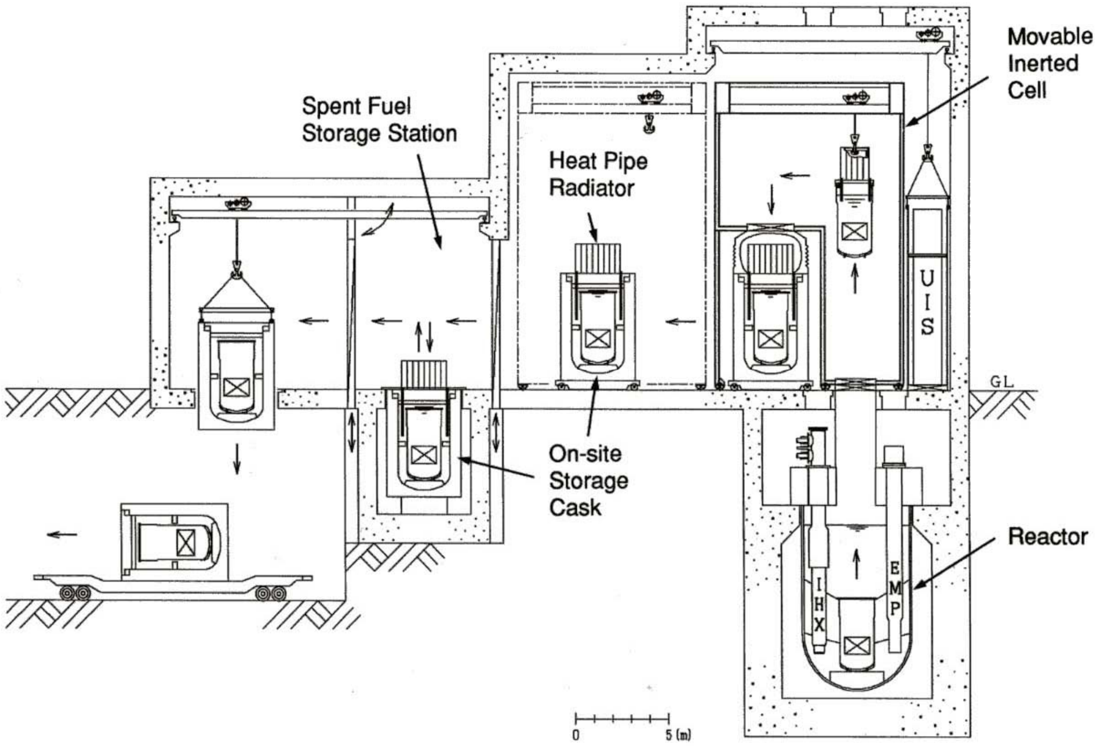


FIG. XVII-7. RAPID refuelling concept.

XVII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The cartridge-type design of the fuel assembly insures less volume and weight of the core structural materials activated under irradiation, which may contribute to minimizing the volume and weight of radioactive waste at the source.

Being a fast reactor with a breeding ratio of ~1.0016, the RAPID could also contribute to the effective use of uranium resources, once a closed nuclear fuel cycle is established.

XVII-1.6.3. Safety and reliability

Safety concept and design philosophy

The design objective of the RAPID is to exclude human errors in the reactor operation by making it essentially operator-free, see Fig. XVII-8.

Provisions for simplicity and robustness of the design

No moving mechanical parts are provided in the RAPID. The LEMs, LIMs and LRMs are passive systems that are driven by natural phenomena, such as volume expansion of lithium-6 and meltdown of the freeze seal. The reactor will be equipped with flow meter(s) and thermocouple(s) to monitor the primary flow rate and core outlet temperature, however, this instrumentation is only to monitor the reactor and has nothing to do with the performance of safety functions, Fig. XVII-8.

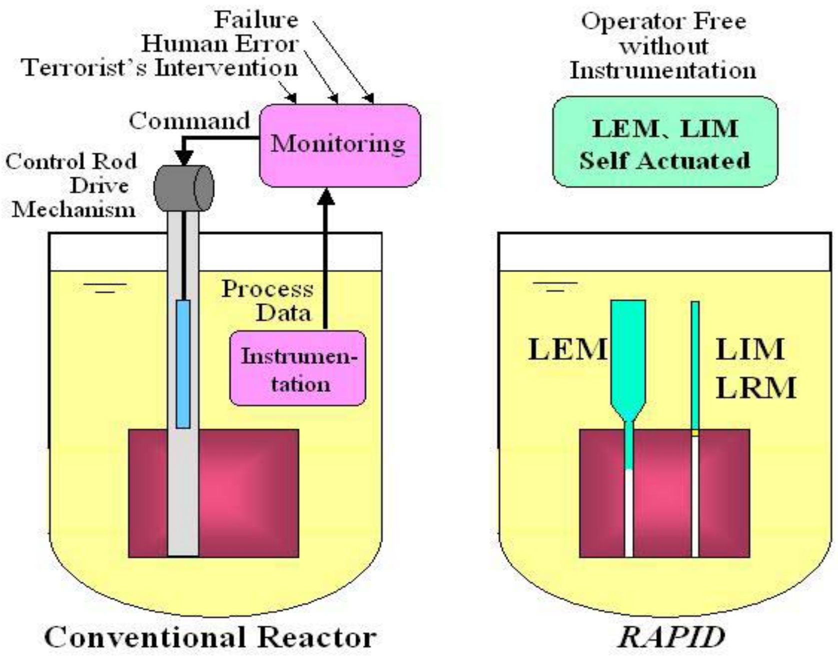


FIG. XVII-8. Safety concept of the RAPID vs. conventional reactors.

Active and passive systems and inherent safety features

LEM, LIM and LRM are passive systems; LEM is actuated by the volume expansion of lithium-6 and LIM and LRM are actuated by meltdown of the freeze seal. The RAPID has no active systems employed in the reactor operation.

Structure of the defence-in-depth

No information was provided.

Design basis accidents and beyond design basis accidents

The analytical results for unprotected loss of flow and unprotected transient overpower as design basis accidents are presented below; beyond design basis accidents were not considered at this stage. However, risk of a core disruptive accident due to local blockage of the core is expected to be low since there are no wrapper tubes in the RAPID core.

Unprotected loss of flow

Figure XVII-9 illustrates the effect of quick LEMs on an unprotected loss of flow (ULOF) transient as obtained in dynamic analysis of the plant. In this particular case, the primary coolant flow rate was supposed to decrease to 10% of the nominal value within 2 seconds.

Unprotected transient overpower

An important characteristic of the LEM reactivity control system is redundancy. Since the RAPID has no control rods, UTOP transients due to faulty handling of the control rods do not arise. However, a failure of one of the LEM envelopes could be anticipated. This failure causes positive or negative reactivity insertion, depending on the LEM status (location of gas-liquid interface just before the failure). A maximum reactivity addition is anticipated in the case of a 40% primary flow rate in the beginning-of-life core, when each slow LEM is in a state of full insertion of the ^{6}Li liquid poison.

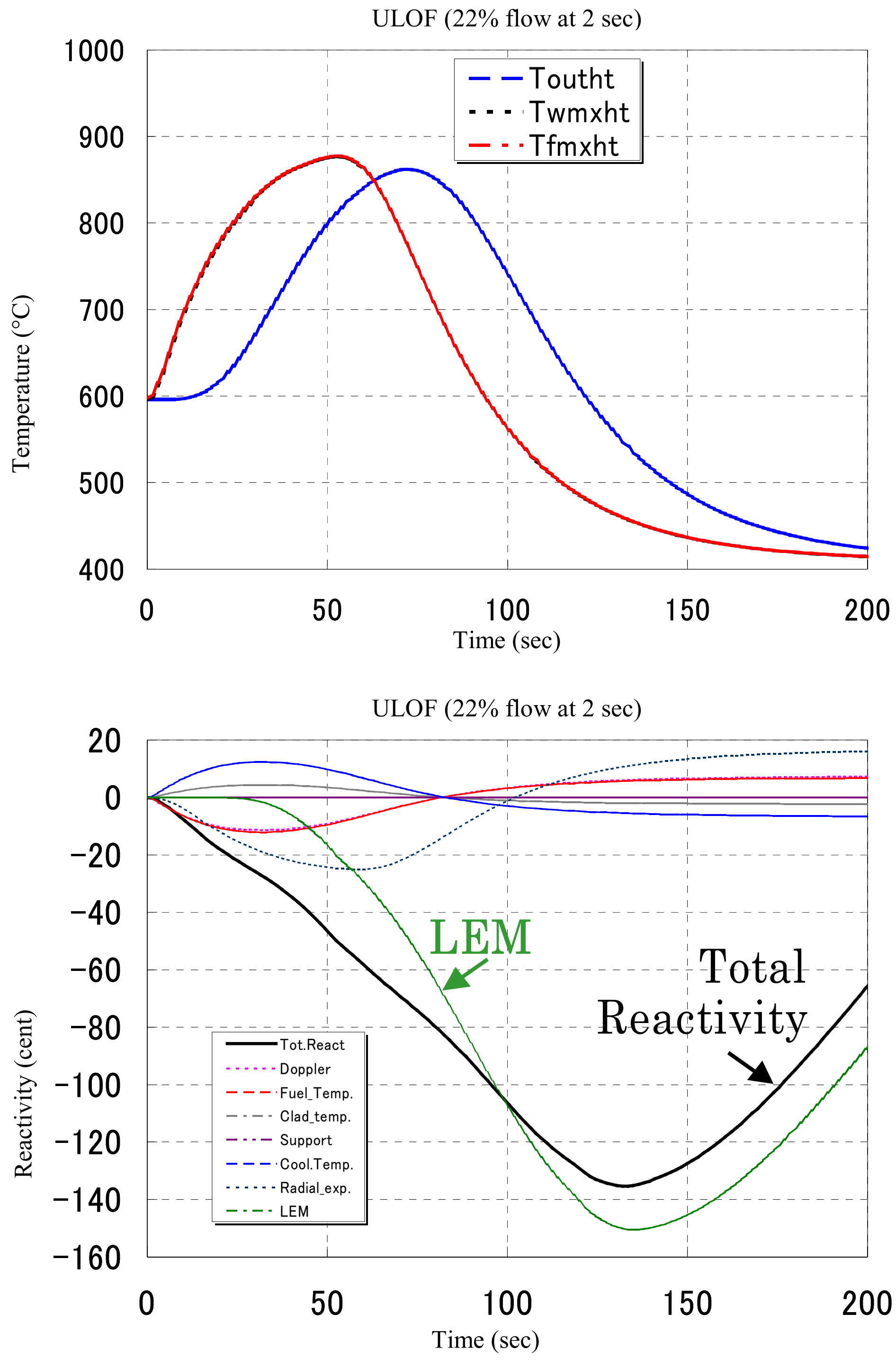


FIG. XVII-9. Analytical results for ULOF.

If the envelope of a slow LEM fails, liquid poison in the envelope would be quickly replaced by sodium coolant or by a void. The latter causes more severe consequences: there is a positive reactivity addition of 16.2 cent (5.67 \$/35) per single slow LEM. Time to replace ^{6}Li by void depends on the magnitude of the envelope failure. For the above-mentioned reactivity inserted in 3 s, the transient response is illustrated by the results plant dynamic analysis shown in Fig. XVII-10. This reactivity addition is more severe than the UTOP in conventional reactors (typically 3 cents/s), but is quickly compensated by other quick LEMs.

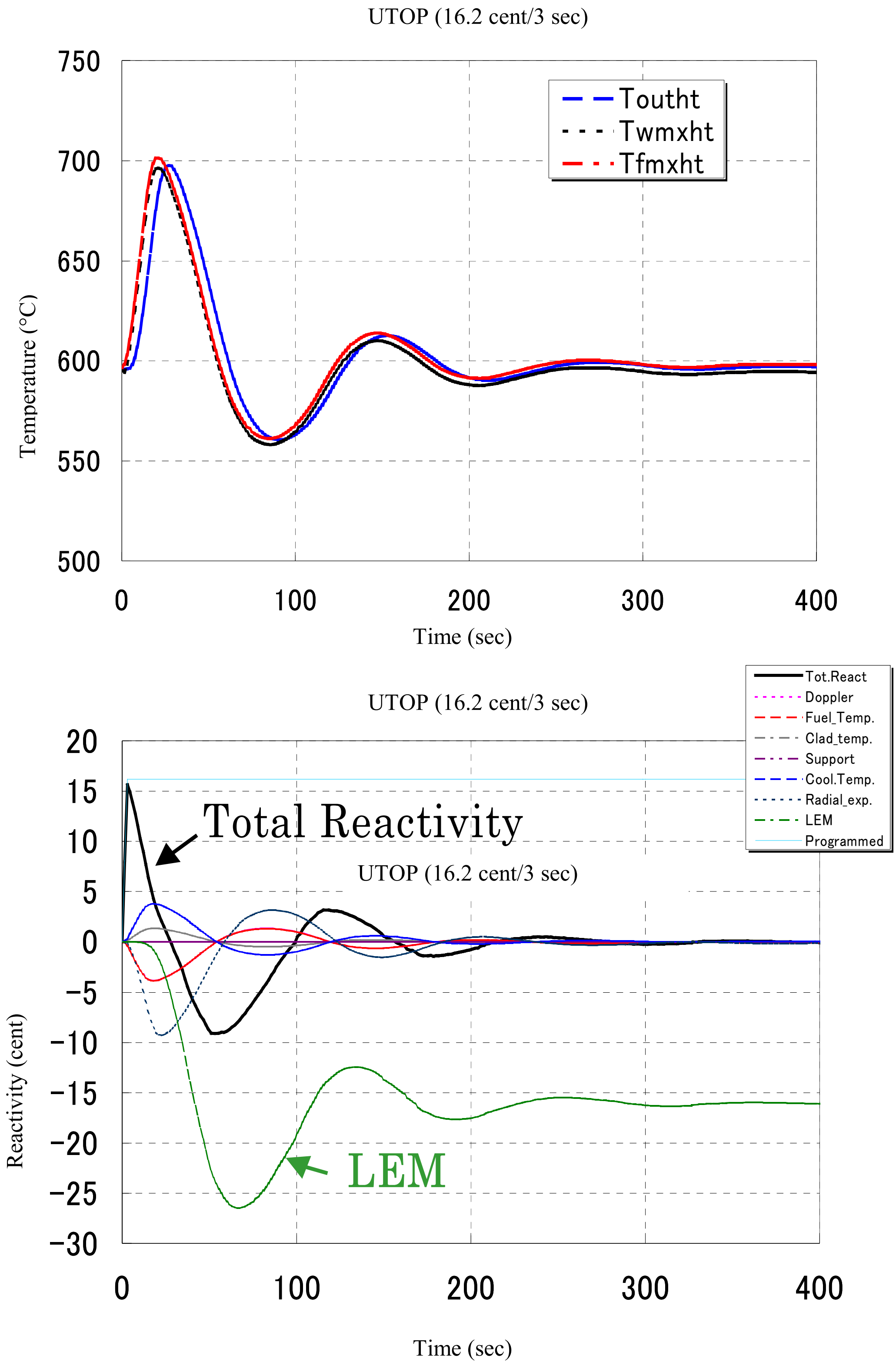


FIG. XVII-10. Analytical results for UTOP.

On the contrary, LEM failures in which they provide positive reactivity (for example, failure of burn-up compensation by LEM in the end-of-life core) will cause negative reactivity insertion. This results in only a slight decrease in reactor power.

Other accidents

The RAPID passive systems play a similar role in less stringent ULOHS (unprotected loss of heat sink) incidents.

Provisions for safety under seismic conditions

The essential feature of LEMs, LIMs and LRMs is that they are effectively independent of the magnitude and direction of the gravity force. LIMs ensure quick negative reactivity insertion even during seismic vibration because of liquid poison being used. The gas-liquid interface of LEM can never be broken in the anticipated seismic vibration[§]. In a beyond-design-basis earthquake, the gas-liquid interface of LEMs might be broken and lithium-6 is released in the active core region. Then, the reactor is shut down and its start-up is impossible until the used LEMs are replaced with new ones.

Probability of unacceptable radioactivity release beyond the plant boundaries

No information was provided.

Measures planned in response to severe accidents

No information was provided.

XVII-1.6.4. Proliferation resistance

The core consists of 14 000 fuel elements (pins) assembled into a single fuel cartridge. Spent fuel (the cartridge) is stored in an on-site storage cask (OSSC). The dissipation of decay heat will solidify sodium in the OSSC one year after refuelling. Frozen sodium as well as the OSSC provides a barrier preventing the diversion of nuclear materials.

The RAPID fuel cartridge is assembled in the factory, leaving no chances to install target materials during its presence in the reactor. Shutting the reactor down, then inserting target materials into the integral fuel assembly and restarting the reactor is impossible, because once LRM is released to shut the reactor down, it is clumped at the bottom and impossible to pull out again. Reactor start-up is only possible by installing a new fuel cartridge, which can only be obtained from the factory.

The spent fuel cartridge is stored in an on-site storage cask (OSSC) with sodium being solidified one year after refuelling. Accounting of fuel elements (pins) is not required in the RAPID refuelling concept because no one can remove the fuel cartridge from the frozen sodium encased in the OSSC. Sealing the cask lid could be helpful for verification.

XVII-1.6.5. Technical features and technological approaches used to facilitate physical protection of RAPID

Any fault of the operator or human intervention of malevolent character would result in a change in the reactor power or shutdown. In a sudden decrease or increase of the primary flow rate, the reactor power will decrease/ increase and gradually approach the value roughly proportional to the primary flow rate. This characteristic is similar to that of the RAPID-L; the results of plant dynamic analyses for the RAPID-L are shown in Fig. XVII-11.

[§] 0.3 g is assumed as seismically isolated response.

XVII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of RAPID

No information was provided.

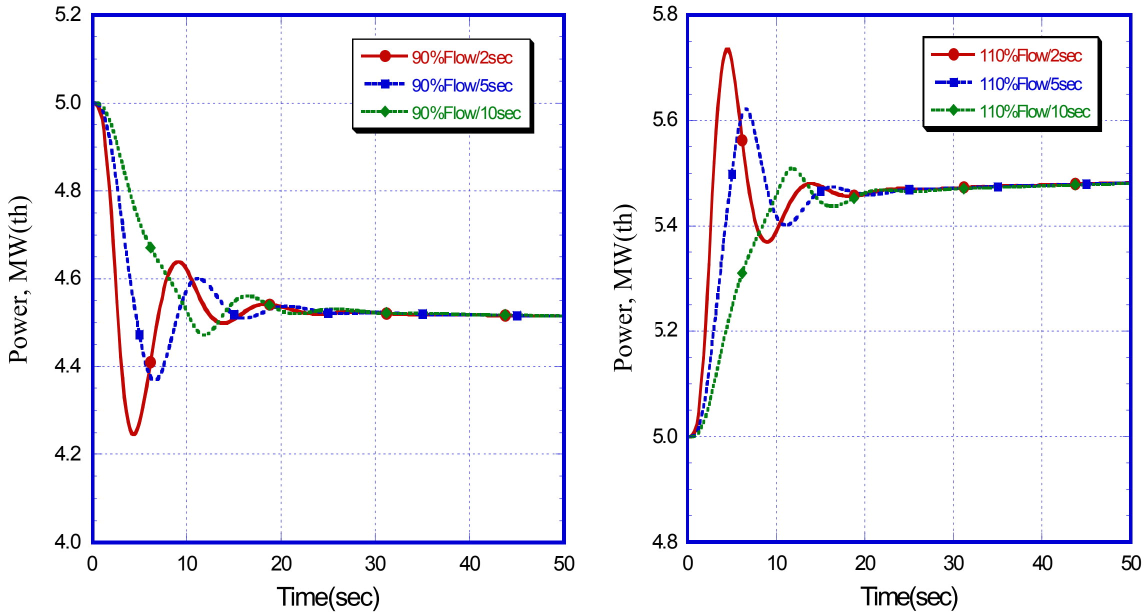


FIG. XVII-11. RAPID-L transient response in a primary flow rate decrease (left) and increase (right).

XVII-1.8. List of enabling technologies relevant to RAPID and status of their development

LEM, LIM and LRM are the essential technologies to realize the RAPID; they have been acquired through the following R&D conducted in 1999-2001 with the support of Japan Atomic Energy Research Institute (JAERI, currently JAEA — the Japan Atomic Energy Agency):

(1) Performance tests and visualization by neutron radiography at JRR-3M reactor of JAERI [XVII-3], Fig. XVII-12 and XVII-13;

(2) Criticality test to evaluate the reactivity worth of lithium-6 [XVII-5];

(3) The development of in-service-inspection (ISI) techniques [XVII-9]; an eddy current coil was designed and manufactured to detect Li in the LEM and LIM envelope.

The following further R&D on LEM, LIM and LRM are necessary:

(1) Endurance tests of LEM and LIM using full scale models (out of pile);

(2) LEM transient performance test using an experimental fast reactor;

(3) An LEM irradiation test to investigate (n, α) reaction on ^{6}Li using an experimental fast reactor;

(4) LIM freeze seal irradiation test using an experimental fast reactor.

The principle of LEM has been verified by neutron radiography conducted at JRR-3M of JAEA. The gas-liquid interface was shown to move up and down in accordance with temperature, Fig. XVII-12.

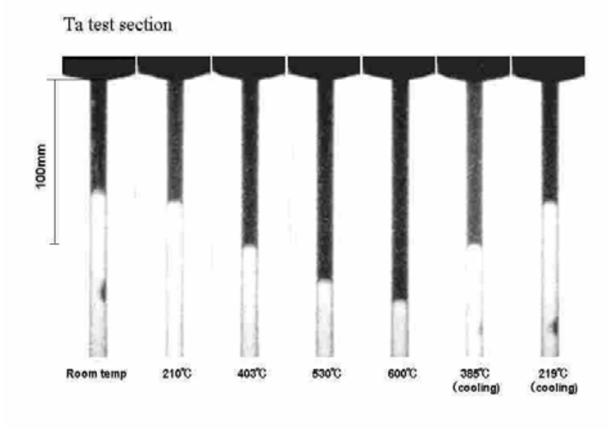


FIG. XVII-12. Neutron radiography of the LEM specimen.

The LIM performance was also demonstrated by neutron radiography as shown in Fig. XVII-13. The time required for the reactivity insertion by LIM is 0.4 s, which is much shorter than the free drop of conventional scram rods (by as much as 2 s).

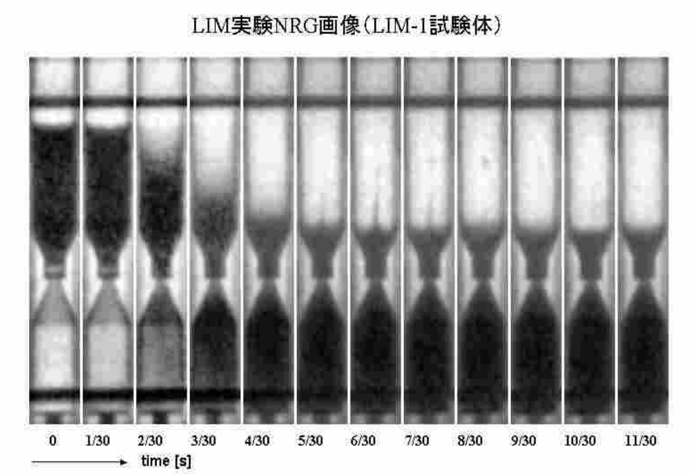


FIG. XVII-13. Neutron radiography of the LIM specimen.

XVII-1.9. Status of R&D and planned schedule

In fiscal years 1999–2001, the Japan Atomic Energy Agency (JAEA) supported the RAPID-L project; this predecessor of RAPID was designed for a lunar-based power system.

Since fiscal year 2002, the Central Research Institute of Electric Power Industry (CRIEPI) has supported the RAPID project. The CRIEPI is the leader of the project; Mitsubishi Research Institute (Japan) participates.

RAPID is currently at the conceptual design stage. Under favourable conditions of financing and with participation of industry, the detailed design of a RAPID prototype could be developed within 6 years.

As of 2005, the R&D on key technologies essential for the RAPID design (such as LEM and LIM) was ongoing, supported by CRIEPI internal funding.

XVII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The operator-free RAPID reactor without control rods is the first-of-a-kind design in the world. Therefore, a substantial R&D, feasibility tests and construction of a prototype or a demonstration plant would be needed.

XVII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The design and R&D for the 200 kW(e) lithium cooled, uranium-nitride fuelled fast reactor RAPID-L, intended for a lunar-based power system, was also conducted by the CRIEPI.

XVII-2. Design description and data for RAPID

XVII-2.1. Description of the nuclear systems

Reactor core and fuel design

The integrated fuel assembly is shown in Fig. XVII-14 and XVII-15. The fuel cartridge, 1.6 m in diameter and 3.9 m long, involves approximately 14 000 fuel elements, 41 LEMs, 12 LIMs and 9 LRMs, as shown in Fig. XVII-15.

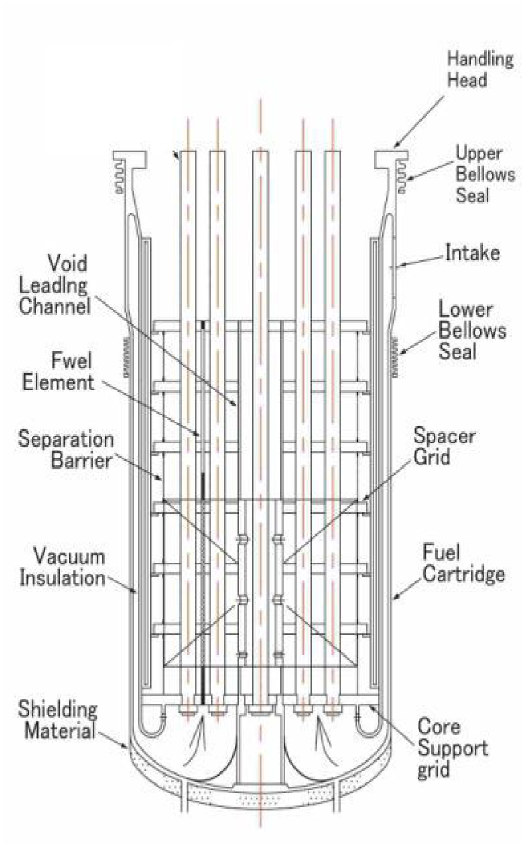


FIG. XVII-14. Integrated fuel assembly (cut view).

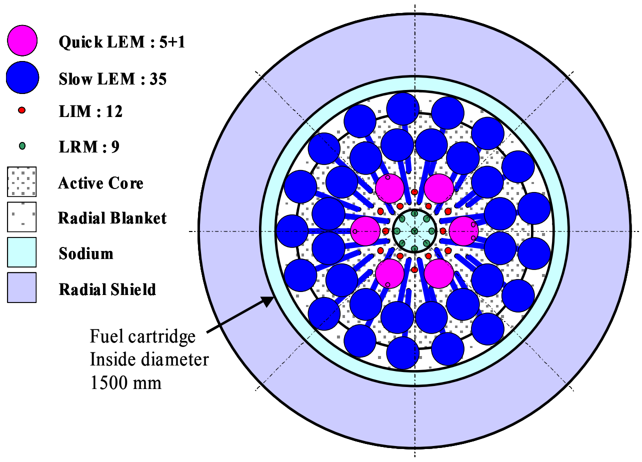


FIG. XVII-15. Integrated fuel assembly (horizontal view).

The reactor core of the RAPID is a homogeneous design with two regions. A two-dimensional R-Z model of the core is shown in Fig. XVII-16.

Some design characteristics of the fuel cartridge and core are given in Table XVII-6, with more data being provided in Table XVII-1 and in Fig. XVII-15 and XVII-16.

TABLE XVII-6. FUEL CARTRIDGE AND CORE DESIGN DATA

ITEM	SPECIFICATION
Fuel pin outer diameter (mm)	8.0
Fuel pin pitch (mm)	9.04
Number of fuel pins	14 000
Core volume fractions (fuel/coolant/structure), %	52/32/16

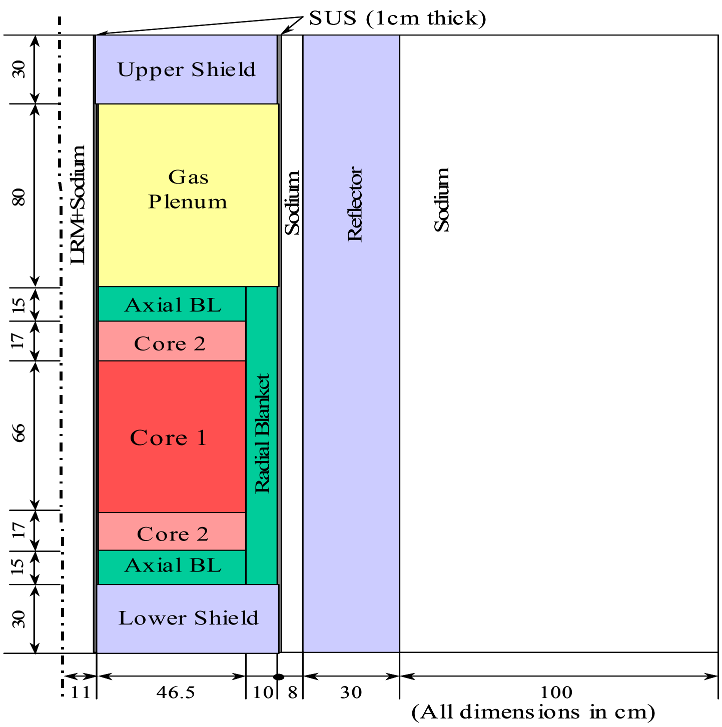


FIG. XVII-16. Two-dimensional R-Z model of the core.

Main heat transport system

The heat removal paths of the RAPID, under normal operation and in accidents, are shown in Fig. XVII-17.

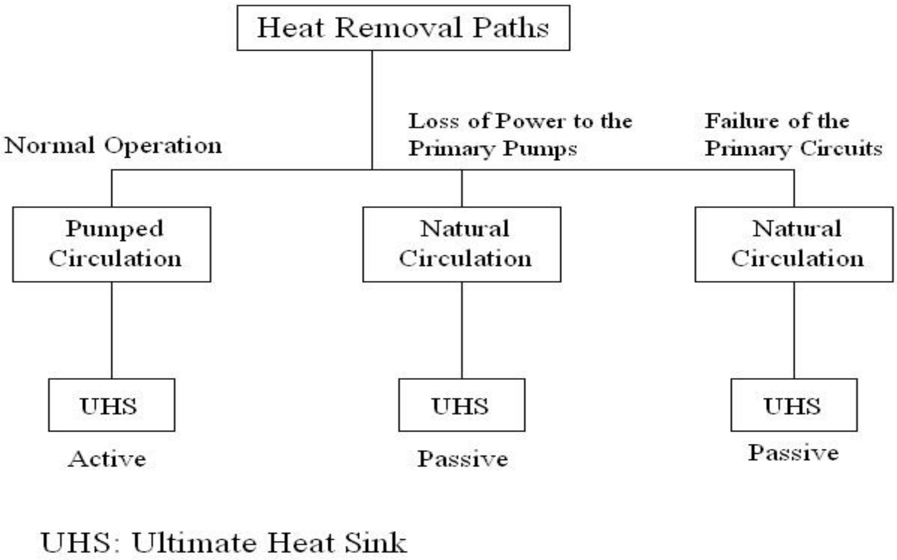


FIG. XVII-17. Heat removal paths of the RAPID.

Intermediate circuit

The RAPID has no intermediate circuit.

XVII-2.2. Description of the turbine generator plant and systems

The RAPID power plant adopts a thermoelectric power system instead of turbine generators, as shown in Fig. XVII-1.

XVII-2.3. Systems for non-electric applications

A seawater desalination plant is under consideration. No further details were provided.

XVII-2.4. Plant layout

No information was provided.

REFERENCES

[XVII-1] KAMBE, M., TSUNODA, H., NAKAJIMA, K., IWAMURA, T., RAPID-L and RAPID operator-free fast reactors combined with thermoelectric power conversion system, Journal of Power and Energy, pp. 335–343 (August 2004).

[XVII-2] KAMBE, M., TSUNODA, H., NAKAJIMA, K., IWAMURA, T., RAPID-L and RAPID operator-free fast reactor concept without any control rods (Int. Conf. on Advanced Nuclear Power Plants and Global Environment, 15–19 September 2003), GENES4/ANP2003, pp. 1039.

[XVII-3] KAMBE, M., TSUNODA, H., NAKAJIMA, K., IWAMURA, T., RAPID-L operator-free fast reactor concept without any control rods, Nuclear Technology, Vol. 143, pp. 11–21 (July 2003).
[XVII-4] KAMBE, M., TSUNODA, H., NAKAJIMA, K., IWAMURA, T., RAPID-L highly automated fast reactor concept without any control rods, (1) reactor concept and plant dynamics analyses, ICAPP'02 (Proc. Int. Congress on Advanced Nuclear Power Plants, Florida, USA, 9–13 June 2002) pp. 1035.
[XVII-5] TSUNODA, H., et al., RAPID-L highly automated fast reactor concept without any control rods, critical experiment of lithium-6 used in LEM and LIM, ICAPP'02 (Proc. Int. Congress on Advanced Nuclear Power Plants, Florida, USA, 9–13 June 2002) pp. 1036.
[XVII-6] KAMBE, M., et al., Coming down to earth, Nuclear Engineering International, Vol. 47, No. 579, pp. 26–31 (October 2002).
[XVII-7] KAMBE, M., et al., RAPID-L: an operator-free reactor for the moon, Nuclear Engineering International, Vol. 49, No. 596, pp. 24–25 (March 2004).
[XVII-8] KAMBE, M., UOTANI, M., Fast reactor concept RAPID for improvement of reactor performance and proliferation resistance, Nuclear Engineering and Design, No. 170/1–3, pp. 9–19 (July 1997).
[XVII-9] KAMBE, M., UOTANI, M., Design and development of fast breeder reactor passive reactivity control systems: LEM and LIM, Nuclear Technology, Vol. 122, pp. 179–195 (May 1998).
[XVII-10] KAMBE, M., Fast reactor passive shutdown system: LIM, ICONE-7 (Proc. of the 7th Int. Conf. on Nuclear Engineering, Tokyo, Japan, April 1999) pp. ICONE-7069.
[XVII-11] KAMBE, M., RAPID-A fast reactor concept without any control rods, Nuclear Technology, Vol. 128, pp. 12–24 (October 1999).
[XVII-12] KAMBE, M., High energy density thermoelectric energy conversion systems by using FGM compliant pads for space and terrestrial applications, (Proc. on the 48th Int. Astronautical Congress, Turin, Italy, October 1997), pp. IAF-97-R.1.02.
[XVII-13] KAMBE, M., SHIKATA, H., Intensive Energy density thermoelectric energy conversion system by using FGM compliant pads, (Proc. of the 52nd Int. Astronautical Congress, Toulouse, France, pp. IAF-01-R.2.06, October 2001),
[XVII-14] KAMBE, M., SHIKATA, H., Intensive energy density thermoelectric energy conversion system by using FGM compliant pads, Acta Astronautica (Pergamon Press), Vol. 51, No. 1–9, pp. 161–171 (2002).
[XVII-15] KAMBE, M., SHIKATA, H., Thermoelectric power conversion systems by using compliant pads for space and terrestrial applications, (Proc. of the 54th Int. Astronautical Congress, Bremen, Germany, pp IAC-03-R.2.01, 1614001, October 2003).

ANNEX XVIII

TRANSPORTABLE MODULAR SODIUM COOLED REACTOR WITH GAS-TURBINE GENERATOR (BN GT-300)

Institute of Physics and Power Engineering (IPPE),
Russian Federation

XVIII-1. General information, technical features and operating characteristics

XVIII-1.1. Introduction

The BN GT-300 is a transportable modular nuclear cogeneration plant based on a small sodium cooled reactor with fast spectrum of neutrons and a gas turbine cycle for energy conversion [XVIII-1].

Design development for the BN GT-300 was carried out in compliance with the Russian standard 2.118-73; however, specialized design organizations such as chief design organization, designers of the reactor, gas turbine and generator equipment, and architect and engineering companies have not been involved so far.

The experience gained in the Russian Federation with reactors based on sodium coolant and having a fast neutron spectrum (the so-called BN type reactors) has proven that high levels of safety, reliability and environmental benignness can be achieved with such reactors. But their serial production is still not planned due to higher initial cost and higher fuel cost than for conventional pressurized water reactors. For example, the construction cost of a new BN-800 plant of 800 MW(e) is close to or even higher than that of a serial VVER-1000 of 1000 MW(e), and the primary electricity generation cost is also 15–20% higher. This means that a traditional design approach to the BN type reactors may put limitations on their economic acceptability. With an economy of scale approach being applied, the BN-1800 plant of 1800 MW(e) may have the economic characteristics close to those of the VVER-1500 of 1500 MW(e), but offers no sound guarantees that the latter would be surpassed.

Another limitation for further development of the traditional BN type reactors may be due to the current electricity market structure. When electricity consumption growth rate is slow, the demand for large capacity nuclear power plants (NPPs) will be limited also. Meanwhile, small NPPs could be used not only for gradual increase of the generating capacity but for the replacement of the outdated fossil fuel and nuclear power plants (such replacement is often referred to as renovation). The reactor installations with an installed capacity of 300 MW(e) could be used both to replace large-scale NPPs and for autonomous operation within small local grids. The latter may have even small capacity requirements and, therefore, smaller versions of the BN GT plant (down to 5 MW(e)) are being examined currently.

The sites of the plants to be renovated often pose challenges to new construction, for example, due to limited space or location in the vicinity of a populated area, which restricts the possible radioecological, thermal and other impacts of a new plant. To meet these stringent requirements, small NPPs of a new generation should be:

- Sufficiently safe and ecologically clean to be placed within city borders;
- Easy in manufacturing of the components;
- Easy in transportation of the components;

- Provide for an easy construction of a NPP protective enclosure, with a possibility of use of the existing older buildings;
- Provide for easy assembly and testing of components on the site;
- Economically competitive over the whole NPP lifetime, even if fossil fuel costs go down during this period;
- Protected from the external events including aircraft crash and human actions of malevolent origin;
- Provide for a simple and ecologically safe decommissioning.

Most of these properties could be achieved with NPPs of a transportable design with a high grade of factory fabrication.

Floating NPP options, such as provided for in the KLT-40S [XVIII-2] and VBER-300 [XVIII-3] projects, require the construction of special coastal facilities large enough to accept barges with a displacement of 20.000–50.000 tons. Even among seacoast (outer) sites, only a few would be able to accommodate such ship sizes, and nearly all inner sites appear to be not appropriate for this purpose.

Therefore, rail transportable units were chosen for the BN GT plant. Six to eight factory fabricated modules are enough to deliver a complete BN GT-300 plant to a site. In regions without proper railways, main equipment modules could be transported by barge. By combining rail and water transport, the BN GT-300 could be delivered to almost every site.

Design and technology development of the BN GT-300 has been partially funded by the Government of Russia via Rosatom of Russia and partially by the Institute of Physics and Power Engineering (IPPE, Obninsk), which is the only stakeholder at this phase of the project.

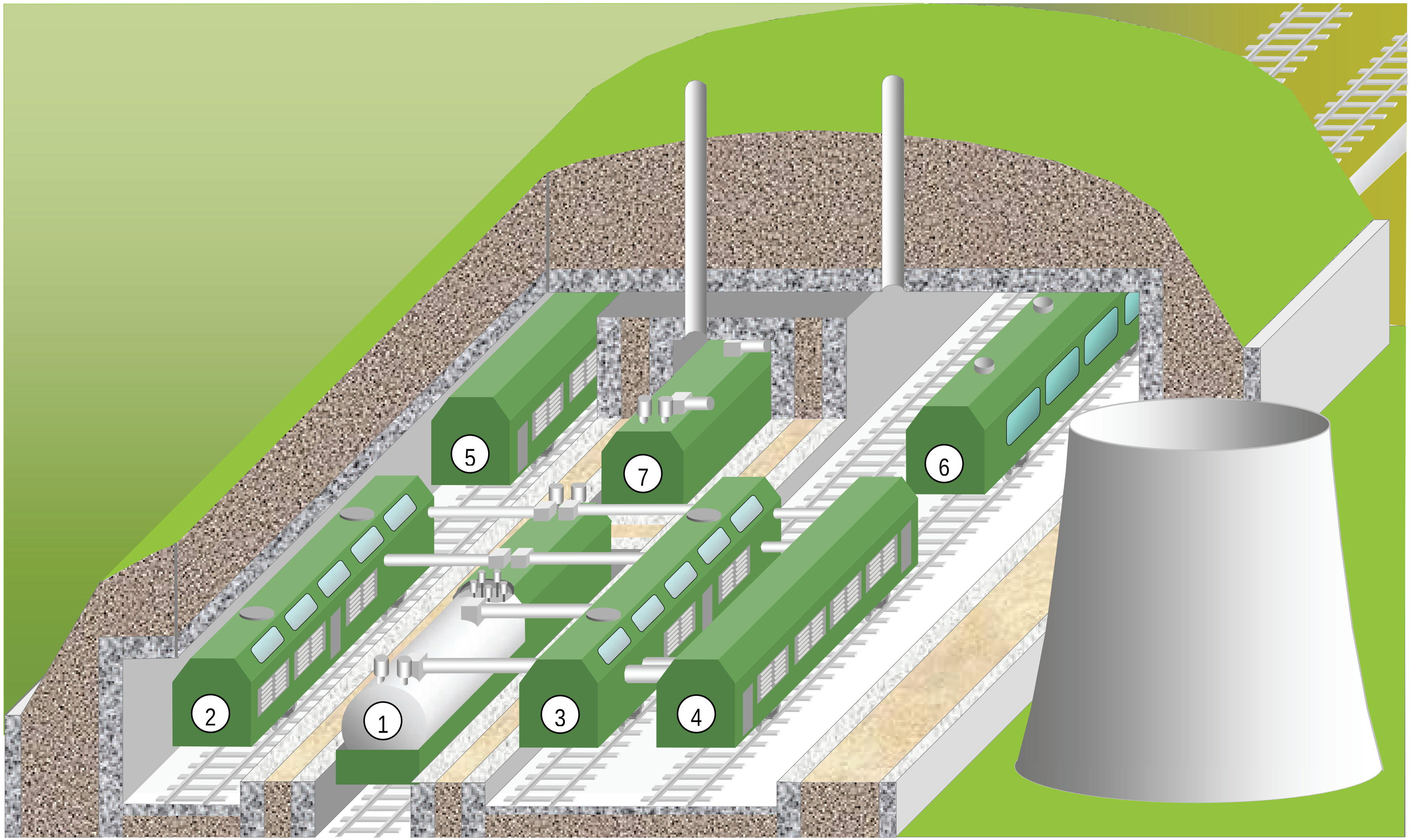
XVIII-1.2. Applications

The BN GT-300 reactor installation is being designed for a cogeneration plant producing up to 300 MW of electricity and up to 100 MW(th) of heat for district heating. These outputs could be varied; with an electricity production decrease down to 220 MW(e), the heating output could be increased up to 440 MW(th). In addition, the heating output could be redirected for potable water production if transportable modules of a desalination plant are delivered to the site.

XVIII-1.3. Special features

The BN GT-300 basic design is that of a land-based nuclear power station. Such option was selected in pursue of simplified equipment manufacture and broader siting options and simpler and less costly physical protection. The main equipment modules are prefabricated and rail transportable. Railways, shelter buildings, cooling towers and some other equipment are definitely not transportable; they are stationary.

Seven rail transportable modules of the BN GT-300 in a shelter building are shown in Fig. XVIII-1. The main equipment includes a second reactor unit to increase the availability factor. This permits a reduction in the refuelling period by reconnecting power circuit equipment from one reactor unit to another.



1 – Rail transportable reactor module; 2 – Generator module; 3 – Compressor module; 4 – Heat exchanger module;
5 – Auxiliary equipment module; 6 – Control room and reserve equipment module; 7 – Reserve reactor module.

FIG. XVIII-1. BN GT 300 single-unit nuclear power plant (NPP); section of the shelter building.

The operation mode presumes no on-site refuelling and fresh or spent fuel storage facilities on the site. Fuel management for serial BN GT-300 plants would be concentrated at a single (regional) site. The reactor modules for refuelling (de facto, replacement) will be delivered to and moved from the operation site in conventional spent fuel containers.

Placing of the BN GT-300 main equipment on a floating platform is also possible but a land-based plant option is viewed as safer and cheaper. Land-based plant option does not exclude the delivery of certain equipment items by water transport.

Operation of the pilot BN GT-300 plant would require about 70 qualified and 12–15 highly qualified staff members permanently present at the site. Experience in the initial plant operation could help reduce this figure with a possibility of totally remote control for future serially produced plants.

XVIII-1.4. Summary of major design and operating characteristics

Installed capacity

The BN GT-300 is designed to yield maximum output within the limitation of rail transportability. A number of combinations were tested including different reactor and turbine types. The combination of sodium cooled small reactor and specially designed gas turbine has shown promising results. Further development of this combination led to the BN GT-300 concept and design. The BN GT-300 reactor produces 730 MW(th), enough to co-generate 300 MW of electric power and 100 MW of heat for district heating.

Mode of operation

For the first-of-a-kind BN GT unit, base load mode of operation was selected; further BN GT units could be operated in load follow modes within electric output variation between 90 and 300 MW(e). The prerequisites for this are fast neutron spectrum (no effect of xenon poisoning in reactivity) and the use of a special gas turbine. To realize load follow operation modes, a demonstration of fuel element reliable performance under multiple power ramps and associated thermo-cycling would be needed, which could be accomplished during the operation of a first-of-a-kind plant.

Load factor/ availability

The load factor targeted for the first-of-a-kind plant with the BN GT-300 is 80%; in further plants it could be improved through learning. The theoretical limit for availability factor is close to 98%; it could be considered as a target for the BN GT-300.

Major design characteristics of the BN GT-300 are given in Table XVIII-1.

TABLE XVIII-1. SUMMARY OF MAJOR DESIGN CHARACTERISTICS

ATTRIBUTES	DESIGN PARTICULARS
Thermal output, MW	730
Electric output, MW - Gross - Net	 305 300
Plant net efficiency on electricity generation, %	~ 41
Plant net thermal efficiency on electricity generation and heat production %	~ 55

ATTRIBUTES	DESIGN PARTICULARS
Useful heat power in cogeneration mode[1], MW	100
Maximum useful heat power in cogeneration mode[2], MW	440
Reactor type	Integral primary circuit design
Reactor layout	Mono-block with horizontal double cylindrical vessel and vertical reactor vessel welded into it from inside
Reactor vessel dimensions, height × diameter	22 × (4.4–5.6) m
Core configuration	Vertical cylinder; triangular lattice of cylindrical fuel pins in fuel assemblies
Core diameter	2.5 m
Active fuel length	1.1 m
Fuel	First load: UO_2; next loads: MOX or UN
Enrichment	Different for different fuel loads; ~ 17 % of ^{235}U for uranium fuel load
Fuel pin cladding	Stainless steel
Reactor structural materials	Stainless steel
Moderator	None
Number of circuits	- Primary sodium circuit; - Gas turbine power circuit; - District heating circuit.
Primary coolant	Liquid sodium
Secondary coolant	Mixture of argon and nitrogen
Coolant of the district heating circuit / temperature	Water / 120°C
Normal mode of core cooling	Forced circulation; four shaft pumps with electric drives
Mode of decay heat removal	Natural convection
Number of loops in primary (sodium) circuit	2
Number of loops in secondary (gas turbine) circuit	1
Primary shutdown system	Mechanical; 12 independent shutdown control rod clusters

[1] - Without electric output reduction.
[2] - With an appropriate electric output reduction.

ATTRIBUTES	DESIGN PARTICULARS
Secondary shutdown system	Mechanical; 12 independent control rod clusters of the reactor control system
Operation period between refuellings, years	4.5
Average burn-up of discharged fuel, MW day/kg	46
Design service lifetime of reactor vessel, years	45
Estimated fabrication cost for the first / serial BN GT-300, US$ million	~ 178/ ~ 143
Estimated specific fabrication cost for the first / serial BN GT-300, US$/kW	~ 593/ ~ 477
Estimated primary electricity generation cost for the first / serial plant, cent/kW h	~ 1.0/ ~ 0.95

Simplified schematic diagram

For operation, all transportable units of the BN GT-300, including the turbo-generator, auxiliary electric machinery and others, are placed into a special building capable of withstanding an aircraft crash or a terrorist attack, Fig. XVIII-1. The same approach, i.e., providing shelter for the main equipment, would be applied for multi-module BN GT plants.

The operation scheme assumes the presence of two reactor modules on the site: one (1 in Fig. XVIII-1) is in normal operation; the other (7 in Fig. XVIII-1) is either a module with spent nuclear fuel awaiting for a reduction in the decay heat and transport to a refuelling factory, or a module with a 'fresh' core awaiting the end of operation of the operating module. When an active reactor core reaches the limits of energy generation, it gets shutdown. The rest of the main equipment is reconnected to another reactor module, which incorporates a fresh fuel load and is waiting in the second reactor compartment of the building. It is anticipated that reconnecting operations could be completed within 1 month. A newly shutdown reactor is then stored in the shelter for a sufficient reduction in decay heat to allow for its transport to a refuelling site. No refuelling is necessary during operation and no on-site refuelling is allowed.

As it was already mentioned, the BN GT main equipment could be placed on the site of a former power plant or on a proper floating platform. In these cases, the NPP general layout should be modified to best fit the new conditions. The minimal required equipment set (modules 1–4 in Fig. XVIII-1) could be placed on a properly designed barge with a full displacement of 3.500–4.500 metric tons.

Neutron-physical characteristics

Simplified design of the modular and transportable BN GT-300 might reduce the manufacture and construction costs to an extent when the fuel share in electricity generation cost would rise up to 55–65%. Therefore, the reactor physics of the BN GT-300 is currently being optimized for the fuel cost impact reduction. Such optimization targets achievement of the highest energy production per initial core cost in a fuel cycle without reprocessing. As a result of this optimization, the fuel share in primary electricity generation cost was reduced down to 46%.

The neutron-physical characteristics of the BN GT-300 make it easy to operate with MOX fuel containing plutonium and actinides from the spent fuel of light water reactors (LWRs). MOX fuel could also be used to immobilize weapons-grade plutonium.

The BN GT core could also be adapted for the uranium nitride (UN) fuel use.

Different variants of the core load differ in temperature and coolant density reactivity coefficients, but their sign and the void reactivity effect are always negative. The burn-up reactivity swing for the fuel lifetime varies from –7.0 β_{eff} to 0 (for UN fuel).

The radial power peaking factor is –1.23, due to the use of 3 axial zones with different enrichments. An increase in the number of axial zones could additionally reduce radial power peaking but also leads to an increase in core cost.

Reactivity control mechanism

The reactor control and protection system includes the emergency shutdown system and normal operation system.

The emergency shutdown system consists of 12 independent control rod clusters, with each 6 being capable to shut down the reactor.

The normal operation reactivity control system consists of 12 independent control rod clusters. Only 3 of them are simultaneously in use, providing a maximum reactivity margin of less then 0.8 β_{eff}. The drop of reactivity control clusters also ensures the reactor shutdown.

All fast neutron spectrum reactors are sensitive to reactivity effects due to spectrum shifts caused by a prompt introduction of moderating materials to the core. Adding a spectrum sensitive burnable poison to the BN GT fuel secures negative reactivity effects even when the core is flooded with water completely.

Cycle type

The electricity generation cycle is indirect; heat from the primary (sodium) circuit is transferred, via heat exchanger, to the secondary power circuit and heats gas that turns the turbine. The turbine outlet gas is used to deliver heat for district heating or seawater desalination. This indirect system minimizes the potential impact of reactor radioactivity on the common use water. The thermodynamic efficiency is 42%.

Thermal-hydraulic characteristics

The circulation is forced in both circuits. Maximum pressure in the primary (sodium) circuit is 0.8 MPa, in the secondary (gas) circuit – 15 MPa.

Maximum/average discharge burn-up of fuel

The average discharge burn-up for UO_2 fuel is 5.0 weight %.

Fuel lifetime/period between refuellings

The fuel lifetime is equal to the period between refuellings and constitutes 1750 effective full power days.

Mass balances/flows of fuel and non-fuel materials

The mass balances and flows of fuel and non-fuel materials associated with the BN GT-300 construction and operation are given in Tables XVIII-2 and XVIII-3.

TABLE XVIII-2. ESTIMATED MASS FLOWS OF MATERIALS ASSOCIATED WITH THE BN GT-300 CONSTRUCTION

MATERIAL	REQUIREMENT	REMARKS
Stainless steel	116.3 kg/MW(e)/year	Total: 1570 tons
Concrete	1.1 m^3/MW(e)/year	Total: 15 000 m^3
Electric machinery, cables and others	16.3 kg/MW(e)/year	Total: 220 tons

TABLE XVIII-3. ESTIMATED MASS FLOWS OF MATERIALS ASSOCIATED WITH THE BN GT-300 OPERATION (ASSUMING THE USE OF UO_2 FUEL WITHOUT REPROCESSING)

MATERIAL	REQUIREMENT	REMARKS
UO_2	15 kg/MW(e)/year	
Including ^{235}U	2.5 kg/MW(e)/year	About 0.3 kg/MW(e)/year with MOX recycling
Stainless steel	10.2 kg/MW(e)/year	
B_4C	0.11 kg/MW(e)/year	Natural composition

Design basis lifetime for reactor core, vessel and structures:

Main equipment of the rail transportable module set	- 45 years
The core	- 4.5 years with the possibility of prolonging up to 6 years
Shelter building (in the basic operation scheme)	- up to 135 years

Design and operating characteristics of systems for non-electric applications

The output temperature of heat for district heating is up to 120°C. Heat of the same output could be used for potable water production, which would require some additional modules beyond the minimal working configuration shown in Fig. XVIII-1 (such option is under investigation currently).

Economics

The specific capital cost is estimated at up to 470 US$/kW(e) in serial production (for the basic operation scheme). The estimated construction period is 18 months in serial production.

The estimated electricity generation cost for a serial plant in the basic operating scheme is 0.009 US$/kW(e)/hour. This value could be reduced when using MOX or UN fuel in a closed fuel cycle.

XVIII-1.5. Outline of fuel cycle options

For the initial period of operation of the first BN GT-300 plant, a once-through UO_2 based fuel cycle is assumed.

For the BN GT plants operating outside Russia, a once-through UO_2 based fuel cycle with the refuelling performed in Russia or at a special protected site on a national territory, under monitoring by the Russian refuelling service team, is assumed.

For serial plants with the BN GT-300 on Russian sites, MOX fuel recycled from pressurized water reactor (PWR or VVER) or RBMK spent fuel could be used.

For mass-produced BN GT plants, the uranium nitride fuel with or without reprocessing could be used.

Any cost-effective reprocessing method could be applied within a closed fuel cycle. For the near-term, the considered UO_2 and MOX fuel loads are fully compatible with the existing reprocessing capacities for the BN type reactors.

It is assumed that there would be a centralized reactor module-reloading site for every 10–20 plants. The existing Russian fuel recycling plants would be capable to manage and recycle the discharged fuel (if the recycling alternative is chosen).

The BN GT spent nuclear fuel has a lower burn-up than that of the BN-600 reactor but is comparable in other characteristics. Therefore, the existing infrastructure for sodium cooled fast reactor spent nuclear fuel management could be used; it also could be modified gradually to increase cost effectiveness.

XVIII-1.6. Technical features and technological approaches that are definitive for BN GT-300 performance in particular areas

XVIII-1.6.1. Economics and maintainability

The BN GT-300 design development targets all potential markets. Initially, the BN GT design was intended to provide electricity and heat to medium sized cities. Detailed investigation has shown that the incorporated maximum degree of plant prefabrication and transportability would allow the construction of the BN GT in many countries. Flexibility in module location allows fitting to any site of a proper size with an access to water. The projected low electricity generation costs could ensure plant competitiveness even with a decrease in organic fuel costs.

Different from many sodium cooled reactor designs, the BN GT-300 employs a two circuit scheme with no intermediate heat transport system.

The BN GT plant could be easily relocated from sites with emergent economic, political or other situations. Plant removal from the site does not mean plant demolition; if removed, the main equipment set does not loose reliability or availability and could be easily assembled for further use at another site. Smaller versions of the BN GT for autonomous operation are currently under investigation.

The BN GT-300 design employs a high grade of standardization, factory fabrication and transportability. The main equipment lifetime could be prolonged by the duration of one additional fuel lifetime, assuming that present day safety requirements would be valid and would not become stricter during the whole projected period of plant operation. In this aspect, a better approach may be to use two reactor modules with the state-of-the-art safety and economy characteristics and a 45-year lifetime, than to try to build reactors capable of meeting safety standards that would be valid in the second half of the 21st century.

The BN GT shelter building could be re-equipped or re-constructed for locating another rail transportable NPP set after a 90-year initial use. The BN GT plant provides for no operations with fuel at a site, which may help reduce the operation and maintenance costs.

The best currently foreseen fuel cost reduction option is related to the use of UN fuel in a closed fuel cycle; however, this is not a commercial technology currently.

XVIII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

A fast neutron spectrum potentially allows the BN GT-based system with fuel recycling to produce energy consuming only depleted uranium, but such a system is currently less economically competitive than that employing spent PWRs' MOX fuel as the initial fuel load.

Compared with PWR reactors, the BN GT electricity generation efficiency of 41–42% allows a 13–15% reduction of fission products generation per unit of electricity produced. In a cogeneration mode, the specific fission products generation could be 30–50% lower, as the plant efficiency taking into account district heating load would be up to ~ 55%.

Reactors of the BN type have very low dose impact characteristics. The secondary circuit of the BN GT has a low level of induced radioactivity and thus practically zero dose impact. Centralized refuelling would also contribute to the reduction of dose impacts.

Being a fast reactor, the BN GT-300 could contribute to sustainability via effective use of the uranium resources, through high conversion ratio and operation in a closed fuel cycle. Further stages of design optimization would address recycling options for the BN GT-300 in more detail.

XVIII-1.6.3. Safety and reliability

Safety concept and design philosophy

A single critical point of sodium cooled reactors of the BN type is the possibility of water-to-sodium leakage and sodium-water reaction. The BN GT-300 retains all positive features of previous sodium cooled reactors but makes use of a gas turbine, which eliminates leakage as a problem. In addition, primary circuit parameters (such as coolant flow rate, neutron flux, fuel burn-up and others) of the BN GT-300 were selected lower than those in conventional BN type reactors and also improve the BN GT safety in transient modes.

Provisions for simplicity and robustness of the design

The use of a two circuit scheme instead of a 3-circuit one (elimination of intermediate heat transport system) and the absence of on-site refuelling facilitate simplification of the primary circuit design. The absence of steam generators, steam condensers, water chemistry and purification units and others significantly simplifies the design of the turbine circuit. As a result, the relative weight of the BN GT main equipment is about 5 t/MW(e) while serial PWRs have 15–20 t/MW(e) and some APWR designs over 25 t/MW(e).

Active and passive systems and inherent safety features

In the current design, all reactivity control rod clusters are built as active. Passive shutdown rods actuated by flow decrease beyond a certain limit (the same as proposed for the BN-800) are under investigation.

The thermodynamics of the selected gas turbine allows it to operate with decreased efficiency at 0.1–10% of the reactor nominal power. Therefore, as long as the reactor generates thermal power of any magnitude, the gas turbine could be used for the reactor after-cooling, i.e. the gas turbine itself could be classified as a passive cooling system. Unlike a steam turbine, the selected gas turbine provides the reactor cooling even under a complete loss of coolant from the secondary circuit into the atmosphere.

To provide for redundancy of emergency cooling systems, the primary circuit is equipped with an auxiliary air cooling system based on multiple small size heat pipes.

As long as the gas turbine serves as a passive cooling system, it can generate electricity for the NPP electric machinery and systems even after the reactor shutdown; in other words, the gas turbine also provides the BN GT plant with a passive source of electricity. Electricity supply redundancy is achieved by placing an accumulator battery and an oil-fired emergency generator in the transportable module of the main equipment.

Fast neutron spectrum appears as a source of several inherent safety features of the BN GT-300. It ensures the absence of core poisoning both in normal operation and in accidents and simplifies the reactivity control scheme. Specifically, it allows a high breeding ratio, contributing to a reduced burn-up reactivity swing. For UN-PuN fuel, the burn-up reactivity swing is close to zero. As a result, the BN GT-300 reactivity control system is designed to be unable to release positive reactivity greater than 0.8 β_{eff}. The fast neutron spectrum also ensures smaller impact of the control rods on power flattening, contributing to an improved thermo-hydraulics of the primary circuit in normal operation and in accidents.

Structure of the defence-in-depth

The structure of the defence in depth is quite typical of many reactors; it includes the following main barriers:

- Fuel composition;
- Fuel rod cladding;
- The integral design of the primary circuit (reactor vessel);
- A second (guard) vessel surrounding the main vessel;
- A reactor cavity inside the shelter building;
- The shelter building itself.

Design basis accidents and beyond design basis accidents

These were under study at the time of this report; no detailed information was provided.

Provisions for safety under seismic conditions and protection against external events

Preliminary studies have shown high seismic resistance of the BN GT-300, which is due to simplicity of the protective structures and their low axial profile. No further details were provided.

The main equipment of the BN GT-300 is surrounded by a shelter building. The shelter building is designed to withstand the fall of a Boeing-class airplane with a full fuel load; it protects the BN GT both from direct hit of a plane and from the following fire resulting from the plane fuel.

The shelter building also protects the plant against external explosions and impacts of the cumulative weapons.

Probability of unacceptable radioactivity release beyond the plant boundaries

Very low values are projected taking into account both internal and external events and their reasonable combinations; no further details were given.

Measures planned in response to severe accidents

The most severe accident with radioactivity release is that with gas leakage from the secondary circuit. As the activation of gas will be low, no special measures would be required beyond the plant boundary

XVIII-1.6.4. Proliferation resistance

Technical features of the BN GT-300 that contribute to an enhanced proliferation resistance are the following:

- With the use of uranium based fuel, the maximum fuel enrichment is less than 20%;
- The absence of on-site refuelling means that there are no fresh or spent fuel storage facilities on the site and no access to the core during the whole period of reactor operation; with this, all accounting and verification procedures are concentrated at special refuelling sites;
- Burnable poison mixed with the fuel also reduces its attractiveness for weapon programmes.

XVIII-1.6.5. Technical features and technological approaches used to facilitate physical protection of BN GT-300

The actions of personnel could be remotely monitored with high degree of detail due to simplicity of the main equipment and a reduced number of personnel on the site. Normal operation of a serial BN GT plant would require the simultaneous on-site presence of only 12–15 qualified staff members.

XVIII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of BN GT-300

For users outside the Russian Federation, only fuel leasing would be allowed because the lifetime of the sodium coolant circuit may be reduced dramatically if an uncertified fuel load is used. Fuel leasing for users abroad could be controlled via centralized refuelling.

NPP leasing might be an option with a proper bank guaranty. Because the BN GT plant can be easily relocated, it could be leased for a period shorter than the total plant lifetime.

In addition to the abovementioned, smaller versions of the BN GT under investigation could offer autonomous operation in distant or poorly populated areas (northern or deserted regions, remote islands).

XVIII-1.8. List of enabling technologies relevant to BN GT-300 and status of their development

The key enabling technologies of the BN GT-300 result from the extensive Russian experience in fast-spectrum sodium cooled reactors and the special design of the gas turbine.

Several other technologies contributing to high reliability and efficiency of the plant were under patent clearance at the moment of this report; therefore, no additional information was provided.

XVIII-1.9. Status of R&D and planned schedule

The research and development (R&D) for the BN GT-300 is only partially supported (funded) under a national industrial programme titled “Innovative reactor technologies and technologies of closed fuel cycle”. The current design phase is that of an early conceptual design. International cooperation is foreseen as an option starting from the basic design development phase.

At the time of this report, only the State Scientific Centre Institute of Physics and Power Engineering (IPPE) was involved in the R&D and design.

Under favourable conditions, it may take 3 years to complete the design project. Licensing of the first plant may take place in parallel with the project work. Construction of the first plant would take 2 years.

It would require an estimated 4.5 years (operation with the first fuel load) for full-scale plant testing and optimization of the operating mode. As a result of operating mode optimization, the plant could be licensed for production in series.

In the terminology applied, the first BN GT-300 plant is not a prototype. It is the first-of-a-kind pre-serial plant, which differs from a serial plant only in the uncertainties concerning the operating mode. After the operating mode is optimized, it could operate as the serial plant.

The estimated R&D costs to complete the BN GT-300 design project are US$ 35 000 000. The estimated cost of the first-of-a-kind plant manufacturing is US$ 171 000 000.

The estimated cost of in series rail transportable plant fabrication and on-site placement is US$ 148 000 000. This figure does not include the expenses associated with plant transportation.

Financial information relevant to the BN GT-300 project and projected effectiveness of the first-of-a-kind plant is given in Tables XVIII-4 and XVIII-5; more details are given in [XVIII-4].

TABLE XVIII-4. FINANCING REQUIRED TO BUILD FIRST-OF-A-KIND BN GT-300

YEARS, STARTING FROM 2006	FINANCING REQUIREMENTS FOR PROJECT PHASES	
	Financing required, million US$	*Project phase name*
3	35.0	Basic design, detailed design, all project related tests and site evaluation
4	136.0	Component manufacturing
	6.5	Site surveys, shelter building and stationary components
5	0.5	Delivery of mobile unit set to the site; balance and commissioning works on the site

TABLE XVIII-5. ESTIMATED EFFICIENCY OF THE FIRST BN GT-300 PLANT OPERATING IN CURRENT RUSSIAN ELECTRICITY MARKET [XVIII-5]

Net profit value (NPV), million US$	475.2
Internal rent ratio (IRR), %	10.4
Profit index (PI)	4.2
Payback period (PB), years - taking into account the discount rate - without discount rate	 12.2 10.7
Payment for credit, US$ million	69

XVIII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The BN GT-300 plant is based mostly on proven technical features and technologies that had been used previously in nuclear installations of different destinations.

However, the first-of-a-kind plant will be needed because the BN GT-300 incorporates a new combination of these technologies and design features, the combination that has never been applied before.

XVIII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

Being a small sized reactor plant, the BN GT-300 has certain similarities in the design philosophy, design approaches and certain technologies with other SMRs, such as VBER-300 [XVIII-2, XVIII-3]; VK-300 [XVIII-2, XVIII-3]; GT-MHR [XVIII-3]; KLT-40 [XVIII-2]; SVBR-75/100 [XVIII-6, XVIII-7]; and others [XVIII-8, to XVIII-10].

XVIII-2. Design description and data for BN GT-300

XVIII-2.1. Description of the nuclear systems

Reactor core and fuel design

The circumscribed diameter of the core is 2.4 m. The core and primary system design data is summarized in Table XVIII-6.

TABLE XVIII-6. CORE AND PRIMARY CIRCUIT DESIGN DATA

CHARACTERISTIC	VALUE
^{235}U load, kg,	~ 3470
Average specific power density, kW/l	147
Number of fuel elements in the core	~ 52 000
Number of control rod clusters in the core	24
Volume of primary coolant (sodium), m^3	~ 20
Nominal pressure in the gas cavity over the coolant, MPa	0.4

Each cluster of the control group of control rods replaces 91 fuel elements in the core. Each cluster of the shutdown group of control rods replaces 19 fuel elements in the core. The fuel element design for a first-of-a-kind BN GT is similar to that of the BN-600 reactor [XVIII-11], including the materials used.

Main heat transport system

Heat removal in both normal operation and in accidents is accomplished via the use of the staff heat removal systems and equipment, as shown in Fig. XVIII-2.

After a reactor shutdown, the heat and power plant is brought to the mode of reactor core aftercooling.

To reduce dose impacts on the personnel, aftercooling within the first month is performed without disconnecting the turbine plant cars from the reactor car. During this time, a reduced power operation of the gas turbine plant is used to remove heat, via heat exchangers, from the primary coolant. In this, turbo-machines could be used both jointly and separately, within a simple gas turbine cycle. Separate use of the turbo-machines makes it possible to organize two independent channels for aftercooling.

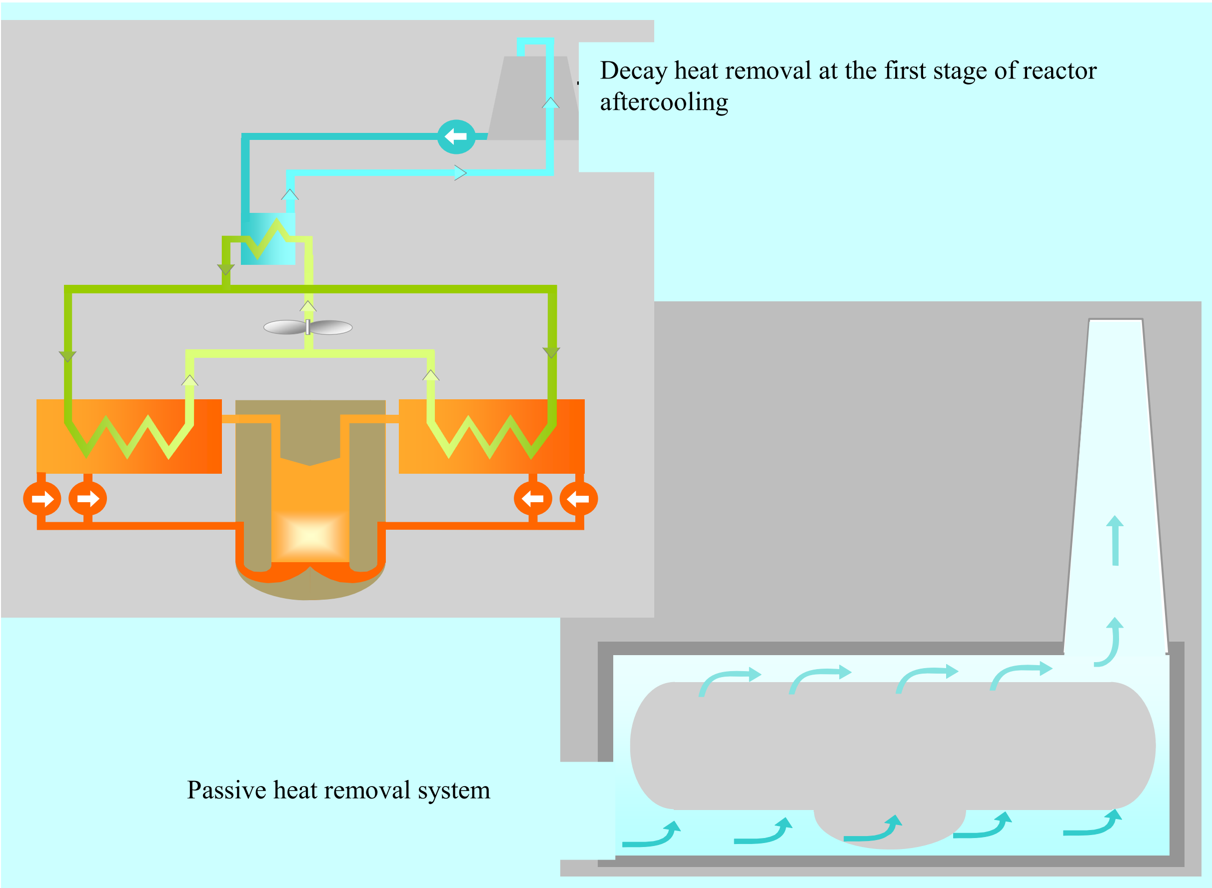


FIG. XVIII-2. Schematic of the BN GT-300 main heat removal system.

To limit maximum temperature of the reactor module in the first period of aftercooling, forced circulation of the primary sodium is employed, using the sodium circulation pumps. In this, the power for the pumps is supplied either from the main generator or from small-power generators connected to the shafts of the turbo-machines, or from reserve power sources – on the total, there is a five-fold redundancy of power supply for the main circulating pumps.

In addition to the turbo-machines, a gas-blower blowing the gas through gas heaters (heat exchangers) could be used at the first stage of an aftercooling.

Subsequently, aftercooling is performed via natural convection of the atmospheric air, heated in a gap between the guard vessel and the surrounding non-sealed shell.

Intermediate circuit, if any

There is no intermediate circuit in the BN GT-300 design.

XVIII-2.2. Description of the turbine generator plant and systems

The power circuit is a gas turbine one with closed circuit and complex thermodynamic cycle. As it was already mentioned, the operating medium is a mixture of argon and nitrogen gases. The thermodynamic cycle efficiency is 42%.

XVIII-2.3. Systems for non-electric applications

The BN GT-300 is being designed to allow for co-production of electricity and heat that could be used for district heating or seawater desalination. Standard systems for non-electric applications are planned to be used; therefore, they are not described here.

Heat for district heating and potable water could be produced using off-peak electric power without reducing the electricity generation of the plant. In a cogeneration mode, an electric output reduction grants substantial thermal energy that could be used for non-electric applications.

XVIII-2.4. Plant layout

General philosophy governing plant layout

Independent of the mode of energy production, only certified sites meeting certain requirements can be used for this purpose.

As a rule, the site should be located near the existing transport and network infrastructure, should provide for a sufficient amount of water resources to cool process equipment of the plant, etc.

The BN GT-300 heat and power plant may have many options for siting, because in the Russian Federation there are 5–10 times more potential sites for allocation of 300 MW(e) NPPs than those able to accommodate 3 GW(e) combinations of power units; on the total, there may be 25–100 sites for 300 MW(e) NPPs.

Another siting option is to use the BN GT modules for replacement of the decommissioned power units of previous generations; such an approach is referred to as renovation.

For the latter option, complications may come from the site being packed with still available (eventually demolished) buildings and structures. Modular approach with railcar delivery of 100% factory ready units of the BN GT-300 plant may help comply well with the siting conditions that offer limited space. The simplicity of the BN GT shelter building makes it possible to build it upon the foundations or structures remaining from plants previously located on the site, suppose they still have a sufficient lifetime margin.

Replacement of the decommissioned VVER-440 and RBMK-1000 reactors with the use of the BN GT-300 modules may offer certain advantages related to a high thermodynamic efficiency of the gas turbine cycle (~42%), e.g. help increase the cumulative power of the plant keeping the same level of heat discharges to the atmosphere. As the BN GT plant is capable to produce up to 100 MW(th) of heat power without reducing the electric output, the effective level of waste heat per unit of the produced electric power could be made even lower. Using ecological and other criteria, it may also be possible to use the BN GT-300 heat and power plants for renovation of the old fossil fuel power stations, suppose their sites meet licensing requirements for NPPs.

Reactor building and containment layout

As it was already mentioned, placing not only the reactor but other equipment of the heat and power plant under an earth-berm and concrete shelter (Fig. XVIII-1 and XVIII-3) provides a reliable protection against aircraft crash, external explosions and impacts of cumulative weapons.

Basic scheme of the BN GT-300 operation, which provides for no operations with nuclear fuel at the site, facilitates securing a restricted access to fuel, the more so as there are no fresh or spent fuel storage facilities at the site. In turn, an attempt of unauthorized removal of the reactor-module car from the plant could be effectively prevented by heavy weight of the car (~500 t) and by the staff replacement of wheels by the foundation blocks, as well as by 'tight' design of the stationary radiation shielding (part of the shelter building) enveloping the car.

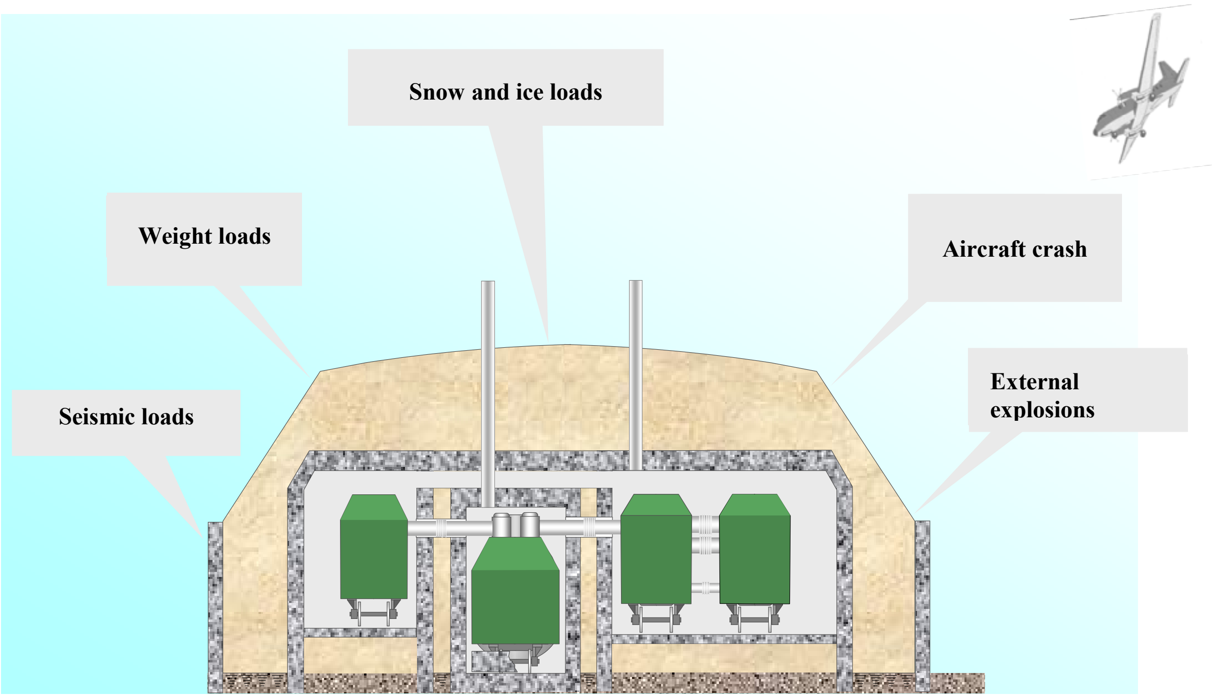


FIG. XVIII-3. Lateral section of the BN GT-300 shelter building.

Plant plot, if available

The allocation of the main equipment within the shelter building of the BN GT-300 heat and power plant is shown in Fig. XVIII-1 and XVIII-3.

REFERENCES

[XVIII-1] TREVGODA, M.M., BUDYLOV, EU.G., OSHEYKO, YU.V., Modular-transportable NPP BN GT-300/100 based on sodium-cooled reactor and gas turbine, IPPE report (Obninsk, Russian Federation, 2003).

[XVIII-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs 2004, IAEA-TECDOC-1391, Vienna (May 2004).

[XVIII-3] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Innovative Small and Medium Sized Reactor Designs 2005: Reactors with Conventional Refuelling Schemes, IAEA-TECDOC-1485, Vienna (March 2006).

[XVIII-4] BUDYLOV, EU.G., TREVGODA, M.M., OSHEYKO, YU.V., Some economical considerations about small NPP construction, Small Sized Power Plants 2004 (Proc. Int. Scientific and Practical Conference, 11–14 December 2004, Moscow, Russian Federation), ISBN 5-9900048-5-0 (on a CD, in Russian).

[XVIII-5] Lending Rates for LIBOR-Based Single Currency Loans, http://treasury.worldbank.org/Services/Financial%2bProducts/Lending+Rates+and+Loan+Charges/usdpostL.htm

[XVIII-6] ZRODNIKOV, A.V., et al., NPPs based on reactor modules SVBR-75/100, Atomnaya Energiya, Vol. 91, Issue 6, p. 415 (December 2001).

[XVIII-7] IGNATENKO, E.I., et al., Use of SVBR-75 reactor plant in design of renovation of NPP units of the first generation after termination of their service life, Heavy-Liquid Metal Coolants in Nuclear Technology (Proc. of Conf.), Vol. 2, Paper No. B.5, p. 366 (Obninsk, Russia, 1999).

[XVIII-8] “Boiling water reactor VK-12” http://www.grpress.podolsk.ru/Designs.html#s8 (in Russian).

[XVIII-9] “Modular-transportable NPP ANGSTREM” http://www.grpress.podolsk.ru/Designs.html#s9 (in Russian).

[XVIII-10] DAVIDOV, A.N., ADAMOVICH, L.A., Small NPPs for remote regions of Russia” http://www.rosteplo.ru/Tech_stat/stat_shablon.php?id=266&name=nie_mgrI9n.htm (in Russian).

[XVIII-11] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Liquid Metal Cooled Reactor Technology, IAEA-TECDOC-1083, Vienna (April 1999).

LEAD AND LEAD-BISMUTH COOLED SMALL REACTORS

LEAD BISMUTH COOLED FAST REACTOR SVBR-75/100

IPPE, EDO “Gidropress”,
Russian Federation

XIX-1. General information, technical and features and operating characteristics

XIX-1.1. Introduction

The SVBR-75/100 is a modular multi-purpose lead-bismuth cooled fast reactor of the equivalent electric power between 75 and 100 MW(e), depending on steam parameters.

The technical basis of the SVBR-75/100 design is as follows:

(1) The experience of 50 years in design and operation of reactor installations with lead-bismuth coolant for nuclear submarines. The operation experience of this type of reactors equals to 80 reactor-years [XIX-1];

(2) The experience in construction and operation of sodium cooled fast reactors;

(3) The experience of mastering lead-bismuth heavy liquid metal coolant technology.

In the course of this effort, the following scientific and technical problems specific to the lead-bismuth coolant have been solved:

- The problems of lead technology, corrosion resistance of structural materials and mass transfer in the primary circuit were comprehensively addressed and resolved; when operating second generation reactor installations, there were no problems with corrosion resistance of structural materials or in meeting the requirements for coolant and circuit quality [XIX-2];
- The radiation safety was ensured under the generation of polonium-210; during the operation of lead-bismuth cooled reactor installations, including primary circuit equipment repair and removal of spilled lead-bismuth coolant, there was no personnel irradiation over the permissible limits for this radionuclide [XIX-3];
- The problem of maintaining the reactor equipment reliability under multiple “freezing/de-freezing” of lead-bismuth coolant was solved; this successful solution was demonstrated at large-scale facilities and in the reactor installations of nuclear submarines [XIX-4].

In 1997, through the co-operation of experts from the Federal State Unitary Enterprise (FSUE) Experimental Design Organization “Gidropress” (EDO “Gidropress”), FSUE State Scientific Centre of the Russian Federation Institute of Physics and Power Engineering (FSUE SSC RF IPPE), and FSUE “Atomenergoproekt” the design drawings of the SVBR-75/100 for renovation of the 2nd, 3rd and 4th units of the Novovoronezhskaya NPP [XIX-5] were completed.

In 2001, the abovementioned experts developed a conceptual design of the two-unit NPP based on thirty two SVBR-75/100 reactor modules; the output of each of the units is 1600 MW(e) [XIX-6].

At the beginning of 2004, the development of multi-purpose nuclear power sources based on the SVBR-75/100 reactor modules has been being initiated by FSUE SSC RF IPPE, FSUE EDO “Gidropress”, and FSUE “Atomenergoproekt”.

XIX-1.2. Applications

Standardized SVBR-75/100 reactor modules will incorporate many inherent safety features and would make it possible to construct nuclear power plants (NPPs) of different power and purposes, such as the following:

(1) The most economically effective area of SVBR-75 use could be the renovation of NPP units with thermal reactors after the expiration of their lifetime. Such renovation could be performed by installing the SVBR-75/100 modules in the steam-generator (SG) and the main circulation pump (MCP) compartments of these NPPs after the decommissioning and dismantling of the equipment installed previously. The results of technical feasibility study and economic evaluation of the renovation of the 2nd, 3rd, and 4th units of the Novovoronezhskaya NPP with the SVBR-75 reactor modules show that the specific capital costs can be reduced by a factor of two compared with the construction of new replacement power capacities [XIX-5];

(2) In regions with cold climate, it could be expedient to build nuclear heat power plants of 200-600 MW(e) in locations near the cities to minimize heat transport [XIX-7];

(3) In countries with large power systems, it could be expedient to build large-power modular NPPs [XIX-6];

(4) The results of development of the SVBR type reactors for application within the floating NPP KRUIZ-50 are presented in reference [XIX-8]. A specific feature of the floating NPP with any kind of nuclear reactor is diving service of the ship vessel, difficulties with the physical protection system (construction of a stationary base location protected against diving saboteurs), protection against external natural events (such as tsunami or typhoon) that require additional capital expenditures. For the abovementioned reasons, the concept of a stationary coastal nuclear power plant consisting of the “nuclear island” including a transportable (floating) reactor unit based on the SVBR-75/100 (or one of its lower power versions) and the stationary on-shore turbine-generator system plus a system of heat removal to the ultimate heat sink, etc. has been proposed. In this case, traditional methods could be used to ensure safety and security of the plant;

(5) In developing countries, the use of the SVBR type reactors is possible through leasing of the transportable reactor unit to provide locally built power complexes with steam for generating electricity, heat and potable water [XIX-9]. In this case, the requirements for non-proliferation of fissile materials are ensured by using a uranium enrichment of less than 20% and by the absence of refuelling in the user-country; the refuelling, performed once in 10 years, is accomplished by transporting the reactor unit to its country of origin (the Russian Federation) in a safe and secure state provided by “freezing” of the lead-bismuth coolant together with the core within a mono-block vessel;

(6) Currently, activities have been started to determine the possibility of using the SVBR type reactors as a power sources to produce synthetic motor fuel and oil from “brown” coal with a hydrogenation method;

(7) It is possible to use the SVBR reactor as a sub-critical blanket of a proton accelerator driven system for transmutation of long-lived radioactive waste [XIX-10].

XIX-1.3. Special features

The SVBR-75/100 project has been devised using a conservative approach. This presumes to use the primary and secondary circuit design and operation parameters already proven in practice, the mastered technologies of fuel and structural materials, and those principal engineering solutions regarding the equipment components and the reactor scheme that have been verified by operating experience.

Such an approach is to assure that the technical solutions used in lead-bismuth cooled nuclear submarine reactors are to a great extent inherited by the SVBR-75/100 design. Adhering to this approach could reduce the implementation terms, the R&D scope and the cost and the investment risk, and would favour the improved reliability and safety of the reactor installation.

The specific features of SVBR-75/100 are as follows:

- Whole core refuelling, which is performed at the end of each fuel lifetime; the discharge of fuel is performed cassette-by-cassette; fresh fuel is loaded as a single cartridge with the help of a refuelling equipment set common to all reactor modules within a given NPP;
- It is possible to use different types of fuel, e.g., UO_2; mixed oxide (MOX) fuel with weapon-grade or reactor plutonium; MOX fuel with minor actinides (MA); or nitride fuel; without changing the reactor design and without sacrificing the reactor safety;
- Factory fabrication of the reactor mono-block will assure high quality of work and significantly reduced fabrication costs because the mono-blocks would be produced in large quantities;
- The possibility of reactor mono-block transport by railway or any other transport, which would support multi-purpose use of the reactor installation and reduce the terms of the construction and assembly work.

On the basis of the “standard” reactor module SVBR-75/100, it is possible to construct modular type power units with a varied number of modules and, therefore, a varied total power capacity.

The principle of modular design of nuclear steam-supply systems (NSSSs) is most economically effective for the reactors in which inherent safety features against severe accidents have been realized to the maximum possible extent. Primarily, this should be attributed to accidents with coolant loss, such as LOCA. To cope with these accidents in light water reactors, many safety systems are needed that are not necessary for the SVBR-75/100. This considerably simplifies the technology of construction and assembly and reduces the scope of construction for the reactor compartment.

With long reactor operation without refuelling, the modular design of the NSSSs of a power unit makes it possible to provide a load factor of not less than 90%. When each reactor installation is shut down in turn for refuelling, the decrease of the total power would be insignificant.

Licensing of the construction of a modular type large power unit will be essentially simplified if an industrial prototype of the reactor or a first-of-a-kind small modular power unit would be licensed and constructed first. Correspondingly, the low power of a reactor installation would determine a comparatively low cost of construction.

The conservative approach adopted for development of the SVBR-75/100 provides for a potential to improve its technical and economic parameters in the future; this could be realized through further evolutionary improvements in the design performed after accomplishing certain additional R&D.

XIX-1.4. Summary of major design and operating characteristics

The SVBR-75/100 concept has been developed to meet the following requirements:

- Meeting the demands of a maximum possible number of users for the competitive nuclear power source in which safety characteristics meet in full the regulatory requirements and even surpass them in severe accidents;
- The possibility of a direct demonstration of reactor resistance to personnel errors, equipment failures and their multiple combinations, as well as malevolent actions;
- The capability of fissile self-sustainable regime (core breeding ratio ~1) in a closed nuclear fuel cycle with mixed uranium-plutonium fuel (oxide or nitride);
- The capability to burn-out effectively both self-generated minor actinides and minor actinides from spent fuel of light water reactors (LWRs);
- Maximum use of the structural materials and maximum conformity with the primary and secondary circuit operation parameters that had already been proved in operation of marine and land-based power plants with lead-bismuth cooled reactors;
- An option of future use of new materials (after corresponding development and testing), new equipment structures and improved operating parameters, without changing the basic vessel unit design.

The SVBR-75/100 design incorporates the following features to meet the abovementioned requirements:

(1) It is a small reactor (~100 MW(e)) with an integral (mono-block) design of the primary circuit based on the mastered lead-bismuth coolant technology. The physical features of small power reactors, the natural properties of lead-bismuth coolant and a mono-block type design (location of all primary circuit equipment in a pool with complete elimination of valves and lead-bismuth coolant pipelines) make it possible to eliminate the initiating events for most severe accidents, to reduce the number of safety systems, localizing accident systems, control and protection systems and, in this way, to ensure high economic competitiveness;

(2) The core dimensions and fuel volumetric fractions ensure the core breeding ratio ~ 1 in operation with MOX or a more dense mixed nitride fuel;

(3) The levels of natural circulation in the heat-removal circuits are provided sufficient to ensure decay heat removal without unacceptable over-heating of the core;

(4) The reactor mono-block is installed in the water tank; the tank is a seismic-resistant support structure shouldering the functions of radiation protection and providing an entirely passive cooling of the reactor installation;

(5) The use of a safeguard vessel, with the gap between the main and the safeguard vessels providing circulation continuity in the event of postulated leak-tightness failure in the main vessel;

(6) A ductless design of the fuel sub-assemblies is applied, ensuring the high lateral heat-mass-exchange in the core and eliminating considerable over-heating of the fuel elements in blockages of the flow at the core inlet;

(7) A two-circuit scheme of heat removal is used that secures the design compactness;

(8) The steam-generator (SG) operates according to a multiple natural circulation scheme to produce saturated steam; this offers the best lifetime and operating characteristics, e.g., a reliable reactor operation at any power level, simplicity in maintenance of the liquid state of lead-bismuth coolant at low power levels, and simplification of the water chemistry regime of the SG;

(9) A slow-rotating gas-tight electric engine is used for the main circulation pump, with its power not exceeding 500 kW; this eliminates the necessity of tightening the rotating shafts, enables the use of ball bearings with grease and helps prevent cavitations at the suction of the main circulation pump impeller that might occur due to hydrostatic pressure of the lead-bismuth coolant column.

(10) Easy replacement and repair of the reactor installation equipment is provided.

The principal scheme of the SVBR-75/100 is presented in Fig. XIX-1 borrowed from [XIX-11].

Each reactor installation includes:

- A reactor mono-block;
- The equipment and pipelines of the primary circuit gas system;
- The equipment and pipelines of the secondary circuit;
- The equipment and pipelines of the passive heat removal system;
- The equipment of the control and diagnostics system;
- The control sub-systems of the reactor installation components;
- The transport and technological equipment of the nuclear fuel management system (for use in developing countries, the equipment of the nuclear fuel management system is not supplied to the user), the equipment for reactor assembly and balance and commissioning, and the equipment for repair and maintenance; for the modular power units it is acceptable to have a single set of equipment regardless of the number of such modules in the power unit.

The basic equipment of the SVBR-75/100 is installed in a tight box-confinement; 11.5 m in height (see Fig. XIX-2).

The number and configuration of the box-confinements can differ depending on the plant power; however, the arrangement of equipment in box-confinements is always the same.

A concrete well is formed in the lower part of each box to install the tank of passive heat removal system (PHRS). The reactor mono-block is installed inside the PHRS tank and is fixed on the support ring of the tank cover. Twelve immersible vertical heat exchangers are also installed in the PHRS tank to transfer heat from the PHRS tank water to the cooling water.

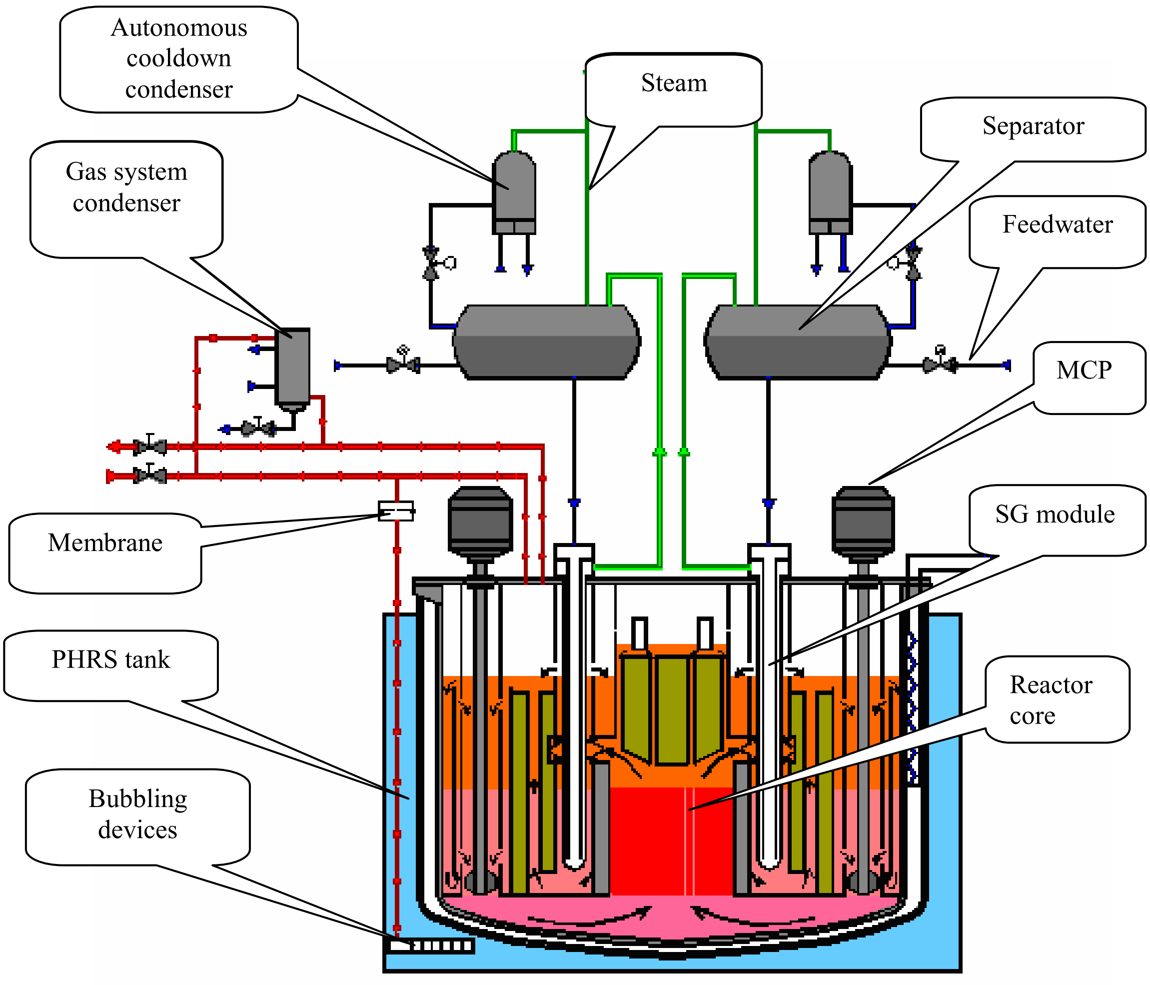


FIG. XIX-1. Principal hydraulic scheme of the SVBR-75/100.

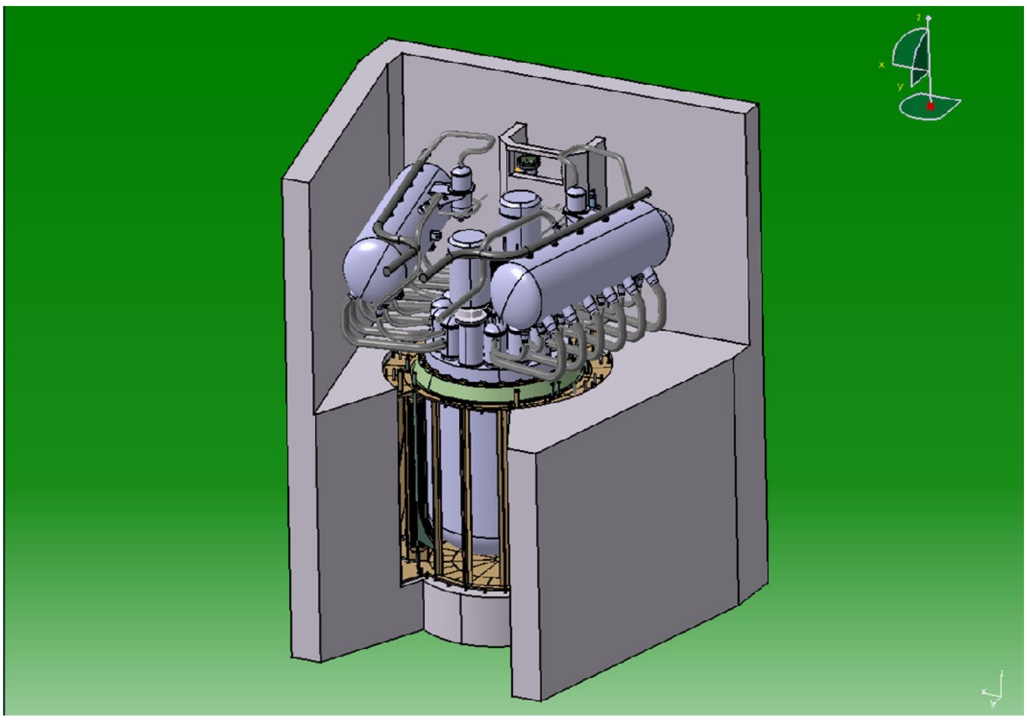


FIG. XIX-2. SVBR-75/100 equipment arrangement.

In the upper part of the box, over the PHRS tank, the reactor equipment is installed, which is not a design component of the reactor mono-block; it includes two steam separators and two cooling condensers “hung” on them. The height of the separators location is selected to provide the level necessary for natural circulation of the secondary circuit coolant in all operating modes of the reactor.

The gas system condensers are installed in the upper part of the box in a separate concrete compartment.

Major design and operating characteristics of the SVBR-75/100 reactor installation are given in Table XIX-1.

TABLE XIX-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF SVBR-75/100 [XIX-12]

PARAMETER	VALUE
Installed capacity, MW(th)	280 *
Electric power, MW(e)	101.5 *
Operation mode of an NPP with the SVBR-75/100	Base load; load follow is possible
Load factor	0.9
Availability factor	0.95
Dimensions of the reactor vessel: (diameter × height), m	4.53×6.92
Core dimensions: (diameter × height), m	1.645×0.9
Number of fuel elements	12 114
Average volumetric power density of the core, kW/litre	140 *
Average linear heat load of the fuel element, kW/m	~ 24.3 *
Fuel (UO_2): U-235 loading, kg U-235 enrichment, %	~ 1 470 16.1
Total fuel loading (UO_2), kg	9144
Core lifetime, effective full power days (EFPD)	~ 2200*
Maximum fuel burn-up, % fissile materials (FIMA)	10.0*
Average fuel burn-up, % FIMA	6.7*
Radial power peaking factor K_r^{max}	1.25
Maximum temperature of fuel element cladding, °C	600 *
Volume peaking factor for fission product build-up K_v^{max}	1.5
Fuel cycle type	Once-through cycle at the initial stage
Lead-bismuth coolant volume in the primary circuit, m^3	18
Mode of the primary circuit coolant circulation	Forced circulation
Lead-bismuth coolant temperature, °C: - At core outlet - At core inlet	 482 * 320
Primary circuit coolant flow rate, kg/s	11 760

PARAMETER	VALUE
Excess pressure of the protective gas over the lead-bismuth coolant level, kPa	10
Number of main circulation pumps (MCPs)	2
Power of the MCP electric driver, kW	450
Head of the MCP, MPa	~ 0.55
Number of steam generators	2
Number of steam generator modules	2×6
Mode of the secondary circuit coolant circulation (in the separator – SG section)	Natural circulation
Steam production rate, t/hour	580
Steam parameters: - Pressure, MPa - Temperature, °C	 9.5 307 *
Feedwater temperature, °C	241
Repetition factor of the secondary circuit coolant circulation	2
Reactor installation operating lifetime, years	50
Specific consumption of natural uranium in a once-through uranium fuel cycle, t/GW(e)/year	487
Specific consumption of natural uranium in a closed fuel cycle with MOX fuel, t/GW(e)/year	~ 0.9
Specific consumption of natural-enrichment boron carbide, t/GW(e)/year	5
Specific consumption of stainless steel, t/GW(e)/year	~ 90
Estimated construction cost of the industrial prototype with the SVBR-75/100 reactor installations at the second unit of the Novovoronezhskaya NPP in Russia, US$ million	~100
Implementation period, including design development and industrial prototype construction, years	6
Fuel cost	In accordance with the world costs of fuel
Electricity cost for a modular NPP (in compliance with [XIX-8]), US$/kW-hour	0.0146
Cost of potable water for desalination and power complex (in compliance with [XIX-13]), US$/$m^3$	0.74
Cost of electricity for desalination and power complex (in compliance with [XIX-13]), US$/kW-hour	0.035

* These data correspond to a conservative variant of SVBR-75/100 using available SGs and saturated steam turbine, with the maximum temperatures of fuel element claddings not exceeding 600°C. Currently, the work to increase the temperature of the fuel element cladding up to 650°C is being conducted that would make it possible to increase the reactor thermal power by ~15% and enhance the possibility of a transfer to a superheated steam turbine cycle that would result in the increase of the thermodynamic cycle efficiency by ~15%. In calculations of the presented technical and economic parameters, an additional margin of 17% over the normative parameter values has been introduced, which corresponds to 60% of the reactor installation equipment cost.

For lead-bismuth cooled reactors, the reliability of fuel elements is in many respects defined by maximum cladding temperature. In accordance with previous R&D, corrosion resistant steel has been selected for the SVBR-75/100 design, corresponding to the maximum temperature of 600°C. Short-time increases of the fuel element cladding temperature up to 800°C without damage are permitted.

The basic neutron-physical characteristics of the reactor were calculated using R-Z geometry with homogenization of materials in the core, in the reflectors, and in the structural elements surrounding the core. For the reactor lifetime calculation, a two-dimensional diffusion code with a system of 26-group cross-sections was used. The calculation of reactivity change under total removal of coolant from the reactor (void reactivity effect) was performed using a Monte Carlo code.

Five variants of the core were considered, different in fuel types; they were as follows:

(1) Uranium dioxide, UO_2, with an effective density of γ_{eff} = 9.65 g/cm^3; hereinafter, "effective density" refers to fuel composition homogenized over internal volume of the fuel element cladding;

(2) Vibro-packed MOX fuel, PuO_2+UO_2, with the addition of depleted metal uranium (10% by weight); γ_{eff} = 9.7 g/cm^3;

(3) Another variant of MOX fuel, including minor actinides such as Np and Am; this composition is referred to as TRUOX fuel;

(4) Uranium mono-nitride, UN, with the density γ_{eff} = 12.5 g/cm^3;

(5) A mixture of plutonium and depleted uranium mono-nitrides (PuN+UN); γ_{eff} = 10.9 g/cm^3. Such fuel composition with a low effective density was selected as a result of reactivity vs. burn-up calculations because it assured the smallest reactivity change during the lifetime.

The isotopic content of plutonium used in the calculations of the abovementioned variants 2 and 5 approximately corresponded to that in light water reactor (LWR) spent fuel. The total quantity of plutonium and minor actinides in variant 3 was taken in accordance with the data of [XIX-10], corresponding to LWR spent fuel after long cooling (~15 years). The data on isotopic content of the plutonium fuel compositions are summarized in Table XIX-2.

The power profile along core radius was shaped to flatten power distribution in all considered variants. The radial non-uniformity of power distribution is reduced by changing the content of fissile material in the fuel, which increases from the core centre to the periphery. The maximal radial power peaking factor K_r^{max} was less than or equal to 1.25 in all calculations described below.

TABLE XIX-2. ISOTOPIC CONTENT OF PU AND MINOR ACTINIDES IN FUEL COMPOSITIONS (ATOMIC %)

ISOTOPE	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{237}Np	^{241}Am	^{243}Am
Variants 2, 5	1	59	22	13	5	-	-	-
Variant 3	1.6	51.5	21.4	6.6	5	5.5	7.4	1

The lifetime calculations were performed for the five variants highlighted above. For variants with uranium fuel (1, 4), the lifetime duration was presumed to be 2200 effective full power days (EFPD); for variants with plutonium fuel (2, 3, 5), the lifetime duration was presumed to be 3200 EFPD. The K_{eff} changes over the lifetime are shown in Fig. XIX-3.

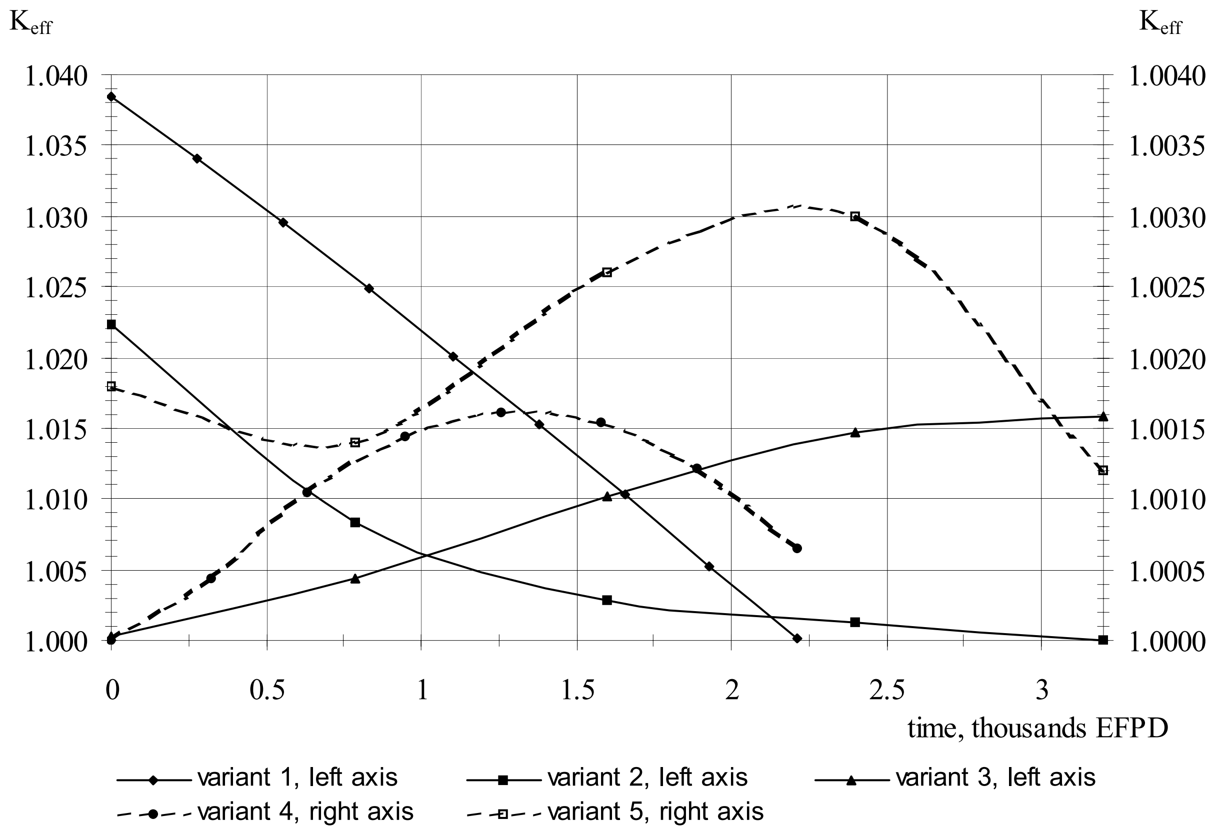


FIG. XIX-3. K_{eff} as function of time for different types of fuel load.

Neutron-physical characteristics for the considered 5 variants of core load are summarized in Table XIX-3.

Variant 1 of the SVBR-75/100 core load, which has maximum reactivity margin, uses the reactivity control system composed of 37 control rods. The channel of each control rod displaces 19 fuel elements in the core; the absorbing material is enriched boron carbide.

The functional classification of rods is as follows: 29 compensating rods (CR), 2 regulating control rods (RCR), and 6 emergency protection (EP) rods. The data on control rod worth are summarized in Table XIX-4.

Table XIX-4 indicates that the maximum value of a single-rod worth does not exceed β_{eff}.

XIX-1.5. Outline of fuel cycle options

The approach to the SVBR-75/100 fuel cycle organization takes into account the features of the reactor as well as economic parameters of different stages of its nuclear fuel cycle.

Certain specific features of the reactor, such as small burn-up reactivity swing, provide a longer fuel lifetime even with the uranium dioxide fuel when the breeding ratio is below one. Once-at-a-time infrequent whole core refuelling makes it possible to change considerably the fuel load characteristics in each subsequent refuelling and to use the type of fuel that is most economically effective at a given stage of nuclear power development.

TABLE XIX-3. NEUTRON-PHYSICAL CHARACTERISTICS OF SVBR-75/100

PARAMETER	VARIANT				
	1	2	3	4	5
Fuel composition	UO_2	MOX	TRUOX	UN	(Pu, U)N
Effective density of fuel, g/cm^3	9.65	9.7	9.7	12.5	10.9
Lifetime, EFPD	2200	3200	3200	2200	3200
Loading of fissile materials, kg Beginning of lifetime: - Uranium-235 - Plutonium - Minor actinides End of lifetime: - Uranium-235 - Plutonium - Minor actinides	 1470 - - 906 393 7	 - 1306 - - 1350 47	 - 1360 218 - 1441 165	 1763 - - 1 207 412 6	 - 1415 - - 1471 41
Effective fraction of delayed neutrons, β_{eff} - Beginning of lifetime - End of lifetime	 0.00722 0.00585	 0.0041 0.0036	 0.0039 0.0035	 0.0074 0.0062	 0.0042 0.0038
Maximum fuel burn-up, % FIMA	10.0	13.5	13.5	7.0	11.5
Maximum reactivity change during lifetime, %	3.8	2.2	- 1.52	- 0.16	- 0.17
Doppler coefficient (average value within the range of fuel operating temperatures) $\alpha_{Doppler}$, 10^{-5}/°C	- 0.74	- 0.77	- 0.58	- 0.97	- 1.1
Reactivity coefficient on coolant temperature $\alpha_{coolant}$: - At T = 200°C (increase of reactor power from the sub-critical state) - At operating temperature	 $- 2.2\times10^{-5}$ $- 1.4\times10^{-5}$				
Reactivity change at total removal of coolant from the reactor (void effect) $\Delta\rho_{void}$, % $\Delta\rho_{void}$, β_{eff}	 - 2.75 - 3.80	 - 1.65 - 4.00	 - 1.1 - 2.9	 - 2.2 - 3.0	 - 1.5 - 3.6

TABLE XIX-4. CONTROL ROD WORTH

DISTANCE FROM THE REACTOR CENTRELINE, mm	ROD TYPE	NUMBER OF RODS	WORTH, $\beta_{eff.}$
0	CR	1	0.28
223.9	CR	6	0.33
387.8	CR	4	0.33
387.8	RCR	2	0.33
447.9	CR	6	0.32
592.3	CR	12	0.32
671.2	EP	6	-
(29 CR + 2 RCR) system, $\beta_{eff.}$		8.95	
6 EP system, $\beta_{eff.}$		1.9	

The analysis performed shows that SVBR-75/100 can operate using both pure uranium fuel and mixed uranium-plutonium fuel, which would require no changes in the design and would not result in sacrificing the safety performance. Specifically, provided for are the following types of fuel and fuel forms:

- For uranium fuel – the uranium dioxide fuel produced in industrial scale quantities as well as the uranium mono-nitride fuel mastered in experimental production;
- For mixed uranium-plutonium fuel – MOX fuel with weapon-grade plutonium (MOX-W); MOX fuel with reactor-grade plutonium (MOX-R), either extracted in chemical reprocessing of LWR spent fuel or self-generated; TRUOX fuel (MOX fuel with the addition of minor actinides extracted from LWR spent fuel); and mixed uranium-plutonium nitride fuel.

Certain parameters of the SVBR-75/100 cores with different types of fuel are summarized in Table XIX-5. The economy features for different fuel cycle options are summarized below, based on the results of [XIX-14].

TABLE XIX-5. SVBR-75/100 CHARACTERISTICS FOR DIFFERENT FUEL TYPES

Fuel type	UO_2	MOX	TRUOX	UN	UN-PuN
Total core load, kg	9144	9590	9650	12 100	-
Average enrichment by ^{235}U or Pu, or Pu and minor actinides, %	16	11.4	14	13.2	-
Calculated lifetimes of the core, EFPD	2200/ 3300	3200	3200	2200/ 6250	Up to 8300
Maximum/ average discharged fuel burn-up, % FIMA;	10/6.7	15/9.5	14/9.0	8.5/5.5	-
Core breeding ratio (CBR)	0.87	1.005	1.028	0.91	1.13
Reactivity margin for fuel burn-up, %	3.8	2.2	-1.52	-0.16	-
Worth of the passive emergency protection (EP) rods, %	4.6	> 2.0	> 2.0	2.0	-

Due to the current low costs of natural uranium and uranium enrichment, the use of uranium dioxide fuel with postponed reprocessing and spent fuel storage on the nuclear power plant site, are economically preferable for SVBR-75/100 at the moment. The duration of the benefits of this fuel cycle option depends on the available uranium resources and nuclear power deployment scale. In any case, the existing uranium resources are sufficient to achieve the realistic scenario of nuclear power development until the year 2050. The costs of natural gas could be expected to increase more intensively than the costs of natural uranium. This will ensure the NPP competitiveness even with a considerable increase in uranium prices, because the structure of electricity cost is different for NPPs and fossil-fuelled heat power plants.

At this stage, the key to improving the economic parameters of the fuel cycle will be extending the core lifetime (to increase fuel burn-up), as experience in operability of the fuel elements is gained. Further on, the reprocessing and recycle of the uranium could be applied, and the plutonium, minor actinides and fission products could be extracted and then stored until their recycle becomes economically efficient. The duration of the uranium stage may be extended upon a transition to the uranium nitride fuel.

In a more distant future, it will be necessary to change to an entirely closed nuclear fuel cycle. The time period for this change would be defined by the industrial development of economically effective spent fuel reprocessing technologies that should also be acceptable from the standpoint of non-proliferation and radioactive waste minimization.

Changeover to a closed fuel cycle will enable the economically effective use of spent fuel from thermal reactors (such as VVER and RBMK), as part of a fuel make-up for the SVBR-75/100. It is estimated that the fraction of spent fuel from thermal reactors in the fresh fuel of a SVBR-75/100 operating in a closed fuel cycle could be ~10–12% and, with the fraction of plutonium in the thermal reactor spent fuel less than 1%, the effect of the plutonium isotopic vector of the thermal reactor spent fuel on the isotopic vector of the fresh SVBR-75/100 fuel would be negligible. Therefore, in the future, the SVBR-75/100 could make it possible to develop an essentially new strategy of a closed nuclear fuel cycle – the strategy of direct use of LWR spent fuel, requiring no expensive reprocessing of such fuel just to extract 1% of the plutonium to supply it to fast reactors [XIX-15].

The flexibility of the SVBR-75/100 in relation to fuel cycle technologies is realized in compliance with the principle: "To operate using the type of fuel and fuel cycle that are most efficient today" makes it possible to postpone the construction of specialized fuel cycle factories for several decades after the first NPP unit with the SVBR-75/100 modules is launched. For example, after the introduction of about 10 GW(e) using the SVBR-75/100 and repaying the NPP construction costs, a share of the profits could be spent to develop the industry for spent fuel reprocessing and MOX fuel fabrication.

The most economically effective will be a large centralized fuel-reprocessing factory located in an industrially developed nuclear country.

After launching such factory, the cost of the core will only be determined by the operating costs of spent fuel reprocessing and the costs of fuel assembly fabrication. If the pyro-electrochemical fuel reprocessing methods developed by the State Scientific Centre of the Russian Federation Research Institute of Atomic Reactors (SSC RIAR) are used, the contribution of fuel costs to the cost of SVBR-75/100 will be even less than in the basic variant using a once-through cycle with the uranium dioxide fuel. This will make it possible to improve considerably the NPP competitiveness. The abovementioned approach to the construction of capacities for reprocessing and fuel assembly fabrication presumes that the owner of the NPPs would also be the owner of the fuel cycle factories.

The attractive features of the RIAR reprocessing method are as follows [XIX-4, XIX-5]:

- High chemical and radio-chemical stability of the medium (the electrolyte is an ion liquid);
- Elimination of additional neutron moderators (the electrolyte is a melt of the mixture of alkali metal chlorides);
- High electrolyte capacity for fissile materials (over 30% in terms of weight);
- Enhanced protection against the undeclared use of fissile materials (high residual gamma-activity of reprocessed products);
- The technological cycle with sufficient flexibility of the process is realized in a single apparatus;
- Minimum volume of high level waste;
- Direct use of the final product (the oxide fuel granules for which the density practically equals the theoretical density of the crystals) for fabrication of the vibro-packed fuel elements; and
- Practical absence of irretrievable losses of potential fuel.

The following procedure is presumed for the SVBR-75/100 spent fuel storage prior to reprocessing. After a fuel assembly with spent fuel has been extracted from the reactor, it is installed in a capsule in which lead has been preliminarily heated in an electric oven to temperatures over the melting point. Then the capsule is sealed and transported to a "dry" repository with natural air-cooling, where the lead in the capsule will quickly solidify.

In this case, the following four barriers block the path for radioactivity release into the environment: fuel matrix; fuel element cladding; solidified lead, and a capsule vessel. The solidified lead that contacts the fuel element claddings made of steel eliminates any kind of corrosion.

For reprocessing of the SVBR-75/100 spent nuclear fuel, it is presumed that the extracted fission products are vitrified and, after necessary cooling and enclosure in special containers providing a multi-barrier shielding, they are transported for final disposal in deep geological formations. Minor actinides (except curium) are not separated from plutonium and are used in the reactor as a fuel component. Due to the high power release caused by alpha decay, curium is extracted and transported to the temporary repository for 100–200-year cooling. After cooling, all curium isotopes (except curium-245) are transformed into plutonium isotopes, and this isotopic mixture is transported back to the reactor for further burning. A principal scheme of the fuel cycle with thermal reactors and fast reactors of the SVBR type with direct utilization of thermal reactor spent fuel is shown in Fig. XIX-4.

XIX-1.6. Technical features and technological approaches that are definitive for SVBR-75/100 performance in particular areas

XIX-1.6.1. Economics and maintainability

For developed countries with large-scale power systems, it may still be economically effective to use large modular power plants. The maximum possible capacity of a modular type power plant depends only on the number of modules.

For a large modular power-unit equipped with a single turbine installation, having in mind the existing technical level of turbine-construction in the Russian Federation, the power capacity can be taken as 1600–1800 MW(e). FSUE SSC RF IPPE, FSUE EDO "Gidropress" and FSUE "Atomenergoproekt" developed a conceptual design for a two-unit nuclear power plant (NPP), in which the power unit includes a nuclear steam-supply system (NSSS) consisting of 16 SVBR-75/100 reactor modules and one turbine installation of 1600 MW(e) [XIX-6]. In this case, a direct comparison of the economic parameters of such NPP with those of the NPPs based on a single VVER-1500 reactor becomes possible.

When making decisions on NPP unit capacity, it was taken into account that the specific capital costs of the reactor compartment ("nuclear island") would decrease with increasing the unit capacity. This is conditioned by the fact that under increasing the number of modules in the reactor compartment, the cost of the equipment and systems installed beyond the reactor module compartments would increase only slightly. For this reason, its contribution to the specific capital costs of the NPP will decrease.

Such shared systems and equipment include the refuelling equipment, the coolant intake equipment, the equipment for conveying coolant to the reactors at the initial filling, etc.; and their contribution to the specific capital construction costs of an NPP will decrease correspondingly.

Since the reactor module has only two states: operating and shut down, the modular NSSS is controlled by an operator using the common control panel (located in a single control room). If there is a fault in one of the reactor modules, this module is automatically removed from operation and can be aftercooled autonomously by the turbine plant systems. In conjunction with this failure, the NSSS power would be slightly reduced.

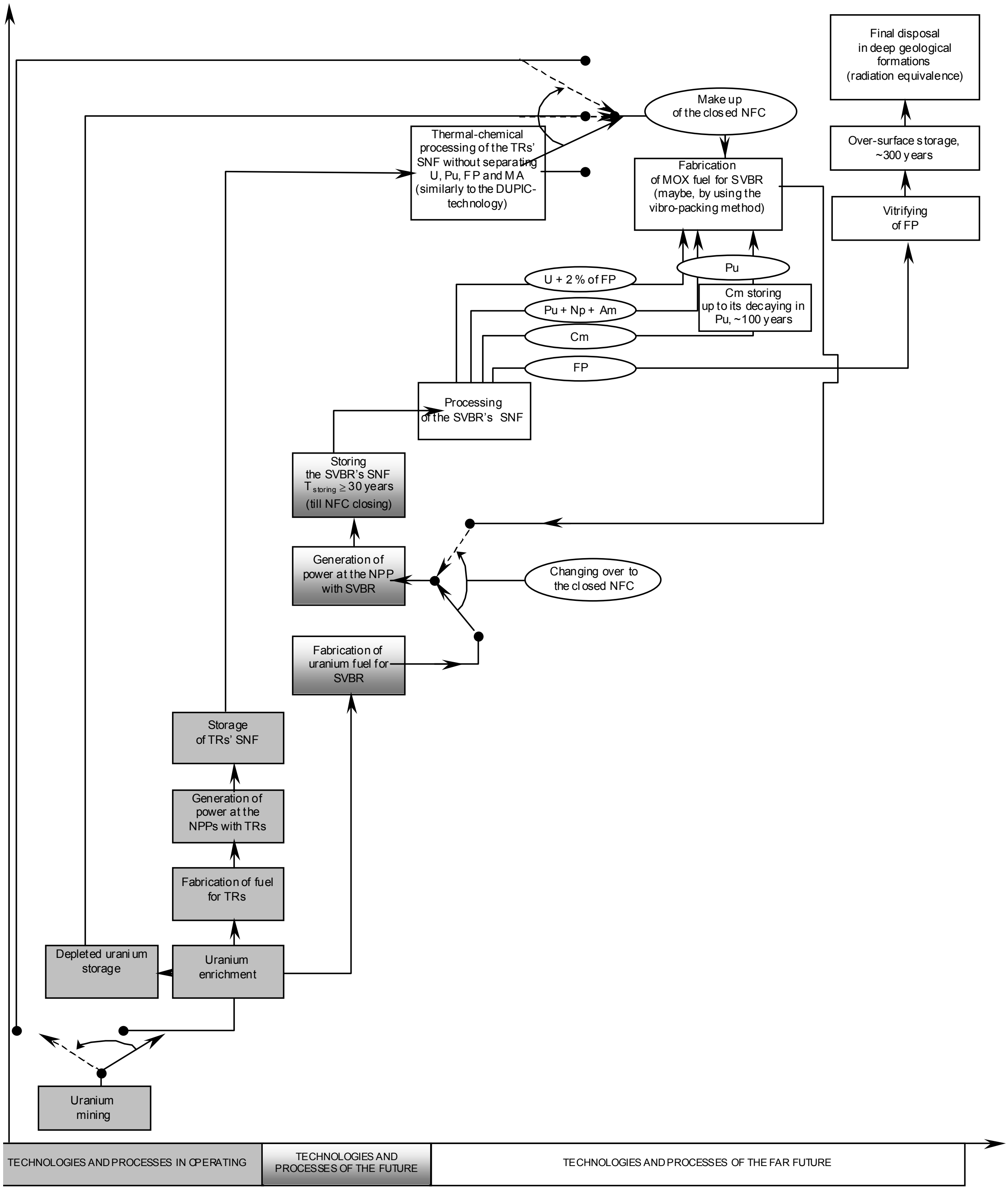


FIG. XIX-4. Nuclear fuel cycle scheme for a nuclear energy system with thermal reactors (TRs) and fast reactors (FRs) of the SVBR type with direct utilization of the thermal reactors' spent nuclear fuel (SNF).

A simple scheme of the reactor module and the fact that all modules installed are of the same type make it possible to reduce the number of personnel for operation and maintenance of the modular NPP as compared with the NPP unit comprising one large-power reactor that incorporates many safety systems, such as protection systems, localizing accident systems, and control and auxiliary systems. For example, the safety systems of the AP-1000 reactor [XIX-16] have 184 pumps, 1400 valves, and 40 km of pipelines and cables [XIX-16].

The basic technical and economic parameters of the two-unit NPP based on the SVBR-75/100 modules in comparison with two-unit NPPs with the VVER-1500, VVER-1000 (V-392), and BN-1800 reactors and with the heat power plant (HPP) with 10 PGU-325 steam-gas units are summarized in Table XIX-6 [XIX-6].

TABLE XIX-6. PARAMETERS OF DIFFERENT POWER PLANTS

Parameter	NPP with SVBR-75/100	NPP with VVER-1500 [XIX-17]	NPP with VVER-1000	NPP with BN-1800 [XIX-18]	HPP with PGU-325
1. Installed capacity of the power-unit, MW(e)	1625	1550	1068	1780	325
2. Number of power-units at the plant; number of modules / unit	2; 16	2; 1	2; 1	2; 1	10; 1
3. Electric power necessary for the plant's own needs, %	4.5	5.7	6.43	4.6	4.5
4. Net efficiency of the plant (power-unit), %	34.6	34.4	33.3	43.6	44.4
5. Specific capital investment in the industrial construction of the plant, US$/kW(e) (1991 prices)	661.5	680	819.3	860	600
6. Cost of produced electricity, US$ cent/kW·h	1.46	1.62	2.02	1.6	1.75

XXIX-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

In the course of operation of the SVBR-75/1000 based NPP, liquid radioactive wastes are produced in very low quantities; this has been verified in the operation of the lead-bismuth cooled reactors of nuclear submarines.

The NPP design incorporates an installation for the concentration and solidification of small quantities of liquid radioactive wastes.

After an expiration of the reactor lifetime, the radioactive lead-bismuth coolant can be recycled many times in new reactor installations. In 1000 years of irradiation, the residual long-lived radioactivity of lead-bismuth coolant caused by ^{208}Bi and ^{210m}Bi will be lower than natural radioactivity of the uranium ore (measured as that of U_3O_8).

In this respect, the disposal of lead-bismuth coolant in the form of solid radioactive waste in deep geological formations will not disturb natural radioactivity equilibrium. The low chemical activity of lead and bismuth eliminates the possibility of radioactivity release into the biosphere; therefore, the radio-ecological consequences of the disposal would be of zero risk for next generations.

The quantity of tritium released into the environment due to unavoidable water losses in the secondary circuit is within the limits of normal disposal of tritium within liquid wastes from operating NPPs worldwide (except those with heavy-water reactors, in which tritium release is by an order of magnitude higher).

The closed fuel cycle of the SVBR-75/100 does not provide for transmutation of certain elements of fission products because this process is rated as low efficient.

Considering that the half-life of most fission products does not exceed 30 years (except for, mainly, long-lived technetium-99, iodine-129, and caesium-135), it is assumed that after being vitrified they will be placed in a long-term repository for about 500 years. After storage in the repository, the vitrified fission products could be deposited in deep geological formations, meeting the natural radioactivity equilibrium requirements.

Management of the transuranic (TRU) elements presumes that their release beyond the fuel cycle should be excluded (except for very low losses at the stage of radioactive waste chemical reprocessing).

To estimate the environmental impact caused by nuclear fuel cycle of the SVBR-75/100, the value of specific radiotoxicity of the produced transuranic elements (neptunium, plutonium, americium and curium) and long-lived fission products (technetium-99, iodine-129 and caesium-135) was taken as a criterion, as a function of the electric energy produced. When this value decreases with energy production, the environmental impact of the nuclear fuel cycle can be considered "friendly". The radiotoxicity characteristic adopted was the volume of water necessary to dilute some quantity of radionuclides to the concentrations for which the specific radioactivity of the solution meets the sanitary requirements for drinking water.

The specific radiotoxicity was defined as the radiotoxicity of the spent nuclear fuel under a given power production divided by the power produced.

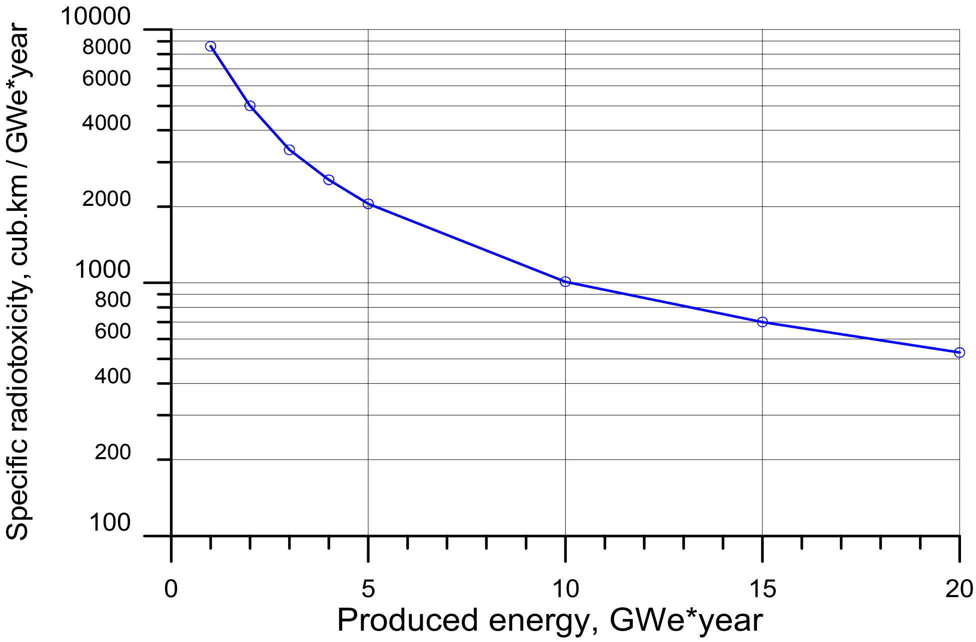


FIG. XIX-5 Specific radiotoxicity as a function of produced energy.

The radiotoxicity was calculated using the following assumptions:

(1) The first load of the reactor is MOX fuel based on plutonium extracted from spent nuclear fuel of light water reactors;

(2) At the end of each lifetime and after a three-year cooling period, the SVBR-75/100 spent fuel is reprocessed. For this:

- The fission products are extracted and vitrified and, after necessary cooling in the temporary repository (~500 years), enclosed in special containers providing a multi-barrier shielding, and transported for final disposal in deep geological formations. This method of managing long-lived fission products was found cheaper, more reliable and safer than the transmutation;
- The radiotoxicity of the bulk of fission products with half-lives less than 30 years is not taken into account since, after 500 years of cooling, this radiotoxicity will be very low;
- Curium is extracted and transported for 100-150-year cooling in the repository. After the cooling, all curium isotopes (except curium-245) are transformed into the plutonium isotopes. This isotopic mixture is then transported back to the reactor for further burning;
- A mixture of plutonium, neptunium and americium with the remaining uranium and necessary addition of depleted or natural uranium is used to fabricate the fuel load for the next lifetime.

Figure XIX-5 presents the specific long-lived radiotoxicity of spent nuclear fuel within the nuclear fuel cycle as a function of energy produced by the SVBR-75/100 reactors. These calculations show that specific radiotoxicity of the technetium-99, iodine-129 and caesium-135 before the final disposal is 0.014 km^3/GW(e)/year, without taking into account the losses in reprocessing. It is nearly equal to the specific radiotoxicity of natural uranium extracted from Earth and added to the fuel cycle each year.

Analysis of the results points to environmental “friendliness” of the SVBR-75/100 nuclear fuel cycle, resulting from the fact that the specific radiotoxicity decreases with increase of the energy produced. This is because the hard neutron spectrum of the reactor facilitates efficient burning of both self-generated minor actinides and those accumulated in light water reactors.

XIX-1.6.3. Safety and reliability

Safety concept and design philosophy

The SVBR-75/100 safety concept is based on the following principles:

- The reactor type, the primary circuit coolant and the design features are selected to maximize the role of inherent safety features in ensuring the reactor safety;
- The reliability and safety of the reactor installation are enhanced due to a reduction and simplification of the safety systems and by assigning certain safety functions to the systems of normal operation.

The reactor installation with SVBR-75/100 incorporates a conservative design approach and provides for simplicity of design and operation achieved through strong reliance on inherent safety features and passive systems for decay heat removal, and ensures a slow pace of accident progression. The SVBR-75/100 incorporates the design and technological features proven in practice, such as a long grace period before personnel intervention in accidents, etc.

Provisions for simplicity and robustness of the design; inherent safety features

Certain features of fast reactors, such as lack of poisoning effects, low values of the negative temperature reactivity effect, and self-compensation of burn-up reactivity swing by the secondary plutonium build up, make it possible for the operating reactivity margin in the reactor to be less than the delayed neutron fraction (β_{eff}). Therefore, prompt criticality is essentially eliminated.

In an unauthorized insertion of positive reactivity under a postulated failure of all emergency protection (EP) control rod drives, the prevention of transient overpower is ensured by a special algorithm controlling the compensating rods, which is part of the automatic control system. After the reactor operation at nominal power during a certain period (~4 months), the reactivity margin controlled by an operator is much lower than β_{eff}. To match this reduced margin, certain compensating rods are disconnected from the control system. In addition, the efficiency of each rod is selected to be much lower than β_{eff}, and the movement rate of the absorbing rods extracted one-by-one is technically limited. For that reason, any conceivable insertion of the positive reactivity would ensure sufficient time for its compensation by negative feedbacks without an unacceptable increase in the core temperature.

In the case of an emergency protection (EP) system failure caused by events not specified in the regulatory documents (for example, failure of the executing mechanisms), fusible locks connecting a rod with a driver bar are incorporated; when the coolant temperatures exceeds 700°C, the EP rods in the gas-filled sleeves are passively separated from the bars and drop into the core due to gravity.

For the fuel loads considered, the total void reactivity effect of the reactor is negative; the local positive void reactivity effect is lower than β_{eff} and cannot be realized due to the coolant's very high boiling point and lack of the possibility for gas or steam bubbles to arise in large quantities and volumes.

The selection of a lead-bismuth coolant for the SVBR-75/100 has been conditioned by the natural properties of lead and bismuth, such as:

- The high boiling point and practical impossibility of coolant boiling that enhance the reliability of heat removal from the core and exclude the phenomenon of heat exchange crisis. Additionally, there is no necessity to maintain high pressure in the primary circuit. These factors result in simplification of the reactor installation design, increase its reliability and practically eliminate the possibility of primary circuit over-pressurization or thermal explosion of the reactor in the event of an emergency over-heating of the coolant;
- Lead-bismuth coolant reacts very slightly with water and air. Progression of accidental processes caused by failures of primary circuit tightness and steam generator (SG) inter-circuit leaks takes place without hydrogen release or exothermic reactions. In addition, there are no materials within the core and the reactor installation that release hydrogen as a result of thermal or radiation effects, or chemical reactions with the coolant. Therefore, the likelihood of chemical explosions and fires as internal events is virtually eliminated.

The possibility of water or steam penetration into the core, e.g. caused by a large SG leak, and the consequent over-pressurization of the reactor mono-block vessel (designed to be resistant against the maximum possible pressure under these conditions) are eliminated by the selected circulation scheme of lead-bismuth coolant. This scheme ensures that steam bubbles are thrown out into the gas volume on the coolant free-level by the upward movement of the lead-bismuth coolant flow. Then the steam goes to the gas system condensers. In the event of postulated failure, the steam goes through the rupture membranes to the bubbler devices of the tank of the passive heat removal system (PHRS).

In the case of a failure of all active aftercooling systems accompanied by the total blackout of the power-unit, the prevention of a core melting (potentially resulting from residual power release) and maintenance of the integrity of the mono-block vessel are ensured in an entirely passive way, through heat removal via the reactor mono-block vessel to the water of the PHRS tank and accumulation of heat in the in-vessel structures and lead-bismuth coolant. For this, the grace period of about five days is provided, defined by the time for water to evaporate from the PHRS tank.

Leak of coolant from the reactor mono-block and termination of lead-bismuth coolant circulation through the core in a failure of tightness of the reactor mono-block vessel (a beyond design basis accident) cannot occur due to the presence of the guard vessel outside the reactor mono-block main vessel, with a small free gap between the guard vessel and the main vessel being provided.

The analyses performed show a high safety potential of the reactor installations of the considered type. For example, even in the event of a postulated combination of such initiating events as the containment destruction, the damage of the reactor installation box-confinement connections and a serious tightness failure of the primary circuit gas system with direct contact of the lead-bismuth coolant surfaces with the atmospheric air, neither a prompt criticality nor the explosion or fire would occur due to the internal causes, and the radioactive release would be lower than that requiring the evacuation of population from the neighbouring territory.

Structure of the defence-in-depth

The defence-in-depth principle is incorporated in the SVBR-75/100 design to ensure protection against radioactivity release into the environment. The following barriers exist on the potential pathway of radioactive release: fuel matrix; cladding of the fuel element; primary circuit coolant, reactor vessel, guard vessel, tight box of the reactor installation and the containment of the plant.

ACTIVE AND PASSIVE SYSTEMS

Emergency shut down system

The system consists of six emergency protection (EP) rods installed in the "dry" channels. The rods are equipped with springs and electromagnetic locks. The rods are inserted in the core driven by gravity and are actuated by melting of the locks in response to a control system signal, or under de-energization and emergency overheating.

Additionally, thirteen reactivity-compensating rods of the RCR group are equipped with springs and electromagnetic locks. These rods are inserted into the core in response to a control system signal or under de-energization. To avoid floating upward in the lead-bismuth coolant, the rods are weighted with tungsten or uranium. This technical feature makes it possible to consider reactivity compensation rod (RCR) system as the second emergency protection (EP) system.

Autonomous heat removal from the reactor installation disconnected from turbine-generator plant

The reactor installation has two heat removal channels; each is able to remove up to 3% of the nominal reactor power. Each channel consists of a cooling condenser connected with a separator and cooled by water, and a condensate drainage pipeline with a direct-acting control valve, which opens if the pressure in the separator exceeds the nominal value.

In normal operation, the autonomous heat removal circuit is used in the start-up and aftercooling modes.

In the waiting mode, the condensers are flooded with water, and there are almost no heat losses. When the steam pressure is increased up to a certain value (for any reason), the valves open and the condensate is drained to the separator. This sets the heat exchange surface free, and the steam begins to condense until the steam pressure decreases to a certain pre-determined level.

The function of passive heat removal from the reactor mono-block vessel in case of failure of all reactor systems

If the water tank of a passive heat removal system (PHRS) is included as the reactor installation component, it is possible to remove heat from the core via the wall of the reactor mono-block vessel. In the failure of all reactor systems, the PHRS tank offers passive heat removal from the mono-block vessel through the evaporation of water from the tank (by boiling) and steam disposal via the air tubes to the atmosphere. The quantity of water stored in the tank is sufficient to remove heat from the reactor over a 5-day period without any damage to the reactor core.

Figure XIX-6 (a) and (b) presents the scheme of heat removal to the PHRS tank and the dynamics of the reactor parameters (the quantity of water in the tank, and maximum temperature of the fuel element cladding) during 120 hours of the accident, corresponding to the water tank volume of 250 m^3.

In normal operating conditions, not more than 0.2% of the nominal reactor power is removed via the PHRS tank; the tank also performs the function of neutron protection.

The water tank also makes it possible to remove residual heat during scheduled repair and maintenance operations with the dried secondary circuit.

Steam generator leak localization function

In the event of a steam generator (SG) leak, it is necessary to prevent such consequences as over-pressurization of the reactor vessel by steam and the ingress of steam-water mixture into the core.

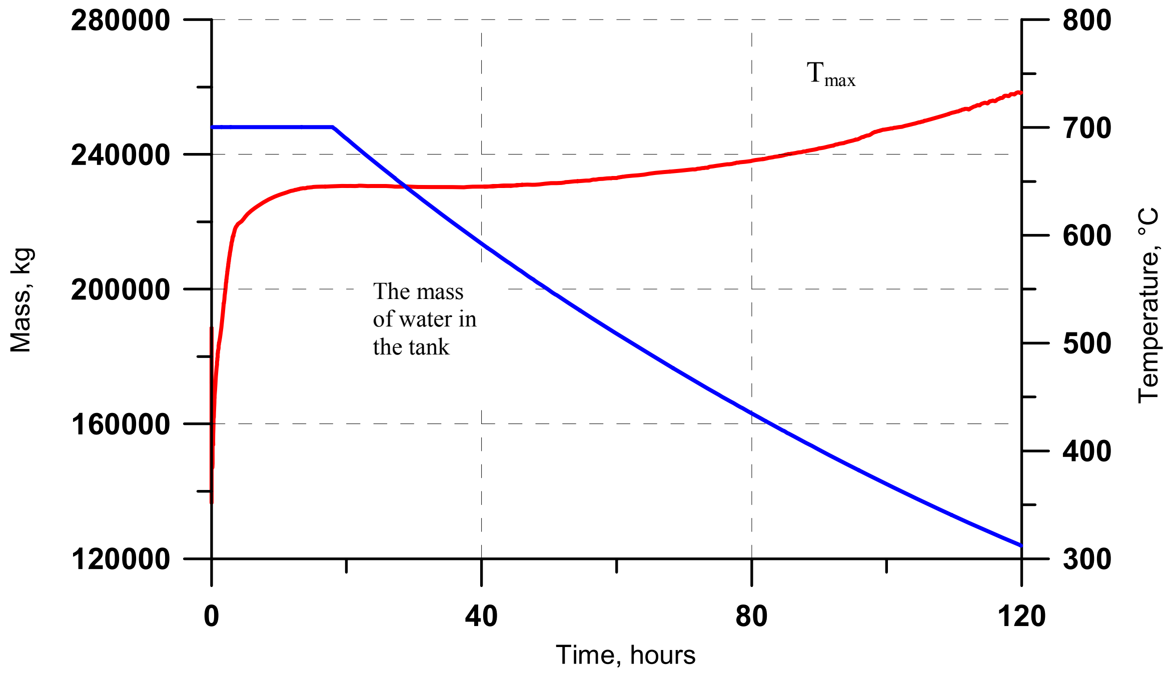
The principal element to localize SG leaks is the lead-bismuth circulation circuit with the streams going upward to the free coolant level. It provides reliable separation of the steam-water mixture and prevents steam penetration into the core with the descending coolant stream of the primary circuit.

In the case of small SG leaks (up to the complete rupture of a single SG tube), two water cooled emergency condensers of the gas system are used. The capacity of the condensers makes it possible to keep the reactor installation gas system pressure within 0.5 MPa.

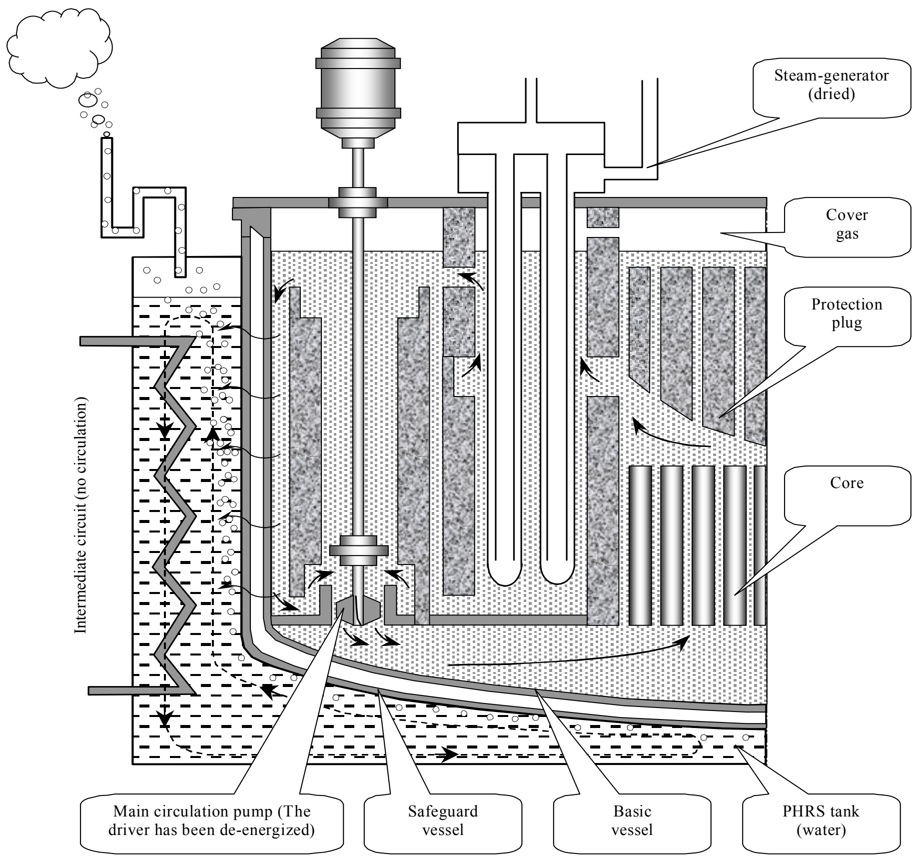
To withstand large SG leaks (a postulated rupture of several SG tubes, i.e., a beyond design basis initiating event), the gas system is connected with the passive heat removal system (PHRS) tank. The connection line is blocked by a membrane device designed for rupture under a gas system pressure of 1 MPa that is not dangerous for the reactor mono-block vessel. In the event of membrane rupture, the steam will be condensed in the water of the PHRS tank. In this, the volatile radionuclides of the cover gas remain in the tank water and the radioactive uncondensed gases will be released into the atmosphere via the filtered ventilation system. The radioactivity release will not exceed the permissible levels.

Design basis and beyond design basis accidents

The dynamic behaviour of the SVBR-75/100 parameters for several design basis and beyond design basis accidents is presented in Fig. XIX-7 through XIX-10.



(a)



(b)

FIG. XIX-6. Dynamics of the SVBR-75/100 parameters during 120 hours of the accident with failure of all reactor systems (a); and scheme of heat removal to the PHRS (b).

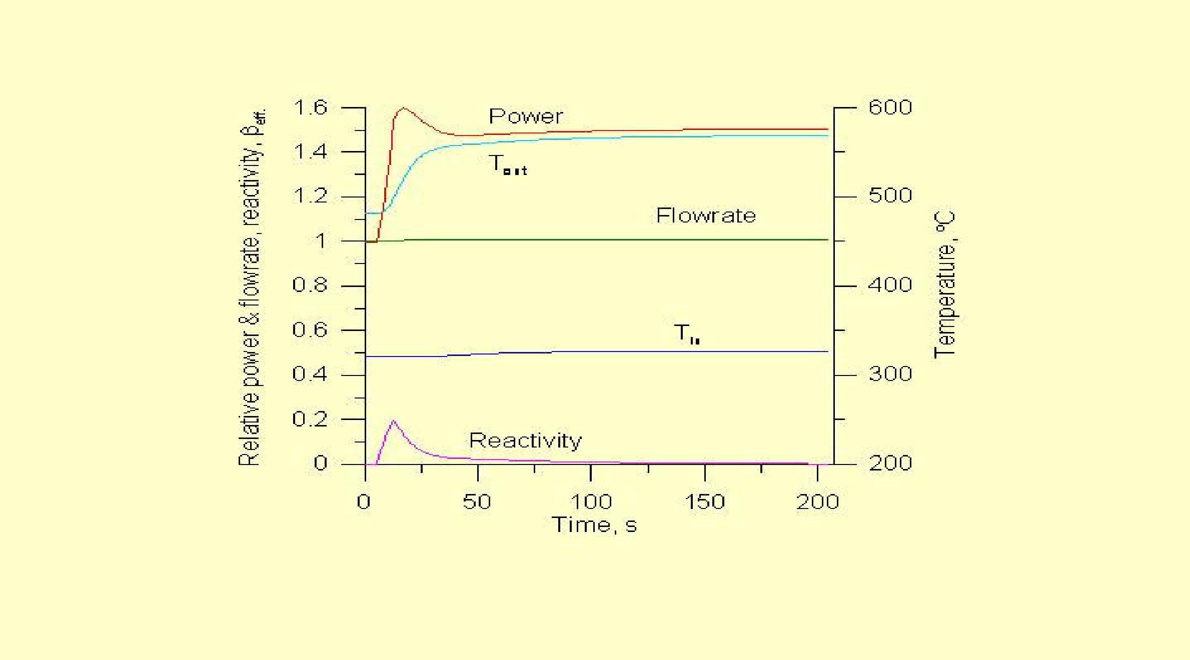


FIG. XIX-7. Scenario with the ejection of all rods of the control system normally available to the operator (without actuation of the emergency protection system (EP); starting from the reactor operation at nominal power).

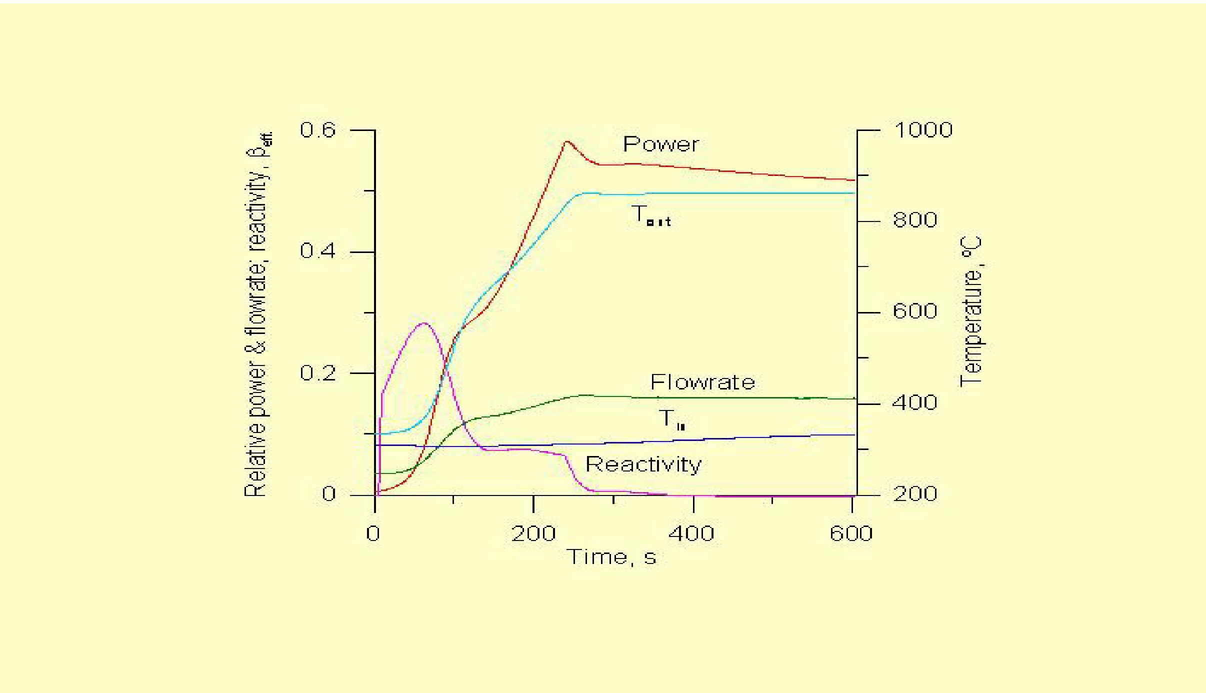


FIG. XIX-8 Scenario with the ejection of all rods of the control system normally available to the operator (without actuation of the emergency protection system (EP); starting from a minimum controlled power level (a critical state)).

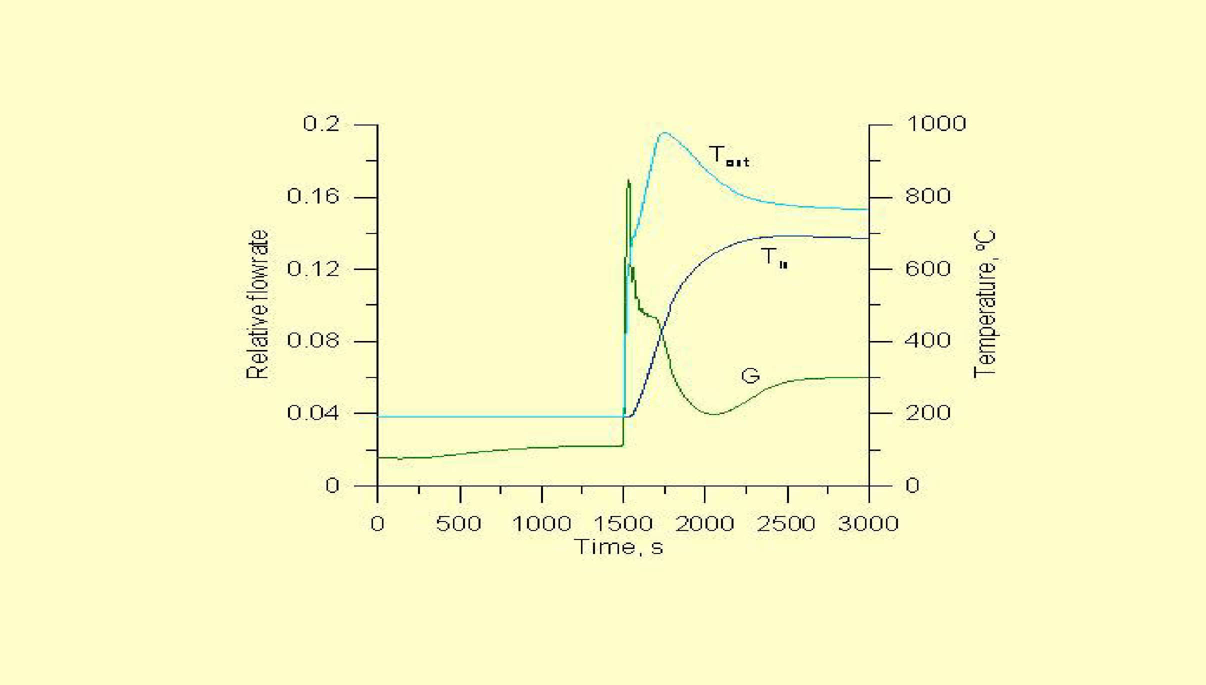


FIG. XIX-9 Scenario with the ejection of all rods of the control system normally available to the operator (without actuation of the emergency protection system (EP); starting from a subcritical state).

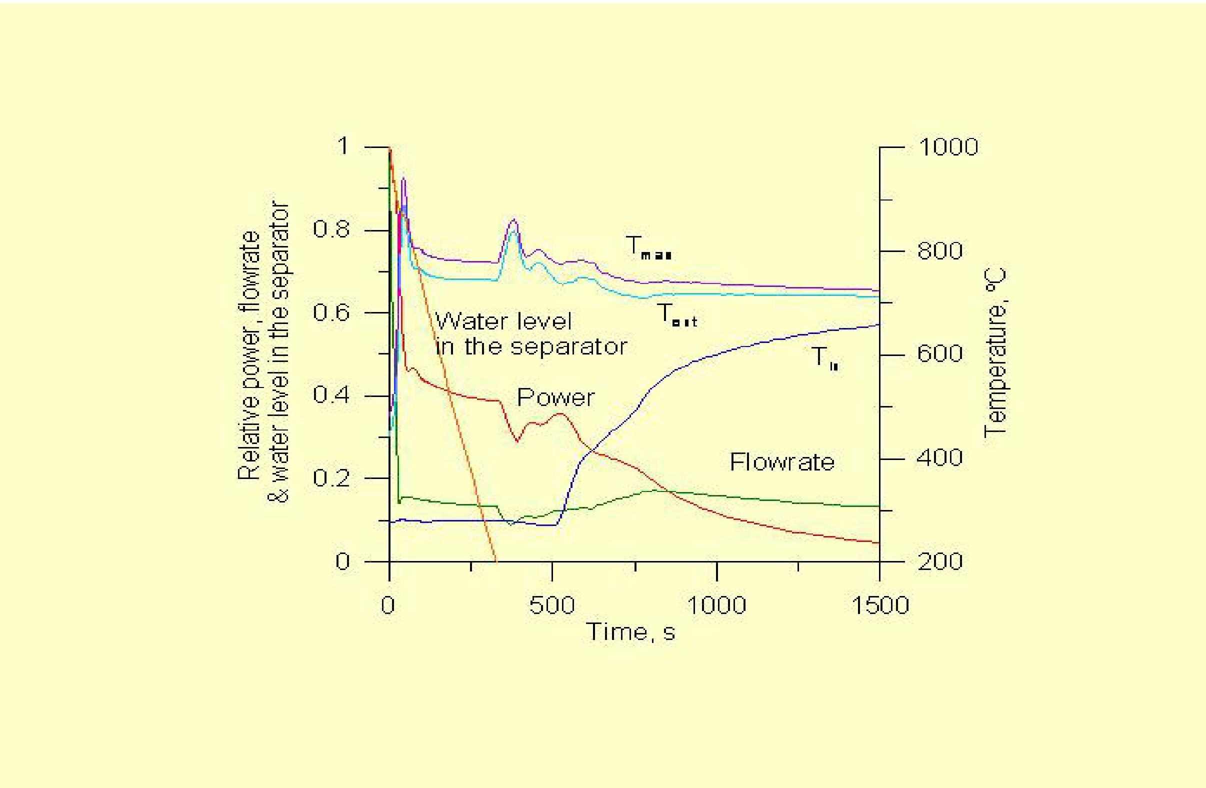


FIG. XIX-10 Scenario with total NPP blackout without actuation of the emergency protection system (EP); starting from the reactor operation at nominal power. In this scenario, the reactor power decreases due to negative reactivity feedbacks.

The analysis of the SVBR-75/100 safety performance indicates that this reactor installation has robust safety features, which make it possible to cope not only with single failures of the equipment and separate personnel errors but with multiple combinations thereof that might be typical of a scenario induced by malevolent human actions.

XIX-1.6.4. Proliferation resistance

The features contributing to an enhanced proliferation resistance of the SVBR-75/100 are as follows:

(1) The enrichment of the initial uranium dioxide fuel load by ^{235}U does not exceed 20% (by weight), which exempts this load from the category of direct use materials as defined by the IAEA;

(2) At the stage of spent fuel storage, the accumulated plutonium is present in the spent fuel together with highly radiotoxic fission products ("spent fuel standard"). Any undeclared operations with spent fuel can easily be detected by tracking the distribution of gamma-radiation;

(3) At the stage of spent fuel reprocessing, the accumulated plutonium is separated from uranium together with the accumulated minor actinides, which makes such plutonium ineffective for weapon devices. Also, the isotopic content of plutonium does not meet the requirements to weapons-grade plutonium;

(4) As comes to the stages of U-Pu fuel fabrication and transport, two percent of fission products and all minor actinides remain in the fuel after the reprocessing. Such fuel requires remote handling, which impedes theft and facilitates movement inspection. Such fuel could be supplied to many countries since fuel management is only possible using special equipment;

(5) The design of the SVBR-75/100 does not require partial refuelling so it is possible to extend the core lifetime up to 15 years; the operation with weld-sealed vessel becomes possible, which could facilitate monitoring of unauthorized access to fissile materials;

(6) Safeguards verifications by the IAEA are provided for at all stages of the SVBR-75/100 fuel cycle.

XIX-1.6.5. Technical features and technological approaches used to facilitate physical protection of SVBR-75/100

Physical protection of the reactor installation against possible human induced events of malevolent character, e.g., sabotage, terrorism, etc., depends to a great extent on the inherent safety features and passive systems of the reactor installation.

A specific feature the SVBR-75/100 is improved resistance to external impacts achieved through the physical characteristics typical of a fast reactor, through natural properties of lead-bismuth coolant, through integral design of the primary circuit incorporating a safeguard vessel of the reactor mono-block, and through passive actuation of safety systems in an event of failure of the active safety systems (e.g., in the event of total blackout or unauthorized withdrawal of certain equipment from operation).

Reactor installations of this type are tolerant not only to single failures of the equipment or personnel errors but to multiple combinations thereof. For example, the emergency protection system is passively actuated each time when lead-bismuth coolant temperature is increased above the preset limit (passive actuation is achieved via melting of the fusible locks); the residual power is released to the passive heat removal system (PHRS) tank via the reactor vessel, providing a grace period of about five days; direct-action rupture membranes are used in the SG localizing system; the reactor has low operating reactivity margin and negative reactivity feedback on temperature.

As it was mentioned, even in the event of the postulated destruction of the reactor compartment containment and other protective structures accompanied by a serious failure of the primary circuit gas system with direct contact of lead-bismuth coolant surface with the atmospheric air, neither explosion nor fire (as caused by internal initiators) would occur, and the radioactivity released into the environment will not require evacuation of the population beyond the plant boundary.

XIX-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of SVBR-75/100

In recent years, several enterprises affiliated to the Rosatom of Russia and led by SSC RF IPPE have developed a conceptual design of the nuclear desalination and power complex (NDPC) based on the lead-bismuth cooled SVBR-75/100 reactors. The results of this design development are presented in reference [XIX-9].

The NDPC proposed includes two parts:

(1) A stationary complex including the protective containment for the reactor installation, the auxiliary reactor systems and equipment, and the installations for power generation and desalination. It is assumed that this complex can be constructed, owned and operated by a user-country, which would also finance all these activities;

(2) A transportable (floating) reactor unit based on the SVBR-75/100 factory assembled and fuelled reactor; this unit will be constructed at the factories of the supplier country. Afterward, it would be delivered to the user-country and leased for a long period. The factories of the supplier-country would commission and operate the transportable unit in the containment of the stationary complex and, upon the expiration of the reactor core lifetime, the reactor unit will decommissioned and replaced by a new one;

(3) The supplier country could seek crediting for the construction of a transportable reactor unit. Annual payments for the loan under different terms of crediting of the transportable reactor unit construction are presented in Table XIX-7 and in Fig. XIX-11; these data correspond to reference [XIX-13].

TABLE XIX-7. THE AMOUNTS OF ANNUAL PAYMENTS FOR THE LOAN, US$ million/year

INTEREST ON THE LOAN, %	TERM OF REPAYMENT, YEARS			
	3	5	8	10
0	14.53	8.72	5.45	4.36
5	16.01	10.07	6.75	5.65
8	16.92	10.92	7.59	6.50
10	17.53	11.50	8.17	7.10
15	19.10	13.01	9.72	8.69
20	20.70	14.58	11.36	10.40

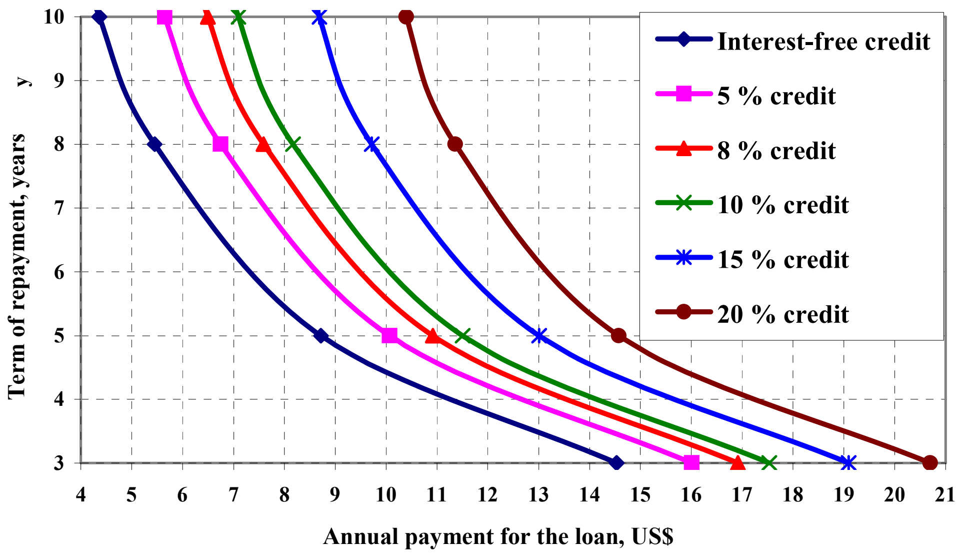


FIG. XIX-11. The amount of annual payments for the loan under different terms of crediting.

For the supplier of a transportable reactor unit, the acceptable term for loan repayment shall not exceed the core lifetime i.e. ~8 years for the first transportable reactor unit and ~12 years for the following ones. If a user-country accepts the rent of ~US$ 12 million/year, this would make it possible for the transportable reactor unit supplier to attract financing for the construction even on the conditions of commercial loans currently offered in the Russian Federation (a 10–20% annual interest rate), under the repayment terms acceptable to the creditors (5–8 years).

Figure XIX-12 shows the calculated discounted payback period of the NDPC versus the local tariff on potable water under different terms of crediting (interest rate on the loan was assumed to be equal to the discount rate (Rd)). The calculations were performed using the DEEP code provided by the IAEA.

Based on the data of Fig. XIX-12, the user may determine the NDPC payback period for a specified NDPC site depending on the local tariff on potable water, or vice versa. For instance, if the cost of potable water is 1 US$/m^3 and the term of repayment is 12 years, the acceptable annual interest rate on the loan may be up to ~10%.

For a user, the costs of construction and operation of a NDPC based on the SVBR-75/100 transportable reactor units would be:

Capital costs ~US$ 260 million, including:

- Coastal construction – US$ 60 million; and
- Desalinating equipment installation – US$ 200 million;

Annual costs ~ US$ 30 million/year, comprising:

- The rent including transportable reactor unit transport ~ US $12 million/year; and
- The cost of NDPC operation and maintenance ~ US$ 18 million/year.

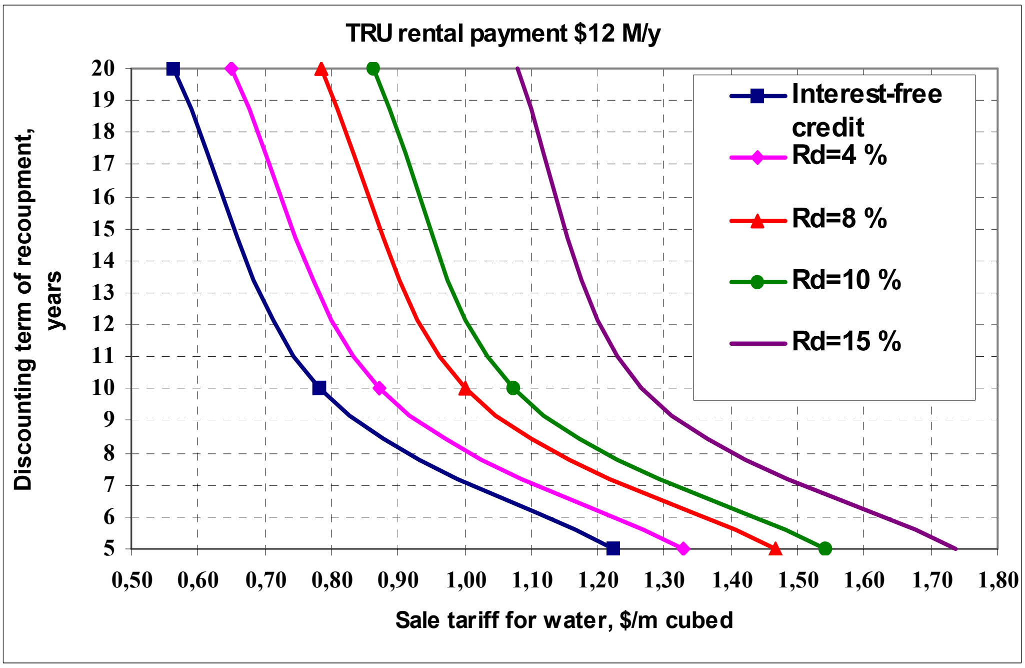


FIG. XIX-12. Discounted payback period of the NDPC vs. tariff on potable water, under different interest rates on the loan (taken equal to the discount rate Rd); rental payment for the transportable reactor unit is fixed at US$ 12 million/year.

XIX.8. List of enabling technologies relevant to SVBR-75/100 and status of their development

The list of major enabling technologies for the SVRB-75/100 reactor installation is given in Table XIX-8.

TABLE XIX-8. LIST OF ENABLING TECHNOLOGIES FOR SVBR-75/100 AND THEIR DEVELOPMENT STATUS

DESIGN OBJECTIVE OR DESIGN SUBJECT AREA	ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Reliability of the primary circuit operation	Technology of maintaining the required quality of lead-bismuth coolant during operation	Available; proven by operating experience of the reactor installations for nuclear submarines
Fabrication technology for fuel elements to achieve reliable operation	Ribbed tubes for fuel elements fabricated from EP-823 steel	Available; mastered at an industrial scale
	Conventional pellets with oxide fuel proven in operation for fast reactors and widely used in the reactors of VVER type	
	Conventional container-type fuel rods	
Corrosion resistant performance of steel in lead-bismuth coolant	Use of EP-823 stainless steel	Available; the production is mastered at an industrial scale

DESIGN OBJECTIVE OR DESIGN SUBJECT AREA	ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Corrosion resistant performance of steel in both, lead-bismuth coolant and steam-water coolant	The technology of fabrication and welding of the bimetallic tubes for steam generators (SGs) that are corrosion resistant in both, lead-bismuth coolant and steam-water coolant	Mastered at an industrial scale
SG design to achieve reliable performance with lead-bismuth coolant	Annular tube type SG	Available; proven by operating experience of the BN-350 reactor
I&C components and systems	No information was provided.	Relevant technologies have been developed; facility and full-scale tests have been completed; industrial-scale fabrication has been mastered
Radiation resistant performance of the reactor vessel and in-vessel structures	Use of in-vessel shielding based on boron carbide blocks	Available; proven by operating experience
Operability of the reactor primary circuit components under multiple “freezing/de-freezing” of the lead-bismuth coolant	Optimized temperature-time curve of “freezing/de-freezing”	Proven by the results of R&D, testing and demonstrations
Reactor start-up and cooling in the absence of connection to the turbine plant	Use of the autonomous cooling system with constant steam pressure in the SG maintained by passive-type devices	Available; proven by operating experience

XIX-1.9. Status of R&D and planned schedule

A prototype plant with the SVBR-75/100 can be constructed in 6–8 years. The cost of construction including costs of the associated R&D and tests is estimated at about US$ 100 million if the prototype is installed in the building of the shut down second unit of the Novovoronezhskaya NPP and the existing infrastructure of this unit is used.

Design development for the prototype plant is performed by the Russian organizations with the responsibilities assigned as follows:

FSUE EDO “Gidropress” – Chief designer of the reactor installation;

FSUE SSC RF-IPPE – Scientific leader;

FSUE “Atomenergoproekt” – Architect and engineering design of the NPP.

The list of basic R&D and tests to be performed at the detailed design stage has been prepared. The current design stage is that of early detailed design[f].

[f] On 15 June 2006 the Scientific and Technical Council No. 1 of the Rosatom of Russia supported the continuation of works for the detailed design of the SVBR-75/100 plant with a link to a certain deployment site.

XIX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

Despite the fact that the proposed reactor technology is backed by a long design and operating experience of the lead-bismuth cooled reactors for nuclear submarines and the fast spectrum reactors with sodium coolant for NPPs; it is still innovative for civil nuclear power. Therefore, additional validation and testing, as well as licensing would be required.

Construction of a prototype plant with the SVBR-75/100 is a necessary and sufficient step in this direction. The necessity of a prototype plant is conditioned by the following reasons.

The operating mode of lead-bismuth cooled reactor installations in nuclear submarines is essentially different from that of NPPs. In a nuclear submarine, the reactor installation operates mainly at low power levels and only occasionally at nominal power; while in an NPP the reactor installation operates mainly at nominal power.

In nuclear submarines, the fuel is a component of the removable reactor unit and is discharged once-at-a-time completely, upon expiration of the core lifetime and after about 1-year cooling period, which is necessary to ensure that decay heat levels are low enough to secure safety in refuelling operations. During this 1-year cooling period, the fuel stays in a shutdown reactor. If applied to a power reactor, such long shutdown for core cooling before unloading could result in a significant reduction of the plant availability. In addition, unloading of the whole core could require adding a significant weight to the refuelling equipment and increasing the lifting capacity of the crane. Due to this, the fuel in the SVBR-75/100 is unloaded cassette by cassette with a 1-month cooling period. Such refuelling technology needs to be validated in a prototype plant.

The reactor barrel extracted from the mono-block after the spent fuel has been reloaded is subject to changes in shape depending upon resistance of the basket material to fast neutron fluence and the effects of thermal strength; therefore, the possibility of a repeated use of the basket needs to be validated.

Reliability must be assured for commercial use of the reactor installation and to facilitate the terms of crediting for the NPP construction; the reliability can only be assured through operation of a prototype plant.

Errors in the cost data produced during the design stages may be noteworthy; the more innovative the technical features of a reactor design, the higher may be the error. Parameter reliability must be proven at the stages of equipment fabrication and during construction and operation of the prototype plant.

Additionally, the prototype plant is necessary to counteract negative public opinion towards nuclear power in the regions or countries where anti-nuclear forces are active. In the presence of observers, a prototype plant with the SVBR-75/100 that incorporates many inherent safety features can be used to demonstrate safety in possible failures of the equipment, personnel errors, and multiple combinations thereof. To perform such experiments under controlled conditions, the prototype should be equipped with additional detectors and control devices.

XIX-2. Design description and data for SVBR-75/100

XIX-2.1. Description of the nuclear systems

Reactor core and fuel design [XIX-10]

Neutron-physical characteristics were calculated for the SVBR-75/100 core of 1645 mm equivalent diameter and 900 mm height. There are ~12 500 fuel elements in the core, arranged in a triangular lattice with a 13.6 mm pitch. There is no partial refuelling in the course of the core lifetime; therefore, ductless fuel assemblies can be used. Regular lattice of fuel elements is used in the core; in this case, the fraction of a more cold coolant flowing through the cells with non-standard geometry (such as those located in the periphery or near the control rod guide tubes) is considerably smaller than that for fuel assemblies with ducts, because of improved lateral agitation. This makes it possible to reduce maximum temperature of the fuel element claddings. Also, the shape of the outer core boundary becomes nearly cylindrical, which facilitates power flattening.

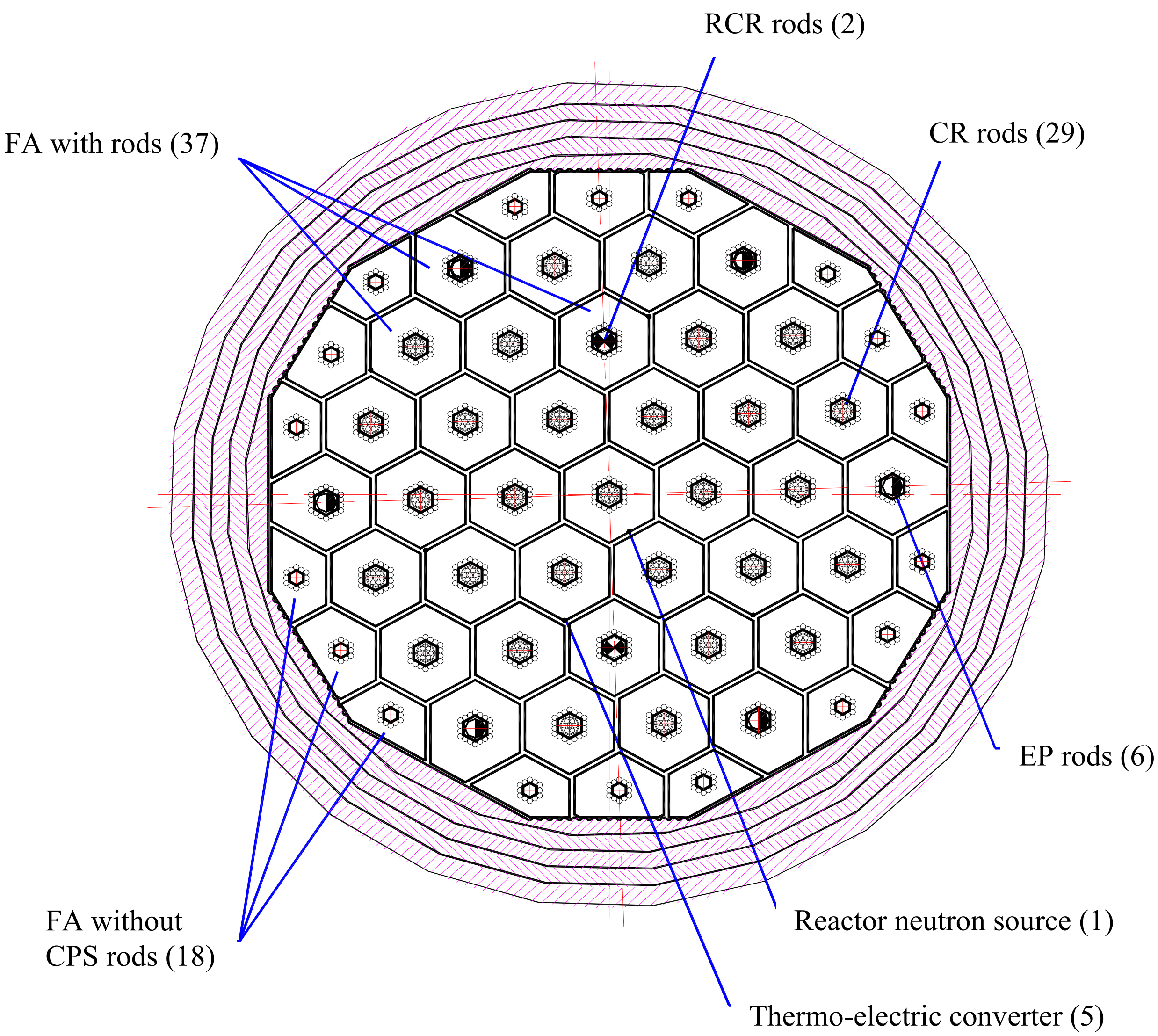
The fuel element is a steel tube of 12 mm outer diameter filled with fuel; it has 4 helical ribs on the outer surface to ensure self-spacing in a bundle. The volumetric fractions of materials in the physical cell corresponding to a single fuel element are as follows:

- Fuel (homogenized over the total volume inside the cladding) ~ 0.615
- Steel ~ 0.105
- Coolant ~ 0.28

In the lower part of the cladding, under the array of fuel pellets, the end reflector made of steel is installed, which has a gas volume to collect gaseous fission products. Over and under the core, mounted are the structures of the removable unit of the reactor, such as support grids for fuel elements and fuel assemblies, coolant inlet and outlet chambers, and components of the upper radiation shielding.

The side surface of the core is surrounded by a steel reflector of ~240 mm thickness; beyond it, the side blocks of the in-vessel radiation shielding are installed.

The core incorporates a system of control rods moving along the fuel assembly axis; these form a triangular lattice of ~224 mm pitch. The number of control rods and their design were selected to comply with the requirement of compensation of reactivity changes during the core lifetime. A cross section of the core with maximum number of control rods (37) is shown in Fig. XIX-13. The fuel assembly design is illustrated in Fig. XIX-14.



FA – fuel assembly RCR – regulating control rod EP – emergency protection rod

CPS – control and protection system CR – compensating rod

FIG. XIX-13. Cross-section of the SVBR-75/100 core.

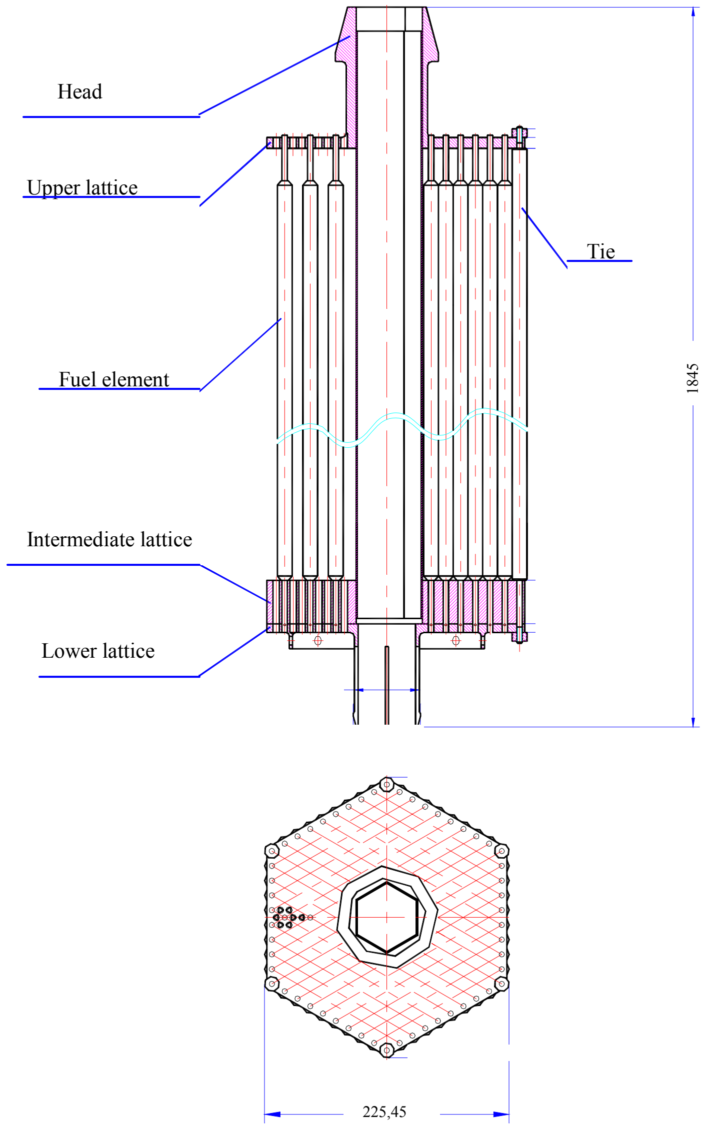


FIG. XIX-14. Fuel assembly design.

Main heat transport system [XIX-11]

Reactor mono-block

The equipment of the primary coolant system is installed inside a strong reactor mono-block vessel. In the central part, a removable unit (a barrel with the core and control rods and a shielding plug) is installed surrounded by the in-vessel radiation shielding with the steam generator (SG) and main circulation pump (MCP) modules mounted upon it (see Fig. XIX-15).

The hydraulic connections between primary system equipment items, which form two circuits for coolant circulation (the main and the auxiliary), are provided exclusively by components and devices located within the reactor mono-block vessel, without using any pipelines or valves.

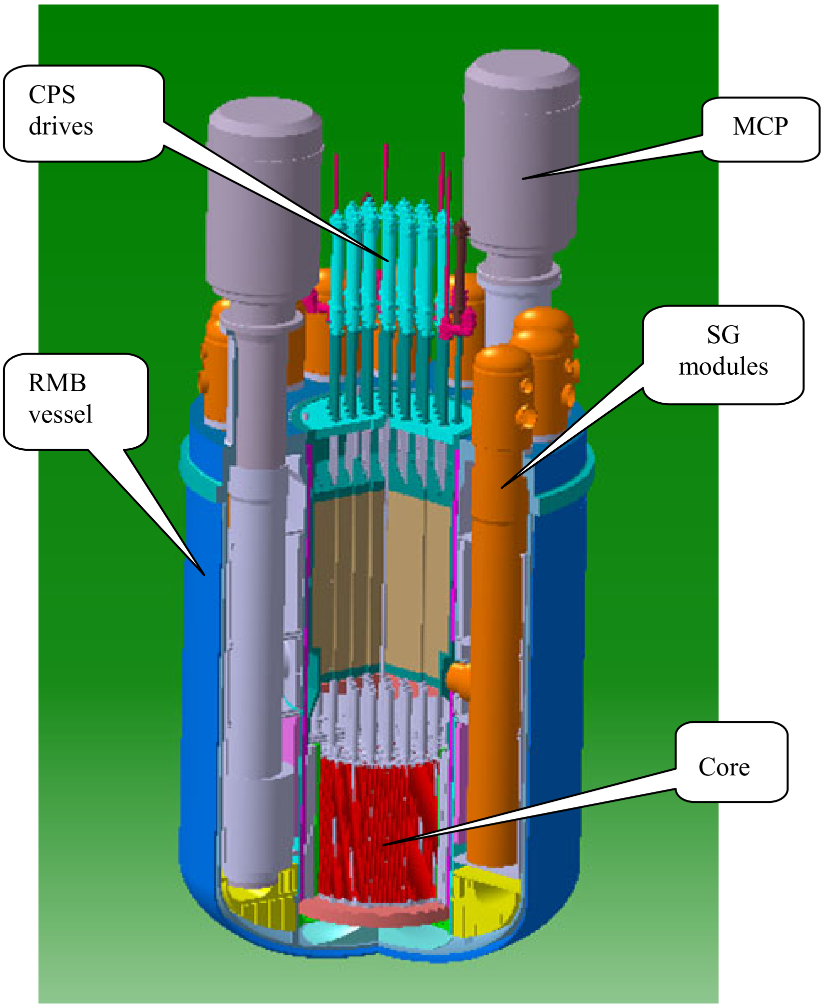


FIG. XIX-15. Reactor mono-block vessel and internals (RMB – reactor mono-block).

The circulation scheme adopted for the main circulation circuit, which provides for free levels of coolant in the upper part of the reactor mono-block and in the SG module channels, coupled with a low coolant velocity in the downcomer sections of the circuit, ensures a reliable separation of the steam-water mixture from the lead-bismuth coolant in the event of an accidental tightness failure in the SG tube system.

The scheme of main heat transport system with specification of heat removal paths in normal operation and in accidents is presented in Fig. XIX-16.

Main circulation pump

The main circulation pump is designed as a pump unit consisting of an axial immersible pump and a gas-tight electric engine with constant revolutions in which the shafts are connected by a spline coupling-clutch. The inner free chambers of the pump and electric engine are filled with inert gas.

Steam generator module

The steam generator (SG) module is designed as a recuperative heat exchanger of the immersible type. The tube still of the SG module that forms a heat-exchanging surface is made of 301 annular tube channels. Each annular tube channel consists of an external tube (26×1.5 mm in diameter) with a bottom and a coaxial central tube (12×1.0 mm in diameter). The channels in the tube cluster are arranged in a triangular lattice. The operating length of the annular channel is ~3.7 m.

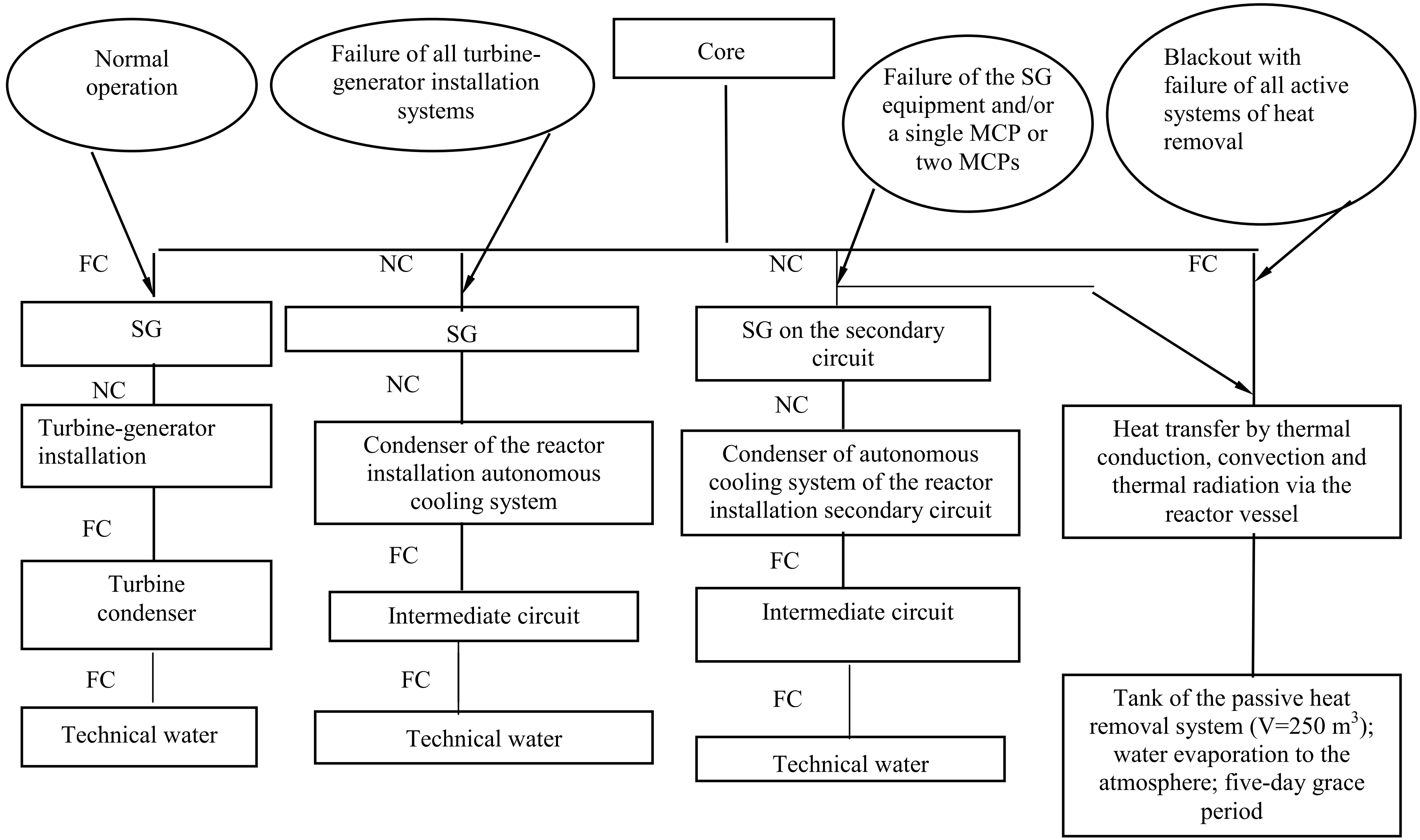


FIG. XIX-16. Main heat removal path of SVBR-75/100: FC – forced circulation; NC – natural circulation.

XIX-2.2. Description of the turbine generator plant and systems

Conventional turbine generators developed for NPPs will be used in the SVBR-75/100.

XIX-2.3. Systems for non-electric applications

As it was already highlighted, the SVBR-75/100 can be used as a power source for the following:

- Nuclear heat and power plants (NHPP) of 400 MW(e) located near the cities for the advantage of short heat transport and reduced energy losses;
- Construction and leasing of transportable reactor units to supply land-based energy conversion systems with steam to produce power, heat and potable water.

Inherent safety features of the SVBR-75/100 coupled with passive principles of operation of its safety systems make it possible to locate the plant near the consumers of heat and potable water.

XIX-2.4. Plant layout

A plan and a longitudinal section of the main building of the nuclear steam supply system (NSSS) of a 1600 MW(e) modular power unit based on the SVBR-75/100 reactors are shown in Fig. XIX-17.

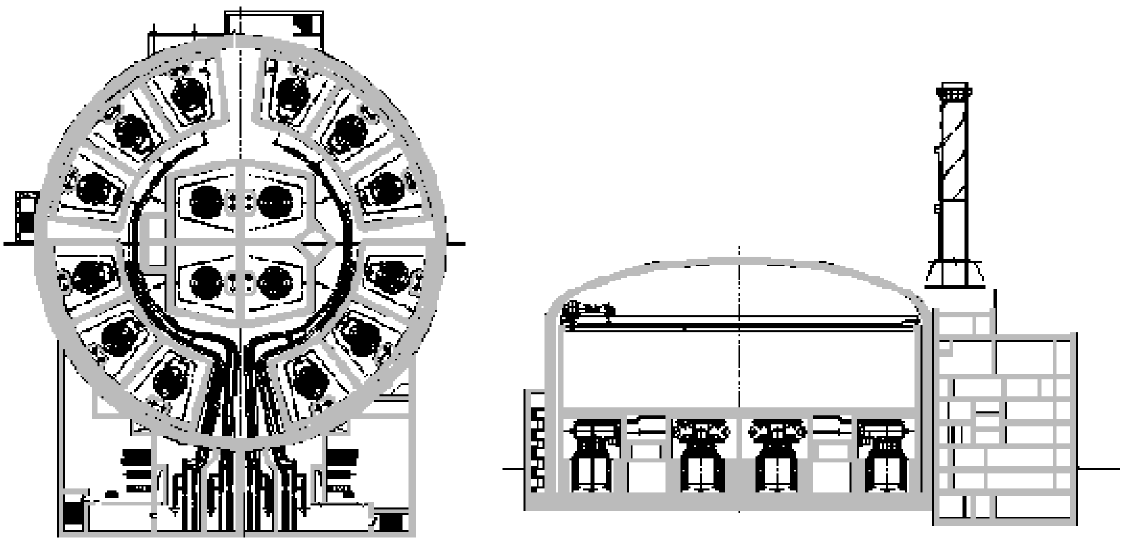


FIG. XIX-17. A plan and a longitudinal section of the main building of the NSSS for a 1600 MW(e) plant with the SVBR-75/100 modules.

The results of technical and economic evaluations (see Table XIX-6), which correspond to the stage of conceptual design, indicate that the economic parameters of a NPP with two units of 1600 MW(e) each based on the SVBR-75/100 reactor modules are superior to those of a NPP based on traditional large-capacity power reactors with thermal and fast neutron spectrum and those of heat power plants (HPP) with 10 steam-gas units (PGU-325) operating on natural gas. The construction period for the SVBR-75/100 based modular plant could be ~3.5 years. Taking into account the additional expenditures related to credit service (in case the construction is performed with the use of a loaned capital), the advantages of a modular NPP may be even higher.

The general plan of a desalination and power complex based on the SVBR-75/100 reactor installations is presented in Fig. XIX-18. Figure XIX-19 shows the reactor compartment of the second unit of the Novovoronezhskaya NPP when renovated with four modules of the SVBR-75/100.

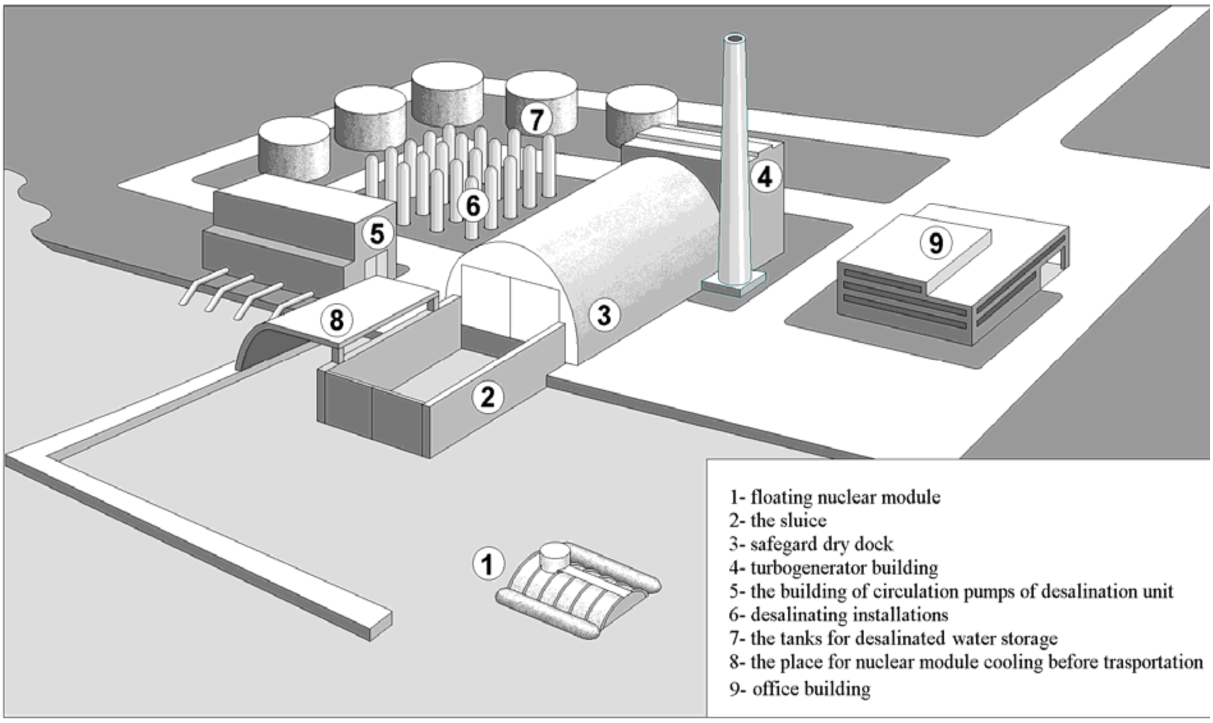


FIG. XIX-18. General plan of the nuclear desalination and power complex based on SVBR-75/100.

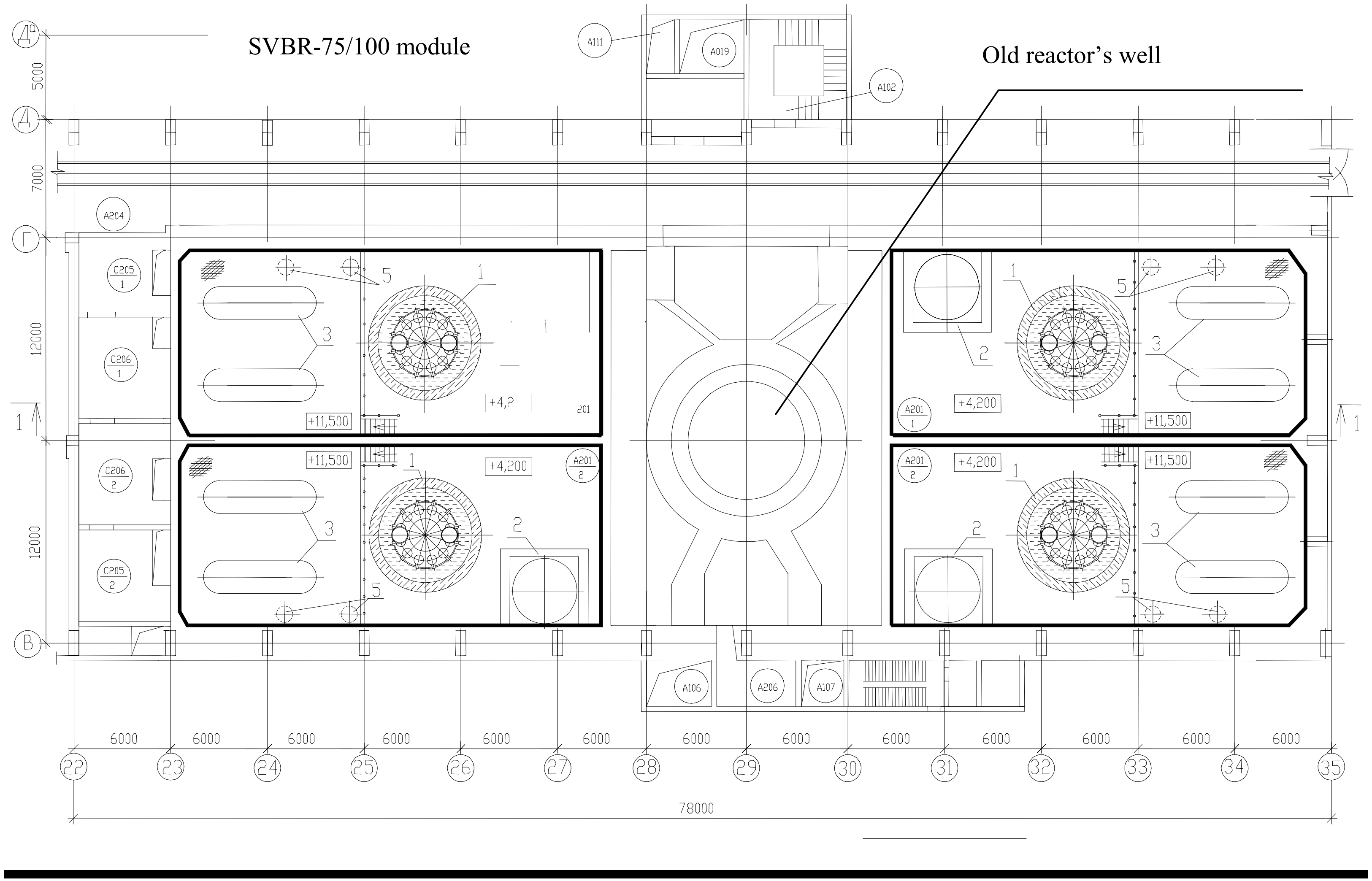


FIG. XIX-19. Reactor building of the second unit of the Novoronezhskaya NPP renovated with SVBR-75/100.

REFERENCES

[XIX-1] GROMOV, B.F., TOSHINSKY, G.I., STEPANOV, V.S., et al., Use of lead-bismuth coolant in nuclear reactors and accelerator-driven systems, Nuclear Engineering and Design, 173 (1997), pp. 207–217.

[XIX-2] GROMOV, B.F., ORLOV, YU.I., MARTYNOV, P.N., GULEVSKY, V.A., The problems of technology of the heavy liquid metal coolants (lead-bismuth, lead), Heavy-Liquid Metal Coolants in Nuclear Technology, (Proc. of Conf.) Vol. 1, Paper No. 13, p. 87 (Obninsk, Russia, 1999).

[XIX-3] PANKRATOV, D.V., YEFIMOV, YE.I., TOSHINSKY, G.I., RYABAYA L.D., Analysis of polonium hazard in nuclear power installations with lead-bismuth coolant, CD-ROM, Russian Scientific and Technical Forum. Fast Neutron Reactors (To commemorate the 100 birthday of A. I. Leypunsky). Heavy Liquid Metal Coolants in the Nuclear Technologies HLMC-2003 (Proc. of Conf. Obninsk, Russia, December 11–12, 2003) Paper 2401.

[XIX-4] PYLCHENKOB E.H., The issue of freezing-defreezing lead-bismuth liquid metal coolant in the reactor facilities circuits, Heavy-Liquid Metal Coolants in Nuclear Technology (Proc. of Conf.), Vol. 1, Paper No. 15, p. 110 (Obninsk, Russia, 1999).

[XIX-5] IGNATENKO, E.I., et al., Use of SVBR-75 reactor plant in design of renovation of NPP units of the first generation after termination of their service life, Heavy Liquid Metal Coolants in Nuclear Technology (Proc. of Conf.), Vol. 2, Paper No. B.5, p. 366 (Obninsk, Russia, 1999).

[XIX-6] ZRODNIKOV, A.V., et al., NPPs based on reactor modules SVBR-75/100, Atomnaya Energiya, Vol. 91, Issue 6, p. 415 (December 2001).

[XIX-7] GROMOV, B.F., et al., Use of the technology of ship reactors with liquid metal coolant for construction of the high safety regional NHPP, Innovative Development: Scientists of the Kaluga Region for the National Economy, (Proc. of Conf.) p. 83, Obninsk, Russia, 2000.

[XIX-8] ZAMUKOV, V.V. et al. Floating nuclear power plant “CRUISE-50”. Heavy-Liquid Metal Coolants in Nuclear Technology (Proc. of Conf.), Vol. 2, Paper No. B.22, p. 494 (Obninsk, Russia, 1999).

[XIX-9] GROMOV, B.F., et al., Nuclear power complex based on SVBR-75/100 small reactors cooled with lead-bismuth liquid metal coolant. Competitiveness, simplified life cycle, safety, non-proliferation, International Seminar on Status and Prospects for Small and Medium Sized Reactors (Proc. of Conf. Cairo, Egypt, 2001) IAEA C&S Paper Series 14/P, Report IAEA-SR-218/46,Vienna (August 2002).

[XIX-10] DEDUL, A.V., et al. Conceptual design of the heavy-liquid target complex of the integrated type for multipurpose ADS, CD-ROM, Russian Scientific and Technical Forum. Fast Neutron Reactors (To commemorate the 100 birthday of A. I. Leypunsky). Heavy Liquid Metal Coolants in the Nuclear Technologies HLMC-2003 (Proc. of Conf. Obninsk, Russia, December 11–12, 2003), Paper 2308.

[XIX-11] STEPANOV, V.S., KLIMOV, N.N., BOLVANCHIKOV, S.N., KULIKOV, M.L., Reactor module SVBR-75/100: Improved safety, transportability, opportunities of multipurpose usage, CD-ROM, Russian Scientific and Technical Forum. Fast Neutron Reactors (To commemorate the 100 birthday of A. I. Leypunsky). Heavy Liquid Metal Coolants in the Nuclear Technologies HLMC-2003 (Proc. of Conf. Obninsk, Russia, December 11–12, 2003), Paper 2103.

[XIX-12] ZRODNIKOV, A.V., et al., Multipurpose lead-bismuth cooled small power modular fast reactor SVBR-75/100, IAEA International Conference on Innovative Nuclear Technologies and Innovative Fuel Cycles, (Proc. Int. Conf. June, 2003) Report IAEA CN-108-36, Vienna (2003).

[XIX-13] BARANAEV, YU.D., PEKANOV, A.A., TOSHINSKY, G.I., Estimation of the economical effectiveness of the NDPC based on RI SVBR-75/100 with use of code DEEP, CD-ROM, Small Power-2003 (Proc. of Conf., November 12–13, 2003, Obninsk, Russia), p. 64.

[XIX-14] GRIGORIEV, O.G., KOMLEV, O.G., NOVIKOVA, N.N., TOSHINSKY, G.I., Neutronic and physical characteristics of reactor SVBR-75/100 with different types of fuel, Workshop on Advanced Nuclear Reactor. Safety Issues and Research Needs, (Paper presented at OECD Workshop, Paris, France, 2002).

[XIX-15] ZRODNIKOV, A.V., et al., Multipurpose small power fast reactor SVBR-75/100 and its possible fuel cycles, International Conference on Nuclear Power and Fuel Cycles, CD-ROM, Paper No. 18, Moscow, NIKIET, December 1–2, 2003 (Proc. of Conf. Moscow-Dimitrovgrad, Russia, 2003).

[XIX-16] PAULSON, C.K., Westinghouse AP-1000 advanced plant simplification results, measures and benefits, ICONE-10 (Proc. 10th Int. Conf. on Nuclear Engineering, Arlington, VA, USA, 2001), Report ICONE10-22784.

[XIX-17] ROGOV, M.F., KORNIENKO, A.G., Advanced and innovative projects in the development strategy of concern “Rosenergoatom”, International Conference on Nuclear Power and Fuel Cycles, CD-ROM, Paper No.7 Moscow, NIKIET, December 1–2, 2003, (Proc. Conf. Moscow-Dimitrovgrad, Russia, 2003).

[XIX-18] POPLAVSKY, V.M., et al., Advanced sodium fast rector BN-1800 meeting the requirements to the 21st century nuclear power, CD-ROM, Russian Scientific and Technical Forum. Fast Neutron Reactors (To commemorate the 100 birthday of A. I. Leypunsky). Nuclear Power Technologies with Fast Neutron Reactors (Paper presented at Conf., Obninsk, Russia, December 9–10, 2003), Paper 1301.

ANNEX XX

ENCAPSULATED NUCLEAR HEAT SOURCE (ENHS)

The University of California at Berkeley,
United States of America

XX-1. General information, technical features and operating characteristics

XX-1.1. Introduction

The Encapsulated Nuclear Heat Source (ENHS) Reactor concept [XX-1 to XX-22] was formulated in an evolutionary process that started in 1996 when Lawrence Livermore National Laboratory (LLNL) supported by the University of California, Berkeley (UCB), Nuclear Engineering Department, initiated a Laboratory Directed Research and Development (LDRD) programme to investigate the possibility of developing a nuclear power system concept that would be highly proliferation resistant, more user-friendly and available to developing countries [XX-23]. The LLNL effort, headed by Neil Brown, contributed significantly to developing the concept of nuclear battery type reactors and to the definition of attributes now specified for the Generation IV reactors.

During the two year study period of FY97-FY98, the UCB/LLNL team identified and studied the potential for a small liquid metal cooled reactor (LMR) to satisfy the requirements developed by LLNL to improve proliferation resistance [XX-24 to XX-26]. The work was initially considering the long-life core super-safe, small and simple (4S) reactor design [XX-27 to XX-29] of the Central Research Institute of Electric Power Industry of Japan (CRIEPI) and Toshiba. The UCB/LLNL study identified new core designs for long life, including the fuel self-sustaining core that was later adopted for the ENHS.

In the summer of 1998, LLNL organized a consortium of national laboratories, industries and universities to jointly prepare proposals for a secure, transportable, autonomous reactor (STAR) system concept [XX-30] for the forthcoming nuclear energy research initiative (NERI) programme. The encapsulated nuclear heat source (ENHS) reactor concept [XX-1] was invented in the summer of 1998 by David Wade of the ANL and by Ehud Greenspan of the UCB while brainstorming on the liquid-metal cooled reactor concept to propose as a STAR for the NERI program. The NERI proposal of the ENHS combined an innovative idea of corrugated confinement wall with the long-life core conceived at UCB and with the control elements of the 4S reactor [XX-27 to XX-29].

The ENHS NERI proposal was approved and the feasibility of the ENHS reactor concept was assessed by an international project team from the fall of 1999 through the end of 2002. The US team studying the feasibility of the ENHS reactor concept consisted of the University of California, Berkeley (leading), Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and Westinghouse. Collaborating with the US team were three Korean organizations: Korean Atomic Energy Research Institute (KAERI), Korean Advanced Institute for Science and Technology (KAIST) and the University of Seoul. Throughout the project we had useful interaction with CRIEPI, and to a lesser extent, Toshiba researchers that were involved in the design of the 4S reactor. In the second part of our project CRIEPI undertook to perform an independent evaluation of the transient behaviour and safety characteristics of the ENHS reactor. The number of researchers that had some participation in the ENHS study is approximately 40, of which 15 were students. In the course of the three years of ENHS NERI project there have been several modifications to the original ENHS concept.

Following the termination of the NERI project of three years, the UCB continued studying improvement possibilities in the ENHS reactor concept. Some research was done in parallel in Korea (KAIST) and in Japan (CRIEPI and JNC). In 2003 DOE selected six reactor categories as candidates for Generation IV reactors. One of the six categories was lead fast reactors (LFR) that include nuclear battery type reactor concepts like the ENHS. Since 2003 the UCB studies were supported by the Generation IV programme through ANL and LLNL.

XX-1.2. Applications

The ENHS reactor is intended to provide electricity, seawater desalination, process-heat and, possibly, district heating for developing countries and for remote sites that are not connected to a central electricity grid [XX-10, XX-31]. Multi-ENHS module power plant can also be of interest to industrial countries.

XX-1.3. Special features

Special features of the ENHS reactor include the following:

- The ENHS modules are fabricated, fuelled and weld sealed in the factory. They are transported to the power plant site with the fuel embedded in solid lead-bismuth;
- At least 20 years of full power operation without refuelling. No on-site refuelling or fuelling hardware is required;
- At end of life, the ENHS module serves as a spent fuel storage cask and, later, as a spent fuel shipping-cask for return to a regional fuel cycle centre.

XX-1.4. Summary of major design and operating characteristics

The design features of the ENHS reactor include the following:

- A small modular reactor; nominal power level is 125 MW(th) with possible upgrade to ~180 MW(th);
- There are no fuel assemblies in the core; each fuel rod is anchored in the factory to the grid plate;
- Nearly constant fissile fuel content and neutron multiplication factor; hence, very small excess reactivity built-in and very simple reactor control system that requires adjustment for burn-up only once every few years;
- Nearly constant power density shape across the core throughout its life;
- There are no pumps or valves in the primary and secondary coolant loops. The coolants flow by natural circulation;
- The natural circulation results in passive load following capability and autonomous control. It also makes loss-of-flow accidents inconceivable;
- There are no special decay heat removal systems other than a reactor vessel air-cooling system (RVACS) that uses natural air draft for the heat sink;
- As all structural components exposed to neutrons are disposed off with the ENHS module once in more than 20 years, the ENHS reactor lifetime might exceed 100 years.

Figures XX-1 and XX-2 show simplified schematic views of the ENHS reactor. The reference ENHS reactor has two coolant circuits, both being of a pool type; the primary coolant circulates inside the ENHS module while the secondary or intermediate coolant circulates in the pool the ENHS module is inserted in. The two coolants interface each other across the intermediate heat exchanger (IHX) that is an integral part of the ENHS module. This arrangement was selected for a couple of reasons:

(1) There are no mechanical connections between the ENHS module and the energy conversion system; this simplifies interfacing the module with the power plant.

(2) Neither the ENHS module vessel nor the IHX have to withstand large pressure differentials. This simplifies the IHX design as well as the ENHS vessel design.

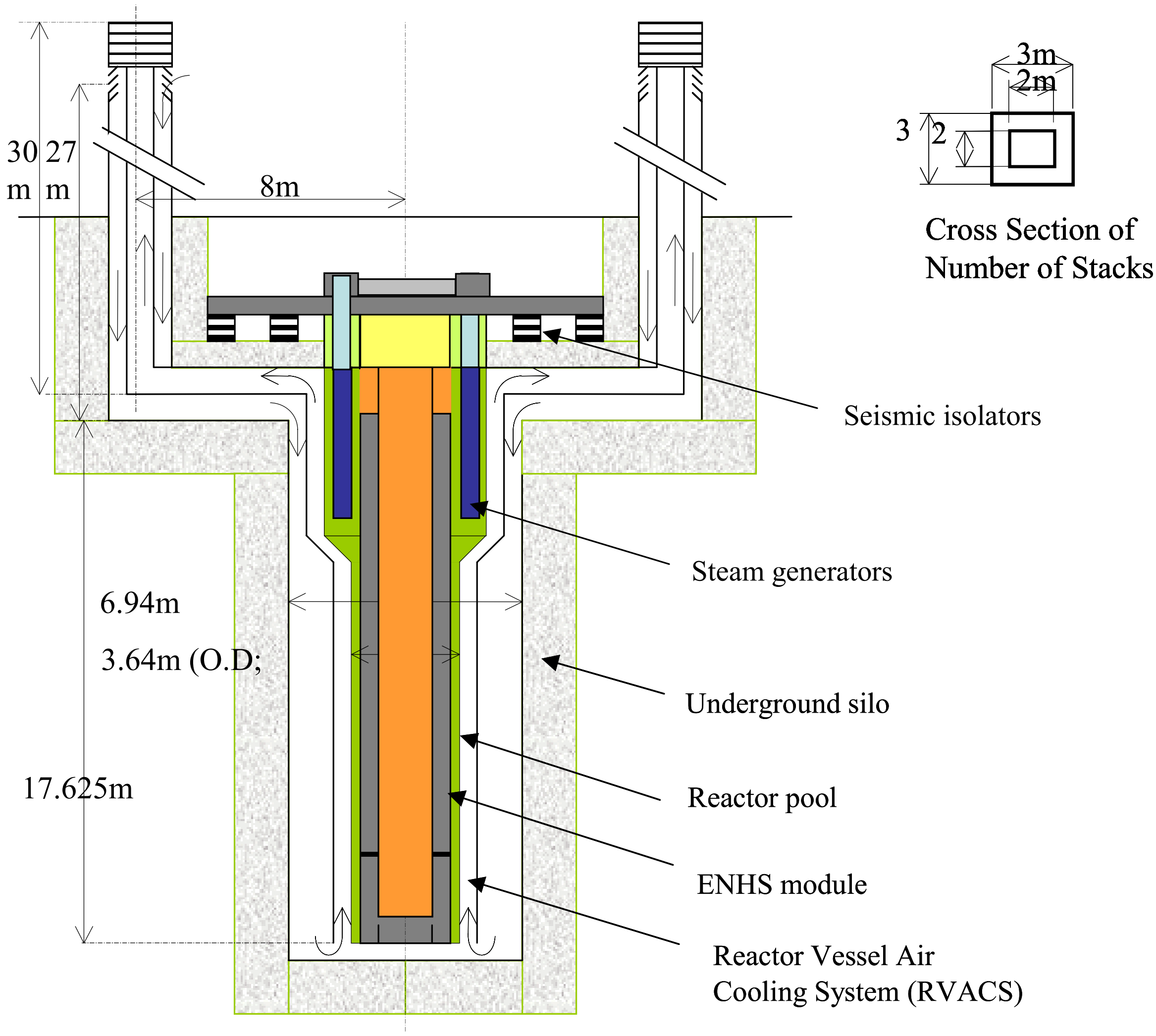


FIG. XX-1. Schematic view of the ENHS reactor.

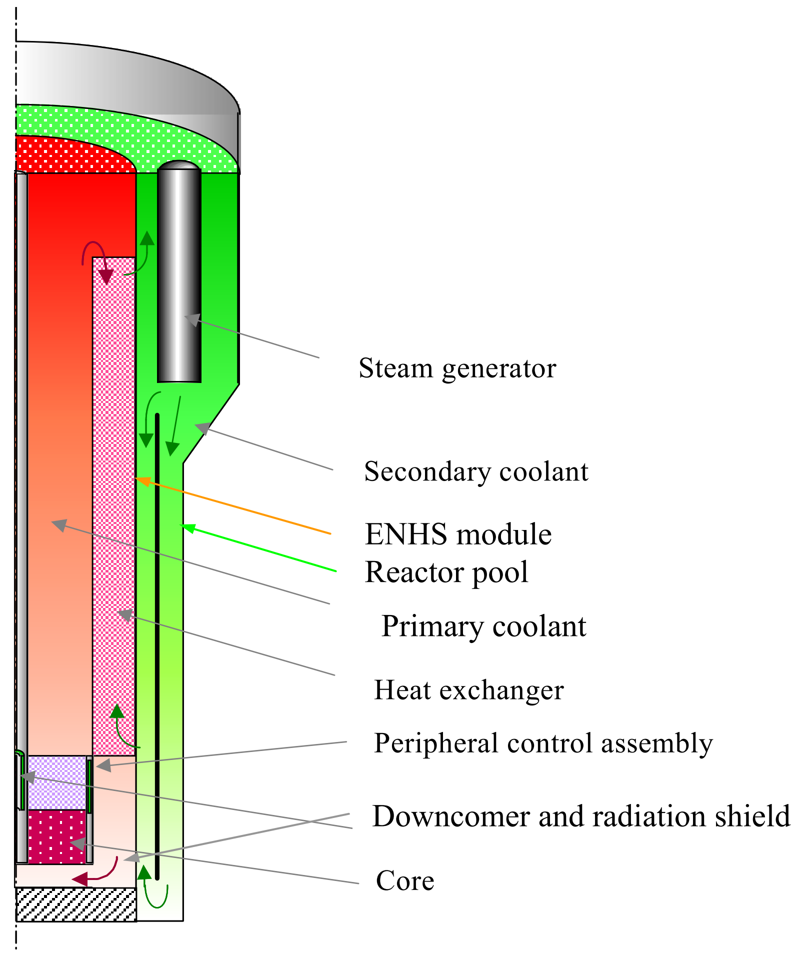


FIG.XX- 2. Schematic vertical view of the ENHS module and pool.

The ENHS module is designed to be as simple, robust and proliferation resistant as possible. There are no moving components except for the control (6 pieces) and safety (1 piece) elements drives. The core is an annular cylinder made of uniform lattice of fuel rods that are individually tied up to the lower grid plate; there are no fuel assemblies. There are no blanket and no solid reflector assemblies. The safety element is located in the coolant-filled cavity at the core centre; the diameter of this cavity is determined so that the reactivity worth of the safety element will be adequate. The central cavity can also be used for flattening of the radial power distribution across the core.

The nuclear steam supply system (NSSS) consists of one ENHS module and eight small steam generators. There is no mechanical connection between the module and the steam generators. Both primary and secondary coolants flow by natural circulation. The primary coolant that is heated in the core flows up the riser, turns over into the intermediate heat exchanger (IHX) and flows back into the coolant plenum underneath the core. The secondary coolant flows from the pool outside of the vessel into the bottom of the IHX and exits back to the pool near the top of the IHX. The IHX, depicted in Fig. XX-2, consists of rectangular channels that are connected at their top and bottom to a tube sheet (not shown in Fig. XX-2). The 4 mm thick rectangular channel walls provide the barrier between the primary and the secondary coolants whereas the inner and outer walls provide the structural support. More conventional IHX made of circular tubes could be used as well. Relative to circular tube IHX, the rectangular channel IHX features close to an order-of-magnitude smaller number of channels and smaller friction losses due to elimination of grid spacers. The mean temperature difference between the primary and secondary coolants does not exceed 50°C.

Major design and operating characteristics of the ENHS are summarized in Table XX-1. The neutron-physical characteristics are presented in Tables XX-2 and XX-3. A simplified schematic diagram of the ENHS energy conversion system is presented in Fig. XX-3.

TABLE XX-1. SUMMARY TABLE OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

Characteristic	Value
Rated thermal power	125 MW(th) nominal; can be upgraded to 180 MW(th)
Rated electrical power	50 MW(e) nominal; can be upgraded to 75 MW(e)
Availability (target)	~99%
Fuel type	(1) Pu-$U_{Dep.}$-Zr(10%), (2) U-Zr(10%), (3) PuN-$U_{Dep.}$N or (4) UN; $U_{Dep.}$ denotes the depleted uranium
Fuel enrichment	(1) 12.2 weight % Pu, (3) 13.1 weight % Pu; N enriched to 99% ^{15}N
Primary coolant	Pb-Bi eutectic; Na is an option
Secondary coolant	Pb-Bi eutectic;
Structural material	HT-9 ferritic-martensitic steel
Core type and dimensions	Annular cylinder; Inner/outer effective radius - 16.41/111.83 cm; Uniform composition fuel; No control elements in the core; no solid reflector elements; Lattice - hexagonal; P/D - 1.36; Fuel rod cladding outer diameter - 1.56 cm; Cladding thickness - 0.13 cm; Fuel smear density - 75%; Active fuel length - 125 cm; Fission gas plenum length - 1.25 cm.
Type of reactivity control	One central absorber; inner/outer radius - 10.05/15.05 cm; Six annular segments of peripheral absorber; inner/outer radius-116.83/121.83 cm.
Reactor module dimensions	Height -19.6 m (10.1 m)[a] Outer diameter -3.52 m (3.72 m)[a] Riser diameter - 214.8 cm (204.6 m)[a] Number of rectangular channels in IHX - 135 (435)[a] Inner dimensions of IHX channels (cm×cm) 39.2×2.5 (49.2×0.5)[a] IHX channel length - 11 m (3 m)[a]
Cycle type	Water Rankine cycle; Supercritical CO_2 Brayton cycle is an option.
Number of loops	Single secondary loop; 8 small steam generators.
Energy conversion efficiency	120 bars: 38.4% (net) with single reheat; 38.2% with no reheat. 180 bars: 40.7% (net) with single reheat; 40.4% with no reheat.
Module life	> 20 effective full power years
Reactor life	Target: > 100 years

[a] Corresponding to an alternative design variant that uses a cover gas lift pump to assist the circulation of the coolant.

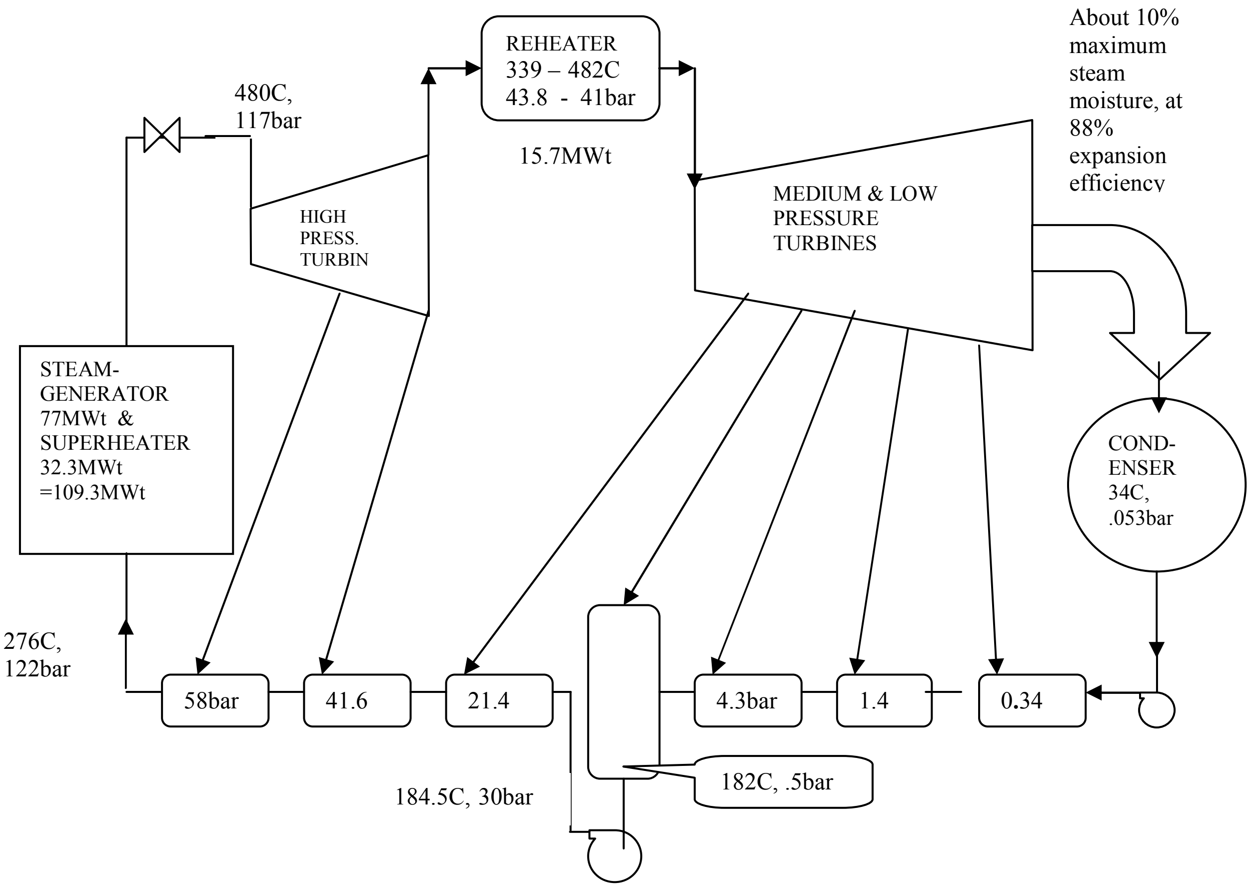


FIG. XX-3. Reference energy conversion system [XX-9].

TABLE XX-2. NEUTRON-PHYSICAL CHARACTERISTICS AND CONTROL ROD WORTH; METALLIC FUEL [XX-13, XX-15, XX-19]

CHARACTERISTIC	BOL	EOL
Pitch-to-diameter ratio	1.36	
Pu weight %	12.20	
Maximum burn-up reactivity swing $\Delta\rho$ (%)	0.221	
Peak-to-average power	1.829	1.863
Peak burn-up after 20 (20.9*) EFPY (GW·day/t HM**)	99.89 (104.4)	
Average burn-up after 20 (20.9*) EFPY (GW·day/t HM)	50.80 (53.1)	
Peak fast (E>0.1MeV) neutron flux (n/cm^2-s)	5.96E+14	6.06E+14
Peak fast (E>0.1 MeV) fluence at 20 EFPY (n/cm^2)	3.829E+23	
Peak linear power (W/cm)	184.1	187.6
Conversion ratio	1.0446	1.0482
Effective delayed neutron fraction	0.00388	
Doppler effect (δk/k k’, °C)	-5.2442E-06	-4.2202E-06
Axial fuel expansion (δk/k k’, °C)	-4.6379E-06	-4.6057E-06
Coolant expansion (δk/k k’, °C)	+1.0917E-07	+5.6866E-07

CHARACTERISTIC	BOL	EOL
Grid-plate radial expansion (δk/k k', °C)	-8.0679E-06	-6.4097E-06
Cold (350°C) to hot (480°C; fuel: 700°C) δk/k k':		
- Doppler effect	-1.9600E-03	-1.5399E-03
- Axial fuel expansion	-6.1421E-04	-6.0921E-04
- Coolant expansion	+2.6400E-05	+8.6558E-05
- Grid-plate expansion	-1.0516E-03	-8.3430E-04
- Total	-3.5994E-03	-2.8968E-03
Coolant void reactivity effect (δk,%):		
- Voiding inner 1/3 core (+upper gas plenum)	+2.718 (+1.516)	+2.809 (+1.618)
- Voiding middle 1/3 core (+upper gas plenum)	+0.689 (-0.068)	+0.659(-0.043)
- Voiding outer 1/3 core (+upper gas plenum)	-0.694 (-1.045)	-0.668 (-1.009)
- Voiding whole core (+upper gas plenum)	+2.555 (+0.424)	+2.644 (+0.572)
Coolant loss reactivity effect (δk/k k', %):		
- Coolant level drops to bottom of fission gas plenum	-1.869	
- Coolant level drops to 1/3 below core top	-6.261	
- Coolant level drops to 2/3 below core top	-12.219	
- Coolant level drops to core bottom	-14.825	
Peripheral absorber reactivity worth (δk,%)	1.990	1.817
Central absorber reactivity worth (δk,%)	4.138	4.104
Peripheral + central absorber reactivity worth (δk,%)	6.811	6.562
Total heavy metal inventory (kg)	17505	16564
Fissile-to-total plutonium ratio	0.6437	0.6649

* Limited by radiation damage to cladding of 4×10^{23} fast (E>0.1 MeV) neutrons per cm^2.
** Heavy metal; including all the actinides in the fuel (HM)

TABLE XX-3. NEUTRON-PHYSICAL CHARACTERISTICS AND CONTROL ROD WORTH; NITRIDE VERSUS METALLIC FUEL AT BOL [XX-18, XX-19]

CHARACTERISTIC	FUEL TYPE		
	Pu-U-Zr	PuN-UN	$Pu^{15}N$-$U^{15}N$
Pitch-to-diameter ratio	1.36	1.21	1.45
Pu weight %	12.20	12.36	13.08
Maximum burn-up reactivity swing Δρ (%)	0.221	0.287	0.368
Peak-to-average power	1.829	1.893	1.856
Peak burn-up after 20 EFPY (GW·day/t HM)	99.89 (104.4)	122.5 (153.1)	127.4 (175.5)
Average burn-up after 20 EFPY (GW·day/t HM)	50.8 (53.1)	59.8 (74.8)	63.7 (87.7)
Peak fast (E>0.1MeV) neutron flux (n/cm^2-s)	5.96E+14	4.7708E+14	4.4334E+14
Peak fast (E>0.1MeV) fluence at 20 EFPY(n/cm^2)	3.829E+23	3.200E+23	2.904E+23
Peak linear heat rate (W/cm)	184.1	190.9	186.9
Conversion ratio	1.0446	1.0529	1.0399
Effective delayed neutron fraction	0.00388	0.00403	0.00382
Doppler effect (δk/k k', °C)	-5.2442E-6	-1.2218E-5	-8.9799E-6
Axial fuel expansion (δk/k k', °C)	-4.6379E-6	-2.1778E-6	-2.7625E-6
Coolant expansion (δk/k k', °C)	+1.0917E-7	-3.4331E-8	+1.520E-6
Core radial expansion (δk/k k', °C)	-8.0679E-6	-5.4325E-6	-7.2268E-6

CHARACTERISTIC	FUEL TYPE		
	Pu-U-Zr	PuN-UN	$Pu^{15}N-U^{15}N$
Cold (350°C) to hot (480℃; fuel:700°C) δk/k k': - Doppler effect - Axial fuel expansion - Coolant expansion - Grid-plate expansion - Total	 -1.9600E-3 -6.1421E-4 +2.6400E-5 -1.0516E-3 -3.5994E-3	 -4.4977E-3 -2.8840E-4 -1.0000E-6 -7.1610E-4 -5.5032E-3	 -3.2837E-3 -3.6698E-4 +2.050E-4 -9.4410E-4 -4.3898E-3
Coolant void reactivity effect (δk;%): - Voiding inner 1/3 core/+ upper gas plenum - Voiding middle 1/3 core/+ upper gas plenum - Voiding outer 1/3 core/+ upper gas plenum - Voiding whole core/+ upper gas plenum	 +2.718/+1.516 +0.689/-0.068 -0.694/-1.045 +2.555/+0.424	 +1.465/+1.042 +0.435/+0.168 -0.276/-0.411 +1.575/+0.803	 +3.114/+1.905 +0.978/+0.224 -0.508/-0.839 +3.462/+1.374
Coolant loss reactivity effect (δk/k k',%): Coolant level drops to bottom of gas plenum Coolant level drops to 1/3 below core top Coolant level drops to 2/3 below core top Coolant level drops to core bottom	 -1.869 -6.261 -12.219 -14.825	 -0.856 -3.195 -6.389 -7.469	 -1.638 -5.183 -9.114 -10.562
Peripheral absorber reactivity worth (δk; %)	1.990	1.43	2.070
Central absorber reactivity worth (δk; %)	4.138	3.05	3.600
Peripheral +central absorber worth (δk; %)	6.811	5.02	6.320
Total heavy metal inventory (κg)	17505	19213	19212
Fissile-to-total plutonium ratio	0.6437	0.6437	0.6437

Currently, the basic design of the control and shutdown systems used for the ENHS was adopted from the 4S reactor of Toshiba-CRIEPI [XX-27 to XX-29]. The central absorber has an electromagnetic latch that does not engage until the start-up temperature of 350°C is achieved. At this temperature the assembly can be withdrawn. Normal operational shutdowns can be accomplished with the peripheral absorbers. The reactor is brought critical by a hydraulic system that moves the peripheral absorbers up at 1 mm/sec to compensate for the negative temperature coefficient of reactivity. At the full power position, the peripheral absorber segments are stopped from further upward movement by mechanical stoppers whose movement is established by high-reliability gear drives. These drives restrict the rate of movement of the peripheral absorber segments to approximately 1 mm/day. It is anticipated that the height of the peripheral absorbers will be adjusted once a year or two to compensate for a slight drift in reactivity due to fuel burn-up. During shipping and reactor installation the absorbing elements are securely latched in place.

The mass balances of fuel materials are presented in Table XX-4. Table XX-5 gives thermal hydraulic characteristics of the ENHS.

Results of preliminary economic analysis performed for the reference ENHS reactor ("Base Case") [XX-32] are summarized in Table XX-6 below. This table also gives results of the sensitivity of the cost of electricity to uncertainties in different assumptions. Definition of the scenarios considered for the sensitivity analysis is given in Table XX-7. Assumptions made for the Base Case are summarized in Section XX-2.1 of reference [XX-32]. A breakdown of the Base Case cost-by-cost category and by major components is given in the pie-chart diagrams of Figures XX-4 and XX-5.

TABLE. XX-4. FUEL MASS BALANCE OVER 20 EFPY OF OPERATION [XX-13, XX-15, XX-19].

ISOTOPE	INVENTORY (KG)	
	BOL	EOL MASS INCREASE
^{234}U	0.0	7.42
^{235}U	30.74	-12.98
^{236}U	0.0	2.53
^{238}U	15339	-1066.3
^{238}Pu	67.99	-22.6
^{237}Np	0.0	3.08
^{239}Pu	1203	177.9
^{240}Pu	568.4	-2.58
^{241}Pu	171.3	-109.0
^{242}Pu	124.5	-8.28
^{241}Am	0.0	75.8
^{242m}Am	0.0	2.4
^{243}Am	0.0	9.33
^{242}Cm	0.0	0.557
^{243}Cm	0.0	0.013
^{244}Cm	0.0	1.23
^{245}Cm	0.0	0.0845
^{246}Cm	0.0	0.002
Initial loading of depleted U, kg/GW(e)	15369.7	
Consumption rate of depleted U, (kg/GW(e)/year)	941.4	

TABLE XX-5. THERMAL-HYDRAULIC CHARACTERISTICS [XX-4, XX-14]

CHARACTERISTIC	REFERENCE DESIGN	LIFT-PUMP DESIGN
Coolant	LBE/Lead	LBE/Lead
Roughness height, μm	10	10
Gas injection above core/ lift pump	No	Yes[b]
Cladding-fuel gap thickness (Na bonded), cm	0.0871	0.0871
Number of fuel rod spacer grids	3	3
Core peak-to-average power	1.767	1.767
Peak-to-average power factor at top of core	0.942	0.942
Riser height, m	13	3.0
HX channel height, m	11	1.75
Primary coolant thermal centres separation height, m	8.125	3.50
HX channel width, cm	2.5	0.5
HX channel thickness, cm	39.2	49.2
Number of HX channels	135	434
Number of SG modules	8	8
SG height, m	4.8	4.8
SG tube outer diameter, cm	2.0	2.0
Number of SG tubes per SG module	613	613
Height difference between SG inlet and HX outlet, m	1.15	1.15

CHARACTERISTIC	REFERENCE DESIGN	LIFT-PUMP DESIGN
Intermediate coolant thermal centres separation, m	4.25	-0.375
Core flow area, m^2	1.83	1.83
Primary coolant HX flow area, m^2	1.86	1.87
Intermediate coolant HX flow area, m^2	1.32	1.07
Intermediate coolant SG flow area, m^2	2.04	2.04
Core-to-HX flow area ratio	0.980	0.975
Primary coolant flow rate, kg/s	8320/8220	12100/12200
Intermediate coolant flow rate, kg/s	5970/5870	7840/7910
Water flow rate in SG, single reheat, kg/s	52.3	52.3
Coolant velocity in core, m/s	0.447/0.426	0.646/0.630
Primary coolant velocity in HX, m/s	0.438/0.418	0.630/0.615
Intermediate coolant velocity in HX, m/s	0.440/0.418	0.715/0.697
Intermediate coolant velocity in SG, m/s	0.285/0.271	0.374/0.364
Reynolds number in core	47400/30800	66900/43800
Primary coolant Reynolds number in HX	159000/103000	65400/42800
Intermediate coolant Reynolds number in HX	128000/80700	42900/27500
Intermediate coolant Reynolds number in SG	44000/27800	56500/36300
Fraction of primary coolant pressure drop in HX	0.370/0.374	0.303/0.307
Fraction of intermediate coolant pressure drop associated with HX	0.701/0.699	0.790/0.788
Primary coolant temperature rise, °C	103/104	70.8/69.8
Intermediate coolant temperature rise, °C	143/144	109/107
Primary coolant outlet temperature, °C	503/504	471/470
Primary coolant inlet temperature, °C	400	400
Intermediate coolant outlet temperature, °C	471/477	443/445
Hot channel primary coolant temperature rise, °C	147/149	102/101
Hot channel outlet temperature, °C	547/549	502/501
Hot channel peak cladding outer surface temperature, °C	568/567	520/516
Hot channel peak cladding inner surface temperature, °C	579/579	531/527
Hot channel peak sodium bond temperature, °C	583/582	534/531
Hot channel peak fuel temperature, °C	622/621	575/572
Intermediate coolant inlet temperature, °C	328/333[(c)]	334/338[(c)]
Primary-to-intermediate coolant temperature difference at HX top, °C	31.4/26.60	27.7/24.5
Primary-to-intermediate coolant temp. difference at HX bottom, °C	71.6/67.4	65.7/61.6

[b] Primary two-phase region height is 1.75 m; intermediate two-phase region height is 2.0 m; void fraction is 0.1.
[c] Impractical to use Pb as a secondary coolant; too close to its melting temperature (327°C).

The economic analysis assumes assembly line mass production of the ENHS modules. Hence, the capital cost accounts for investment in the factory and depends on the number of modules the factory produces per year. The fuel cycle cost is considered as a component of the capital cost as the fuel lasts as long as the module; it is an integral part of the module. The costs are in US$.

TABLE XX-6. RESULTS OF SENSITIVITY ANALYSIS FOR ECONOMIC CHARACTERISTICS

	Base Case	Site Labour 2X	Fact. Labour 2X	High SWU Price	High U_3O_8 Price	High Inter. Rate	Low Capac. Factor	Long Const. Period	High Prod. Rate
Unit capital cost, $/kW(e):									
Without fuel	914	926	921	924	932	958	914	1,524	911
For fuel	1087	1087	1087	1207	1302	1087	1087	1087	1087
Total unit capital cost	2001	2013	2009	2131	2234	2045	2001	2611	1999
Levelized costs, $M/year:									
Without fuel	3.96	4.01	4.00	4.00	4.03	5.01	3.96	6.43	3.95
For fuel	5.54	5.54	5.54	6.15	6.63	6.39	5.54	5.54	5.54
Total levelized capital cost	9.50	9.55	9.53	10.15	10.66	11.40	9.50	11.96	9.49
O&M costs, M$/year	3.93	7.83	3.93	3.93	3.93	3.93	3.93	3.93	3.93
Busbar costs, ¢/kWh									
Capital	1.01	1.02	1.01	1.02	1.02	1.27	1.13	1.63	1.00
O&M	1.00	1.99	1.00	1.00	1.00	1.00	1.12	1.00	1.00
Fuel	1.40	1.40	1.40	1.56	1.68	1.62	1.58	1.40	1.40
Total	3.41	4.41	3.42	3.57	3.70	3.89	3.83	4.03	3.40

TABLE XX-7. SCENARIOS CONSIDERED FOR SENSITIVITY ANALYSIS

CASE	VARIATION
Base	
Site labour 2X	Site labour cost is doubled
Factory labour 2X	Factory labour cost is doubled
High SWU price	SWU price is set to $100/SWU
High U_3O_8 price	U_3O_8 price is set to $50/kg
High interest rate	Interest rate is set to 10%
Lower capacity factor	Capacity factor is set to 80%
Longer construction period	Construction period is set to 8 years, plus six months for testing

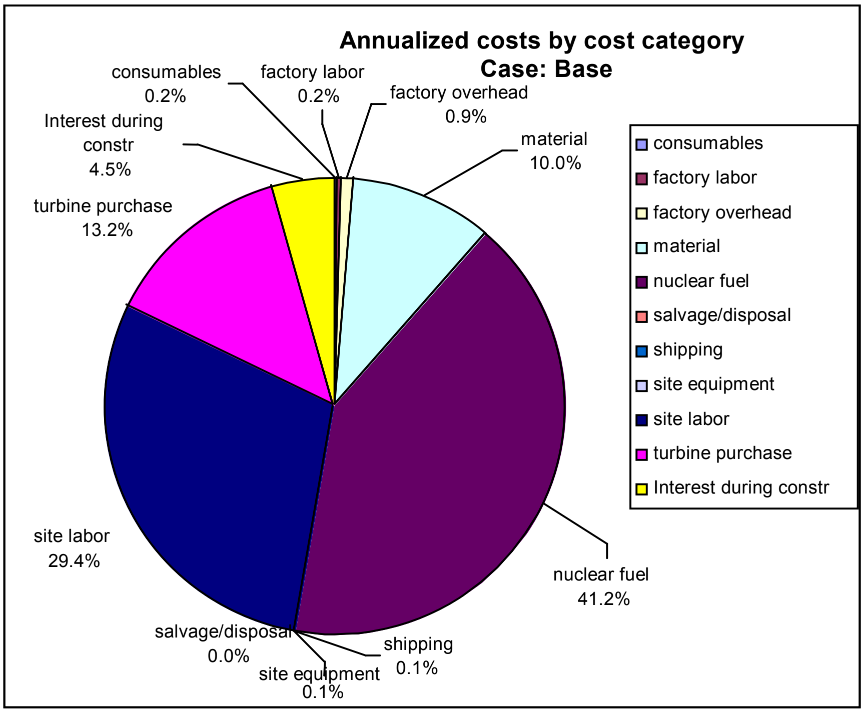


FIG. XX-4. Breakdown of annualized costs by cost category for Base Case.

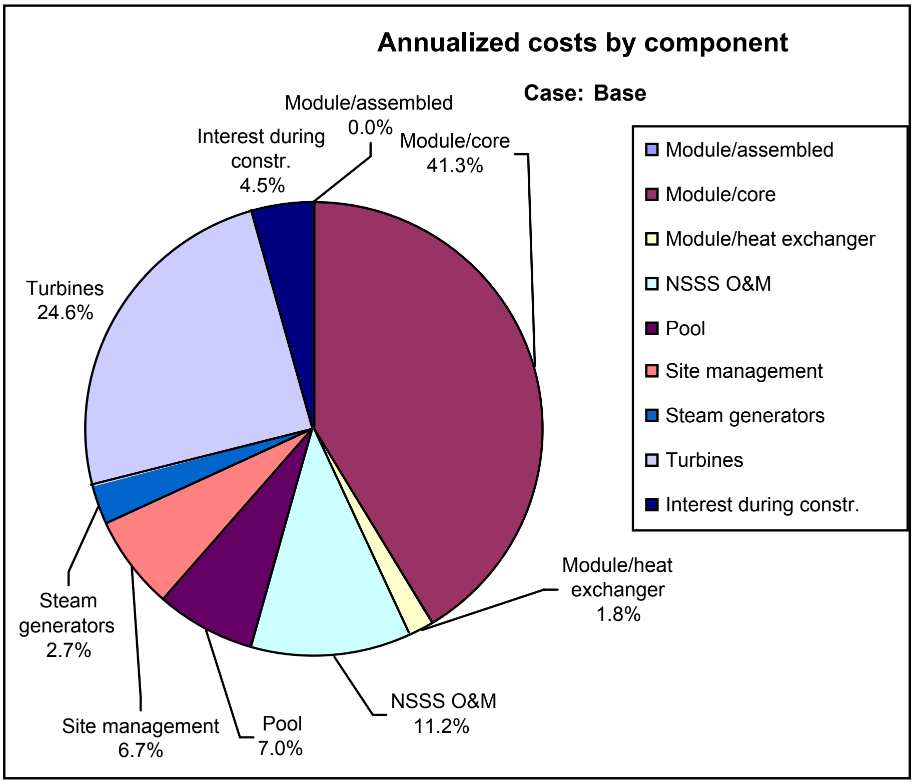


FIG. XX-5. Breakdown of annualized costs by generating unit component.

XX-1.5. Outline of fuel cycle options

The basic ENHS is a small nuclear reactor suitable for remote regions and countries with limited or developing energy infrastructures. The ENHS reactor and its fuel cycle are intended to make a highly secure nuclear energy system (HSNES) so as to reduce the complexity and expense of eliminating concerns about nuclear proliferation and severe nuclear accidents, even when the user is in the initial phase of developing an energy infrastructure [XX-31]. It is the objective of the HSNES to make the environmental, energy security and reliability benefits of nuclear energy available to all at a competitive cost. It may provide an appealing energy alternative because it does not require the user country to assume the many burdens normally associated with acquisition of nuclear energy, yet it could assure energy supply security due, in part, to the long core life - at least 20 years of effective full power operation.

Most of the countries currently using nuclear energy have had to establish their own nuclear infrastructure. In many cases this included the complete nuclear fuel cycle with the exception of uranium mining and enrichment. Others have not included fuel fabrication, but have nevertheless faced the need to develop a substantial nuclear technology capability. Users have typically established their own operator training, licensing and security organizations. Many large users of nuclear power are addressing the issues of reprocessing and waste disposal internally, while others have not yet reached a decision on these issues and are likely to seek to resolve them with external solutions.

The HSNES would address all the issues up front and could substantially reduce the burden on the user. It is envisioned that the supplier of the reactor system would retain responsibility for the maintenance and recycling of the nuclear components and the waste disposal. The relationship between the user and supplier countries could be similar to that found in the commercial airline business where the user provides trained operators for equipment licensed and constructed by the supplier. The equipment may also simply be leased, reducing the initial investment requirements for the user.

The long term fuel cycle envisioned for the ENHS reactor is a closed, fuel self sustaining (FSS) Pu-U cycle the initial feed for which is the fuel discharged from light water reactors (LWR) [XX-8, XX-31, XX-33]. The ENHS core is designed to maintain the fissile-fuel content nearly constant with burn-up; that is, to have a breeding ratio that is few percent above unity. The fuel k_∞ is nearly independent of the location in the core and of burn-up. There is a slight build-up of fissile fuel with burn-up that is used to compensate for the negative reactivity effect of the fission products that accumulate during the cycle. The fuel is discharged from the FSS core when reaching its radiation damage limit. What is necessary for reusing this discharged fuel is to remove all or part of the fission products, mix the heavy metal (HM) with makeup fuel and re-fabricate fuel elements. The make-up fuel, approximately 5% to 7% of the fuel loading, can be either depleted uranium or natural uranium or spent fuel from LWRs. The reactivity worth (or k_∞) of the resulting fuel is almost the same as the reactivity worth of the previous fuel loading and of the beginning of life (BOL) fuel loading. This is because the reactivity released by the removal of fission products is used to compensate for the low reactivity worth of the makeup fuel. The net result is that such a FSS can recycle its actinides many times in a highly proliferation resistant manner - without partitioning of actinides and without using uranium enrichment services. Relatively simple and proliferation resistant processes could be used for the extraction of fission products.

One of the processes proposed [XX-8, XX-33] for consideration for the extraction of fission products is an atomics international reduction oxidation (AIROX) like process [XX-34] that removes only volatile (^{3}H, I, Xe, Kr, ^{14}C) and semi-volatile (Ru-90%; Cd-75%; Te-75%)

fission products (FPs) from the discharged fuel such as the one being developed in the direct use of PWR fuel in CANDU reactors (DUPIC) programme [XX-35] in the Republic of Korea. The AIROX process, originally developed in the USA, has enhanced proliferation resistance features for FP removal process since (a) it leaves significant concentration of FPs in the fuel, thus making the fuel more radioactive, and (b) it can not partition actinides, since the separation of the FP is based on volatility induced by heating rather than by chemical or electrochemical means. For this reason the US State Department supports the DUPIC programme in the Republic of Korea that is aiming at recycling LWR spent fuel into CANDU reactors. As these water reactors do not breed, the fuel will be recycled only one time and the added burn-up is limited to the order of 50%. In the case of ENHS like reactors, on the other hand, the fuel could be recycled many times leading to a dramatic reduction in the volume and toxicity of the high-level waste along with a dramatic increase in the uranium utilization.

There remains the question of how to come by the first core loading without separation of Pu. One possibility [XX-8, XX-33] is to use LWR spent fuel as the feed material and to remove from it only part of the uranium and part or all of the FP. For example, if the LWR spent fuel contains 1% Pu and minor activities (MA), it is necessary to remove approximately 90% of the uranium to make a fuel with 11 to 12 % of Pu and MA by weight. This could hopefully be done using a highly proliferation-resistant process, possibly a combination of an AIROX process and a fluoride volatilization process or a simplified version of the UREX process. Another feed option that could be considered is the spent fuel from MOX fuelled LWRs. The transuranium isotopes (TRU) content in such spent fuel can be approximately half of that needed for ENHS like reactors. Hence, only ~50% of the uranium need be extracted along with FP to make fuel for ENHS like reactor. The latter is likely to offer a more economical fuel cycle.

The resulting multi-recycle fuel cycle features a high fuel utilization - an order of magnitude higher than in LWR, along with a great reduction in the inventory of the high level waste that need be disposed of in an underground repository.

An energy system based on the proposed combination of FSS reactors and proliferation-resistant multi-recycling using spent fuel from LWRs as the feed is environmentally attractive for three reasons:

(1) It converts most of the TRU discharged from LWRs into high quality fuel.
(2) It generates nuclear energy without increasing the total inventory of Pu. The total Pu and MA inventory in the FSS based energy system of a given capacity is nearly constant. Most of the Pu and MA inventory is well secured inside the reactor core.
(3) It maximizes utilization of the existing uranium resources - possibly generating ~50 times more energy per unit weight of mined U than generated in an LWR operating with the once-through fuel cycle.

The only waste anticipated from the above proposed energy system consists of the following components:

(1) The FPs extracted in the dry process.
(2) Transuranium elements (TRU) and FPs that cannot be recovered from the spent fuel cladding material.
(3) TRU and FP waste from the fuel fabrication process.
(4) Coolant and structural material activation products. This waste has not yet been quantitatively assessed.

Ideally, the fuel recycling and module fabrication facilities are to be collocated as illustrated, schematically, in Fig. XX-6. The fuel feed to such a site could be spent fuel from LWRs and, when needed, natural or depleted uranium. The only fuel sent out from these sites will be embedded in ENHS modules. The radioactive waste from the fuel processing and from the used ENHS modules will be shipped for disposal in regional or international waste centres.

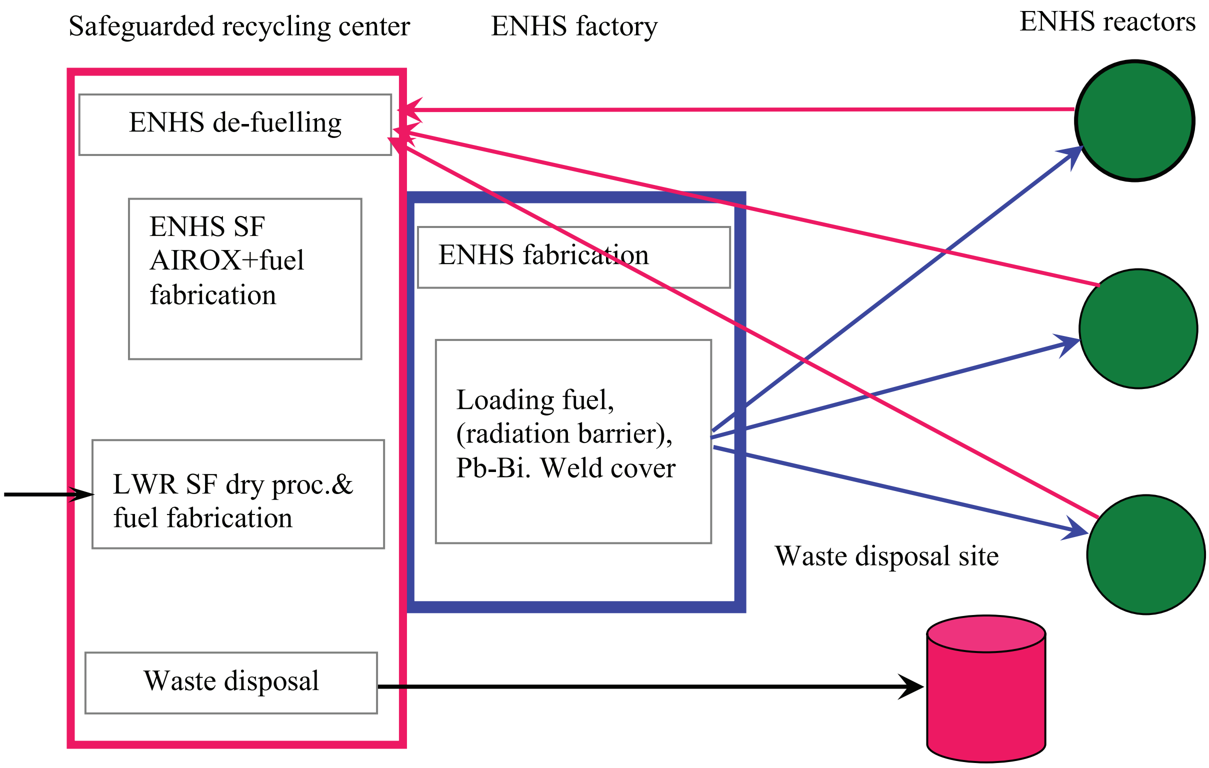


FIG. XX-6. A schematic illustration of ENHS based nuclear energy systems in which the ENHS module fabrication and fuel processing and handling facilities are collocated.

An alternative initial fuel loading option for the ENHS reactor is enriched uranium. With approximately 12.7 weight % ^{235}U the ENHS core will be FSS [XX-36]. The discharged fuel could be recycled many times using a similar strategy outlined above.

XX-1.6. Technical features and technological approaches that are definitive for ENHS performance in particular areas

XX-1.6.1. Economics and maintainability

Specific technical features and technological approaches of the ENHS reactor that contribute to its economics and maintainability include the following:

- Factory fabrication and fuelling of the reactor module combined with very short - less than 2 years, on site power plant construction time. The economics benefits from assembly-line mass production of many modules of identical designs.
- Once for life (>20 effective full power years (EFPY)) core along with no refuelling or fuel shuffling on site; the entire ENHS module is replaced when the core reaches end of life. Very high availability is expected. Plant life target is on the order of 100 years.

Inventory of lead or lead-bismuth from the decommissioned reactor can be used as the coolant of a new reactor.

- Very simple design with relatively few components. This is due to the superb passive safety features of the reactor coupled with the lack of refuelling, nearly zero burn-up reactivity swing over core life and autonomous load following capability.
- Fuel self-sufficiency (FSS) - no need for uranium enrichment or for plutonium supply beyond the first fuel loading. Simplified recycling.
- No need for nuclear technology infrastructure and no need for central electricity transmission lines in owner country.
- Low investment risk due to relatively small unit cost, factory assembly line fabrication, short construction time of ~2 years and high level of safety. Leasing the ENHS power plant and paying for the electricity is an option.
- Close match between demand and supply by adding ENHS modules to a given power plant to match the growth in demand.
- A rough order of magnitude cost estimate has been completed [XX-32] based on factor assembly and rapid installation at pre-approved sites. The results indicate that the ENHS can be cost competitive with alternative forms of energy in the markets it is targeted to serve.

XX-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

Provisions for sustainability, waste management, and minimum adverse environmental impacts include the following:

- The ENHS reactor is FSS. That is, the in-core fissile fuel inventory is kept nearly constant over core life and then recycled as many times as one wishes. The only feed after the initial fuel loading is either depleted uranium, uranium extracted from fuel discharged from LWRs, or natural uranium.
- Fuel discharged from LWRs is to provide the initial fuel loading for the ENHS core. The LWR discharged fuel is converted to the ENHS feed fuel by removing most of the fission products and approximately 90% of the uranium. The entire inventory of plutonium and minor actinides to be accumulated in the USA until 2030 could be "stored" in the cores of ~400 ENHS modules.
- Only fission products and trace amounts of actinides need to be stored in a high-level waste repository.
- Early deployment of ENHS reactors could provide a reasonable solution to the spent LWR fuel that will enable to handle all the high level waste to be accumulated in the USA in a single high-level waste repository - Yucca Mountain.

XX-1.6.3. Safety and reliability

Safety concept and design philosophy

Design philosophy for the ENHS is as follows:

- Very simple and robust design to minimize accident-initiating possibilities.
- Minimum reactivity available for reactivity insertion accidents.

- Low pressure single-phase coolant that does not strongly react with water and air eliminates stored energy driven accidents.
- Low pressure system along with double walled pool vessel[(a)] and underground silo walls make loss of coolant accidents inconceivable.
- Natural circulation coupled with negative temperature reactivity coefficient and large heat capacity make loss of flow accidents inconceivable; enables decay heat removal using RVACS only; and enables autonomous control and high tolerance to human errors.
- High heat capacity and very high boiling temperature coolant make core-voiding accidents via coolant boiling inconceivable.
- No on-site fuelling or fuel shuffling eliminates fuel handling accidents.
- No need for exclusion zone beyond the plant boundary and no need for evacuation plans. The site boundary can be identified as low population zone per 10CFR100. Emergency planning will be limited to on-site responses and informing authorities of the plant status even following a severe accident.
- It is possible to experimentally demonstrate safety characteristics of the ENHS reactor.

Provisions for simplicity and robustness of the design

The provisions for design simplicity and robustness are as follows:

- There are no pumps or valves in the primary and secondary coolant systems. Loss of flow accidents are not conceivable. The possibility of flow instabilities at off-nominal operating conditions needs to be more thoroughly investigated.
- Reactor is very simple and robust. It has small number of components. There are no moving components except for the control (6 pieces) and safety (1 piece) element drives; these drives need to be actuated only seldom. The only special safety system is the Reactor Vessel Air Cooling System (RVACS); this system is passive.
- There is no fuelling hardware on site. The module is removed and shipped in a special cask with fuel frozen in the primary coolant.
- Reactor module is a standard design and factory fabricated and assembled with high quality control.

Inherent and passive safety features

The ENHS inherent and passive safety features are:

- Passive cooling via natural circulation of both primary and secondary coolants.
- The reactor is located in an underground silo and has a guard vessel/containment. Loss of primary or secondary coolants is inconceivable.
- Very high heat capacity that, along with the natural circulation, makes temperature changes due to accidents relatively slow and small.
- Very small excess reactivity available at any time once the reactor is at full power. Accidents involving large reactivity insertion are inconceivable.
- Autonomous load following capability due to natural circulation coupled with negative temperature feedback. Reactor is very tolerant to operators' errors.

[(a)] The guard vessel is considered as the second wall.

Active and passive systems

The ENHS safety design includes the following active and passive systems:

- The active systems include 1 safety element (at core centre) and 6 control elements; they are made of 6 segments of an annular cylinder at the core radial periphery.
- The passive safety grade decay heat removal system is the reactor vessel air cooling system (RVACS). The steam generators (immersed in the secondary coolant pool) provide another heat rejection path for decay heat removal.
- The guard vessel is a safety measure against accidental loss of secondary coolant. The limited volume of the underground silo in which the reactor pool is located provides another safety measure against loss of coolant accident.

Defence in depth structure

The defence in depth structure is as follows:

- Barriers to fission products include fuel matrix, fuel cladding, primary coolant, primary coolant boundary, secondary coolant, secondary coolant boundary and containment boundary. The ENHS module walls provide boundary between the primary and secondary cooling systems. The pool vessel walls and steam generators provide boundary between the secondary cooling system and reactor building.
- Redundancy in counteracting reactivity insertion accidents (only very small positive reactivity insertion is feasible): (1) central safety assembly; (2) 6 independent segments of peripheral absorber - one of these segments could scram the reactor; and (3) negative temperature coefficient of reactivity.
- Means for decay heat removal in case of an accident: (1) steam generators and (2) VACS. Silo walls also can provide a limited heat sink - via radiation from the guard vessel.

Design basis accidents

The design basis accidents in ENHS are defined as follows:

- Start-up accidents - due, for example, to withdrawal of the central or peripheral absorbers to their fully withdrawn position.
- Transient overpower accidents - due to positive reactivity insertion as a result of a breach in the cladding of a fuel rod that is followed by creation of void in the form of bubbles of gaseous fission products, due to a drastic reduction in the feedwater inlet temperature to the steam generators, or due to inadvertent reactor controller insertion of reactivity.
- Loss of heat sink (LOHS) accidents- due to an accident in the energy conversion system, such as a turbine trip, loss of feedwater, or a steam line break.
- Steam generator tube ruptures.

Beyond design basis accidents

The beyond design basis accidents in ENHS are defined as follows:

- Unprotected transient overpower (UTOP) accidents - a transient overpower accident with failure to scram.

- Unprotected loss of heat sink (ULOHS) - isolation of the balance of plant (BOP) with a failure to scram.
- Postulated core disruption from fuel failure propagation.
- Steam generator tube rupture with failure of pressure relief.

Accident analysis has shown that the deviation of the ENHS from nominal operating conditions is very slow and that all plant components do not reach their design limit temperatures. All the postulated events were mitigated by naturally occurring phenomena - negative reactivity feedback and natural circulation. The unprotected LOHS was found to be the most severe beyond design basis accident. Figure X-7 shows the evolution of power following a ULOHS. Figure X-8 below illustrates the consequence of a complete unprotected LOHS accident on the fuel, cladding and coolant temperatures.

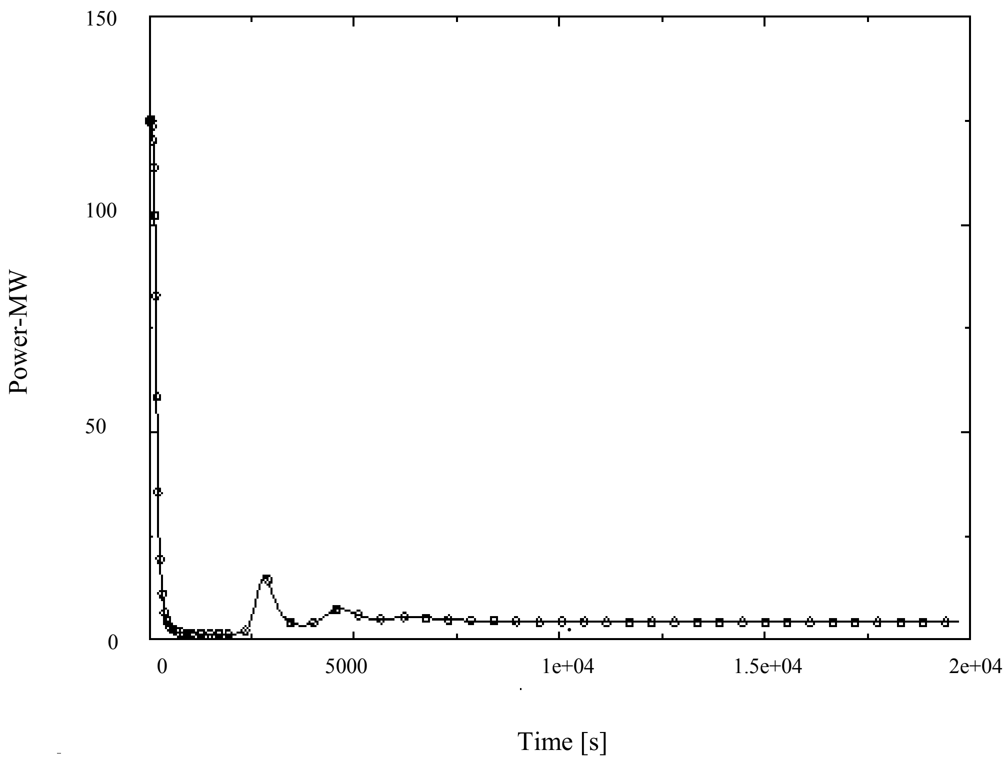


FIG. XX-7. Power evolution following an ULOHS accident.

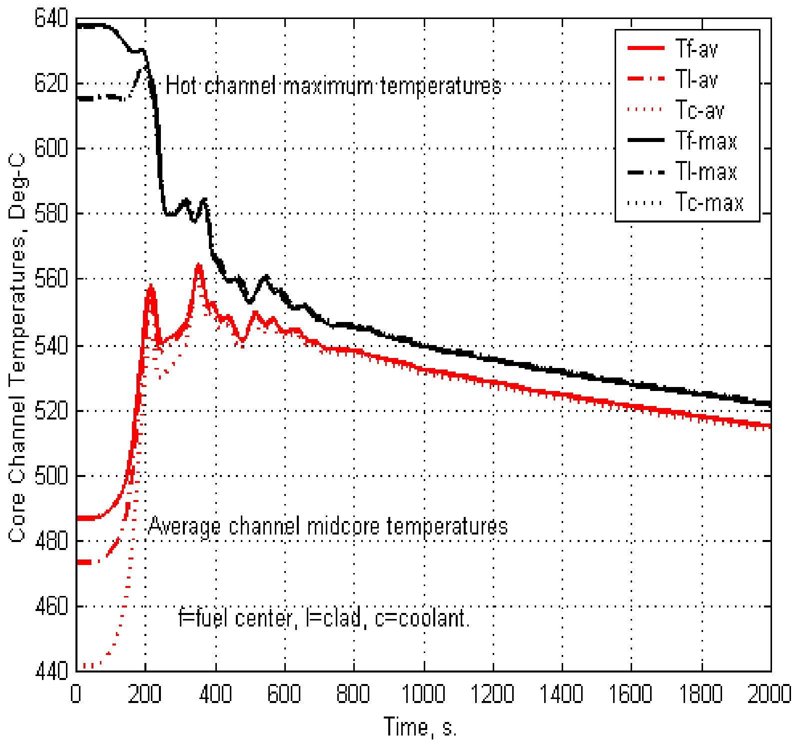


FIG. XX-8. Fuel, cladding and coolant temperature evolution following an ULOHS accident.

Initially the average core temperature gradually increases - a process that is very slow in the ENHS, due to its very large heat capacity. Due to the overall negative temperature feedback the reactivity drops and as a consequence the power starts to drop. As a result of the power drop the core average temperatures start to drop as well. At about 1500 seconds into the accident there is a re-criticality as a result of which there is a small increase in the power level. It takes few additional oscillations of lesser and lesser amplitude for the system to stabilize at a low power level.

XX-1.6.4. Proliferation resistance

The ENHS nuclear energy system is highly proliferation resistant due to a unique combination of technological and material barriers:

- Limited access to fuel - the fuel is loaded into a weld-sealed ENHS module in the factory and is shipped imbedded in solidified Pb-Bi. There is no fuel handling and fuel handling hardware in the power plant. The fuel bearing ENHS module is impossible to divert without notice.
- Limited access to neutrons - there are no blanket elements and there is no technical way to insert fuel or target materials for irradiation in or near the core. Moreover, the small excess reactivity built into the core prevents undeclared production of weapons-usable materials.
- The fuel always contains FPs and thus has a radiation barrier. As the fuel is to be shipped to the site embedded in Pb-Bi, it is possible to seed it with additional radioactive isotopes so as to increase the intensity of the radiation barrier without

interfering with shipment and installation. The minor actinides fraction keeps increasing with irradiation.

- For a given power generating capacity with ENHS reactors, the overall inventory of plutonium and other fissile isotopes is stable. Most of this inventory resides inside the core.
- Installing and operating ENHS reactors will not require the country to obtain sensitive technologies that could be used for clandestine production of strategic nuclear materials. Specifically, no fuel fabrication or handling facilities and no fuel reprocessing capability are needed in the client country. Yet, the ENHS reactors will provide energy security for relatively long time periods exceeding 20 years - the lifetime of the ENHS module.
- Proliferation-resistant multi-recycling without partitioning of actinides and without using uranium enrichment services. This could improve the proliferation resistance of the fuel cycle in industrial countries and in regional fuel cycle centres.

XX-1.6.5. Technical features and technological approaches used to facilitate physical protection of ENHS

The features facilitating physical protection of the ENHS are the following:

- The fuel is fabricated and installed in a sealed module in a secure facility.
- The new fuel is frozen in the lead-bismuth coolant inside the sealed reactor module during shipping to the operating site. This package weights several hundreds of tons.
- The weld-sealed ENHS module is located inside an underground silo in a pool of lead-bismuth.
- There is no fuel handling hardware on site.
- There are no spent fuel storage pools; the spent fuel is contained in the weld sealed reactor module and is embedded in solidified lead. This sealed module is made into a shipping cask.
- It is preferable to have the recycling facility collocated with the module manufacturing facility, but if this is not possible, the shipping of radioactive recycled fuel to the module manufacturer will be done in a multi-ton cask that facilitates security.

XX-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of ENHS

Non-technical factors that could facilitate development and deployment of the ENHS are the following:

- Identification of financial resources to support development of the concept.
- Identification of developing countries that will express interest in ENHS-type reactors, will define user requirements for such reactors, and will provide an estimate for the market size of these reactors. It may eventually be necessary to get a commitment to purchase a large enough number of reactor modules to justify the investment in special fabrication plants of ENHS modules.
- Establishment of international nuclear fuel centres that will be able to handle the used ENHS modules, remove the used fuel, and process the fuel to extract fission products. Such fuel centres could also add makeup of depleted uranium or of uranium from LWR discharged fuel and refabricate fuel rods for loading into new ENHS modules.

- Establishment of institutional agreements that will enable deploying the ENHS reactors in a proliferation resistant manner while providing the host country an increased energy security. Among the issues to be resolved are who will have ownership on the fuel and where the HLW will be disposed.
- Construction of a demonstration plant that will be used to demonstrate to the public the unique safety attributes of this reactor system.
- International collaboration between industrial countries on the development of ENHS reactors and their design.

XX-1.8. List of enabling technologies relevant to ENHS and status of their development

The enabling technologies for the ENHS are as follows:

- Structural materials that could maintain their integrity for over 20 years while in contact with lead or lead-bismuth coolants in the temperature range between 400 and 550°C need be verified or developed. A related issue is that of the chemistry of lead and lead-bismuth coolants, so as to minimize their ability to corrode structural materials or to dissolve certain constituents and transport them from hot surfaces to cold surfaces. As part of the technology developed for its "Alpha" class nuclear submarines, the Russian Federation has developed ferritic-martensitic steels [XX-37] and coolant chemistry control that make this steel compatible with lead-bismuth coolant. There is a need to acquire the knowledge or to develop it independently. A number of research programmes aimed at the identification of suitable structural materials are ongoing in the USA, Germany, Japan and other countries. Alternative structural materials need be considered as well.
- Maintainability of the welded sealed ENHS module over 20 to 30 years of module life needs to be established. Alternately it is necessary to demonstrate that maintenance is unnecessary. The later is the preferred approach and may be possible if the corrosion and mass transferred are demonstrated to be inconsequential.
- There is a need to establish the feasibility of solidifying the Pb-Bi or Pb in a reasonable time after removing the ENHS module from the pool.
- Likewise, there is a need to establish the feasibility of converting the ENHS module into a licensed shipping package for fresh and, primarily, spent fuel.
- There is a need to establish the economic viability of ENHS reactors. This activity is to be coupled with the development of more optimal designs of the ENHS reactor with greater design detail for the subsystems.
- Demonstrate the robust safety performance that will enable the autonomous control and minimum staffing necessary to realize the economy.

In addition to the above mentioned enabling technologies, there are a number of enabling infrastructure development needs including the following:

- Establish the logistics and infrastructure required for the fabrication, fuelling, transporting, installing and operating the ENHS modules.
- Likewise for removing the used ENHS module, converting it to a shipping package, and transporting it to the processing plant.
- License the ENHS reactor design and the shipping package.
- Work out institutional arrangements for financing ENHS power plants for developing countries, for the ownership of the fuel and for the disposition of the nuclear waste.

- It is desirable to construct an ENHS demonstration plant that could be used to demonstrate to the public the unique safety of this reactor concept by actually subjecting the demonstration unit to accidents.

XX-1.9. Status of R&D and planned schedule

The status of R&D for the ENHS is as follows:

- The Lawrence Livermore National Laboratory, Argonne National Laboratory and Los Alamos National Laboratory as well as the University of California at Berkeley are collaborating on the R&D of small, lead alloy cooled battery type reactors. R&D that is specific to the ENHS reactor is being carried out at the University of California at Berkeley and LLNL supported, at a low level, by the US DOE Generation IV programme as part of the work done on lead alloy cooled nuclear battery type fast reactors. The ENHS R&D is also partially supported by the Lawrence Livermore National Laboratory.
- ENHS related R&D is being carried out at the Argonne National Laboratory in the USA and at CRIEPI and Toshiba in Japan. The CRIEPI and Toshiba effort is focused on the 4S reactor concept important elements of which were adopted for the ENHS reactor concept. There is a close collaboration between CRIEPI and the Lawrence Livermore National Laboratory on R&D of small nuclear battery type reactors.
- The ENHS reactor R&D work at the University of California at Berkeley is being partially supported by the Korean Atomic Energy Research Institute.
- The Generation IV planning would place the deployment of an ENHS in the post 2025 time period. If increased funding were available for R&D and licensing, it is estimated that the reactor could be available for deployment in 2020.

XX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

A full-scale demonstration of the ENHS reactor is recommended for the following reasons:

- Demonstrating the superb safety characteristics of the ENHS reactor by subjecting the demonstration reactor to design-basis accidents as well as to off-design basis accidents and illustrating that no damage occurs. A set of such experiments is expected to convince the public in the safety of ENHS reactors that is necessary for public acceptance.
- It is also desirable to verify the autonomous load following capability of the ENHS reactor.

XX-1.11. List of other similar or relevant SMRs for which the design activities are on going

The SMR developments of relevance to the ENHS are:

- The STAR-LM and STAR-H2 R&D activities at the Argonne National Laboratory.
- The 4S reactor R&D at CRIEPI and Toshiba of Japan.
- The SVBR-75/100 in Russian Federation.

XX-2. Design description and data for ENHS

XX-2.1. Description of the nuclear systems

Reactor core and fuel design

The ENHS core is made of uniform composition fuel rods with no blanket assemblies and designed to maintain a nearly constant K_{eff} over 20 years without refuelling or fuel shuffling. The reference fuel is taken to be metallic alloy with 10 weight % of Zr. Both enriched uranium and Pu-U are considered for the heavy metals. The structural material is taken to be HT-9; the ferritic-martensitic stainless steel used for EBR-II. The ENHS module will be manufactured and fuelled in the factory and shipped to the site as a weld-sealed unit with solidified Pb-Bi filling the vessel up to above the fuel rods. A unique feature of Pb-Bi that makes it possible to embed the fuel rods and core structure in solid Pb-Bi without damage is its nearly zero coefficient of volumetric expansion upon phase change [XX-38]. At the end of its core life the module will be removed from the reactor pool after pumping-out Pb-Bi from above the fuel level and Pb will replace the remainder. The module will be stored on site until the decay heat drops to a level that will permit to solidify the lead and to convert the module into a shipping cask.

Figure XX-9 shows the radial location of the control elements relative to the core, while their axial location at full power is shown in Fig. XX-2. The active element for both central and peripheral absorbers is B_4C and tungsten; being heavier than Pb-Bi tungsten can scram by gravity.

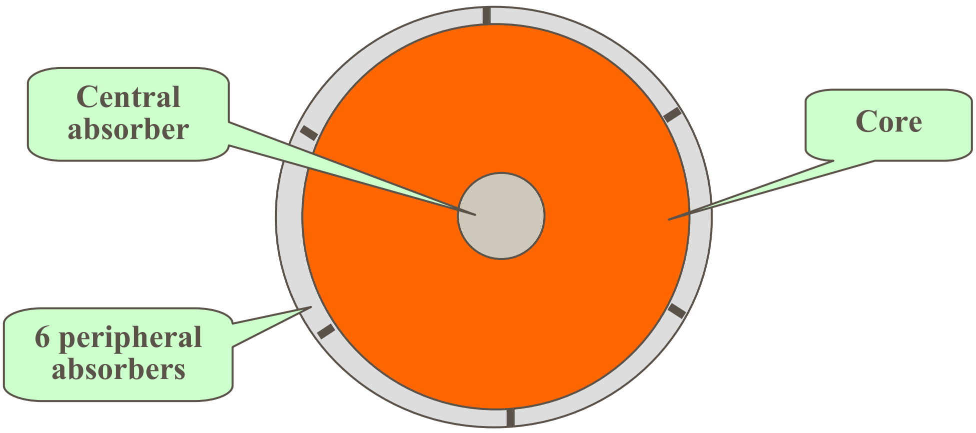


FIG. XX-9. A schematic horizontal view of the ENHS core and control elements. Not to scale.

The control elements are made of six segments of an annulus that surrounds the core. The drive mechanism for these control elements is supported by the inner structural wall of the ENHS module, thus minimizing the structure in the module and minimizing obstruction to the natural circulation flow.

Figure XX-10 shows a vertical cut through the reference ENHS module (the module) while Fig. XX-11 shows an alternative ENHS module design that uses a lift-pump integrated within the riser. Cover gas from the top of the module is injected into the coolant just above the core. The cover gas bubbles reduce the effective density of coolant in the riser, thus increasing the head for coolant circulation. The circulator would be located above the reactor pool, outside of the module vessel. A lift-pump for the secondary coolant is integrated within the module above the IHX. Table XX-8 summarizes selected design and performance parameters of the two reference designs.

TABLE XX-8. SELECTED CHARACTERISTICS OF ENHS REFERENCE DESIGNS

DESIGN PARAMETER	ENHS1	ENHS2
Primary Pb-Bi coolant circulation	100% natural	With lift-pump
Fuel material	U; 12 weight % Pu; 10 weight % Zr	
Structural material	HT-9	
Fuel length (m)	1.25	
Fission gas plenum length (m)	1.25	
Core diameter (m)	2.24	
Fuel rod outer diameter (cm)	1.56	
Cladding thickness (cm)	0.13	
Lattice (hexagonal) pitch (cm)	2.12	
Overall module height (m)	19.6	10.1
Outer module diameter (m)	3.52	3.72
Number of rectangular channels in IHX	135	435
Inner dimensions of channel (cm × cm)	39.2×2.5	49.2×0.5
IHX channel length (m)	11	3
Weight of fuelled module for shipment (t)	~360	~300
Coolant inlet/peak coolant outlet/peak cladding/peak fuel temperature (°C)	400/547/579/622[a]	400/502/531/576[a]
Primary/intermediate coolant outlet temperature (°C)	503/471[a]	471/443[a]
Attainable thermal power (MW(th))	125	190
Net energy conversion efficiency (%)	38	38

[a] At 125 MW(th)

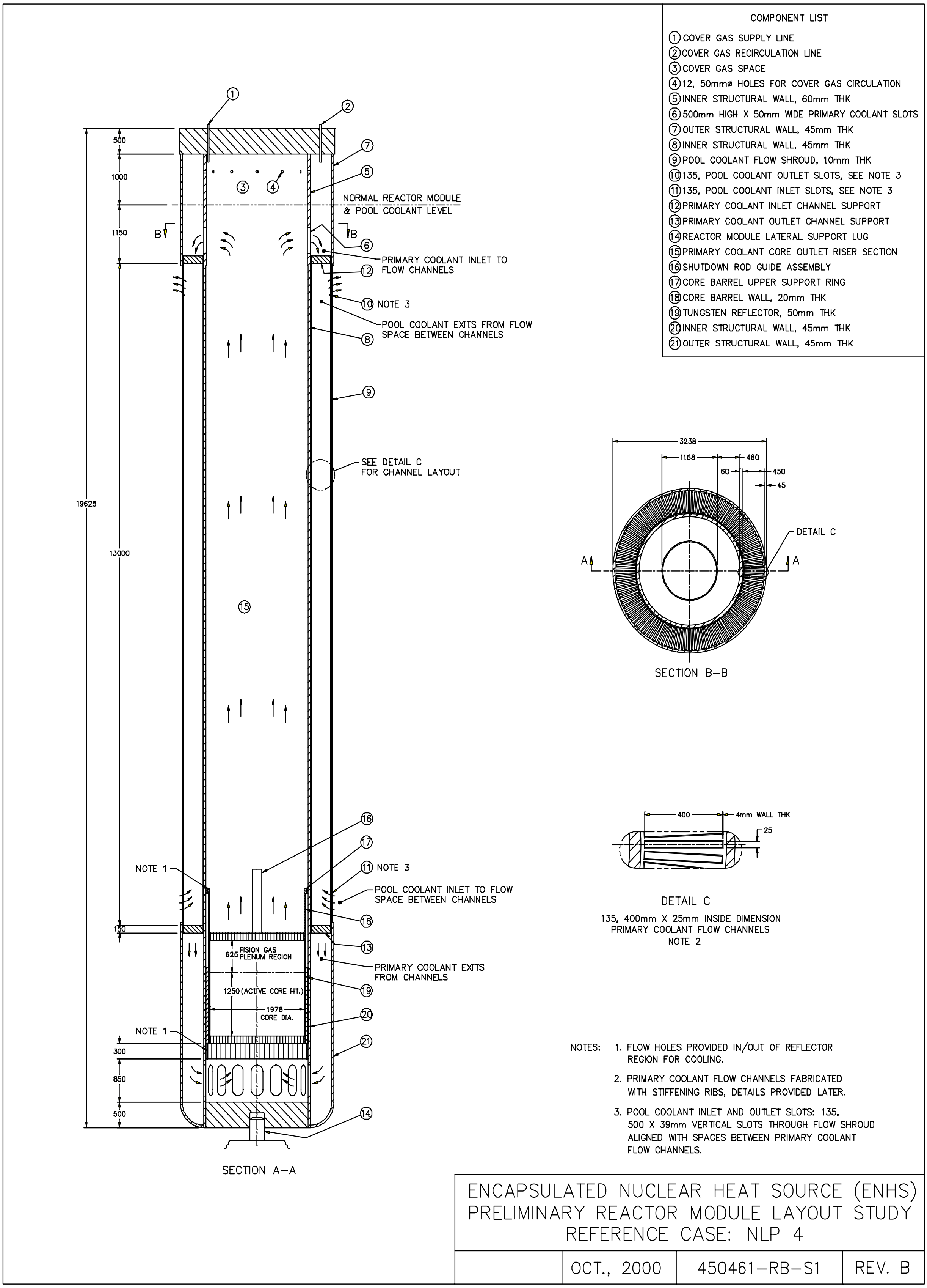


FIG. XX-10. Reference ENHS module featuring 100% natural circulation, ENHS1.

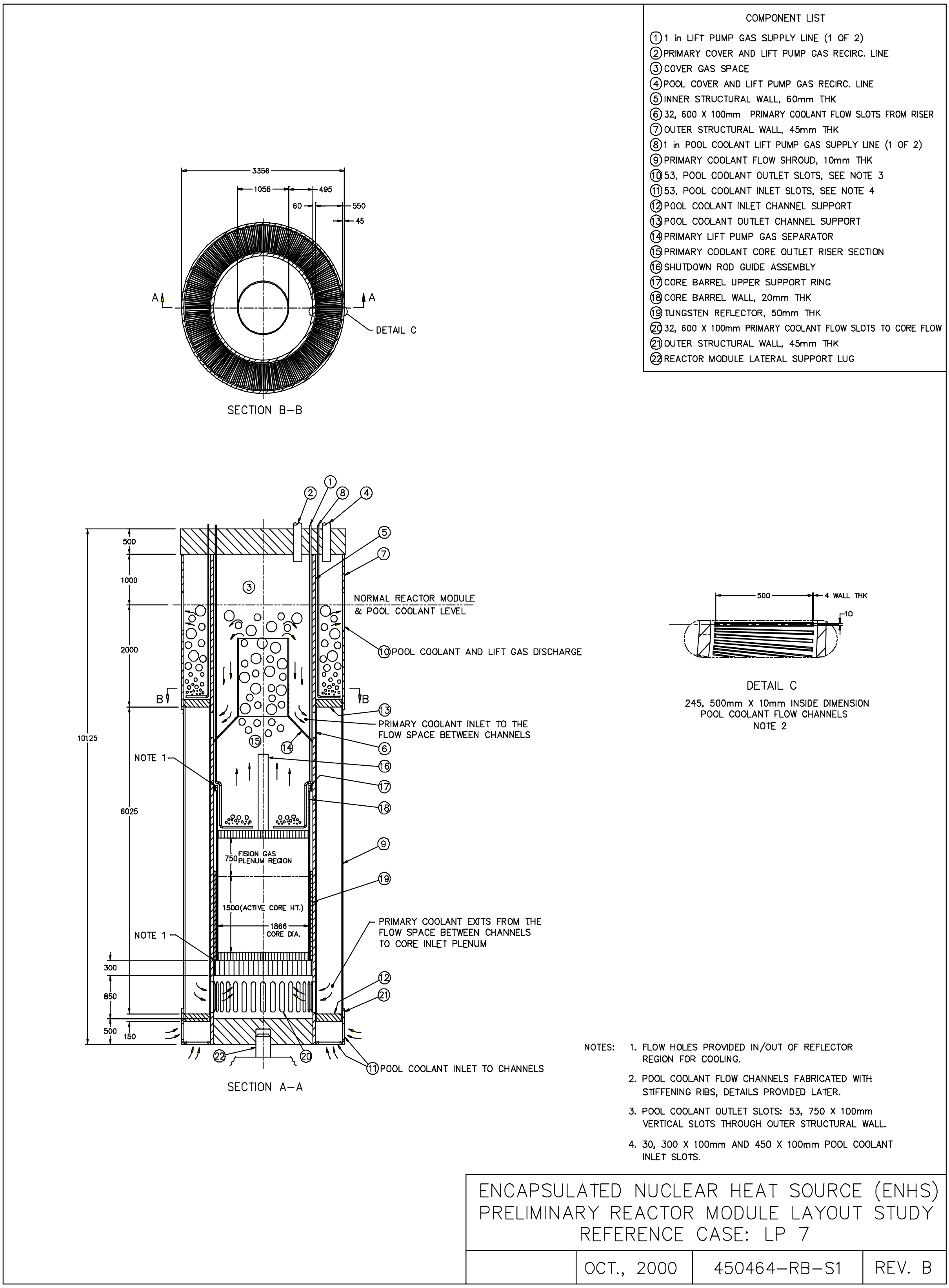


FIG. XX-11. Reference ENHS module featuring cover-gas lift pump, ENHS2. There are no pumps or valves inside the module, but there are gas circulators outside (on top of) the module.

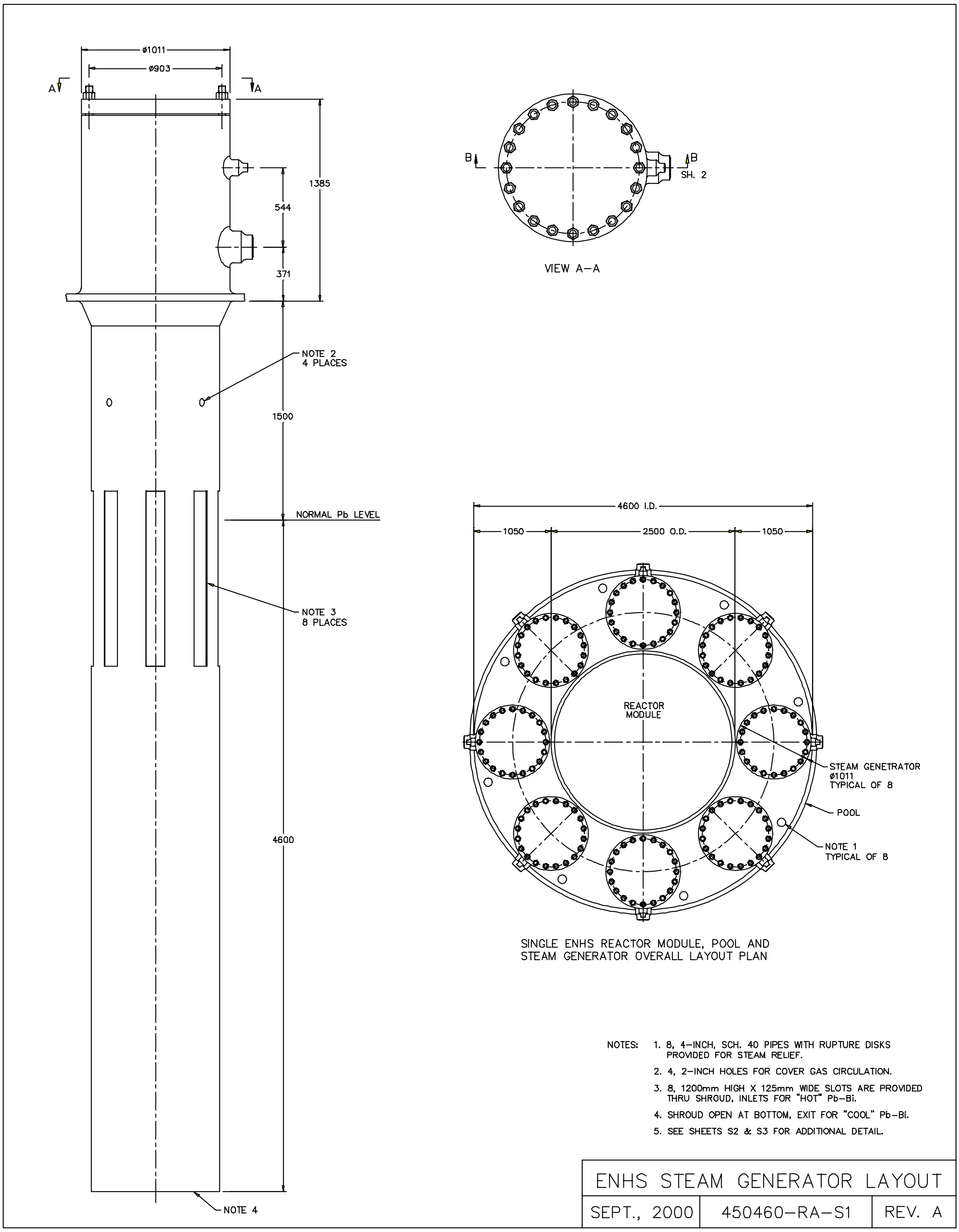


FIG. XX-12. ENHS steam generator overall elevation view and ENHS module layout plan.

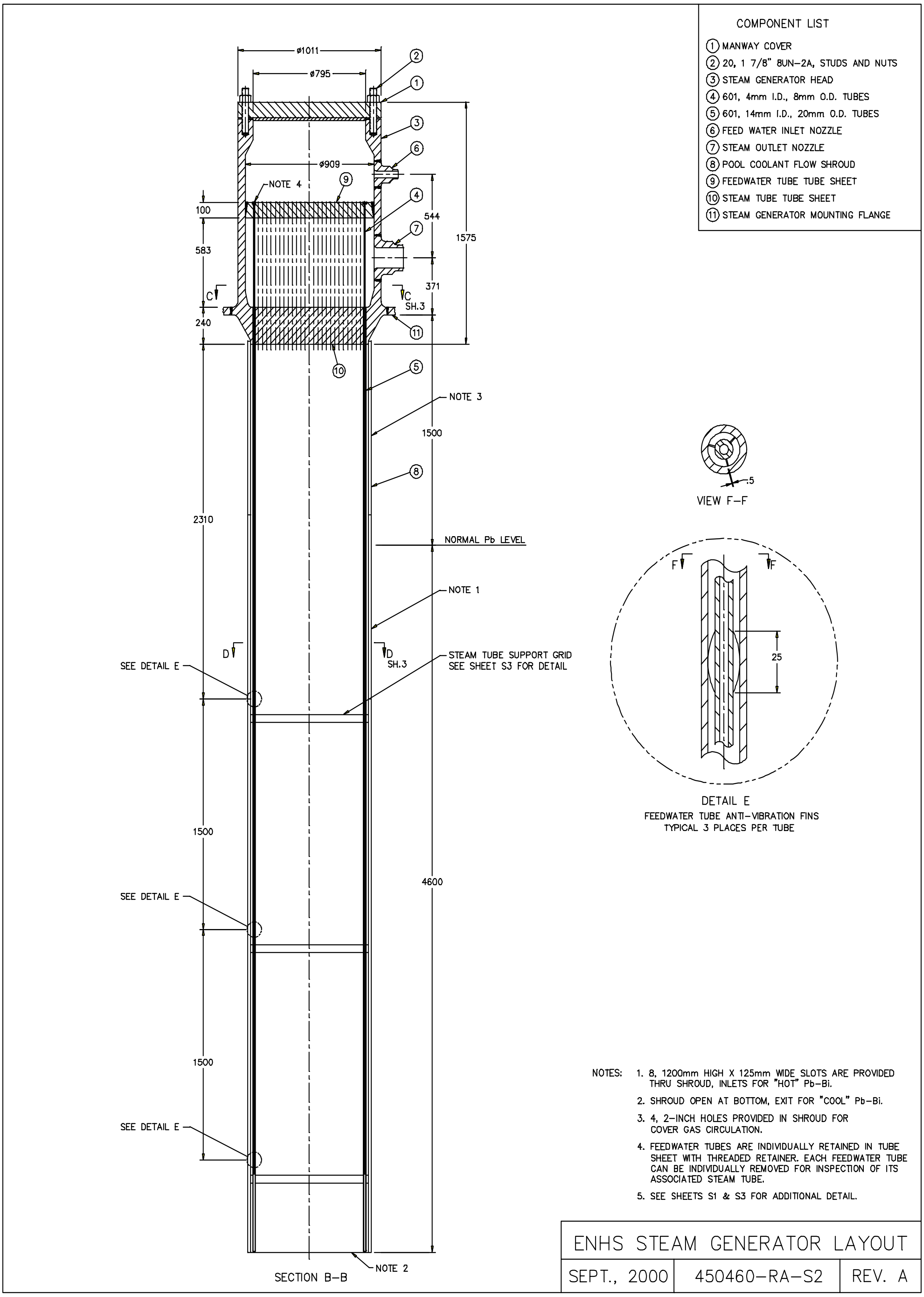


FIG. XX-13. ENHS steam generator cross-section elevation view.

The eight steam generators (SGs) shown in Figures XX-12 and XX-13 are anchored to the support structure that covers the pool and are not mechanically connected to the module. They are designed to meet several unique requirements that are dictated by the ENHS reactor layout: (1) effective utilization of the pool volume surrounding the module; (2) minimum friction losses so as to enable 100% natural circulation of the intermediate coolant; (3) having no mechanical connection with the module; (4) minimum flow rate of water into the intermediate coolant pool in case of a breach in steam generator tube or failure of other water-containing component; (5) accommodation of a large thermal expansion; (6) ease of inspection and maintenance; (7) modular design that is easy to install and replace.

The steam generator is of a once-through tube-in-tube design; feedwater flows in via the inner tube and the steam is generated in the shell between the inner and outer tubes. The liquid metal (either Pb-Bi or Pb) coolant flows outside the tubes. Also, the steam and feedwater piping and nozzles are located outside the ENHS pool, and the feedwater to each steam tube is inherently orificed by a small diameter feed tube. These features all act to minimize the quantity and/ or mass flow rate of water or steam that can be introduced into the pool due to a postulated steam line break, or feed line break, or tube rupture.

A flow partition is extending from below the exit from the steam generators down to close to the bottom of the intermediate coolant pool (not shown in Fig. XX-1). Its function is to force the coolant to flow through the bottom of the pool so as to eliminate stagnation of Pb-Bi.

Main heat transport system

Figure XX-14 defines the heat removal path under nominal operating conditions, at hot low power conditions and during the worst accident that, for the ENHS reactor, is a loss of heat sink accident.

XX-2.1. Description of the turbine generator plant and systems

Although the reference design is for Rankine cycle using water as the working fluid, super-critical CO_2 cycle is a design option. Based on studies done at MIT [XX-39] and ANL [XX-40], the super-critical CO_2 cycle will be the preferred option, provided the predicted performance characteristics and cost will be experimentally demonstrated; it offers higher efficiency and is significantly more compact. Based on the ANL analysis [XX-40], the efficiency of the ENHS reactor using a super-critical CO_2 cycle is estimated to exceed 40%.

XX-2.3. Systems for non-electric applications

No specific design for the ENHS reactor non-electric applications has been worked out, as yet.

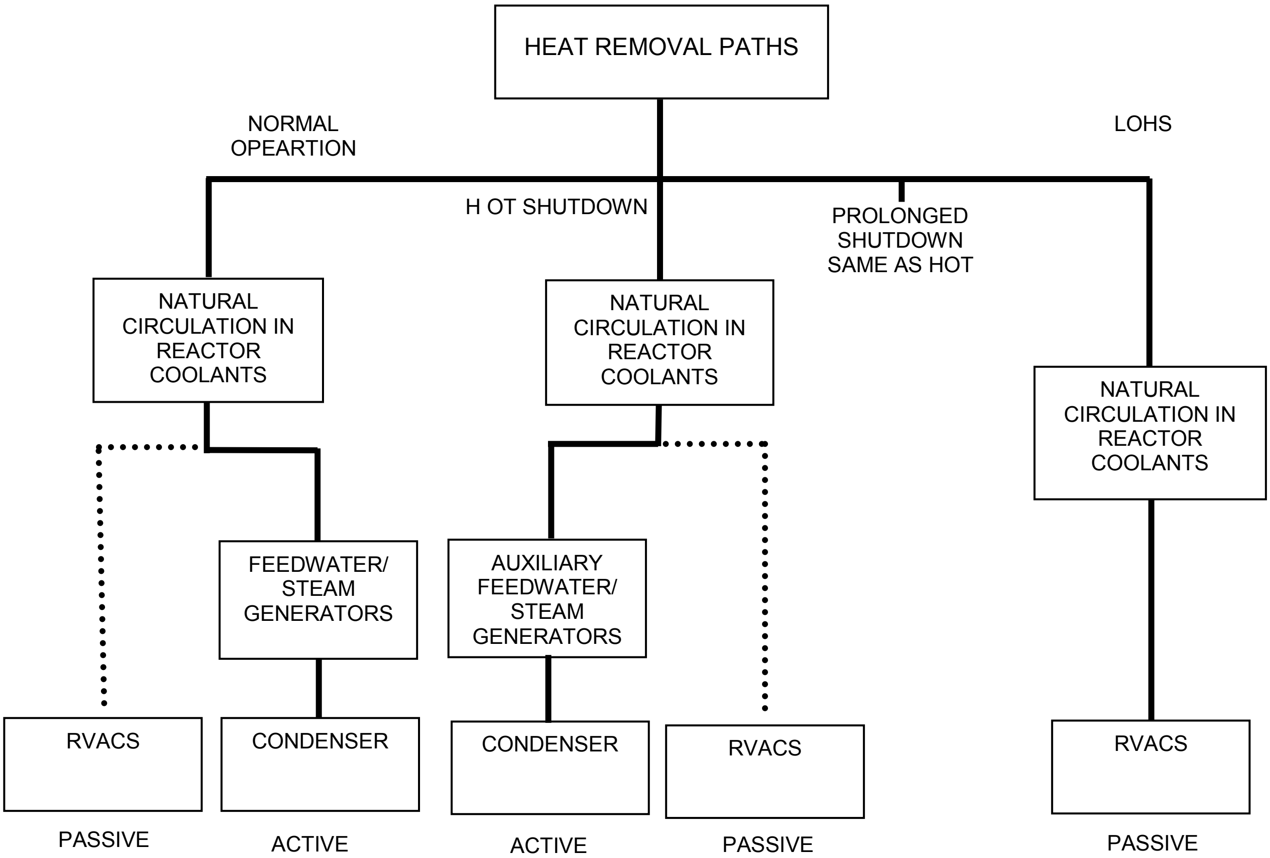


FIG. XX-14. Heat removal paths under different operating conditions.

XX-2.4. Plant layout

A single ENHS reactor consists of nine factory-fabricated modules - one ENHS and eight steam generators. All these modules are transported to the power plant site completely assembled ready to be inserted into the secondary coolant (lead-bismuth eutectic) pool. These modules are supported from the top by a seismically isolated structural platform. There is no mechanical connection between the modules. This makes it easier to install, inspect and replace them in an existing plant. The reactor pool is located inside an underground silo. The pool vessel with the secondary coolant are supported either from the top on the seismically isolated structural platform, or from the bottom. The preferred design choice is yet to be made.

It is possible to construct a power plant made of multiple ENHS reactors. There are a couple of general approaches to the design of a multiple ENHS module plant: (a) install several ENHS modules in a single pool of secondary coolant; (b) use as many independent single module ENHS reactors as desirable in a single power plant.

Approach (a) is illustrated in Fig. XX-15. The secondary coolant pool can have a rectangular cross section and is made of thermally insulated concrete. Such a pool design concept is being proposed for a single core, higher power, BREST reactor [XX-41]. Steam generators, super-heaters, possibly reheaters and possibly secondary coolant pumps are also inserted into the secondary coolant pool. Each of these components is a relatively small module, supported from a seismically isolated platform that covers the pool. There is no mechanical connection between the modules, so it is relatively simple to install and replace these components. In case of some failure in one of the modules, it will be probably most economical to replace it with a new module. If economically justified, the failed module could be fixed "off-line" and used as a spare. The secondary reactor pool of Fig. XX-11 has a BREST-like RVACS system for decay heat removal [XX-42].

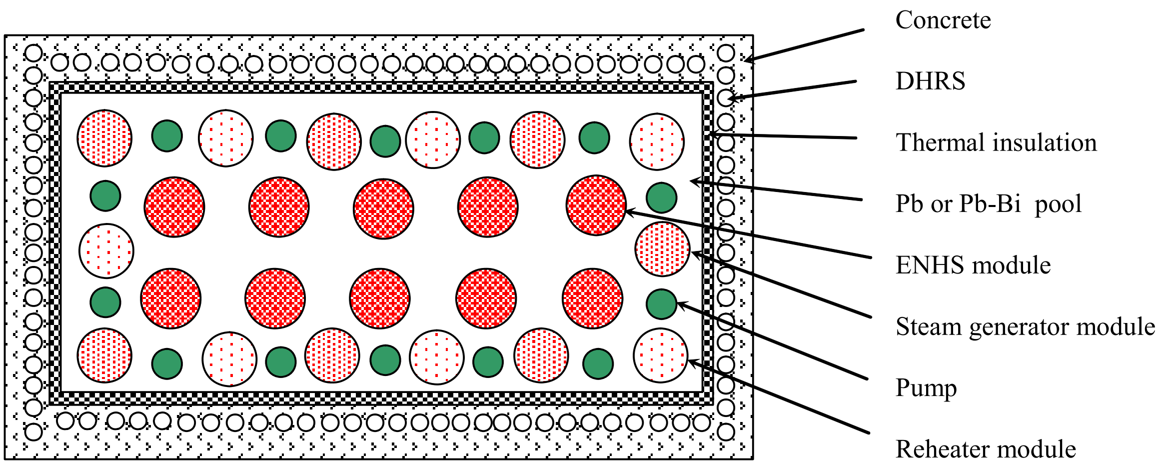
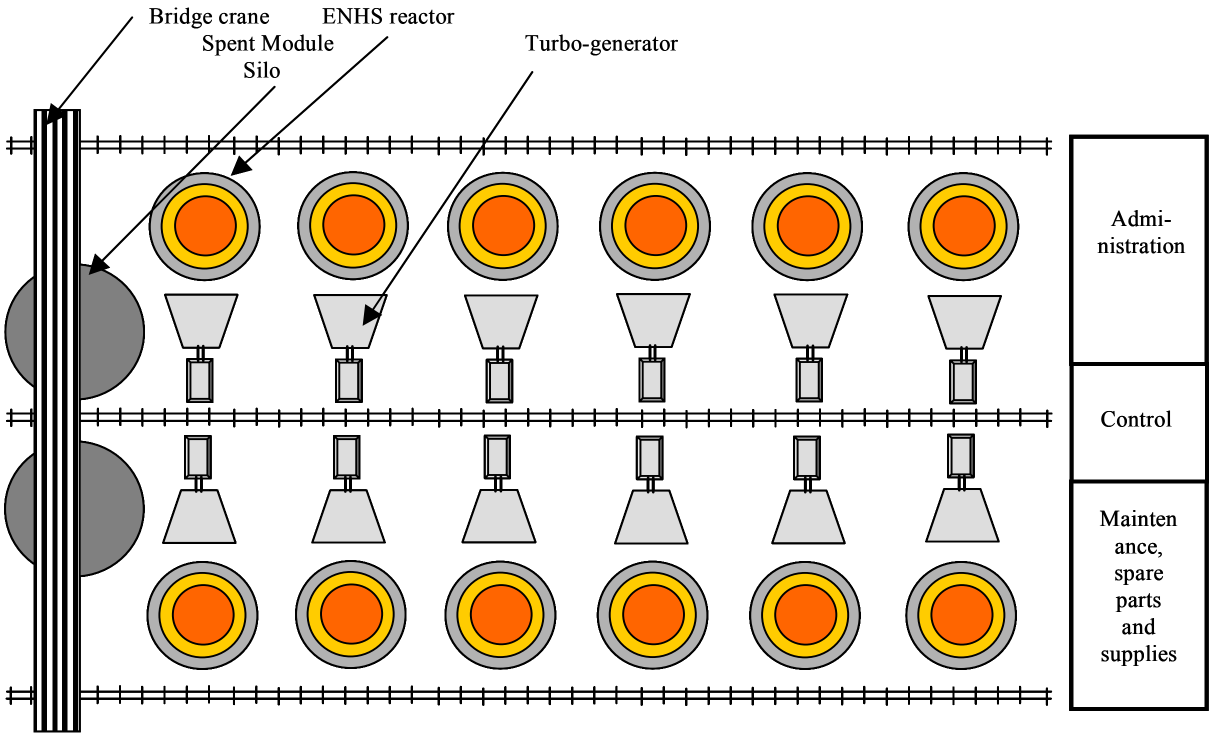


FIG. XX-15. A schematic horizontal view of a 10-ENHS module plant using insulated concrete massive structure for the secondary coolant pool. Not to scale.

Approach (b), the preferred approach at the moment, is illustrated in Fig. XX-16. The power plant consists of 12 ENHS reactors, each including its own reactor pool, steam generators and turbine-generator. The total capacity of this power plant is 600 MW(e). This arrangement provides the utmost level of uniformity and modularity. Additionally, this arrangement is most suitable for a gradual increase in the installed capacity of the power plant so as to best fit the increase in demand for electricity.



Each reactor can be operated as a stand-alone unit. There are common control, services, etc. Not to scale.

FIG. XX-16. A schematic horizontal view of a 12-ENHS reactor power plant for 600 MW(e).

REFERENCES

[XX-1] GREENSPAN, E., WADE, D.C., Encapsulated nuclear reactor heat source module, Transactions American Nuclear Society, 80, 197–99 (1999).

[XX-2] GREENSPAN, E, SHIMADA, WANG, H., K., Long-life cores with small burnup reactivity swing, PHYSOR-2000 (Paper presented at the 2000 Int. Topical Mtg. On Advances in Reactor Physics and Mathematics and Computation into the Next Millennium, Pittsburgh, PA, May 7–11, 2000).

[XX-3] CONWAY, L., et al., Promising design options for the encapsulated nuclear heat source reactor, ICONE-9 (Proc. 9th Int. Conf. on Nuclear Engineering, Nice, France, April 8–12, 2001), Paper ICONE9-417, ASME.

[XX-4] SIENICKI, J.J., WADE, D.C., Thermal hydraulic analysis of the encapsulated nuclear heat source, ICONE-9 (Proc. 9th Int. Conf. on Nuclear Engineering, Nice, France, April 8–12, 2001), ICONE9-113, ASME.

[XX-5] SAPHIER, D., et al., Some safety aspects of the encapsulated nuclear heat source LMR plant, ICONE-9 (Proc. 9th Int. Conf. on Nuclear Engineering, Nice, France, April 8–12, 2001), ICONE9-22, ASME.

[XX-6] SAPHIER, D., DZODZO, M., GREENSPAN, E., Applying the DSNP modular modelling system to transient and accident simulations of lead cooled reactors, ICONE-9 (Proc. 9th Int. Conf. on Nuclear Engineering, Nice, France, April 8–12, 2001), ICONE9-661, ASME.

[XX-7] GREENSPAN, E., et al., The encapsulated nuclear heat source reactor - a generation IV reactor, GLOBAL-2001 (Paper presented at Int. Conf., Paris, France, September 2001), ANS/ENS.

[XX-8] GREENSPAN, E., et al., The encapsulated nuclear heat source reactor for proliferation-resistant nuclear energy, GLOBAL-2001 (Proc. Int. Conf., Paris, France, September 2001), ANS/ENS.

[XX-9] BARAK, A., GREENSPAN, E., ENHS reactor energy-conversion efficiency and steam generators size (Trans. American Nuclear Society, vol. 85, 77–78, Reno, NV, Nov. 2001).

[XX-10] WADE, D.C., et al., ENHS: The encapsulated nuclear heat source – A nuclear energy concept for emerging worldwide energy markets, ICONE-10 (Proc. 10th Int. Conf. on Nuclear Engineering, Arlington, VA, April 14–18, 2002), ICONE10-22202, ASME.

[XX-11] GREENSPAN, E., et al., The long-life core encapsulated nuclear heat source Generation IV reactor, ICAPP'02 (Proc. of the Int. Congress on Advanced Nuclear Power Plant, Hollywood, FL, June 9–13, 2002), paper 1081 (2002).

[XX-12] BUONGIORNO, J., BARAK, A., GREENSPAN, E., Irradiation performance of the metal fuel for the encapsulated nuclear heat source, Trans. American Nuclear Society, 86, 348–349 (Hollywood, FL, June 2002).

[XX-13] HONG, S.G., GREENSPAN, E., KIM, Y.I., Once-for-life core for the encapsulated nuclear heat source (ENHS) reactor, PHYSOR-2002 (Proc. Int. Conf. on the New Frontiers of Nuclear Technology: Reactor Physics, Safety and High-Performance Computing, Seoul, Korea, October 7–10, 2002), paper 10B-04 (2002).

[XX-14] SIENICKI, J.J., Updated thermal hydraulic analyses for recent ENHS design improvements, ICAPP'03 (Proc. Int. Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4–7, 2003), paper 3115 (2003).

[XX-15] HONG, S.G., GREENSPAN, E., KIM, Y.I., Alternative design options for the Encapsulated Nuclear Heat Source Reactor (ENHS), ICAPP'03 (Proc. Int. Conf. on Advances in Nuclear Power Plants, Cordoba, Spain, May 4–7, 2003) paper 3248 (2003).

[XX-16] SHAYER, Z., MILOSEVIĆ, M., BARNES, D., GREENSPAN, E., Shielding optimization for the ENHS reactor, ICAPP'03 (Proc. Int. Conf. on Advances in Nuclear Power Plants, Cordoba, Spain, May 4–7, 2003), paper 3220 (2003).

[XX-17] SAPHIER, D., Safety assessment of advanced nuclear power plants with the DSNP modular modelling systems, ICAPP'03 (Proc. Int. Con. on Advances in Nuclear Power Plants, Cordoba, Spain, May 4–7, 2003), paper 3217 (2003).

[XX-18] HONG, S.G., GREENSPAN, E., KIM, Y.I., KIM, H.R., Encapsulated nuclear heat source (ENHS) core designs using nitride fuel, GLOBAL'03 (Proc. of New Orleans LA, Nov. 16–20, 2003), paper 88255 (2004) ANS/ENS.

[XX-19] HONG, S.G., GREENSPAN, E., KIM, Y.I., The encapsulated nuclear heat source (ENHS) reactor core design, Nuclear Technology, Vol. 149, 1, paper 22–44 (January 2005).

[XX-20] GREENSPAN, E., et al., The encapsulated nuclear heat-source (ENHS) reactor, Summary Report of 1st year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4232 (February 2001).

[XX-21] GREENSPAN, E., The encapsulated nuclear heat-source (ENHS) reactor, Summary Report of 2nd year of NERI Project No. 990154, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4234 (December 2001).

[XX-22] GREENSPAN, E., Editor, STAR: The secure transportable autonomous reactor system encapsulated fission heat source (The ENHS Reactor) - final report, (University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4241 (October 2003).

[XX-23] BROWN, N.W., ELIAS, E., GREENSPAN, E., HASSBERGER, J.A., Proliferation resistant fission energy systems, GLOBAL'97, (Paper presented at Int. Conf. on Future Nuclear Systems, Yokohama Japan, October 5–10, 1997).

[XX-24] GREENSPAN, E., ELIAS, E., Proliferation resistant fission energy systems, GLOBAL'97, (Int. Conf. on Future Nuclear Systems, Yokohama, Japan, Oct. 5–7, 1997), paper 2, p. 879–884.

[XX-25] GREENSPAN, E., Compact once-for-life fuelled reactors for developing countries, Trans. American Nuclear Society, 78, 239v41 (1998).

[XX-26] GREENSPAN, E., ELIAS, E., KASTENBERG, W.E., STONE, N., Non-proliferation liquid metal cooled reactors for developing countries, Center for Nuclear and Toxic Waste Management, University of California, CNTWM 98-2 (Feb. 15, 1998).

[XX-27] HATTORI, S., HANDA, N., Use of super-safe, small and simple LMRs to create green belts in desertification area, Transaction Amer. Nucl. Soc., 60, 437 (1989).

[XX-28] HATTORI, S., MINATO, A., Current status of 4S plant design, ICONE-2 (Paper presented at 2nd Int. Conf. on Nuclear Engineering, California, Mar. 21–24, 1993) ASME-JSME.

[XX-29] HATTORI, S., MINATO, A., A large modular LMR power station which meets current requirements, ICONE-3, (Paper presented at. 3rd Int. Conf. on Nucl. Eng., Tokyo, Oct. 1993) ASME-JSME.

[XX-30] BROWN, N.W., et al., The secure, transportable, autonomous reactor system, GLOBAL'99, (Paper presented at Int. Conf. on Future Nuclear Systems, Jackson Hole, Wy, USA, Aug. 29–Sept.3, 1999), ANS/ENS.

[XX-31] STEWART, J.S., et al., An economic analysis of generation IV small modular reactors, Lawrence Livermore National Laboratory Report UCRL-ID-148437 (March 1, 2002).

[XX-32] BROWN, N.W., et al., Liquid metal cooled reactors and fuel cycles for international security, ICONE-11 (Proc. Int. Conf. on Nuclear Engineering, Tokyo, Japan, April 20–23, 2003) ICONE11-36339, ASME.

[XX-33] GREENSPAN, E., Generation-IV reactors and nuclear waste minimization, (Proc. of the 14th Pacific Basin Nuclear Conference, pp 032, Honolulu, Hawaii, March 21–25, 2004).

[XX-34] FEINROTH, H., GUON, J., MAJUMDAR, D., An overview of the AIROX process and its potential for nuclear fuel recycle, (Paper presented at the Global Int. Fuel Cycle Conf., Seattle, 1993).

[XX-35] LEE, J.S., et al., Research and development program of KAERI for DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors), conceptual study on the DUPIC fuel manufacturing technology, GLOBAL'93, (Proc. Int. Conf. on future energy systems: Emerging nuclear fuel cycles and waste disposal options, Seattle, Washington, Sept. 1993), ANS/ENS.

[XX-36] SHIMADA, H., GREENSPAN, E., STONE, N., WANG, S., Long-life lead-cooled core fuelled with enriched uranium, Transactions Am. Nucl. Soc., 80, 199–200 (1999).

[XX-37] GROMOV, B.F., et al., Use of lead-bismuth coolant in nuclear reactors and accelerator-driven systems, Nuclear Eng. and Design, 173, 207–217 (1997).

[XX-38] GROMOV, B.F., et al., Use of lead-bismuth coolant in nuclear reactors and accelerator-driven systems, Nuclear Eng. and Design, 173, 207–217 (1997).

[XX-39] DOSTAL, V., DRISCOLL, M.J., HEJZLAR, P., TODREAS, N.E., A supercritical CO2 gas turbine power cycle for next-generation nuclear reactors, ICONE-10 (Proc. 10th Int. Conf. on Nuclear Engineering, Arlington, April 14–18, 2002), ICONE10-22192, ASME. Also as MIT-ANP-TR-090, CANES (Nov. 2002).

[XX-40] MOISSEYTSEV, A.V., SIENICKI, J. J., WADE, D. C., Turbine design for a supercritical carbon dioxide gas turbine Brayton cycle, Advances in Nuclear Power Plants (Proc. of Int. Congress, Cordoba, Spain, May 4–7, 2003) ICAPP'03 pp 3064 (2003).

[XX-41] ADAMOV, E.O., et al., The next generation of fast reactors, Nuclear Engineering and Design, 173, pp. 143–150 (1997).

[XX-42] BOARDMAN, C.E., et al., A description of the S-PRISM plant, ICONE-8 (Proc. of 8th Int. Conf. on Nuclear Engineering, Baltimore, Maryland, April 2–6, 2000), ICONE-8, pp 8168, ASME.

ANNEX XXI

SMALL LEAD-BISMUTH COOLED REACTOR

Japan Atomic Energy Agency (JAEA),
Japan

Short description

XXI-1. Basic summary

The Small Lead-Bismuth Cooled Reactor is being developed by the Japan Atomic Energy Agency (JAEA).

Core design

The core is of homogeneous type and has 2 regions. With nitride fuel, the core lifetime of 30·years is achieved without reloading or shuffling of fuel.

Plant design

This is a small sized tank-type reactor without an intermediate heat transport system. Steam generator is located inside the reactor vessel. The intermediate heat transport system is eliminated because there is no essential chemical interaction between lead-bismuth and steam.

Safety design

Main and auxiliary cooling systems are driven by natural convection. The inherent safety features of the core are enhanced to avoid a core-disruption accident even in anticipated transients without scram (ATWSs).

XXI-2. Major design and operating characteristics

Main characteristics of the reactor core are summarized in Table XXI-1. Major characteristics of an NPP with the Small Lead-Bismuth Cooled Reactor are given in Table XXI-2. A general view of the reactor is shown in Fig. XXI-1.

XXI-3. List of enabling technologies and their development status

A list of the enabling technologies for the Small Lead-Bismuth Cooled Reactor is presented in Table XXI-3.

TABLE XXI-1. CORE CHARACTERISTICS

ITEMS	SPECIFICATIONS
Fuel type	Nitride type (100% ^{15}N enriched)
Core type	2-region; homogeneous
Fuel assembly type	Ductless type
Number of fuel assemblies (inner/outer core)	84/90
Number of fuel pins	58 per fuel assembly
Enrichment by Pu (inner/outer core)	10.5/18.2%
Fuel burn-up	78 000 MW·day/t
Operation cycle length	30 years
Cladding outer diameter/ lattice pitch	15/18.4 mm
Pitch of fuel assemblies	146 mm
Core circumscribed radius	2.10 m
Core effective height	1.24 m
Average core power density	34.7 W/cm^3
Maximum linear heat rate	198 W/cm
Burn-up reactivity swing	0.93% $\Delta k/k$

TABLE XXI-2. PLANT CHARACTERISTICS

ITEMS	SPECIFICATIONS
Reactor type	Tank type
Electric output	50 MW(e)
Thermal output	132 MW(th)
Primary coolant temperature	505/335°C
Main steam temperature/ pressure	403.5°C/6.5 MPa
Feedwater temperature	220°C
Plant efficiency	38%
Primary coolant circulation	Natural convection
Steam generator	Helical coil type; located in reactor vessel
Decay heat removal system	One primary reactor auxiliary cooling system (PRACS) and one passive reactor vessel auxiliary cooling system (RVACS)
Containment system	Top dome and guard vessel

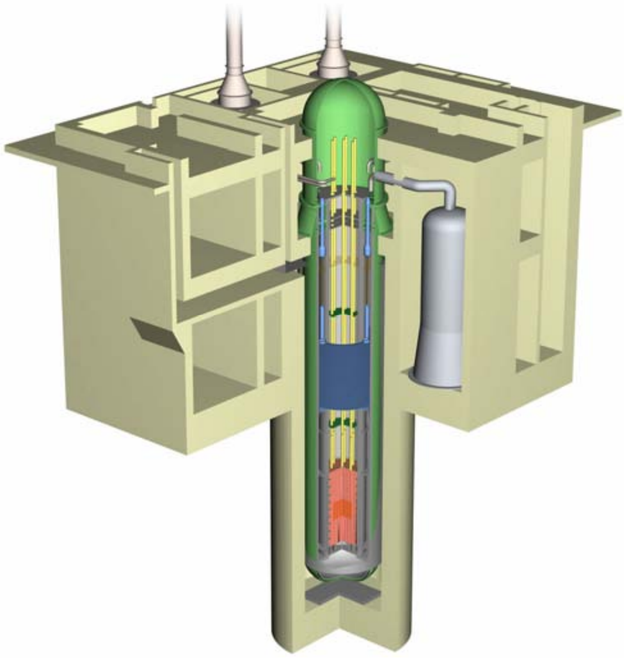


FIG. XXI-1. General view of Small Lead-bismuth Cooled Reactor.

TABLE XXI-3. LIST OF ENABLING TECHNOLOGIES FOR SMALL LEAD–BISMUTH COOLED REACTOR

ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Nitride fuel technology	Conceptual design.
Ductless fuel assemblies	Conceptual design.
Corrosion resistant operation of claddings in lead-bismuth coolant	A corrosion test has been conducted for 10 000 hours in stagnant lead–bismuth.
An oxygen control system to protect structural materials operating in Pb–Bi from corrosion	Conceptual design.
Control rod with a tungsten sinker	Conceptual design.
Three-dimensional seismically isolated reactor building	Experiments on a laboratory scale have been performed.

ANNEX XXII

SMALL SECURE TRANSPORTABLE REACTOR (SSTAR)

**ANL, LLNL, LANL, INL and the University of California,
United States of America**

XXII-1. General information, technical features and operating characteristics

XXII-1.1. Introduction

The Small Secure Transportable Autonomous Reactor (SSTAR) is a lead or lead-bismuth eutectic (LBE)-cooled fast reactor with the power level in the range of 10–100 MW(e). The current reference design is a 20 MW(e) unit that employs a supercritical CO_2 power conversion system. This concept is the latest of a series of concept studies conducted by a team of U.S. national laboratories (Argonne, Livermore, Los Alamos, and Idaho) and the University of California at Berkeley. Several concepts of small heavy liquid metal cooled reactors were developed by members of this team earlier under the Nuclear Energy Research Initiative (NERI). These are STAR-LM, STAR-H2, and the ENHS discussed in other sections of this report. The earlier experience in the NERI projects has played a role in supporting development of the current SSTAR concept under the Generation IV Nuclear Energy Systems Initiative.

The SSTAR, like the ENHS, is targeted for a selected niche in the world market that can benefit from such small power units. These are typically countries with no or small electrical grids. This niche is anticipated to grow significantly in the next 50 years. Other applications include desalination and service in remote regions of developed countries where power is generated locally off the major grids. Studies have been completed that indicate there is a potential worldwide need for several thousand units in the power range of the SSTAR. Most of the identified potential users are not currently using nuclear power and have no or a small technology infrastructure to support nuclear growth. The SSTAR and some of the earlier U.S. heavy liquid metal cooled reactors were specifically developed to address this problem in a cost effective approach that would also reduce concerns about the proliferation of technology and materials used in nuclear weapons.

The word “Secure” has been included in the names of this group of reactors to indicate that the reactor operates in a highly secure nuclear fuel cycle. The reactor is designed to operate in what is now being called a user/supplier arrangement that is in many ways similar to how the suppliers and users of large commercial airliners operate. The user of a SSTAR will have a much smaller investment in the technology used to supply the reactor and fuel cycle services than is required for use of the large plants currently available. In this arrangement, the user need not develop fuel processing plants or waste disposal facilities. The word “Transportable” has been included in the SSTAR name to indicate that the reactor is delivered to the site essentially fully assembled. Ideally, the assembly includes fuel for the life of the plant sealed within the reactor vessel. For such an arrangement to be economically attractive, it is expected that the core life must be very long, – ideally 30 years or more, and the cost of the entire reactor assembly must support its periodic replacement. In addition, the operability of the units must be very reliable with reduced need for in-service inspection for safe reliable operation. Because of this severe challenge, considerations are being given to supplying and removing the core as a single assembly on a shorter core life (e.g. 15 to 20 years). In this case, the core and reactor assembly would be supplied and transported as separate units.

Lastly, the word "Autonomous" has been included in the name to indicate that this reactor and to some extent the balance of plant can operate safely and reliably automatically, requiring only a monitoring function. This requirement is analogous to that placed on a reactor operated in space with the monitoring done at a terrestrial facility. This means that the plant not only performs load following but includes self diagnostic monitoring and controls that respond without operator action as needed to protect the plant and the surrounding environment. It also means that the consequences of operational failures are not severe. Any significant accidental radioactive release is confined to the site.

Both the design and the institutional infrastructure envisioned for the SSTAR are unique and challenging, but if achieved would provide unique proliferation resistant nuclear power to the rapidly growing developing countries that have no alternative for major energy growth other than fossil fuel.

XXII-1.2. Applications

SSTAR at 45 MW(th) and 19.8 MW(e) is the smallest of the Secure, Transportable Autonomous Reactor (STAR) set of concepts which also includes the larger (400 MW(th)) reactors, STAR-LM and STAR-H2 [XXII-1]. SSTAR [XXII-1 to XXII-6] is intended for electricity supply for small towns and villages in off-grid locations such as are prevalent in Alaska, Canada, Siberia, Africa, the interior of South America, and many island nations. Cogeneration options with potable water production or district heating are included. The SSTAR is also suitable for energy supply at off-grid industrial operations such as mining.

XXII-1.3. Special features

Special features of the SSTAR include the following:

- *Long refuelling interval*

The low power density, low pressure drop core has an extremely long refuelling interval of 20 effective full power years. The core is designed as a single large assembly/cartridge and is not composed of individual removable fuel assemblies. To remove the core cartridge, it is necessary to remove the upper head/cover from the reactor vessel. This is done only at the end of the core lifetime when refuelling equipment is temporarily brought on site; the used core and refuelling equipment are then transported back to a regional fuel cycle support centre. Alternatively, the refuelling equipment is moved to another site scheduled for refuelling.

- *Factory fabrication, transportability, and modular assembly at the site*

All nuclear power plant components are factory fabricated to reduce costs and enhance quality control. The components are assembled into fully transportable modules for transport to the site by barge, rail, or by truck. Assembly of modules at the site reduces construction time and costs. Similarly, modular components for the non-safety grade balance of plant can be factory fabricated, and quickly assembled at the site or they can be the responsibility of local companies.

- *Nearly autonomous operation*

The strong coolant temperature-driven reactivity feedback in the fast neutron spectrum core enables autonomous load following whereby the reactor power self-adjusts itself to match heat removal from the primary coolant solely as a consequence of inherent physical phenomena. The system temperatures that are attained following an autonomous power change from the nominal steady state can be optimized through design of the core clamping and restraint approach to enhance the negative reactivity feedback from core radial expansion/flowering, although this additional enhancement is not required for the current SSTAR concept. Autonomous operation reduces operator staffing numbers, workload, and requirements.

XXII-1.4. Summary of major design and operating characteristics

The major features of the SSTAR are as follows:

- *Small reactor size*

The unit power level of 19.8 MW(e) is sized for small towns and villages in off-grid locations and/or to support energy-intensive industrial operations in off-grid locations.

- *Natural circulation primary coolant heat transport*

Natural circulation removes the core power at all levels up to and greater than 100% nominal.

- *High power conversion efficiency at moderate temperatures*

The S-CO_2 gas turbine Brayton cycle power converter for electricity production provides a cycle efficiency of 44% at a lead core outlet temperature of 566°C.

- *Efficient production of fresh water or district heat*

The production of desalinated water or heat for district heating utilizes reject heat from the Brayton cycle and does not degrade the S-CO_2 cycle efficiency for electricity production. The absence of a low pressure turbine and condenser, as in a Rankine steam cycle, means that the CO_2 exiting the low temperature recuperator has an elevated pressure and temperature that facilitates coupling to a bottoming cycle desalination plant or a district heating heat exchanger in which the CO_2 is cooled to 31.25°C immediately above the CO_2 critical temperature in preparation for compression to maximum pressure.

- *Passive safety*

The LFR system provides for ambient pressure single-phase primary coolant natural circulation heat transport and removal of core power under all operational and postulated accident conditions. The high boiling temperature of the Pb coolant enables heat transport by natural circulation of the primary coolant at significantly higher temperatures than with traditional liquid metal cooled reactors. External natural convection driven passive air-cooling of the guard/containment vessel is always in effect and removes power at decay heat levels.

The strong reactivity feedback from the fast neutron spectrum core with transuranic nitride fuel and lead coolant results in passive core power reduction to decay heat power levels while system temperatures remain within structural limits, in the event of loss-of-normal heat removal to the secondary side through the in-reactor lead-to-CO_2 heat exchangers.

Passive safety together with low risk of fission product release could enable siting close to population centres. There may be no need for an emergency planning zone.

- *Sustainable closed fuel cycle*

The fast neutron spectrum with transuranic nitride fuel and lead coolant is fissile self sufficient with a core conversion ratio of unity. This enables a closed fuel cycle based upon a fertile feed stream of depleted or natural uranium and a minimal volume waste stream comprised only of fission products. All fissile material including minor actinides is recycled in the fabrication of new fuel cores and is burned as fuel in STAR reactors.

Fuel cycle and waste management services are outsourced to centralized, economy-of-scale regional centres that operate under international non-proliferation oversight.

The SSTAR concept is illustrated in Fig. XXII-1 through XXII-3. The Pb coolant flows upwards through the core and the above-core riser region interior to the above core shroud, see Fig. XXII-1. Coolant flows through the holes in the shroud and enters the modular in-reactor heat exchangers on the shell side to flow downwards over the exterior of circular tubes arranged on a triangular pitch. The S-CO_2 flows upwards in the tubes and heat is transferred from Pb to S-CO_2 in a counter-current regime. The Pb exits the heat exchangers to flow downwards through the downcomer to enter the reactor vessel lower head. A flow distributor head provides for an approximately uniform pressure boundary condition beneath the core.

The traditional Rankine steam cycle has been replaced with a gas turbine recompression Brayton cycle that utilizes S-CO_2 as the working fluid to achieve a small balance of plant footprint with reduced equipment count and reduced staffing requirements [XXII-7 to XXII-10]. It achieves a cycle efficiency of 44 %. The higher operating temperatures of the Brayton cycle facilitate selection of Pb (T_{melt} = 327°C; T_{boil} = 1740°C) as the primary coolant rather than lower melting point LBE (T_{melt} = 125°C; T_{boil} = 1670°C). Lead is less corrosive to unprotected steel than Bi and LBE [XXII-11] and it therefore provides a development pathway to the high operating temperatures targeted for the STAR-H2 member of the STAR portfolio.

The higher discharge temperature of the Brayton cycle turbine makes SSTAR compatible with bottoming cycles for district heating or desalination. Figures XXII-2 and XXII-3 show that heat rejection from the Brayton cycle occurs over the temperature interval 90°C down to 31°C. In the design illustrated here, that heat is simply rejected to the environment; however it comprises ~55% of the reactor thermal rating, it is delivered at a still-useful temperature and it is therefore available for cogeneration missions including district heating, low temperature process heat applications, or desalination.

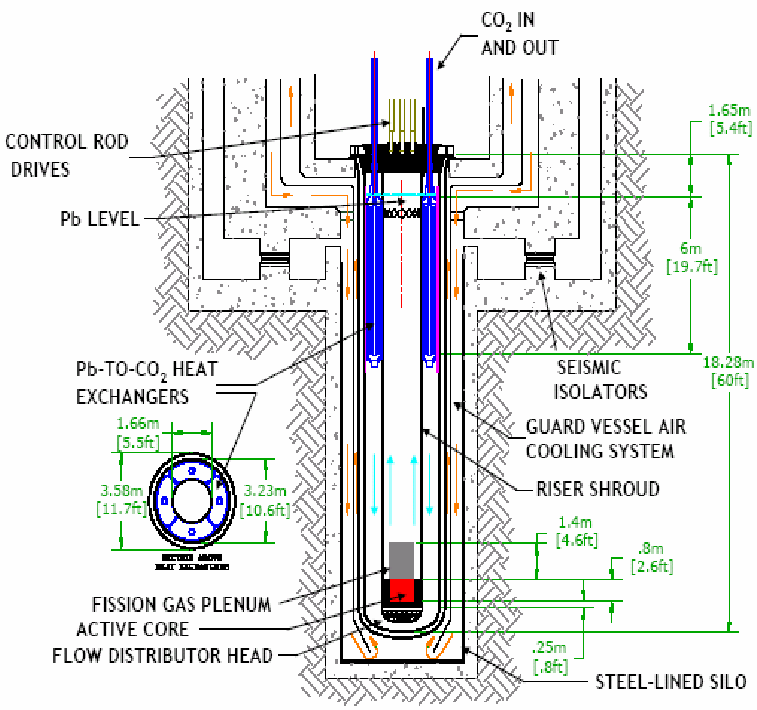


FIG. XXII-1. SSTAR reactor module.

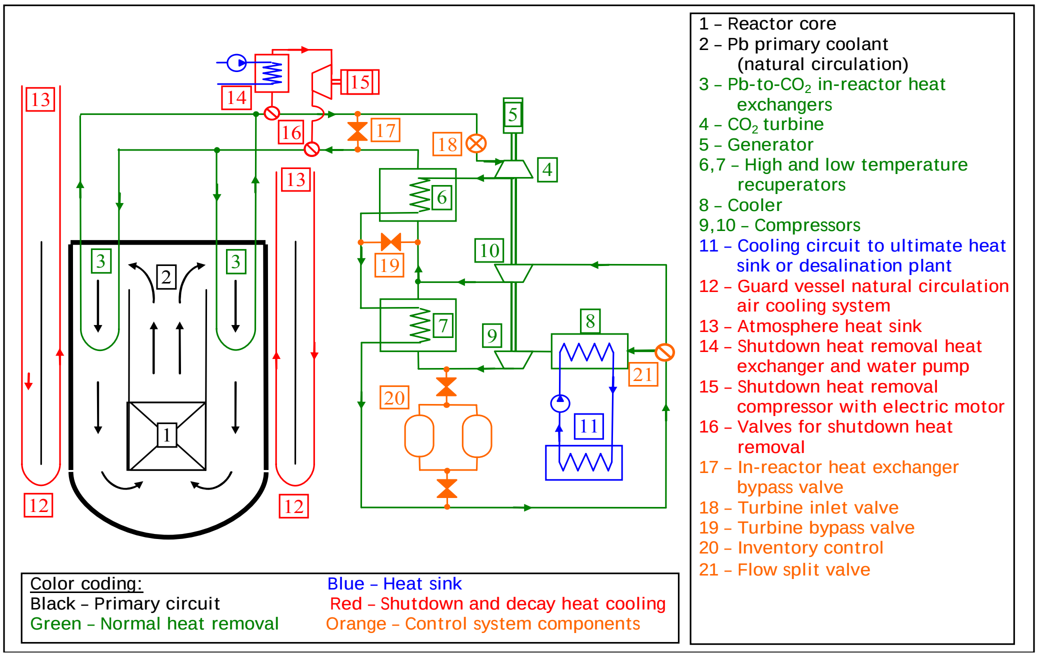


FIG. XXII-2. Schematic of SSTAR coupled to S-CO_2 Brayton cycle showing heat transfer paths.

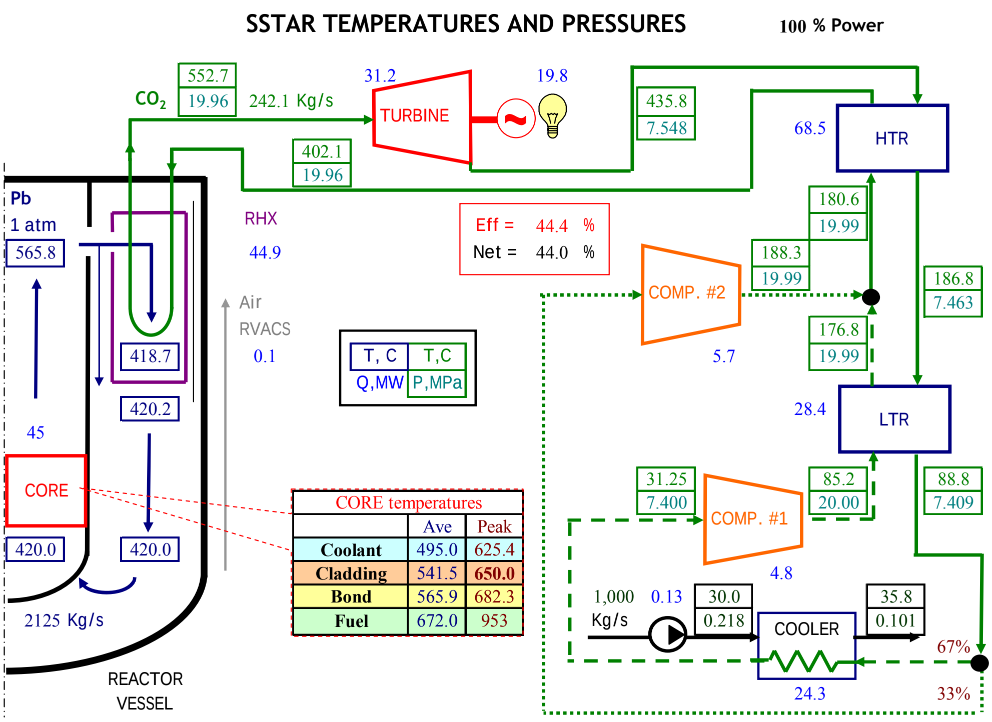


FIG. XXII-3. Schematic of SSTAR coupled to S-CO_2 Brayton cycle showing nominal operating temperatures, pressures and flow rates.

Major design and operating characteristics of the SSTAR are summarized in Table XXII-1.

TABLE XXII-1. SUMMARY OF DESIGN AND OPERATING CHARACTERISTICS

CHARACTERISTIC	VALUE
Installed capacity	45 MW(th) (20 MW(e))
Mode of operation	Autonomous load follow
Load factor/availability	Very high - to be determined. Refuelling only once every 20 years.
Type of fuel	Transuranic nitride clad in cylindrical fuel rods; nitrogen isotope = ^{15}N. (Early development may use enriched uranium nitride fuel.)
Fuel enrichment	Five enrichment zones TRU/HM by zone = 1.7/3.3/16.6/18.3/19.9
Coolant	Lead
Moderator	None
Core structural materials	Ferritic-martensitic stainless steel cladding and structures
In-vessel structural materials	Ferritic-martensitic stainless steel
Core Fuel rod pitch-to-diameter ratio Cladding outer diameter Cladding thickness Fuel pellet-cladding bond Fuel smear density Fuel pellet outer diameter Active core height Active core diameter Above core fission gas	Open-lattice of cylindrical fuel rods on a triangular pitch lattice. 1.121 2.5 cm 1 mm Pb 0.90 2.18 cm 0.8 m 1.02 m 1.4 m
Plenum height Below core axial reflector height Reflector Reflector effective thickness Core diameter with reflector	1.75 times the active core height, based on conservative assumptions on fission gas driven creep rupture 0.25 m 50 volume % ferritic-martensitic SST and 50 volume % Pb 29.5 cm 1.31 m
Reactor vessel Outer diameter Height Thickness Design lifetime	Steel cylinder with curved lower head. 3.23 m 18.3 m 5.08 cm 60 years
Cycle type	Indirect gas turbine Brayton cycle with supercritical carbon dioxide
Circuits	- Lead primary coolant circuit with natural circulation; - Supercritical carbon dioxide secondary circuit with gas turbine Brayton cycle; - Four kidney shaped in-vessel heat exchangers (HXs).
Neutron physical characteristics: Cycle length Coolant void worth	 20 full power years -1.68 \$ at BOC/ -1.83 \$ at EOC
Burn-up reactivity swing = $k_{eff,max} - k_{eff,min}$ during the cycle Peaking factors	 0.85\$ 1.68 BOC/1.64 EOC

CHARACTERISTIC	VALUE
Reduction of peaking	By-enrichment zoning
Reactivity control mechanism	- Shutdown rod for reactor start-up and shutdown. - During operation, reactor power autonomously load follows by means of inherent physical processes without the need for any motion of control rods or any operator actions. - System temperatures change corresponding to reactivity feedbacks from fuel Doppler, fuel and cladding axial expansion, core radial expansion, and coolant density effects. - Control rods for possible fine reactivity compensation during cycle. - Control rods also provide for diverse and independent shutdown.
Maximum reactivity change with burn-up	<1$
Cycle type Cycle efficiency	Indirect cycle with gas turbine Brayton cycle secondary side using supercritical carbon dioxide as the working fluid at ~20 MPa. 44.0%
Thermal-hydraulic characteristics Core inlet temperature Core outlet temperature Primary coolant flow rate Primary coolant cover gas pressure Temperature limit for cladding Maximum fuel temperature Maximum cladding inner surface temperature during normal operation Average fuel temperature Average cladding inner surface temperature	Natural circulation of lead primary coolant. No primary coolant pumps. 420°C 566°C 2125 Kg/s Slightly below 1 atmosphere 650°C 953°C 650°C 628°C 567°C
Maximum/average discharge burn-up of fuel: - Maximum - Average	 122 MW·day/kg 72 MW·day/kg
Fuel lifetime/period between refuellings	20 full power years
Mass balances/flows of fuel materials: 3671 kg depleted U, 750 kg TRU, and 1448 kg HT9 cladding every 20 years; 4.08 kg depleted U/(MW(th)·year); 0.83 kg TRU/(MW(th)·year); 1.61 kg HT9 cladding/(MW(th)·year); 9.18 kg depleted U/ (MW(e)·year); 1.88 kg TRU/(MW(e)·year); 3.62 kg HT9 cladding/(MW(e)·year); Best estimate calculation using DIF3D and REBUS-3 computer codes.	
Design basis lifetime: Core	 20 years (core lifetime based upon fluence limit of 4×10^{23} fast neutrons/cm^2 for HT9 cladding).
Reactor vessel Core shroud In-vessel structures other than core shroud	60 years (reactor vessel lifetime based upon service temperature) 20 years 60 years.
Design and operating characteristics of systems for non-electric applications	Optional desalinated water production using portion of reject heat.
Economics	Estimated: 50 to 80 $/(MW(e) hr)

XXII-1.5. Outline of fuel cycle options

The SSTAR (and the STAR-LM and STAR-H2) concepts utilize transuranic nitride fuel with ^{15}N in a closed fuel cycle in which the fuel cycle feedstock is depleted or natural uranium. Multiple recycle through sequential reloading cycles of the cassette/cartridge core achieves total fission consumption of the feedstock. The effluent stream contains only fission product waste forms and trace losses of the transuranics. The reactor is fissile self sufficient with an internal core conversion ratio of unity. The fuel recycle technology is based on electrometallurgical recycle and remote vibropack refabrication of the transuranic/uranium nitride fuel. The recycle technology produces a co-mixed stream of all transuranics and achieves incomplete fission product removal such that the transuranic materials during fresh and used cassette shipping are always at least as unattractive for military use as light water reactor (LWR) spent fuel.

In the future sustainable world energy supply architecture, recycle could be conducted at secure regional fuel cycle support centres. Each such centre forms the hub of a hub-and-spoke energy supply system of regional centres and surrounding STARs with shipments of fuel occurring on 20 year refuelling intervals.

For initial core loadings of new STAR reactor deployments, transuranic fissile material might be obtained from used LWR fuel, which has been cooled no less than 25 years to allow for ^{241}Pu decay. Alternatively, the initial core loading could utilize enriched uranium or excess plutonium from weapons. In a growing economy, later in the century, the increasing need for fissile material to support new STAR deployments could be covered by fast breeder reactors sited at the regional fuel cycle support centres. Their function would be to produce excess fissile material to fuel the initial working inventories of new reactors.

XXII-1.6. Technical features and technological approaches that are definitive for SSTAR performance in particular areas

XXII-1.6.1. Economics

There is currently little specific information or a detailed design to provide a basis for a cost estimate of the SSTAR. Consequently, current estimates are based on information about the Advanced Liquid Metal Reactor (ALMR) for which a detailed cost analysis has been completed by the GE Nuclear Energy [XXII-12]. Because the ALMR reactor is substantially larger than SSTAR, 622 MW(e) vs. 20 MW(e), the equipment, assembly and installation costs of the ALMR have been scaled to estimate the corresponding costs of the SSTAR. The fact that SSTAR does not require an intermediate coolant loop between the primary coolant and the working fluid was also accounted for in the analysis. In other areas such as staffing, buildings, and fuel cost, direct scaling did not appear to be appropriate and other adjustments were made through a review of the assumptions used in the ALMR.

In the scaling analysis, a 50 MW(e) power level has been assumed for SSTAR. The other costs of the ALMR, such as the operating labour and other operating costs, have been adjusted to reflect the likely operation of the SSTAR.

In addition, the SSTAR is very similar in size and design to the 4S reactor [XXII-13] for which a detailed design has been completed. Many elements of the SSTAR design are similar to the 4S design so that cost scaling when performed at the detailed design level is a reasonable approach. Based on the 4S design, a direct cost estimate for the nuclear steam supply system (NSSS) was derived based on estimated materials and labour effort required to fabricate and install the NSSS. The cost scaling approach and the direct estimate yielded comparable results, suggesting that the cost scaling approach is appropriate for the analysis.

Assessment of the costs of the SSTAR

The cost analysis of the SSTAR is directed toward estimating the cost of energy (COE) and to identify features of the design, construction, and operation that have potential for reducing the costs.

The analysis makes a base case estimate using simple scaling and adjustments. The base case does not account for a number of features of the SSTAR that are expected to be simpler than the ALMR and may reduce the cost of the SSTAR (on an energy cost basis) below what is estimated for the base case. For each of these features an alternative adjustment was identified. Additional estimates were made, each one incorporating one or more of these alternative adjustments. The analyses assess the potential impact of these features on the final cost of energy. Some of them have a negligible effect, while others do indicate significant reductions in the COE could be achieved (i.e. below the base).

Major phases of fabrication and assembly

Modular design and construction of nuclear plants continues to be an important direction for realizing cost reductions. The approach envisioned for the SSTAR is directed toward dramatically reducing the number of modules necessary to install at the site. Depending on the plant size and design, anywhere from 125 to 600 modules may be used in current large LWR plants. In the case of the SSTAR, it is sought to reduce the number of modules by an order of magnitude or more. Specifically, it is envisioned that the reactor module will be assembled in a specially designed assembly factory. This assembly would possibly include installation of the fuel and sealing it into the reactor prior to shipment.

The SSTAR units are physically smaller than the ALMR units, which means that even major building structures can be factory assembled and brought to a site for installation. The modules can be more self-contained leading to a shorter installation time and a simpler installation process. This should lead to less engineering and labour effort on site, and a lower risk of cost over-runs. It is expected that SSTAR will not require fuelling over its lifetime. Consequently, the SSTAR will have a higher capacity factor than the ALMR since it does not need to be shut down for refuelling. It is also anticipated that SSTAR will experience fewer shutdowns for maintenance and repair than a larger, more complex, reactor.

Some components may be fabricated at the dedicated assembly facility; others will be purchased from outside vendors. In addition to the reactor assembly, it is envisioned that the turbine generator and several building structures will constitute the major assemblies. The major assemblies will be delivered to the site as modules. Some components, materials, and services will be purchased locally. The site must be prepared, the modules installed and connected together. Operational costs will include the costs of operating the reactor system and the turbine/generator. Decommissioning costs have also been considered, but on a present value basis they are minimal.

Fuel costs

Two cases have been used in evaluating the fuel cost. The base case assumes that the fuel is reprocessed and fabricated using the ALMR reprocessing and fabrication evaluations directly, in which case the cost per kg is the same as for the ALMR, or US $5.2 million per tonne. The alternative case assumes that SSTAR is fuelled with uranium in which case the reprocessing cost is replaced with the cost of enriching the fuel to 20%. In this case, the cost of fuel is assumed to be US $7.1 million per tonne.

In both cases, all of the fuel in the 20-year life is purchased along with other capital equipment. However, consistent with the approach taken in the ALMR estimate, the cost of the fuel is not included in the capital costs used to compute the indirect costs.

Cases analyzed

The base case analysis uses simple scaling and adjustments from the ALMR design. The discussion above identifies a series of factors that would result in lower energy costs than would be indicated by the simple scaling. A series of additional cases were run to test the impact of these factors. Each factor was evaluated in a case that varied only that factor. Then several cases were run that included more than one factor to assess the impacts of combinations. Table XXII-2 summarizes the cases analyzed.

TABLE XXII-2. SUMMARY OF CASES ANALYZED

CASE	DISCUSSION
Simple scaling from ALMR (base case)	Direct scaling and other simple adjustments
Reduced field labour	For the field installation of the reactor system, the electric plant, turbine generator, and heat rejection system, reduce the field labour effort (and thus cost) by 50%
Reduced contingency	Reduce the contingency cost by 50%
Reduced indirect cost	Reduce indirect costs by 50%
Increased capacity factor	Increase capacity factor from 85% to 95%
Reduced construction time	Reduce construction time from five years to two years
Capacity factor and construction time	Adjust both the construction time and the capacity factor, as above
All benefits	Include all of the adjustments together
All benefits with higher fuel cost	Include all the benefit and increase fuel costs from US \$5.2 million per tonne to US \$7.1 million per tonne
Larger scaling factor	Include all the benefit and increase scaling factor from 0.53 to 0.70. This is closer to linear scaling.
ALMR	Costs taken from the ALMR estimate (i.e., without scaling)

Results

The energy generation costs of SSTAR and the ALMR are shown in Fig. XXII-4, on a per MW-h basis. The far right hand columns in the figure show the breakdown and the total cost per MW-h for the ALMR design. The far left hand columns show the cost breakdown for the SSTAR concept for the base case - not taking into account any of the possible cost benefits identified above. The subsequent columns show the impact of individual benefits and groups of benefits. When all of the expected benefits are realized, the generation costs of SSTAR are higher than in the large ALMR but not substantially higher and perhaps in the competitive range for many of the markets this reactor concept is intended to serve. Many of the cost reductions are the result of the short two-year installation time and reduction in operating staff. These reductions are based on the judgment that the small simplified design and increased safety and security built into the plant will make it possible to realize these targets.

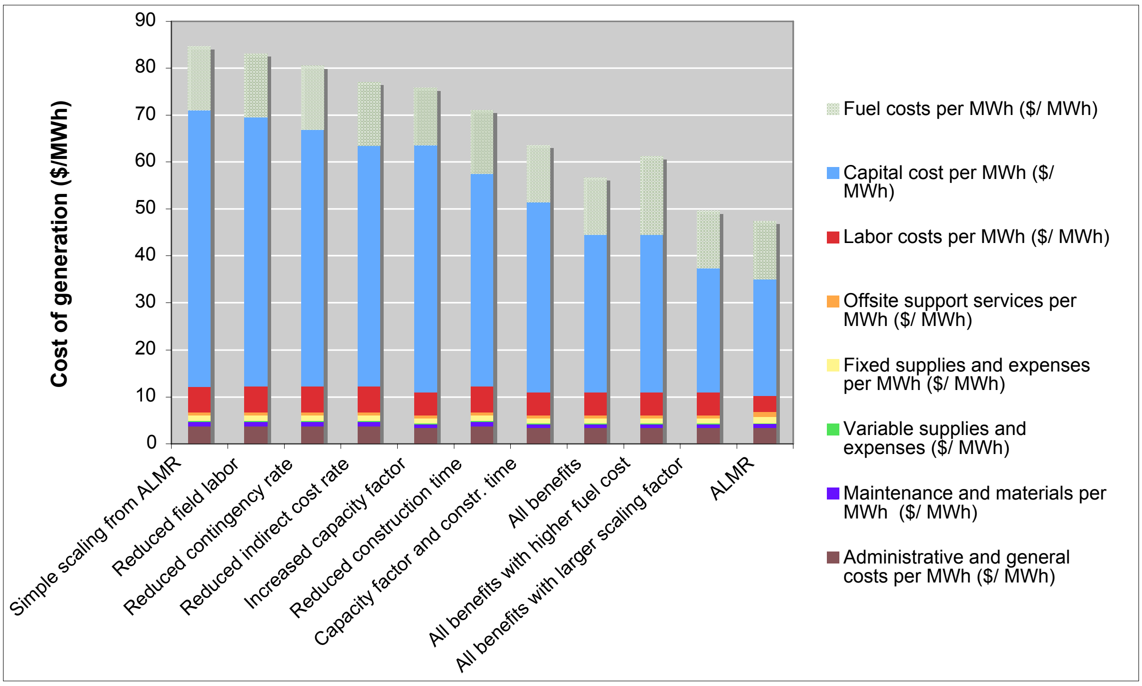


FIG. XXII-4. Energy generation costs by cost element for various cases.

XXII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The future world energy supply architecture based on regional fuel cycle centres and distributed STAR reactors could achieve sustainability. The closed fuel cycle achieves maximum utilization of uranium resources. The fuel cycle feedstock is depleted or natural uranium. Multiple recycle through sequential reloading cycles achieves total fission consumption of the feedstock. The fuel cycle effluent stream is minimized, consisting only of fission product waste forms and trace losses of transuranics.

XXII-1.6.3. Safety

Safety concept and design philosophy

The approach to SSTAR safety design is intended to achieve significant improvements in safety over the current fleet of operating reactors. This improvement is needed to meet the U.S. regulatory objectives for advanced reactors. The properties of lead coolant, nitride fuel, and a fast neutron spectrum core support achieving and ensuring a passively safe plant. One aspect of the approach being used is to eliminate the need for reliance upon active systems, to take advantage of inherent safety features, and to include safety margins that assure acceptable response to even the most improbable postulated accident sequences. Thus, while certain active systems such as active scram systems may be required to meet licensing requirements, the passive safety behaviour of SSTAR does not result in unacceptable power or temperature conditions in selected scenarios involving postulated failure of the active systems to shut down the reactor neutronically.

Provisions for simplicity and robustness of the design; active and passive systems and inherent safety features

Passive safety features were readily analyzed and analyses have revealed that SSTAR is robust to postulated failures. There is no need to rely upon active safety systems. Although scram systems are provided to insert rods to shut down the reactor neutronically, success of scram is not required to prevent unacceptable power or temperature conditions. The SSTAR is designed to respond safely to all operational, unlikely and highly unlikely transients, including selected postulated transients without scram. The atmospheric pressure system uses single-phase natural circulating coolant for heat transport and removal of core power under full power and postulated accident conditions. This is a consequence of the following inherent and passive features:

- Strong reactivity feedbacks from the fast neutron spectrum core with transuranic nitride fuel and lead coolant, possibly enhanced through passive structural design, that enables autonomous load following and passive core power reduction to decay heat level in response to accidents while system temperatures remain within structural limits;
- Excess reactivity limited to less than $1, that permits safe response to postulated insertion of all excess reactivity from power operating conditions;
- Safe shutdown of the reactor due to the passive strong reactivity feedbacks without motion of control rods either due to operator or automatic scram action in response to all credible and postulated accident sequences;
- Natural circulation heat transport of the lead coolant at power levels in excess of 100% nominal that eliminates loss-of-flow accidents;
- External natural convection driven passive air-cooling over the guard/containment vessel (surrounding the reactor vessel) that is always in effect and removes decay heat power levels.

In addition to this:

- The high boiling temperature of the lead liquid metal coolant equal to 1740°C realistically eliminates boiling of the low pressure coolant;
- The lead coolant is inert, i.e. does not react chemically with the power conversion coolant, carbon dioxide, above about 250°C (well below the 327°C Pb melting temperature) and does not react vigorously with air or water;
- Transuranic nitride fuel is chemically compatible with the lead coolant. The high nitride thermal conductivity together with bonding of the fuel and cladding with molten Pb results in low fuel centreline temperatures and small thermal energy storage in the fuel;
- The system has a pool configuration and ambient pressure coolant with a reactor vessel and surrounding guard vessel that eliminates loss-of-primary coolant;
- The high heavy liquid metal coolant density (Δ_{Pb}=10400 Kg/m^3) limits void growth and downward penetration following postulated heat exchanger (HX) tube rupture such that void is not transported to the core but instead rises benignly to the lead free surface through a deliberate escape channel between the HXs and the vessel wall; and
- The compact liquid metal system is seismically isolated on a nuclear island using seismic isolators (see Fig. XXII-1).

In the event of a HX tube rupture, a blow down of secondary CO_2 (initially at 20 MPa) into the lead occurs. Molten lead and CO_2 do not react chemically. Protection against over pressurization of the primary coolant vessel must be provided and activity that is entrained from the lead coolant into the CO_2 must be contained. Thus, a pressure relief system is provided for the primary coolant system. The S-CO_2 secondary circuit incorporates valves to isolate the failed heat exchanger and limit the mass of CO_2 that can enter the primary coolant system. The CO_2 released from the primary coolant system is contained inside of a volume.

Structure of the defence-in-depth

SSTAR incorporates defence-in-depth in providing a containment that surrounds the primary coolant system. The bottom portion of the containment is the steel guard vessel that surrounds the reactor vessel. The guard vessel and reactor vessel are hermetically sealed by the upper head/cover. An additional containment boundary has been provided over the reactor head cover. Pressurization due to depressurization/ flashing of the coolant is precluded by usage of the ambient pressure liquid metal primary coolant. The usage of Pb primary coolant and CO_2 secondary coolant that does not react chemically with Pb precludes the generation of hydrogen or other combustible gases. Thus, there are no combustion or explosion hazards from the generation of combustible or explosive gases. Postulated leakage of the cover gas or CO_2 following a HX rupture event may require this containment boundary.

Design basis accidents and beyond design basis accidents

In consideration of design basis accidents and beyond design basis accidents, the set of possible accident initiators is greatly reduced relative to LWRs. Significantly, the need to consider scenarios involving core uncovery, fuel degradation/melting, combustible gas generation (e.g. H_2), and significant fission product release does not arise.

The following preliminary list of accidents or potential accidents has been identified. Consistent with the philosophy of avoiding reliance upon active safety systems, for the analysis it is assumed that automatic insertion of the shutdown rods does not occur; that is, there is an assumed failure to scram in each accident. Analyses show that unacceptable system temperatures and core damage will be averted by strictly passive means without scram.

Loss of normal heat exchanger heat removal

This is an extreme loss-of-heat sink accident variant in which the removal of heat from the lead coolant by CO_2 through all of the lead-to-CO_2 heat exchangers is assumed to cease. It demonstrates passive safety features of the reactor.

Heat exchanger tube rupture

This scenario must be analyzed for any system that incorporates in-reactor primary-to-secondary coolant heat exchangers [XXII-14].

Transient overcooling

The potential existence of such scenarios needs to be investigated in connection with the realistic behaviour of the S-CO_2 Brayton cycle in response to secondary side accident initiators. Of interest is the potential for cooling part of the lead below its freezing temperature. One such scenario involves a postulated rupture of the CO_2 hot stream piping upstream of the turbine and the recuperators reducing the temperature of CO_2 entering the in-reactor HXs while temporarily increasing the CO_2 flow rate through the HXs.

Loss of generator load

This scenario involves disconnect of the grid from the generator tending to increase the rotational speed of the turbine and reducing heat removal from the in-reactor HXs. The S-CO_2 Brayton cycle control system shall be designed to cope with the event.

Passive safety combined with the provision of a containment structure should result in an extremely low risk of radioactivity release beyond the plant boundaries. Consequently, the goal is to restrict emergency planning to the reactor site.

XXII-1.6.4. Proliferation resistance

The STAR proliferation resistance features include 20 full power years between whole-core refuelling, combined with a regime of fuel cycle services and waste management conducted at centralized facilities. The long refuelling interval offers unprecedented energy security without a need for a nation to deploy an indigenous fuel cycle infrastructure. At the same time it minimizes the frequency of shipping fuel, it ties fissile material up in an inaccessible in-core inventory, and it centralizes bulk fissile handling operations into a small number of centralized facilities under international safeguards oversight.

SSTAR has been designed from the outset to incorporate technical features to prevent the diversion of nuclear materials. One feature is the cassette/ cartridge core that is a single large assembly and is not composed of individual removable fuel assemblies/ bundles. The reactor vessel upper head/ cover does not incorporate openings that could be used to facilitate removal of individual assemblies/ bundles.

Restriction of access to the inside of the reactor vessel also limits access to neutrons escaping from the core that could be used for the undeclared production of direct-use material. The lack of access to the vessel interior can also be verified by means of overhead reconnaissance.

The plant does not incorporate refuelling equipment or provision for on-site nor in-vessel fuel storage. Centralized fuel cycle service personnel bring refuelling equipment onto the site with a reload cassette at the end of the 20-year refuelling interval. Removal of the core requires a crane or other lifting device, and refuelling equipment brought to the site will include a mobile crawler crane; unwarranted presence of a crane at the site could be detected by satellite or other overhead surveillance. The upper portion of the containment is removed to provide access to the reactor vessel. The spent core cassette is removed from the reactor vessel and installed inside a shipping cask (Fig. XXII-5) for transport to a regional fuel cycle support centre, for reprocessing and refabrication and waste management operations. Another cask containing the fresh cassette is mounted above the vessel and the fresh cassette is installed inside the vessel. The upper head is mounted upon the vessel and the seal is welded shut. The refuelling equipment including the heavy lift crane is removed from the site.

Once the cassette is installed inside the cask, Pb filling the coolant flow channels between the fuel pins is solidified. The purpose is to embed the fuel pins in frozen Pb during shipping, to embed shutdown and control rods and to prevent rearrangement of the fuel pins into a possible critical configuration, to embed radionuclides that might be released from failed fuel pin cladding, and to preclude water ingression into the coolant channels between the fuel pins even in the event of a shipping accident. Decay heat is removed by circulation of air from the shipping cask exterior. The fresh core cassette brought to the site is also embedded inside frozen Pb. Once refuelling operations are completed, all refuelling equipment is removed from the site with the spent cassette. It is envisioned that the refuelling equipment will be taken by the itinerant refuelling team successively to other sites as other reactors in the service territory require refuelling.

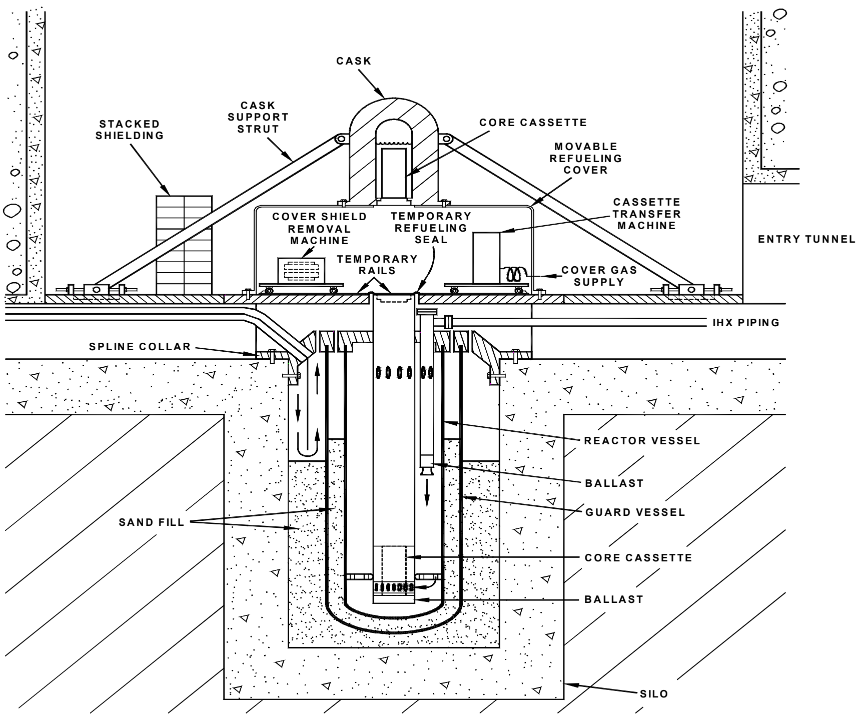


FIG. XXII-5. Illustration of installation of a cask to receive spent core cassette.

As it was already mentioned, recycle and cassette refabrication are carried out inside secure regional fuel cycle support centres under international safeguards oversight. The electrometallurgical-based recycle technology conducted at the central facility does not involve separation of pure plutonium. The plutonium always remains inherently commingled with minor actinides (i.e., americium, curium, and neptunium), uranium, and fission products. The minor actinides contribute substantial decay heat and contamination with alpha, beta, gamma, and neutron radiation emitters. The fresh fuel product during all recycle operations and that present in a reload cassette remains highly radioactive and self-protective in the safeguards sense. In particular, the “spent fuel standard” is met or exceeded throughout the fuel cycle meaning that the material is self-protected by virtue of contained radiation comparable with that of used LWR fuel.

XXII-1.6.5. Technical features and technological approaches used to facilitate physical protection of SSTAR

The reactor is installed inside a silo and under a massive, controlled access berm – affording protection against the effects of an aircraft crash, terrorist acts, and natural disasters (Fig. XXII-6). Emergency decay heat removal from the reactor is provided by natural circulation air-cooling of the containment guard vessel surrounding the reactor vessel – this heat removal process is always in operation. Multiple chimneys, which penetrate the berm, are provided for air inflow and outflow; they are redundant and separated as a measure against the effects of aircraft crash or sabotage.

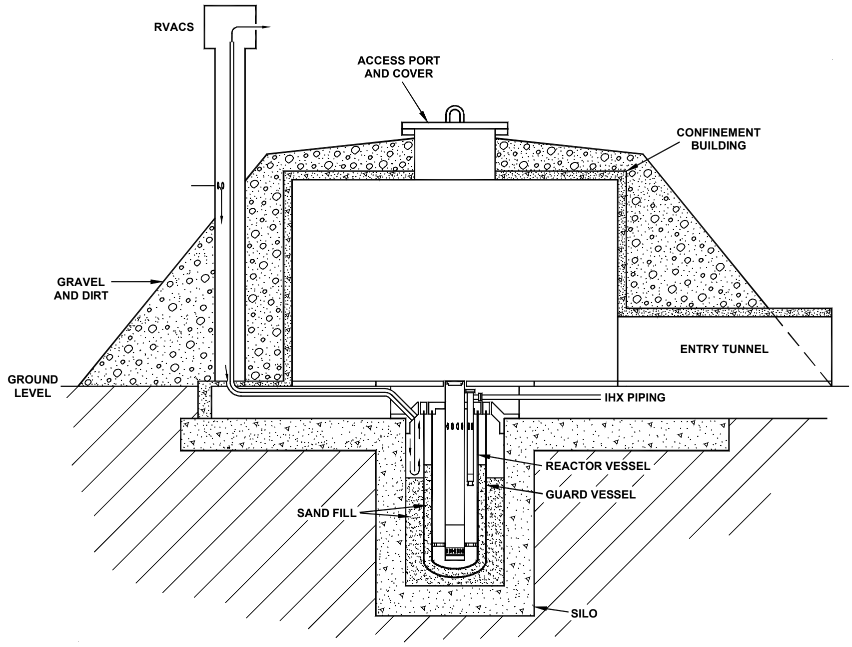


FIG. XXII-6. Possible configuration to enhance protection of STAR-LM.

The autonomous load follow and passive safety behaviour of the reactor enhance its resistance to sabotage or malevolent human-induced events; any combination of mechanical and human malfunctions in the balance of plant are passively accommodated without core damage. Subsequently, as long as both the reactor and guard vessels are not penetrated simultaneously, causing the loss of lead coolant, and as long as the guard vessel external air cooling is not interrupted, the core remains covered and intact, heat is removed from the core by natural circulation of Pb, and is removed from the guard vessel/containment by natural circulation of air.

XXII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of SSTAR

To realize the SSTAR solution for supplying power in developing countries there are two major non-technical areas that need to be developed that are different than the current nuclear business practices. These are: (1) initial development; and (2) commercialization. Initial development has a fairly straightforward solution assuming the commercialization problem is solved, but commercialization has several complexities that are not easily addressed.

First, assuming that commercial incentive and financial support existed, the initial development would simply be a matter of identifying a development programme that the supplier country government and commercial suppliers were willing to support. The development requires support from both government and industry because reducing proliferation concerns is a government driven issue but the design and production of the SSTAR is a commercially driven issue. In addition, there is more than just the SSTAR needed to implement the envisioned proliferation resistant infrastructure. The supplier/ user arrangement and highly secure nuclear fuel cycle must include a government commitment to providing an assured fuel supply and accommodation of the waste at a few centralized locations. Government support for the SSTAR role in proliferation reduction and waste disposal must be established. Government support will also be dependent on the existence of an industry-supported commercial development plan.

Realization of a deployable design could be accelerated through a license-by-test approach involving the construction of a demonstration test reactor that is a prototype for the commercial reactor. Once these commitments are established, the appropriate sharing of the development costs between government and industry will need to be established. The development plan, schedule, and costs would naturally take advantage of the government funded research, completed and ongoing, that is supporting the SSTAR technology.

The extent to which these institutional issues have to be illuminated and committed to will only be determined as the ongoing research identifies the incentives for both government and industry to undertake the development. This must all be done in the political environment of government actions and the competition for the government and industrial development funding. However, given a clear understanding of the proliferation-resistance benefits of the SSTAR and the highly secure nuclear fuel cycle, and with a clear picture of the commercial opportunity, it might be possible to find the development funding.

Commercialization of the SSTAR and the highly secure nuclear fuel cycle is a bit more complex with several different business options available. It shares many of the issues associated with the development phase but also a business model that is different than the one used in the large fossil-based and nuclear energy supply business. The current industry approach mainly involves the operation and construction of large-capacity reactor units that take advantage of economy-of-scale, in order to be economically competitive. The power plants are owned by utilities or generating companies that must raise large amounts of capital to finance their investment and, under the competitive business structure growing throughout the industry, they bear the financial risk. New licensing statutes, although untested, are intended to reduce the financial risk associated with the safety review, provided that criteria to demonstrate safe operation are met.

In the case of SSTAR, deployment may take place in many developing nations and the market would therefore need hundreds or thousands of SSTAR units. The investment, production, installation, and servicing business are expected to be more like the large commercial aircraft business than the current nuclear plant business. To counter the cost penalty associated with small size, it is necessary for the reactor supplier to develop assembly plants that can assemble the entire reactor assembly, including the fuel (as mentioned previously, the core may be a separate module installed at the site but it is desirable to include it sealed in the reactor at the assembly plant, if possible). There may be some gain associated with the cost of unit production in this manner but the real cost savings will result from the speed of installation of this fully assembled unit along with a minimum of other factory assembled building and power generation modules. A two-year installation period may be achieved and the project construction cycle may be closer to that associated with the modern gas fuelled combined cycle plant. This favours the opening of a factory and assembly line for production of a standardized reactor design. In such a scenario, the reactor vendor must make a large investment in the factory and tooling on an expectation of the development of a large future market. The reactor vendor thus bears a major risk in committing investment to cover the costs of development of the modular nuclear plant and factory construction that might not be recovered before the sale of a large number of units. Production of standardized reactor units for deployment in different nations implies that the same standardized design will be deemed acceptable to the regulatory bodies of all of the nations.

Realization of a standardized design deployable in many countries requires the adoption of a licensing approach that incorporates criteria that are accepted multi-nationally. A model for such an approach is provided by the commercial aircraft industry for which the requirements to achieve certification of an aircraft in the U.S. and Europe are more or less universally accepted.

Opening a factory and assembly line for a standardized nuclear power plant is a “chicken and egg” problem for a potential vendor, in the absence of a large confirmed market. Arrangements are sought that can break this dilemma. One might be a “virtual factory” whereby fabrication is performed by a consortium of vendors or by subcontractors that already manufacture components similar to those in the power plant design.

The highly secure fuel cycle needed to support the SSTAR will require multi-national acceptance of the SSTAR approach and national policies that enable its creation and success. In particular, user nations must develop confidence in the fuel supply and spent reactor or core assembly replacement and be willing to rely on a few suppliers to assure continued supply of the skills and materials to maintain a reliable power supply. The fact that the unit has a very long life reduces the frequency of concern, particularly if a 20 year or longer power supply can be purchased up front. These additional facilities will also require capital investment to modify existing facilities or provide new facilities. If a core replacement approach is used, the ability to lease core assemblies supplied by a vendor could further reduce the costs to utilities that operate SSTARs while making ownership of fuel a lucrative business. This approach could be applied to the entire reactor module. Alternatively, there could be a business entity to own and lease all of the SSTAR units similar in conduct to a portion of the commercial airlines, shipping, and railroad businesses. In the context of recycle of fissile self-sufficient units, the fuel that contains as much fissile material as when first fabricated becomes a valuable commodity that can be leased for a profit to utilities for use after which it is returned for reprocessing and fabrication for further use. Leasing arrangements could accelerate or enhance the development and investment in the facilities needed to implement the highly secure nuclear fuel cycle.

There is the difficult issue associated with nuclear waste. The SSTAR approach and the recycling significantly reduce the high level nuclear waste burden but do not eliminate it. It will be necessary for the supplier to develop arrangements for disposal of the recycle waste. Security and safety concerns with the waste are less because the actinides will be ultimately consumed and the hazardous life of the waste will be much reduced. However, these benefits will not be achieved with the SSTAR alone. It is assumed that the unique market for the SSTAR units is imbedded in a growing market for large breeder plants that will use the same or adjacent facilities to recycle their fuel. In fact, implementing the SSTAR is an attractive step toward future deployment of large breeder plants that require much of the same infrastructure.

*XXII-1.8. List of enabling technologies relevant to SSTAR and status of their developmen*t

The following are key enabling technologies relevant to the SSTAR, STAR-LM, and to a large extent STAR-H2:

- For the near term, application of the effective and successful Russian coolant chemistry control approach that monitors and adjusts the dissolved oxygen level in the Pb coolant to maintain the formation of protective Fe_3O_4 and other oxide layers upon steel structure without the formation of solid PbO contaminant is appropriate. However, given the long refuelling interval and simplification goals for STAR concepts, development of a passive corrosion control approach and associated structural materials for use with lead coolant should be pursued for the long term. Small-scale experiments have been conducted at ANL seeking materials that are resistant to attack by molten Pb [XXII-15]. Research on the effects of additives that inhibit corrosion has been carried out at a number of organizations through the years. Suitable materials or additives have not yet been identified;

- Development and demonstration of the S-CO_2 Brayton cycle power converter. Turbine and compressors need to be developed for utilization in S-CO_2 Brayton cycle tests. A key question concerns whether an axial flow compressor can be used near the critical point. The recuperators and cooler in the cycle are compact Printed Circuit Heat Exchangers (PCHEs) for which the performance and efficiency with S-CO_2 needs to be validated. A complete S-CO_2 Brayton cycle power converter including a heat source, turbine, load, high and low temperature recuperators, compressors, cooler, and supporting components for control and operation needs to be constructed and demonstrated at a sufficiently large scale;
- Development of integrated, software-based modularization/ factory fabrication/ logistics and rapid site assembly technologies for overcoming the loss of economy-of-scale. These technologies are already commercialized in shipbuilding, ocean oil rig, aircraft, and other industries, but need to be adapted to the small nuclear reactor field;
- Technology for performing in-service inspection under lead coolant;
- Technology for refuelling the core cassette and cooling the cassette during refuelling operations and shipment. Schemes are under investigation at ANL and The Ohio State University in the USA;
- Initially, SSTAR core loadings can be based on uranium nitride fuel or U/ transuranic/ nitride fuel using transuranics recovered from LWR spent fuel. In the longer term, for recycle of SSTAR spent fuel returns (after 20 years), development and demonstration of electrometallurgical reprocessing for transuranic nitride fuel will be required. A key requirement is to recover the enriched N [XXII-15]. Some theoretical work on fuel recycle and small-scale experiments have been conducted in Japan mainly at the Japan Atomic Energy Research Institute (JAERI).

XXII-1.9. Status of R&D and planned schedule

Development of SSTAR has been supported under the lead fast reactor element of the U.S. Department of Energy Generation IV Nuclear Energy Systems Initiative. Development of the SSTAR small modular fast reactor under Generation IV involves LFR-related funding at Argonne National Laboratory, at Lawrence Livermore National Laboratory, at Los Alamos National Laboratory, and at the Idaho National Laboratory. Development and design of the STAR-LM and STAR-H2 concepts at Argonne National Laboratory was previously supported by U.S. Department of Energy Nuclear Energy Research Initiative (NERI) projects. Institutions involved in the STAR portfolio research and development together with ANL are Oregon State University, Texas A&M University, and The Ohio State University.

As part of the Generation IV work on the SSTAR, it was proposed in 2003 that a lead-cooled demonstration test reactor could be designed, constructed, and ready for operation by about 2015. There is considerable interest in a license-by-test approach that makes use of a demonstration test reactor. The demonstrator would subsequently be operated to support SSTAR commercial deployment in about 2025.

Currently, funding is not available at a level sufficient to make design, construction, and initial operation of a demonstration test reactor feasible within a 2015 timeframe; and there remains substantial uncertainty as to what funding priorities the U.S. Department of Energy would place on this concept.

XXII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

A demonstration prototype test reactor is needed to perform the research, development, and demonstration required to support commercially deployable designs of the SSTAR.

The SSTAR design incorporates a number of specific attributes that justify demonstration in a reactor test facility including:

- Nitride fuel bonded by molten lead to ferritic-martensitic stainless steel cladding;
- Long core lifetime of 15 to 30 years;
- Open-lattice core thermal hydraulics. The velocities through the core from natural circulation are low (e.g. u_{mean}~0.9 m/s). The core design has a significant power spatial distribution (peaking factor of 1.63) that gives rise to significant temperature and velocity profiles across the core;
- Autonomous power-driven natural circulation heat transport. The fast neutron spectrum core has strong reactivity feedbacks to changes in the coolant and fuel temperatures that enable autonomous changes in power in response to changes in load demand/ heat removal (i.e., autonomous load following) and passive shutdown. The coolant flow rate as well as the temperature rise through the core (and, hence, the core coolant temperatures) are directly dependent upon the reactor power. Thus, a perturbation in coolant flow or temperature can cause a perturbation in the core power that, in turn, affects the fuel temperatures, coolant temperatures, and coolant flow rate. Crucial questions, therefore, concern the potential for development of oscillations in power, system temperatures, and flow, and whether small perturbations in power, temperature, or flow are unstable and can grow to significant magnitude. Demonstration of stable behaviour during operation is required;
- Autonomous load following and passive safety. Tests that demonstrate autonomous operation and passive shutdown are required. The core design might incorporate mechanical design features that enhance the negative reactivity feedback from core radial expansion/ flowering to optimize the system temperatures during autonomous changes in power; in particular, the core outlet temperature may be maintained approximately unvarying during autonomous power changes. Such enhancement is not required for the current SSTAR concept;
- Start-up strategy. Demonstration is required of the approach by which the reactor is taken from a deeply subcritical state with external heating sufficient to maintain the Pb coolant in a molten state to the nominal fission power level and primary coolant natural circulation flow;
- Refuelling of the core cassette/cartridge. Successful demonstration of refuelling operations involving the single assembly core cassette is required;
- Lead-to-S-CO_2 heat exchangers. Demonstration of the performance and reliability of the Pb-to-CO_2 heat exchangers is required;
- Removal of reactor afterheat. Tests are required that demonstrate passive natural circulation air cooling of the outside of the containment/ guard vessel, removal of reactor afterheat, and transport to the atmosphere following termination of heat removal through the in-reactor Pb-to-CO_2 heat exchangers;
- Supercritical carbon dioxide Brayton cycle power converter. No information has been found that a commercial scale S-CO_2 gas turbine Brayton cycle power converter has ever been operated. A demonstration of this new power conversion technology coupled to the SSTAR is required.

A SSTAR prototype sized the same as the ultimately selected reference production unit, somewhere between 10 MW(e) and 100 MW(e), is recommended. Because of its small size, a SSTAR prototype would be relatively inexpensive, and in analogy with EBR-II and BOR-60, a massive payoff in technology acquisition would accrue at a relatively low cost. The prototype design would be very close to the production model but would be more heavily instrumented and may include non-prototypical features to implement test servicing and inspections that would not be conducted in the standard design.

Ideally, the test demonstrator reactor would be sufficiently close to the commercially deployable SSTAR design, that it could be used for a design certification license-by-test under 10 CFR 52 – proceeding to authorize construction of multiple replicate commercial SSTAR reactors. This licensing approach is being encouraged in the U.S. as a way to demonstrate significantly improved safety margins that the U.S. NRC desires in advanced reactors. It would also be part of the support for the factory assembly of standard designs that could be used at qualified sites around the world. The demonstration prototype would be capable to experience all of the operational and safety transients, including selected transients that were not protected with scram. Thus, it is expected that the design will have a demonstrated level of safety that has only been demonstrated in EBR-II [XXII-16]. This testing will also confirm that there are parameter measurements that can be made on the production models to assure that adequate safety performance is retained throughout its long life. The demonstration prototype, at a power level between 10 MW(e) and 100 MW(e), can serve as a stepping stone to larger plant sizes if these are determined to be desirable. Based on past sodium reactor experience (EBR-II to FFTF and BOR-60 to BN350), a scale up factor of 6 or 7 may be possible, if the higher power level designs retain the similar features to the tested prototype. It is likely that the first prototype of larger plants would also have to demonstrate its safety performance to be approved as a standard design.

XXII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

Other SMRs closely related to the SSTAR are:

- STAR-LM – Similar to SSTAR at a power rating of 400 MW(th) and 180 MW(e);
- STAR-H2 for production of hydrogen and potable water at a power level of 400 MW(th).

XXII-2. Design description and data for SSTAR

XXII-2.1. Description of the nuclear systems

Reactor core and fuel design

The main design challenges given the small SSTAR power rating stem from the goal to simultaneously retain small burn-up control swing (<1\$); achieve high discharge burn-up (approaching 100 MWt·day/kg) while retaining natural circulation primary coolant flow. It was necessary to employ a very high fuel volume fraction (55 volume %) to achieve adequate neutron economy and to use a very large pin diameter (2.5 cm) to compensate the resulting small pin pitch-to-diameter ratio – thereby achieving a large enough hydraulic diameter to support natural circulation in a rail shippable sized vessel of less that 18 m height. At an average power density of 69 kW/l, it was possible to attain an average discharge burn-up of 72 MW·day/kg HM over a 20 year refuelling interval within a reactivity burn-up swing of less than one dollar.

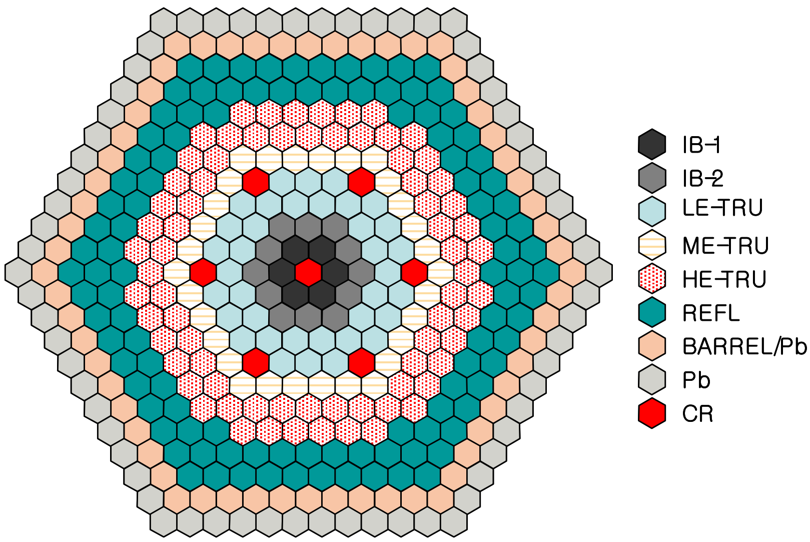
To avoid reactivity loss due to ^{241}Pu decay (~14.5 year half-life) at low power density, the start-up fuel isotopic spectrum has to be low in ^{241}Pu content. Transuranics recovered from LWR spent fuel with no less than 25 years cooling were found to be suitable (subsequently, fuel recycled from the SSTAR used refuelling cassettes in a closed fuel cycle naturally contains a low ^{241}Pu content).

Given the small core size, neutron leakage and power peaking were significant. It was necessary to employ a central blanket incorporating two low-enrichment zones and three radial enrichment zones in the driver fuel to achieve satisfactory power peaking factors.

Once a suitable core design was found, [XXII-2, XXII-3] the remainder of the reactor and power plant design was a straightforward re-optimization of the STAR-LM design to the reduced power rating of SSTAR [XXII-2].

Figure XXII-7 shows the SSTAR core map. The fuel lattice consists of cylindrical fuel rods arranged on a triangular pitch (the hexagonal geometry does not imply that the core is formed of individual hexagonal fuel assemblies or bundles; it merely reflects the assumed nodalization used for neutronics modelling.) A central two low enrichment zones blanket, the three enrichment zones, and locations for shutdown and control rods are indicated in the figure.

Also shown in the core map is a radial reflector assumed to consist of a 50 volume % HT9-50 volume % Pb mixture. The steel shroud surrounding the core is also represented by this region. A steel containing reflector is necessary to reduce the fast neutron fluence at the reactor vessel (lead is a superior gamma shield, but has a low effectiveness in shielding in-vessel structures from fast neutrons.) Flowing lead in the downcomer between the shroud and the reactor vessel is also modelled in the neutronics analysis.



IB-1, IB-2, LE-TRU, ME-TRU, HE-TRU = different enrichment zones of active core
REFL = radial reflector consisting of 50 volume % stainless steel and 50 volume % lead
BARREL/Pb = core barrel/shroud and lead coolant surrounding reflector
Pb = lead coolant in downcomer; and CR = control rod locations

FIG. XXII-7. Core map of the TRU SSTAR core with 20 year refuelling interval.

Table XXII-3 presents the design conditions and neutronics performance results. Figure XXII-8 shows the change in reactivity vs. exposure time – with a peak to minimum swing of less than 1$ over 20 effective full power years (EFPY).

TABLE XXII-3. SSTAR CORE CONDITIONS AND PERFORMANCE

Core diameter, m	1.02
Active core height, m	0.8
Nitride fuel smeared density, %	90
Fuel volume fraction	0.55
Cladding volume fraction	0.16
Bond volume fraction	0.10
Coolant volume fraction	0.28
Fuel pin diameter, cm	2.5
Fuel pin pitch-to-diameter ratio	1.121
Cladding thickness, mm	1.0
Average power density, W/cm^3	69
Specific power, kW/kg HM	10
Peak power density, W/cm^3	119
Average discharge burn-up, MWd/kg HM	72
Peak discharge burn-up, MW d/kg HM	120
Peak fast fluence, n/cm^2	4.0×10^{23}
BOC to EOC burn-up swing, % $\Delta\rho$	0.13
Maximum burn-up swing, % $\Delta\rho$	0.36
Estimated delayed neutron fraction	0.00375
BOC to EOC burn-up swing, $	0.35
Maximum burn-up swing, $	0.96

Kinetics parameters and reactivity feedback coefficients are shown in Table XXII-4 (middle-of-cycle (MOC) is defined as the time at which the multiplication factor attains a maximum). It may be noted that for this small core, the coolant void worth is negative throughout life – at BOL due to radial leakage out of the core and at EOL due to radial leakage into the internal blanket – as the radial power profile shifts from out to in with burn-up.

TABLE XXII-4. REACTIVITY COEFFICIENTS OF 45 MW(th) SSTAR

	BOL	MOL	EOL
Delayed neutron fraction	0.0035	0.0034	0.0034
Prompt neutron lifetime (s)	1.8E–07	1.8E–07	1.8E–07
Coolant density (cents/°C)	–0.002	0.003	0.002
Fuel density (cents/ °C)	–0.28	–0.28	–0.28
Structure density (cents/ °C)	0.03	0.04	0.04
Radial expansion (cents/ °C)	–0.16	–0.16	–0.16
Axial expansion (cents/ °C)	–0.06	–0.06	–0.06
Control rod worth ($/cm)	–0.08	0.16	–0.12
Doppler (cents/ °C)	–0.12	–0.12	–0.11
Coolant void worth ($)	–0.99	–0.45	0.71

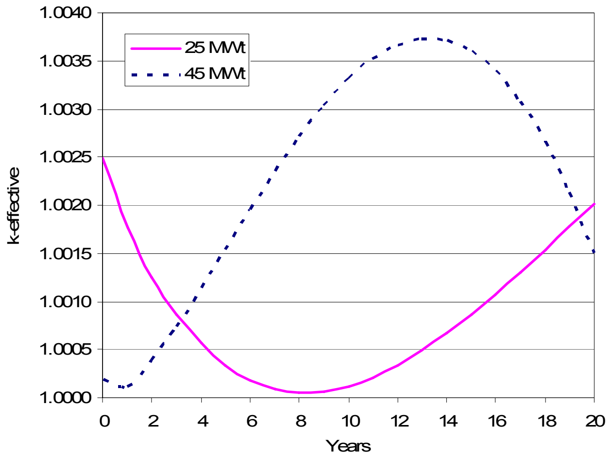


FIG. XXII-8. Burn-up swing of a 25 MW(th) SSTAR and a 45 MW(th) SSTAR design.

Main heat transport system

The SSTAR thermal-hydraulic development has been carried out to meet the following set of requirements and constraints:

- Power level = 45 MW(th);
- Full transportability of the vessel by barge or rail, or road, if possible;
- Natural circulation heat transport of primary coolant at power levels up to and exceeding 100% nominal;
- Core dimensions and fuel volume fraction from the core neutronics analyses;
- Peak cladding temperature equal to 650°C;
- Maximize S-CO_2 Brayton cycle efficiency;
- Pb coolant channels about 1 cm or more in diameter to reduce potential for plugging by contaminants;
- Space for incorporation of a cylindrical liner and annular gap escape path for CO_2 vapour/ gas between in-vessel Pb-to-CO_2 heat exchangers and reactor vessel inner surface;
- Space for multi-plate thermal radiation heat shield between bottom of upper head/cover and Pb free surface;
- Adequate coolant temperature margin above the freezing temperature;
- Removal of decay heat from outside of guard/containment vessel to the atmospheric heat sink by natural circulation of air.

In general, vessel size is constrained by conflicting goals. Rail transportability imposes a size limitation on the reactor vessel and guard vessel of 6.1 m (20 feet) in diameter and 18.9 m (62 feet) in height [XXII-17]. Alternately, the vessel height (18.3 m) and diameter (3.23 m) must be sufficient to fit the following components inside of the vessel and to provide sufficient thermal centres separation driving head for single-phase natural circulation heat transport between the elevations of the in-reactor heat exchangers and the active core:

- 1.02 m active core diameter;
- 0.297 m reflector thickness;

- 2.54 cm core shroud thickness interior to downcomer;
- 5.72 cm thick gap between reactor vessel inner surface and 1.27 cm thick cylindrical liner to provide an escape path to the Pb free surface for CO_2 void, in the event of HX tube rupture;
- 5.08 cm thick reactor vessel;
- Kidney-shaped Pb-to-CO_2 heat exchangers must fit inside of the annulus between the shroud and reactor vessel, and provide sufficient heat exchange performance to realize a significant Brayton cycle efficiency.

In addition, there is an incentive to reduce the reactor and guard vessel heights to reduce the Pb mass, further reduce capital costs, and improve seismic performance. The fuel pin cladding outer diameter is selected to minimize the peak cladding temperature given the fixed fuel volume fraction and fuel smeared density from the neutronics design and a fixed Pb core inlet temperature. Figure XXII-9 shows the relationship between the fuel pin pitch-to-diameter ratio for a triangular lattice and the fuel pin diameter, for a fixed fuel volume fraction equal to 0.55, smear density of 90%, and cladding thickness equal to 1.0 mm. The dependency of the peak cladding temperature (PCT) upon the fuel pin diameter for different core inlet temperatures is presented in Fig. XXII-10. A fuel pin diameter of 2.5 cm is selected. Figure XXII-11 shows the dependency of peak cladding temperature upon core inlet temperature; for a 2.5 cm pin diameter; the PCT equals 650°C for a core inlet temperature of 420°C. As observed from Fig. XXII-12, the peak fuel centreline temperature is equal to 953°C for the selected core inlet temperature, core power rating and fuel pin diameter.

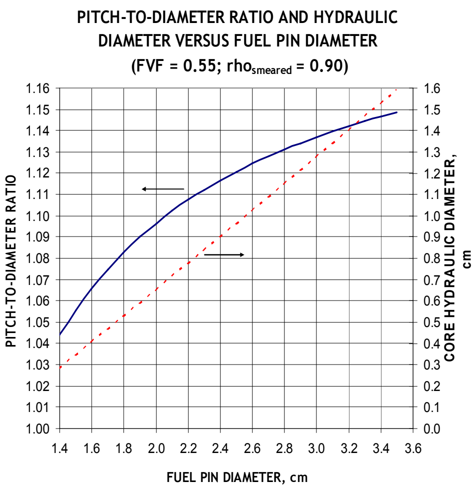


FIG. XXII-9. *Relationship between fuel pin diameter and triangular pitch-to-diameter ratio.*

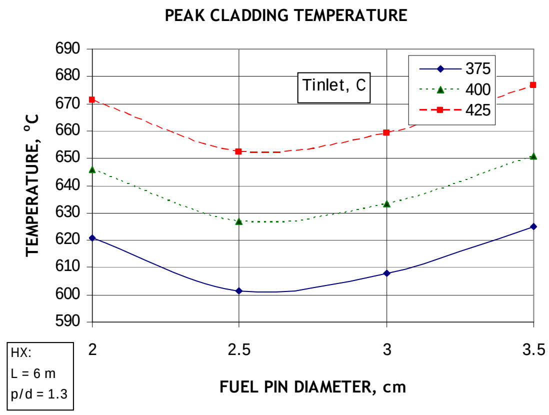


FIG. XXII-10. *Peak cladding temperature versus fuel pin outer diameter.*

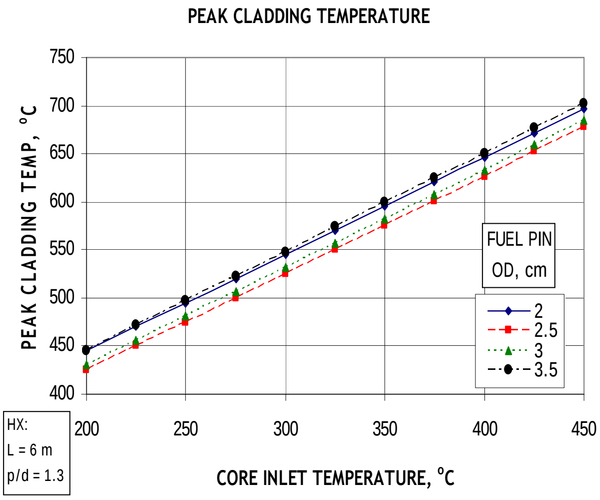


FIG. XXII-11. Peak cladding temperature versus core inlet temperature.

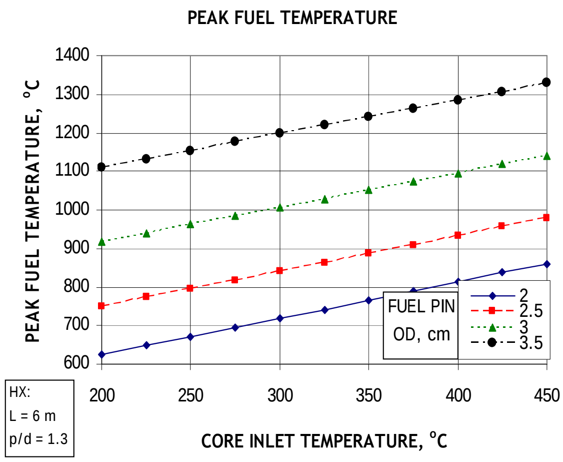


FIG. XXII-12. Peak fuel centreline temperature versus core inlet temperature.

Table XXII-5 lists thermal hydraulic conditions calculations for the SSTAR.

The SSTAR reactor is coupled to a supercritical carbon dioxide (S-CO_2) Brayton cycle power converter. It provides higher cycle efficiency than a helium ideal gas Brayton cycle or a Rankine saturated steam cycle operating at the same core outlet temperature. A key contributor to the high efficiency is the low amount of work (PdV work) to compress S-CO_2 immediately above its critical temperature – due to the high S-CO_2 density. Table XXII-6 compares the densities of S-CO_2 at cycle conditions versus those for helium in the Eskom Pebble Bed Modular Reactor (PBMR) as well as typical liquid coolants; the S-CO_2 density is more like that of an ordinary liquid. Thus, the S-CO_2 temperature and pressure at the low end of the cycle are designed close to but slightly greater than the critical temperature (30.98°C) and pressure (7.373 MPa) to exploit the small PdV work of compression.

Figure XXII-13 is a schematic of SSTAR coupled to the S-CO_2 Brayton cycle showing the heat transfer paths as well alternative control mechanisms for the S-CO_2 Brayton cycle. To facilitate S-CO_2 Brayton cycle components to be isolated for replacement, maintenance, or repair, a shutdown cooling compressor to circulate CO_2 through the in-reactor heat exchangers and a shutdown cooler to reject decay heat is provided.

TABLE XXII-5. SSTAR THERMAL-HYDRAULIC CONDITIONS

Power, MW(e) (MW(th))	19.8 (45)
Reactor vessel height, m (feet)	18.3 (60.0)
Reactor vessel outer diameter, m (feet)	3.23 (10.6)
Active core diameter, m (feet)	1.02 (3.35)
Active core height, m (feet)	0.80 (2.62)
Active core height-to-diameter ratio	0.8
Fuel volume fraction	0.55
Fuel pin outer diameter, cm	2.5
Fuel pin pitch-to-diameter ratio	1.121
Core hydraulic diameter, cm	0.964
Cladding thickness, mm	1.0
Fuel smeared density, %	90
HX tube height, m	6.0
HX tube outer diameter, cm	1.4
HX tube inner diameter, cm	1.0
HX tube pitch-to-diameter ratio	1.242
HX hydraulic diameter for Pb flow, cm	0.983
HX-core thermal centres separation height, m	12.2
Peak fuel temperature, °C	953
Peak cladding temperature, °C	650
Core outlet temperature, °C	566
Maximum S-CO_2 temperature, °C	553
Core inlet temperature, °C	420
Core coolant velocity, m/s	0.896
Pb coolant flow rate, kg/s	2125
CO_2 flow rate, kg/s	242
CO_2 mass in Brayton cycle, kg	8712
S-CO_2 Brayton cycle efficiency, %	44.4
Plant efficiency, %	44.0

TABLE XXII-6. COMPARISON OF DENSITIES

FLUID	LOCATION	PRESSURE, MPa	TEMPERATURE, °C	DENSITY, KG/m^3
S-CO_2 (STAR-LM)	Critical point	7.37	30.98	468
	Cooler outlet	7.40	31.25	369
	Compressor outlet	20.0	84.0	567
	Turbine inlet	19.88	564	121
	Turbine outlet	7.46	439	55.4
Helium (Eskom PBMR)	Cooler outlet/ Compressor inlet	2.6	27	4.17
	Compressor outlet	7.0	104	8.93
Water		0.1	20	998
Lead		0.1	495	10400
Sodium		0.1	420	828

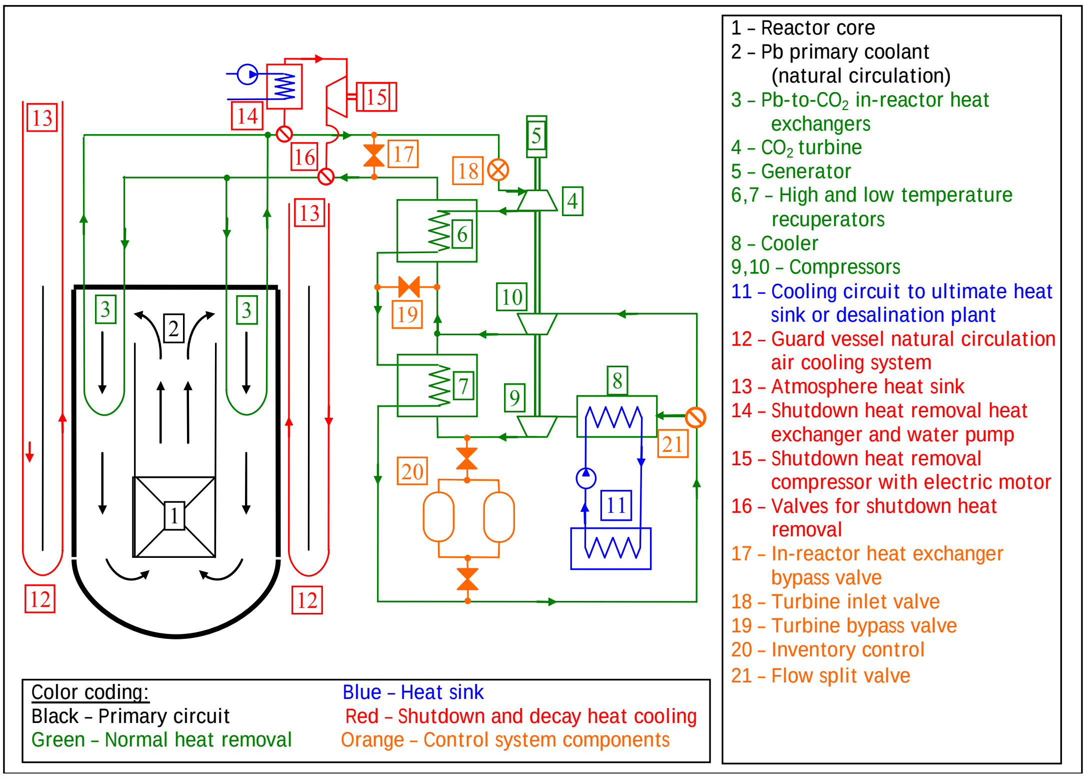


FIG. XXII-13. Schematic illustration of SSTAR coupled to S-CO_2 Brayton cycle showing normal, shutdown, and emergency heat transfer paths.

XXII-2.2. Description of the turbine generator plant and systems

The turbine and two compressors are connected via a common shaft. This enhances the cycle efficiency and reduces the required generator power. Conditions for the turbine and compressors are presented in Table XXII-7. The turbo-machinery components are of remarkably small size – suggesting that the plant footprint might be reduced relative to a ideal gas helium Brayton cycle or a Rankine saturated steam cycle.

TABLE XXII-7. RESULTS OF TURBINE AND COMPRESSOR ANALYSES FOR 45 MW(th) SSTAR

	TURBINE	COMPRESSOR No. 1	COMPRESSOR No. 2
Power, MW	31.2	4.83	5.74
Number of stages	5	10	10
Rotational speed, rev/s	180	180	180
Length without casing, m	0.41	0.26	0.13
Maximum diameter without casing, m	0.38	0.14	0.21
Minimum hub diameter, m	0.214	0.108	0.184
Maximum hub diameter, m	0.288	0.116	0.194
Minimum blade height, cm	3.1	1.3	0.5
Maximum blade height, cm	4.4	1.9	1.0
Minimum blade chord, cm	3.3	1.2	0.6
Maximum blade chord, cm	5.1	1.5	0.8
CO_2 flow rate, kg/s	242	162	80
Efficiency without secondary losses, %	96.0	92.4	90.5
Assumed secondary losses, %	5.0	5.0	5.0
Net efficiency, %	91.0	87.4	85.5

The two recuperators and cooler are Printed Circuit Heat Exchangers (PCHEs) [XXII-18]. It is assumed in the analysis that 1.0 mm semicircular channels are chemically etched into plates that are subsequently hot iso-statically pressured together at sufficiently high temperature and pressure to form a highly compact, rugged heat exchanger. Use of PCHEs offers the potential for up to a factor of three savings in the recuperator and cooler volumes relative to traditional shell-and-tube heat exchangers. It also offers the potential for enhanced reliability and reduced requirements for inspection through elimination of the concerns about tube failures typical of shell-and-tube heat exchangers.

It has been assumed that the etched-plate manufacturing process limits the plate width to 0.6 m. To obtain the required heat exchange area, twelve such PCHEs are incorporated to realize each of the high temperature recuperator (HTR), low temperature recuperator (LTR), and cooler heat exchanger units. A concept was developed whereby the three components are assembled from three transportable modules. Each module consists of twelve PCHEs in total: four PCHEs with 2.0 m long channels belonging to the high temperature recuperator (located at the top); four PCHEs with 2.0 m long channels belonging to the low temperature recuperator (in the middle); and four PCHEs with 1.6 m long channels of the cooler (at the bottom). A steel space frame supports the PCHEs of each transportable module.

Figure XXII-14 shows a plan view of an arrangement of S-CO_2 Brayton cycle components inside of the turbine generator building. The turbine and two main compressors are housed inside of a power conversion unit coupled to a commercially available generator through a gearbox. The inventory control volume consists of a number of cylindrical tanks connected to manifolds. Pressures, temperatures and flow rates calculated for the Pb and S-CO_2 circuits are shown on the schematic in Fig. XXII-3 in the beginning of this design description. A S-CO_2 Brayton cycle efficiency of 44.4% is calculated. Subtracting off the pumping power requirement for the cooling water flowing through the cooler where heat is rejected from the cycle, a net plant efficiency of 44.0 % is obtained.

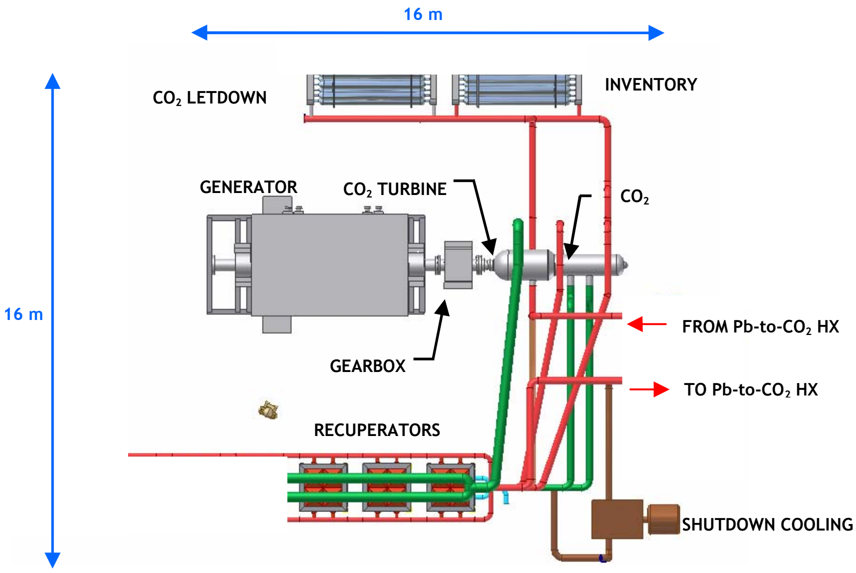
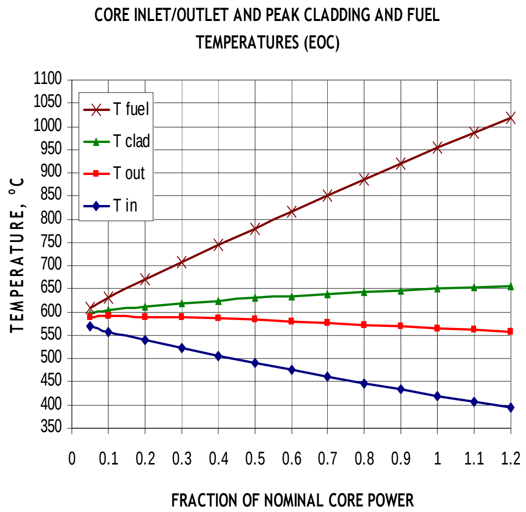


FIG. XXII-14. SSTAR S-CO_2 Brayton cycle layout.

As it was already mentioned, the strong reactivity feedbacks of the fast neutron spectrum core enable autonomous load following whereby the reactor power adjusts itself to match heat removal from the primary coolant solely as a consequence of inherent physical phenomena. Figures XXII-15 and XXII-16 show the autonomous load following behaviour calculated for the SSTAR. In particular, Fig. XXII-15 presents the system temperatures following an autonomous change in power from the nominal steady state to a new steady state. The asymptotic core outlet temperature is observed to rise mildly while the peak cladding temperature decreases as the steady state power is reduced. Thus, under autonomous operation, the steady state maximum cladding temperature at all partial power remains less than that at 100% power. Contributions from individual reactivity feedbacks (Doppler, axial expansion, core radial expansion, and coolant density reactivity feedback) are shown in Fig. XXII-16 where the small reactivity feedback from coolant density changes is evident, and neutron leakage changes due to core radial expansion comprise the dominant thermo-structural reactivity feedback to compensate changing Doppler as fuel temperature changes in response to power density changes.



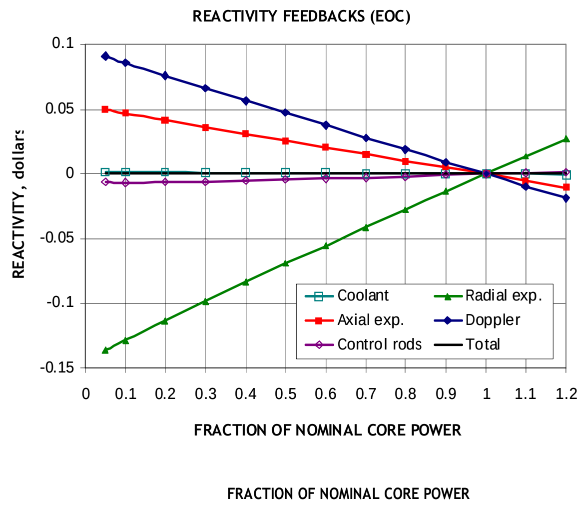


FIG. XXII-15. System temperatures following an autonomous power change from nominal power to a new steady state at end-of-core life (EOC).

FIG. XXII-16. Contributions from individual reactivity feedbacks for an autonomous power change from nominal power to a new steady state at end-of-core life (EOC).

XXII-2.3. Systems for non-electric applications

A desalination plant can optionally be added downstream of the low temperature recuperator – replacing the Brayton cycle cooler.

XXII-2.4. Plant layout

The plant layout is currently under development. The following are considerations in the general philosophy that governs development of the plant layout:

- Seismic isolation using seismic isolators and a seismic island, see Fig. XXII-1. Given the small size of the S-CO_2 Brayton cycle secondary side components (see Fig. XXII-14), a design objective is to locate the reactor, containment, and secondary side all on a common seismic island. This approach would reduce the effects of differential motion between S-CO_2 secondary side components and the S-CO_2 piping to and from the reactor;

- Resistance against an aircraft crash and malevolent human-induced acts; perhaps with silo emplacement under an earthen berm;

- Provision for core cassette refuelling using refuelling equipment including a heavy lift crane that is brought to the site only at the end of the core lifetime and is removed from the site following refuelling; and

- Modular assembly of components at the site to reduce construction/ assembly costs and time. It is planned to use 4^+ D CAD/CAM visualization for development and planning of modularization and site assembly sequences in the development of modules and a plant layout.

REFERENCES

[XXII-1] SIENICKI, J.J., et al., "STAR Performer", Nuclear Engineering International, p. 24 (July 2005).

[XXII-2] SIENICKI, J.J., MOISSEYTSEV, A.V., SSTAR lead-cooled, small modular fast reactor for deployment at remote sites – system thermal hydraulic development, ICAPP 2005, Paper 5426 (Int. Conf. on Advances in Nuclear Power Plants Seoul, May 15–19, 2005).

[XXII-3] YANG, W.S., et al., Lead-cooled, long-life fast reactor concepts for remote deployment, ICAPP 2005, Paper 5102 (Int. Conf. on Advances in Nuclear Power Plants Seoul, May 15–19, 2005).

[XXII-4] SIENICKI, J.J. et al., SSTAR lead-cooled, small modular fast reactor with nitride fuel, ARWIF-2005, Paper 2.03 (OECD Nuclear Energy Agency Workshop on Advanced Reactors with Innovative Fuels, Oak Ridge, February 16–18, 2005).

[XXII-5] SIENICKI, J.J., WADE, D.C., Non-proliferation features of the small secure transport autonomous reactor (SSTAR) for worldwide sustainable nuclear energy supply, American Nuclear Society 2005 Winter Meeting (paper presented at meeting in Washington, November 13–17, 2005).

[XXII-6] SIENICKI, J.J., Safety and licensing aspects of the LFR/SSTAR, American Nuclear Society 2004 (Annual Meeting, Pittsburgh, June 13–17, 2004), Transactions of the American Nuclear Society, Vol. 90, p. 24 (2004).

[XXII-7] MOISSEYTSEV, A., SIENICKI, J.J., Control of supercritical CO_2 Brayton cycle for LFR autonomous load following, American Nuclear Society 2005 Winter Meeting (paper presented at meeting in Washington, November 13–17, 2005).

[XXII-8] MOISSEYTSEV, A., SIENICKI, J.J., WADE, D.C., Lead-to-CO2 heat exchangers for coupling of the STAR-LM LFR to a supercritical carbon dioxide Brayton cycle power converter, ICONE-12 (12th Int. Conf. on Nuclear Engineering, Arlington, April 25–29, 2004), Paper ICONE12-49303.

[XXII-9] MOISSEYTSEV, A., SIENICKI, J.J., WADE, D.C., Turbine design for a supercritical carbon dioxide gas turbine Brayton cycle, ICAPP '03, 2003 (Proc. Int. Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4–7, 2003), Paper 3064.

[XXII-10] MOISSEYTSEV, A., SIENICKI, J.J., WADE, D.C., Cycle analysis of supercritical carbon dioxide gas turbine Brayton cycle power conversion system for liquid metal-cooled fast reactors, ICONE-11 (11th Int. Conf. on Nuclear Engineering, Tokyo, April 20–23, 2003), Paper ICONE 11-36023.

[XXII-11] ILINČEV, G., Research results on the corrosion effects of liquid heavy metal Pb, Bi and Pb-Bi on structural materials with and without corrosion inhibitors, Nuclear Engineering and Design, Vol. 217, pp. 167–177 (2002).

[XXII-12] GE Nuclear Energy (ALMR), (1994 Capital and Busbar Cost Estimates), GEFR-00940, UC-87Ta (March 1995).

[XXII-13] UEDA, N., KINOSHITA, L., MINATO, A., KASAI, S., MARUYAMA, S., Design Study of Sodium Cooled Small Fast Reactor, GENES4/ANP2003, Paper 1114 (Kyoto, September 15–19, 2003).

[XXII-14] FARMER, M.T., SIENICKI, J.J., Analysis of transient coolant void formation during a guillotine-type HX tube rupture event in the STAR-LM system employing a supercritical Brayton cycle, ICONE-12 (12th Int. Conf. on Nuclear Engineering, Arlington, April 25–29, 2004), Paper ICONE12-49227.

[XXII-15] RUNGE, J.M., LEIBOWITZ, L., BARNES, L.A., RARAZ, A.G., MCDEAVITT, S.M., Performance of structural materials in lead-based reactor

coolants at temperatures up to 800 EC, GLOBAL 2003 (Proc. Int. Winter Meeting, New Orleans, November 16–20, 2003), p. 2093, ANS/ENS.
[XXII-16] The Experimental Breeder Reactor-II Inherent Safety Demonstration, EBR-II Division, Argonne National Laboratory, April 1986, Nuclear Engineering and Design, Special Issue, ed. Fistedis, S.H., Vol. 101, pp. 1–91 (1987).
[XXII-17] GYOREY, G.L., HARDY, R.W., MAGEE, P.M., Safety design for the advanced liquid-metal-cooled reactor, Nuclear Safety, Technical Progress Journal, Oak Ridge National Laboratory, USA, Vol. 31, p. 323 (1990).
[XXII-18] DEWSON, S.J., THONON, B., The development of high efficiency heat exchangers for helium gas cooled reactors, Advances in Nuclear Power Plants, ICAPP'03 (Proc. Int. Congress Cordoba, Spain, May 4–7, 2003), Paper 3213. See also www.heatric.com .

ANNEX XXIII

SECURE TRANSPORTABLE AUTONOMOUS REACTOR - LIQUID METAL (STAR - LM)

Argonne National Laboratory (ANL),
United States of America

XXIII-1. General information, technical features and operating characteristics

XXIII-1.1. Introduction

The Secure Transportable Autonomous Reactor - Liquid Metal (STAR-LM) project at Argonne National Laboratory (ANL, USA) was undertaken to develop a modular nuclear power plant for electric power production with optional production of desalinated water that meets the requirements of a future sustainable world energy supply architecture optimized for nuclear rather than fossil energy [XXIII-1]. Those requirements include features that facilitate deployment in developing as well as developed countries such as enhanced proliferation resistance, sustainability, economy, nearly autonomous operation, and a range of plant power levels compatible with widely varying extents of national nuclear infrastructure and local electric grid development.

Awareness of the potential benefits of heavy liquid metal coolants followed the declassification of aspects of the history and technology of their utilization for submarine propulsion in the Russian Federation [XXIII-2, XXIII-3]. Significantly, the feasibility of fast reactor cores cooled with heavy liquid metal is not simply a notion but has been demonstrated through the operation of two prototype land reactors and ten operational submarine reactors in naval service. Early problems ascribed to usage of the LBE coolant were eliminated through the development of coolant technology that resulted in the control of structural material corrosion together with elimination or removal of solid contaminants that could potentially result in plugging of coolant channels.

Historically, an impediment to commercial deployment of liquid metal cooled fast reactors has been relatively higher costs. The STAR-LM development has been attacking the problem of relatively higher costs on a number of fronts. The traditional Rankine steam cycle for electricity production has been replaced with a supercritical carbon dioxide (S-CO_2) gas turbine Brayton cycle in which S-CO_2 is utilized as the working fluid [XXIII-4 to XXIII-7], resulting in an increased efficiency of ~45%. The S-CO_2 Brayton cycle secondary side also offers a significant reduction in the size of the turbo-machinery components (i.e. turbine and compressors) suggesting the prospect of a reduced plant footprint. The passive load follow and passive safety features of the fast reactor core with transuranic nitride fuel and lead coolant offer the prospect of reduced operator workload requirements. The S-CO_2 Brayton cycle thus offers the prospect of reduced capital and operating costs and reduced plant staffing requirements. The STAR-LM also makes use of the potential economic benefits of factory fabrication (quality control), full transportability (barge and railroad for larger components; trucks for smaller components), modular construction (less work and assembly time at a site), and simplification (natural circulation heat transport at power levels in excess of 100%, eliminating primary coolant pumps). Factory fabrication of a standardized design is well suited to a worldwide architecture involving hundreds or thousand of reactor units.

The STAR-LM development has been directed at improvement of the LFR for commercial deployment in the future sustainable world energy supply architecture and to assist in facilitating a transition to such architecture, for which an enhanced proliferation resistance is of paramount importance. Attention has been devoted to a low power density core with an extremely long refuelling interval of 15 to 20 years or longer. The core is designed as a single large assembly/cartridge and is not composed of individual removable fuel assemblies. Thus, the long core lifetime together with the single fuel cartridge design eliminates access to the fuel during the core life. At the end of that lifetime, refuelling equipment is only temporarily brought on site to change the cartridge core. The used fuel cassette and the refuelling equipment are returned to a regional fuel cycle support centre operated under international oversight where the fuel is reprocessed. In the future architecture, regional fuel cycle support centres form the hubs of a hub-and-spoke network of support centres and associated surrounding reactor sites.

Primary stakeholders are the United States Department of Energy that has supported the development of the STAR-LM through the nuclear energy research initiative and Generation IV nuclear energy systems initiative programmes as well as Argonne National Laboratory. International nuclear energy research initiative collaborations are in place or have been proposed, pending approval with the Republic of Korea involving the Korea Atomic Energy Research Institute as well as with the European Commission involving the Joint Research Center of the European Commission, Institute for Energy (Petten, the Netherlands). Additional stakeholders are the Lawrence Livermore National Laboratory, the Los Alamos National Laboratory, and the Idaho National Laboratory that recently combined their relevant technology development efforts together with Argonne towards SSTAR development. Participating U.S. universities in collaboration with Argonne National Laboratory include Oregon State University, Texas A&M University, and The Ohio State University. In Japan, Toshiba and CRIEPI are also interested in small modular liquid metal cooled fast reactors; they have developed the 4S sodium cooled reactor [XXIII-8]. There is considerable interest in a license-by-test approach that involves operation of a demonstration test reactor facility [XXIII-9]; many of the issues involved are common to the LFR and 4S systems.

XXIII-1.2. Applications

The STAR-LM is a secure transportable autonomous reactor (STAR) variant designed for high efficiency electric power production at a power level of 400 MW(th) (178 MW(e)) with optional production of desalinated water.

The production of desalinated water utilizes reject heat from the Brayton cycle and does not degrade the S-CO_2 cycle efficiency for electricity production. The absence of a low pressure turbine and condenser, as in a Rankine steam cycle, means that the CO_2 exiting the low temperature recuperator has an elevated pressure and temperature that permits a natural coupling to the desalination plant in which the CO_2 is cooled to 31.25°C (e.g., by seawater) immediately above the CO_2 critical temperature and is then compressed to the maximum pressure.

XXIII-1.3. Special features

Special features of the STAR-LM include the following:

Option of incremental capacity increase

The incremental unit power level of 178 MW(e) provides flexibility in matching capacity to evolving demand through the deployment of additional units. The reactors require minimal electric grid modification and can be sited closer to cities.

Operation without on-site refuelling

The low power density, low pressure drop core has an extremely long refuelling interval of 15 or 20 years or longer. The core is designed as a single large assembly/cartridge and is not composed of individual removable fuel assemblies. To remove the core cartridge, it is necessary to remove the upper head/cover from the reactor vessel. This is only done at the end of the core lifetime when refuelling equipment is temporarily brought on site; the used core and refuelling equipment are transported to a regional fuel cycle support centre or move on to the next reactor site scheduled for refuelling.

Nearly autonomous operation

The strong reactivity feedback from the fast neutron spectrum core enables autonomous load following whereby the reactor power adjusts itself to match heat removal from the primary coolant solely as a consequence of inherent physical phenomena. The system temperatures that are attained following an autonomous power change from the nominal steady state can be optimized through design of the core clamping and restraint approach to enhance the negative reactivity feedback from core radial expansion/flowering, although such enhancement is not necessary with the current STAR-LM concept. Autonomous operation reduces operator workload and requirements.

Factory fabrication, transportability, and modular assembly at the site

All nuclear power plant components are factory fabricated to reduce costs and enhance quality control. The components are assembled into fully transportable modules for transport to the site by barge or rail for STAR-LM or by truck for smaller LFRs, such as the SSTAR. Assembly of modules at the site reduces construction time and costs. Similarly, modular components for the non-safety grade balance of plant can be factory fabricated, and quickly assembled at the site.

XXIII-1.4. Summary of major design and operating characteristics

The STAR-LM vertical cross-section and schematic diagram are given in Fig. XXIII-1 and XXIII-2, respectively. The STAR-LM is an indirect cycle lead cooled reactor. The traditional Rankine steam cycle has been replaced with a gas turbine recompression Brayton cycle that utilizes S-CO_2 as the working fluid. Its thermophysical properties immediately above the critical point result in a relatively low work requirement to compress the S-CO_2 fluid. The S-CO_2 gas turbine Brayton cycle power converter for electricity production provides a cycle efficiency of 44.9% at a core outlet Pb temperature of 588°C.

Figure XXIII-1 shows the primary coolant system configuration. The Pb coolant flows upwards through the core and the above-core riser region interior to the above core shroud. Coolant flows through the holes in the shroud and enters the modular in-reactor heat exchangers to flow downwards over the exterior of circular tubes arranged on a triangular pitch through which the S-CO_2 flows upwards, see Fig. XXIII-2. Heat is thus transferred from Pb to S-CO_2 in a counter current regime. The Pb exits the heat exchangers to flow downwards through the down comer to enter the reactor vessel lower head. A flow distributor head provides for an approximately uniform pressure boundary condition beneath the core. Consistent with the higher operating temperatures of the S-CO_2 Brayton cycle, Pb ($T_{Melt.}$ = 327°C; $T_{Boil.}$ = 1740°C) has been selected as the primary coolant rather than the lower melting point LBE ($T_{Melt.}$ = 125°C; $T_{Boil.}$ = 1670°C). Lead is also less corrosive to unprotected steel than Bi and LBE [XXIII-10].

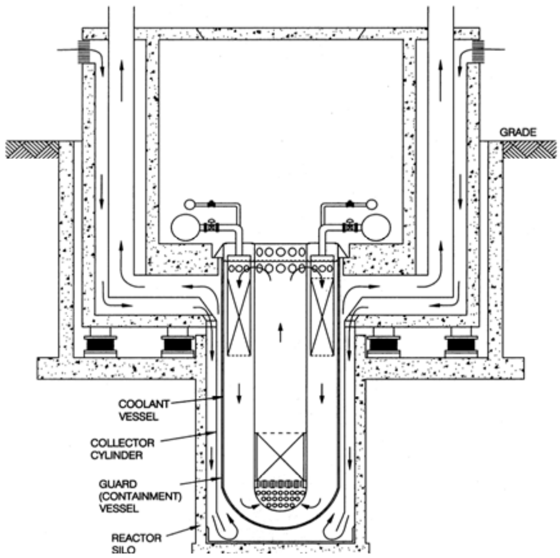


FIG. XXIII-1. Vertical cut of STAR-LM.

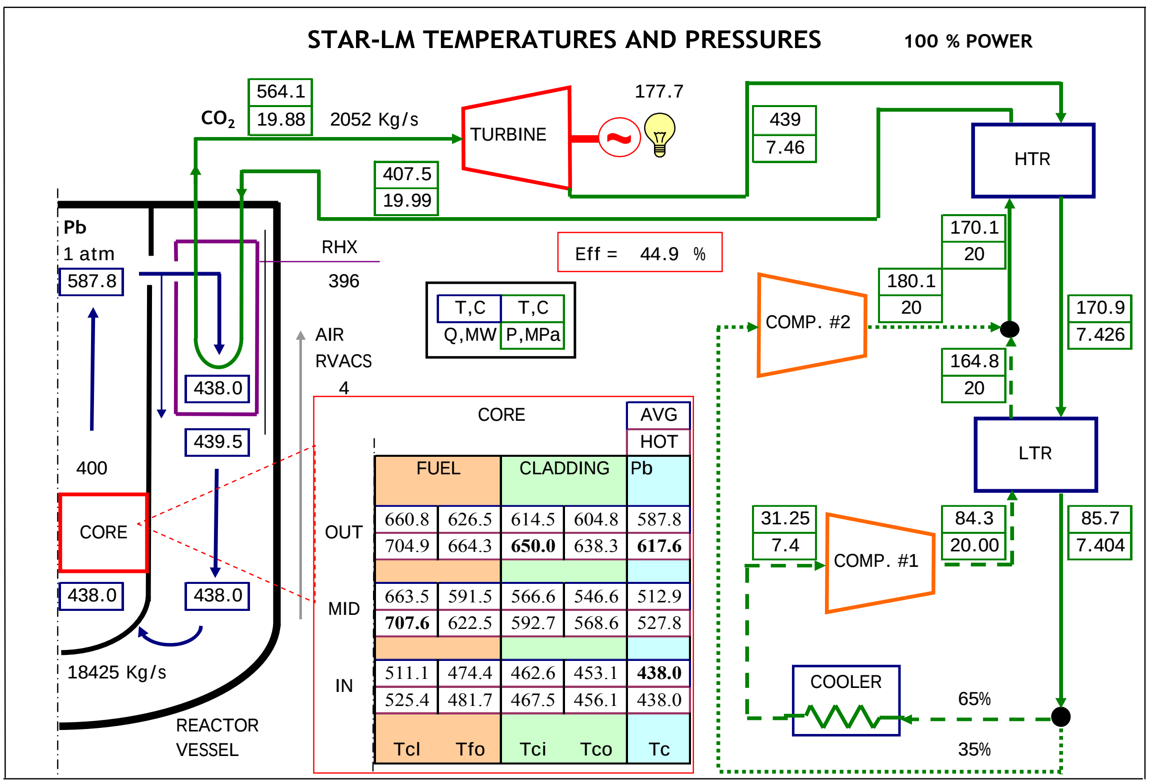


FIG. XXIII-2. Schematic of STAR-LM coupled to S-CO_2 Brayton cycle showing nominal operating temperatures and pressures.

Major design characteristics of the STAR-LM are summarized in Table XXIII-1.

TABLE XXIII-1. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS FOR 400 MW(th) STAR-LM CORE

CHARACTERISTIC	VALUE
Installed capacity	400 MW(th) (178 MW(e))
Mode of operation	Autonomous load follow
Net plant efficiency	44.4%
Load factor/availability	To be determined. Factors contributing to a potentially high load factor / availability: • Refuelling only once every 15 or 20 years. • Possible need to regenerate the coolant every N years.
Type of fuel	Transuranic nitride in clad cylindrical fuel rods. Nitrogen isotope = ^{15}N.
Fuel enrichment	Three enrichment zones: 13.30% TRU/HM, 18.22% TRU/HM and 21.28% TRU/HM
Primary coolant	Lead
Moderator	None
Core structural materials	Ferritic-martensitic stainless steel cladding and structures
In-vessel structural materials	Ferritic-martensitic stainless steel
Core Active core height Above core fission gas plenum height Below core axial reflector height Active core diameter Reflector Reflector effective thickness Core diameter with reflector	Open lattice of cylindrical fuel rods on a triangular pitch (optional square pitch) lattice. 2 m 0.5 m 0.25 m 2.46 m 50 volume % ferritic-martensitic stainless steel and 50 volume % Pb 29.7 cm 3.05 m
Reactor vessel Outer diameter Height Thickness Design lifetime	Steel cylinder with curved lower head. 5.5 m 16.9 m 5.08 cm 60 years
Cycle type	Indirect gas turbine Brayton cycle with supercritical carbon dioxide (S - CO_2)
Number of circuits	Two. Lead primary coolant circuit with natural circulation in primary circuit. Supercritical carbon dioxide secondary circuit with gas turbine Brayton cycle.
Cycle type Cycle efficiency	Indirect cycle with gas turbine Brayton cycle secondary side using supercritical carbon dioxide as the working fluid. 44.9%
NEUTRON PHYSICAL CHARACTERISTICS	
Coolant void worth	11.64 \$ BOC/12.20 \$ EOC
Burn-up reactivity swing	$k_{eff,\ max}$ - $k_{eff,\ min}$ during the cycle = 0.61%k Middle-of-cycle (MOC) is defined as the time at which the multiplication factor attains a maximum.

CHARACTERISTIC	VALUE		
Peaking factors Peak fast neutron fluence, 10^{23} n/cm^2	1.63 BOC/1.64 EOC Reduction of peaking by enrichment zoning 3.90		
KINETICS PARAMETERS AND REACTIVITY FEEDBACK COEFFICIENTS			
Parameter	BOC	MOC	EOC
Delayed neutron fraction Prompt neutron lifetime, μs	0.0035 0.534	0.0032 0.504	0.0031 0.498
Reactivity feedback coefficients Fuel Doppler, cents/°C Fuel and cladding axial expansion, \$/cm Core radial expansion, \$/cm Pb void worth, \$	 -0.12 -0.37 -0.84 11.64	 -0.11 -0.39 -0.90 12.20	 -0.10 -0.40 -0.92 12.20
Reactivity feedback coefficients, cents/°C Fuel Doppler, cents/°C Fuel and cladding axial expansion, cents/°C Core radial expansion, cents/°C Coolant density, cents/°C Enhancement of core radial expansion relative to grid spacer thermal expansion	 -0.12 -0.0685 -0.140 0.151 0.1	 -0.11 -0.0722 -0.150 0.158 0.1	 -0.10 -0.0741 -0.153 0.158 0.1
REACTIVITY CONTROL MECHANISM			
• Shutdown rod for start-up and shutdown. • During operation, reactor power autonomously adjusts to load by means of inherent physical processes without the need for any motion of control rods or any operator actions. • System temperatures change corresponding to reactivity feedbacks from fuel Doppler, fuel and cladding axial expansion, core radial expansion, and coolant density effects. • Control rods for possible fine reactivity compensation during cycle. • Need for fine reactivity compensation to be determined. • Control rods also provide for diverse and independent shutdown.			
THERMAL-HYDRAULIC CHARACTERISTICS			
Primary circuit			
Circulation type	Natural circulation of lead primary coolant. No primary coolant pumps.		
Core hydraulic diameter, cm	2.08		
Core inlet temperature, °C	438		
Core outlet temperature, °C	588		
Mean temperature rise through core, °C	150		
Peak cladding outer surface temperature, °C	638		
Peak cladding inner surface temperature, °C	650		
Average cladding inner surface temperature, °C	567		
Peak fuel centreline temperature, °C	708		
Average fuel temperature, °C	628		
Pb coolant mass flow rate, kg/s	18 400		
Mean Pb coolant velocity in core, m/s	0.605		

CHARACTERISTIC	VALUE
Mean Pb coolant density, kg/m^3	10 430
Primary coolant cover gas pressure	Slightly below 1 atmosphere
Temperature limit for cladding, °C	650
Secondary circuit	
Supercritical CO_2 pressure at top of HX, MPa	19.9
CO_2 temperature at bottom of HX, °C	408
CO_2 temperature at top of HX, °C	564
CO_2 mass flow rate, kg/s	2052
Total CO_2 circulating inventory, kg	55 500
FUEL CYCLE CHARACTERISTICS	
Maximum/average discharge burn-up of fuel	Maximum = 13.6 at. %/ 136 MWd/kg Average = 8.31 at. %/ 83.1 MWd/kg.
Fuel lifetime/period between refuellings	15 full power years
MASS BALANCES / FLOWS OF FUEL	
21470 kg depleted U, 4090 kg TRU, and 4980 kg HT9 cladding every 15 years; 3.58 kg depleted U/MW(th)/year; 0.682 kg TRU/MW(th)/year; 0.830 kg HT9 cladding/MW(th)/year); 8.13 kg depleted U/ MW(e)/year; 1.55 kg TRU/MW(e)/year; 1.89 kg HT9 cladding/MW(e)/year; Best estimate calculation using DIF3D and REBUS-3 computer codes.	
Design basis lifetime Core Reactor vessel Core shroud In-vessel structures other than core shroud	 15 years Core lifetime based upon fluence limit of 4×10^{23} fast neutrons/cm^2 for HT9 cladding. 60 years Reactor vessel lifetime based upon service temperature 15 years 60 years
Design and operating characteristics of systems for non-electric applications	Optional desalinated water production using portion of reject heat.
Economics	To be determined.

XXIII-1.5. Outline of fuel cycle options

The STAR-LM LFR utilizes transuranic nitride fuel with N^{15} in a closed fuel cycle in which the fuel cycle feedstock is depleted or natural uranium. Multiple recycle through sequential reloading cycles of the cassette/cartridge core achieves total fission consumption of the feedstock. The effluent stream contains only fission product waste forms and trace losses of the transuranics. The reactor is fissile self-sufficient with an internal core conversion ratio of unity. The fuel recycle technology is based on electrometallurgical recycle and remote vibropack refabrication of the transuranic/uranium nitride fuel. The recycle technology produces a co-mixed stream of all transuranics and achieves incomplete fission product removal such that the transuranic materials during fresh and used cassette shipping are always at least as unattractive for military use as light water reactor (LWR) spent fuel.

removal such that the transuranic materials during fresh and used cassette shipping are always at least as unattractive for military use as light water reactor (LWR) spent fuel.

In the future sustainable world energy supply architecture recycle is conducted at secure regional fuel cycle support centres. Each centre forms the hub of a network of a hub-and-spoke system of regional centres and surrounding LFRs. For early LFR reactor deployments, transuranic fissile material might be obtained from used LWR fuel. Alternatively, the initial core loading could utilize enriched uranium or excess plutonium from weapons. In a growing economy, later in the century, the increasing need for fissile material to support new LFR deployments would come from fast breeder reactors sited at regional fuel cycle support centres. Their function would be to produce excess fissile material to fuel the initial working inventories of new reactors.

XXIII-1.6. Technical features and technological approaches that are definitive for STAR-LM performance in particular areas

XXIII-1.6.1. Economics and maintainability

The STAR-LM has been designed to incorporate features that will reduce both capital and operations and maintenance (O&M) costs. One driver is the anticipated penalty in fuel costs due to the low power density (i.e. derated) core and the relatively low average burnup of 83 MWd/kg. One of the benefits of the selection of lead coolant is elimination of the need for an intermediate heat transport circuit. This simplification results in capital cost savings, mainly through reduction in the number of heat exchangers, as well as savings in O&M. Reliance upon natural circulation heat transport is another primary coolant simplification that reduces capital and O&M costs through elimination of primary coolant pumps. The core reactivity feedbacks that underlie autonomous load following simplify the requirements for the reactor control system further contributing to both capital and O&M cost reductions. The passive safety features resulting from the reactivity feedbacks could eliminate the need for the secondary heat transport circuit/balance of plant to meet safety grade requirements, thereby providing further savings. Usage of the S-CO_2 Brayton cycle with its remarkably small sized turbine and compressors provides the prospect of significantly reduced capital costs, O&M costs, and plant staffing requirements as the result of a potentially reduced plant footprint with fewer, simpler, and smaller-size components relative to the Rankine steam cycle. Furthermore, the greater efficiency achieved with the S-CO_2 Brayton cycle reduces electricity generation costs. Significant cost reductions are also expected through the use of factory fabrication to reduce costs and enhance quality control, assembly of components into fully transportable modules for transport to the site via barge or rail (STAR-LM) or by truck for smaller LFRs, and modular assembly at the site to reduce construction time as well as costs.

The STAR-LM is targeted at both developing and developed countries. Secondary side and balance of plant components could potentially be manufactured using indigenous factories and labour in some developing countries. This could potentially result in further cost reductions.

The long core lifetime enhances the plant capacity factor and minimizes labour and other costs associated with refuelling. The reactors are fissile self-sufficient and the need for additional fissile material to support the growth in the number of reactors could be met by breeder reactors located at the secure region fuel cycle support centres. The need for fuel enrichment facilities to support STAR reactors would then disappear.

XXIII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The proposed future world energy supply architecture achieves sustainability. The closed fuel cycle achieves maximum utilization of uranium resources. The fuel cycle feedstock is depleted or natural uranium. Multiple recycle through sequential reloading cycles achieves total fission consumption of the feedstock. The fuel cycle effluent stream is minimized, consisting only of fission product waste forms and trace losses of transuranics.

XXIII-1.6.3. Safety and reliability

Safety concept and design philosophy

From the outset, the design and safety philosophy of the STAR-LM has been to take advantage of the particular properties of lead coolant, nitride fuel, and a fast neutron spectrum core to achieve and ensure a strong reliance on inherent safety features and passive protection. One aspect of this philosophy is to eliminate the need for reliance upon any active systems.

Provisions for simplicity and robustness of the design

Passive safety design options were preliminarily analyzed and the analyses have shown the STAR-LM concept to be robust with respect to them. Although scram systems are provided to insert rods to shut down the reactor neutronically, success of scram is not required to prevent the evolution of adverse power or temperature conditions.

Active and passive safety systems and inherent safety features

The STAR-LM LFR system provides for ambient pressure single phase primary coolant natural circulation heat transport and removal of core power without scram under all operational and postulated accident conditions. This is a consequence of:

- The high boiling temperature of the lead heavy liquid metal coolant equal to 1740°C that realistically eliminates boiling of the low pressure coolant;
- The chemical inertness of the lead coolant that does not react chemically with carbon dioxide above about 250°C (well below the 327°C Pb melting temperature) and does not react vigorously with air or water;
- Natural circulation heat transport of the lead coolant at power levels in excess of 100% nominal that eliminates the entire class of loss-of-flow accidents;
- Transuranic nitride fuel that is chemically compatible with the lead coolant. The high nitride thermal conductivity together with bonding of the fuel and cladding with molten Pb results in low fuel centreline temperatures and small thermal energy storage in the fuel;
- External natural convection-driven passive air cooling of the guard/containment vessel (surrounding the reactor vessel) that is always in effect and removes decay heat power levels;
- Strong reactivity feedbacks from the fast neutron spectrum core with transuranic nitride fuel and lead coolant, possibly enhanced through mechanical design means, that enable autonomous load following and result in passive core power reduction to decay heat while system temperatures remain within structural limits, in the event of loss-of-normal heat removal through the in-vessel primary-to-secondary heat exchangers and without any operator action thereby eliminating the potential for

human error. There is no reliance upon the motion of control rods either due to operator action or inherent insertion due to heatup of the control rods or control rod drivelines;

- The system pool configuration and ambient pressure coolant with a reactor vessel and surrounding guard vessel that eliminates accidents with loss of primary coolant;
- The high heavy metal coolant density (Δ_{Pb}=10 400 kg/m^3) that limits void growth and downward penetration following postulated heat exchanger (HX) tube rupture such that void is not transported to the core but instead rises benignly to the lead free surface through a deliberate escape channel between the HXs and the vessel wall; and
- The compact liquid metal system that is seismically isolated on a nuclear island using seismic isolators (see Fig. XXIII-1).

In the event of HX tube rupture, a blow-down of secondary CO_2 (initially at 20 MPa) into the lead occurs. Molten lead and CO_2 do not react chemically. Protection against over-pressurization of the primary coolant vessel must be provided and activity that is entrained from the lead coolant into the CO_2 must be contained. Thus, a pressure relief system is provided for the primary coolant system. The S-CO_2 secondary circuit incorporates valves to isolate the failed heat exchanger and limit the mass of CO_2 that can enter the primary coolant system.

STAR-LM incorporates defence in depth in providing a containment that surrounds the primary coolant system. The bottom portion of the containment is the steel guard vessel that surrounds the reactor vessel. The guard vessel and reactor vessel are hermetically sealed by the upper head/cover. However, an additional containment structure is also provided above the cover. Realistic requirements for containment need to be developed. Pressurization due to depressurization/flashing of the coolant as with a LWR, is precluded by usage of the ambient pressure liquid metal primary coolant. The usage of Pb primary coolant and CO_2 secondary coolant that does not react chemically with Pb precludes the generation of hydrogen or other combustible gases as may exist with a LWR. Thus, there are no combustion or explosion hazards from the generation of combustible or explosive gases. The major consideration may be containment of activity from the lead coolant, in the event of failure of the primary coolant system boundary, or the release of CO_2 secondary coolant with activity entrained from the Pb coolant, in the event of HX tube rupture.

Design basis accidents and beyond design basis accidents

In consideration of design basis accidents and beyond design basis accidents, the set of possible accident initiators is substantially reduced relative to existing power plants. Significantly, the need to consider scenarios involving core uncovery, fuel degradation/melting, and significant fission product release does not arise.

The following preliminary list of accidents or potential accidents has been identified. Consistent with the philosophy of avoiding reliance upon active safety systems, it is assumed that automatic insertion of the shutdown rods does not occur; that is, there is an assumed failure to scram in each accident. Analyses show that unacceptable system temperatures and core damage could be averted by strictly passive means without scram.

Loss of normal heat exchanger heat removal

This is an extreme loss of heat sink accident variant in which the removal of heat from the lead coolant by the CO_2 secondary coolant through all of the lead-to-CO_2 heat exchangers is assumed to cease. It demonstrates passive safety features of the reactor.

Heat exchanger tube rupture

This scenario must be analyzed for any system that incorporates in-reactor primary-to-secondary coolant heat exchangers.

Transient overcooling

The potential existence of such scenarios needs to be investigated in connection with the realistic behaviour of the S-CO_2 Brayton cycle in response to secondary side accident initiators. Of interest is the potential for cooling part of the lead below its freezing temperature.

Loss of generator load

This scenario involves disconnect of the grid from the generator tending to increase the rotational speed of the turbine resulting in bypass of the in-reactor heat exchangers and turbine by the CO_2 flow reducing heat removal from the in-reactor HXs. The S-CO_2 Brayton cycle control system shall be designed to cope with this event.

External reactivity insertion

This may be an incredible accident for the STAR-LM, depending upon the provision and usage of control rods. Traditionally, reactivity insertion accidents result from unanticipated control rod withdrawal. The STAR-LM may operate without reactivity compensation due to fine adjustment of control rods. In such a case, control rods would not be partially inserted in the core such that inadvertent rod withdrawal could not realistically occur. In the past, this scenario has been analyzed for the STAR-LM by postulating a prescribed reactivity insertion at a prescribed ramp rate.

Passive safety features combined with the provision of a containment structure should result in an extremely low risk of radioactivity release beyond the plant boundaries. Consequently, there may be no need for an emergency-planning zone. The substantial elimination of severe accidents could eliminate the need for severe accident management.

Following an accident such as a loss of heat sink without scram in which the reactor power has passively decreased to a low level of afterheat typical of decay heat levels, it may be enough to simply return to power.* Or it may only be required for an operator to ultimately insert the shutdown rod(s) to terminate possible fission power at low afterheat levels and render the core subcritical; i.e. to ultimately shut down the reactor neutronically. Until this action is taken, the reactor would continue to generate power at a low level that is removed by the guard vessel natural convection air-cooling system and transported to the inexhaustible atmosphere heat sink.

XXIII-1.6.4. Proliferation resistance

The electrometallurgical recycle technology does not involve separation of plutonium. The plutonium product is inherently commingled together with minor actinides (i.e. americium, curium, and neptunium), uranium, and fission products. The minor actinides contribute substantial decay heat and contamination with alpha, beta, gamma, and neutron radiation emitters. The fresh fuel product is highly radioactive, which complicates thefts and diversion.

* This was the case during the EBR-II passive safety demonstration tests in 1986 [XXIII-11].

In particular, the "spent fuel standard" is met or exceeded throughout the fuel cycle meaning that the material is protecting itself by virtue of contained radiation comparable with that of used LWR fuel [XXIII-12]. Reprocessing is assumed to be carried out inside secure regional fuel cycle support centres under international oversight.

The STAR-LM has been designed from the outset to incorporate technical features to prevent the diversion of nuclear materials. One feature is the cassette/cartridge core that is a single large assembly and is not composed of individual removable fuel assemblies/bundles. The reactor vessel upper head/cover does not incorporate openings that could be used to facilitate removal of individual assemblies/bundles.

Refuelling is deliberately intended to require the use of a heavy lift crane to raise and lower the core cassette. Refuelling equipment is not provided at the site. It is only brought to the site by the fuel vendor to facilitate refuelling operations at the end of the long 15 or 20 year core lifetime. The required heavy lift capability is provided by the fuel vendor with a transportable crane at this time. The upper portion of the containment is removed to provide access to the reactor vessel upper head. The head is removed and the cassette core is raised up into a shipping cask using the heavy lift crane. The cask is placed upon a transporter (e.g. rail car) and shipped to the regional fuel cycle support centre. Another cask containing the fresh cassette is mounted above the vessel and the fresh cassette is installed inside the vessel. The upper head is mounted upon the vessel and the seal is welded shut. The refuelling equipment including the heavy lift crane is removed from the site.

Any attempt to divert fuel during reactor operation would require shutting down the reactor and commencement of heavy lift operations. The former would be detectable through the provision of monitoring and the latter could be observed by means of overhead reconnaissance. The long core lifetime limits the availability of refuelling equipment at the site to only a relatively short interval every 15 or 20 years.

Restriction of access to the inside of the reactor vessel also limits access to neutrons escaping from the core that could be used for the undeclared production of direct-use material. The lack of access to the vessel interior can also be verified by means of overhead reconnaissance.

XXIII-1.6.5. Technical features and technological approaches used to facilitate physical protection of STAR-LM

The autonomous load follow and passive behaviour of the reactor enhance its resistance to sabotage or malevolent human-induced events. As long as the reactor and guard vessels are not penetrated causing the loss of lead coolant and as long as the guard vessel external air cooling is not interrupted, the core remains covered, heat is removed from the core by natural circulation of Pb, and heat is removed from the guard vessel/containment by natural circulation of air.

The reactor vessel is located inside of a silo. This provides protection against the effects of an aircraft crash; if desired, the silo can be made deeper. The containment and building above the vessel (Fig. XXIII-3) can be buried under a berm of dirt or other material to enhance the resistance to the effects of an aircraft crash or deliberate malevolent acts such as an attack using mortar projectiles. However, unless they are protected, potentially vulnerable locations are the inlets and outlets of the chimneys of the guard/containment vessel natural convection air-cooling system. Blockage of the pathways for airflow could interrupt air natural convection heat removal from the guard vessel and transport to the atmosphere. This vulnerability can be reduced by means of a redundant hardened design for the chimney inlets and outlets, which is more resistant to the effects of an aircraft crash or explosion blast loadings.

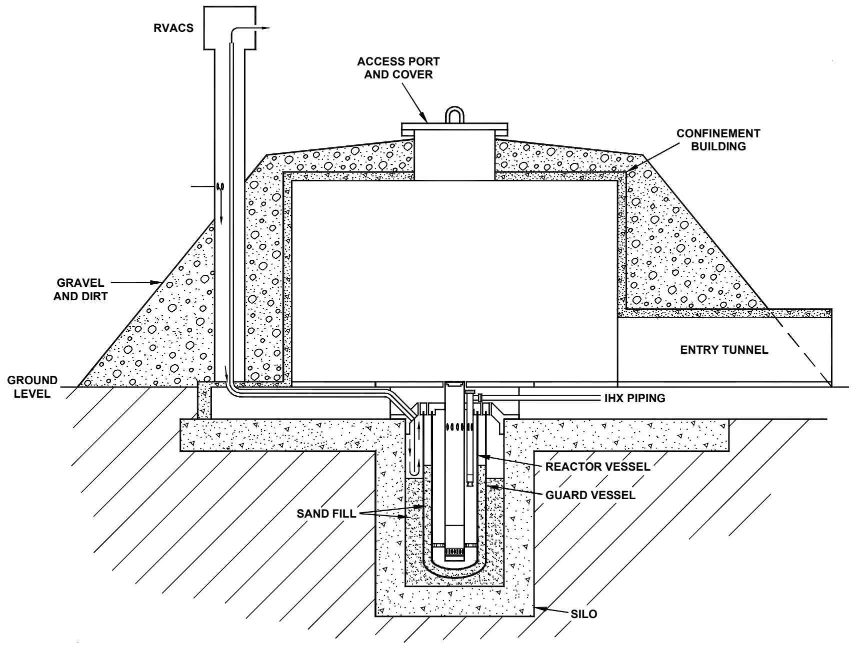


FIG. XXIII-3. Possible configuration to enhance protection of STAR-LM.

XXIII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of STAR-LM

The future world energy supply architecture implies a transition from the current fossil-based energy supply architecture. It also implies a paradigm shift in the structure of the nuclear industry. The current industry approach mainly involves the operation and construction of large capacity reactor units that take advantage of economy of scale, in order to be economically competitive. The power plants are owned by utilities that must raise large amounts of capital to finance their investment. They bear the burden of financial risk; the current regulatory and licensing regime reduces the financial risk in providing certainty of operation, provided that criteria to demonstrate safe operation are met.

In the future world energy supply architecture, deployment could take place in many developing nations. Investment capital may be difficult or impossible to raise. Thus, nuclear power plants that are affordable to purchase and economical to operate are required. Smaller capacity modular plants that can better fit into an existing electrical grid are attractive; additional units can provide for incremental expansion of capacity to meet developing needs. Plants that can operate nearly autonomously can better match the nuclear infrastructure and operations base of some developing countries. Modular plants with extensively implemented passive safety design options could be sited closer to population centres, thereby minimizing the requirements for grid expansion. The proposed hub-and-spoke future energy supply architecture is based upon the existence of regional fuel cycle support centres operated by consortia. This concept enables each country to forsake indigenous programmes of fuel recycle and fuel fabrication, in exchange for assurances of fuel supply from the appropriate centre.

In a market where there is a need for hundreds or thousands of small modular LFRs, unit costs can be reduced by taking advantage of the benefits of factory fabrication, full transportability, and modular site assembly. This favours the opening of a factory and assembly line for production of a standardized reactor design. In such a scenario, the reactor vendor must make a large investment in the factory and tooling on the basis of an expectation of likelihood of the development of a significant market. The reactor vendor thus bears a major risk in committing investment to cover the costs of development of the modular nuclear plant and factory construction that might not be recovered before the sale of a significant number of units. Production of standardized reactor units for deployment in different countries implies that the same standardized design will be deemed acceptable to the regulatory bodies of all of the countries.

To say that the massive restructurings of the world energy supply architecture and nuclear industry discussed above will require many non-technical factors and arrangements is an understatement. One is the development of modular LFRs such as STAR-LM. This could best be accomplished through a programme of national development. Realization of a deployable design could be accelerated through a license-by-test approach involving the construction of a demonstration test reactor that is a prototype for the commercial LFR.

Realization of a standardized design deployable in many countries requires the adoption of a licensing approach that incorporates criteria that are accepted multi-nationally. A model for such an approach is provided by the commercial aircraft industry for which the requirements to achieve certification of an aircraft in the U.S. and Europe are more or less universally accepted.

Opening a factory and assembly line for a standardized nuclear power plant is a “chicken and egg” problem for a potential vendor, in the absence of a significant market. Arrangements are sought that can break this dilemma. One might be a “virtual factory” whereby fabrication is performed by a consortium of vendors or by subcontractors that already manufacture components similar to those in the power plant design.

The secure regional fuel cycle support centre concept will require multi-national acceptance of the concept and national policies that enable its creation and success. In particular, specific countries must be willing to rely upon such centres as a source for nuclear fuel and not regard them as competition to indigenous fuel production programmes.

The ability to lease fuel cassettes supplied by the regional fuel cycle support centres could further reduce the costs to utilities that operate LFRs while making ownership of fuel a lucrative business. In the context of recycle of fissile self-sufficient cassettes, the cassette that contains as much fissile material as when first fabricated becomes a valuable commodity that can be loaned to utilities for use after which it is returned to the fuel cycle support centre for reprocessing and refabrication for further use. Leasing arrangements could accelerate or enhance the development and investment in the regional fuel cycle support centres.

The future needs for significantly greater electricity generation capacity and for fresh water may become so overwhelming that they bring pressures to bear that tend to facilitate the development of modular nuclear power plants and recognition of the benefits of the proposed future world energy supply architecture.

XXIII-1.8. List of enabling technologies relevant to STAR-LM

The following are key enabling technologies relevant to the STAR-LM:

- Development of a passive corrosion control approach and associated structural materials for usage with lead coolant. A passive corrosion control approach would be preferable to the current active approach available in the Russian Federation (which is on itself an effective and successful approach) that monitors and adjusts the dissolved oxygen level in the Pb coolant to maintain the formation of protective Fe_3O_4 layers upon steel structure without the formation of solid PbO contaminant. Small-scale experiments have been conducted at ANL seeking materials that are resistant to attack by molten Pb [XXIII-13]. Research on the effects of additives that inhibit corrosion has been carried out at a number of organizations through the years. Suitable materials or additives have not yet been identified;

- Technology for refuelling the core cassette and cooling the cassette during refuelling operations and shipment. Schemes are under investigation at ANL and The Ohio State University in the USA;
- Technology for performing in-service inspection under lead coolant;
- Development and demonstration of the S-CO_2 Brayton cycle power converter. There is no information that a commercial scale S-CO_2 Brayton cycle has ever been constructed and operated. Turbine and compressors need to be developed for utilization in S-CO_2 Brayton cycle tests. The recuperators and cooler in the cycle are compact Printed Circuit Heat Exchangers (PCHEs) for which the performance and efficiency with S-CO_2 needs to be validated. A complete S-CO_2 Brayton cycle power converter including a heat source, turbine, load, high and low temperature recuperators, compressors, cooler, and supporting components for control and operation needs to be constructed and demonstrated at a sufficiently large scale;
- Development of integrated, software-based modularization/factory fabrication/logistics/and rapid site assembly technologies for overcoming loss of economy-of-scale, These technologies are already commercialized in shipbuilding, ocean oil rig, aircraft, and other industries, but need to be adapted to the small nuclear reactor field;
- Development and demonstration of electrometallurgical reprocessing for transuranic nitride fuel. Some theoretical work on fuel recycle and small-scale experiments have been conducted in Japan mainly at the Japan Atomic Energy Research Institute (JAERI);

XXIII-1.9. Status of R&D and planned schedule

Development of the STAR-LM LFR at Argonne National Laboratory is currently supported by U.S. Department of Energy research and development funds. The main source of support has been a U.S. Department of Energy Nuclear Energy Research Initiative project. Research and development for the smaller-scale SSTAR LFR is also carried out at ANL under the U.S. Department of Energy Generation IV Nuclear Energy Systems Initiative. Institutions involved in the STAR-LM/LFR research and development together with ANL are Oregon State University, Texas A&M University, and The Ohio State University. Development of the SSTAR small modular fast reactor under Generation IV also involves LFR-related funding at Lawrence Livermore National Laboratory, Los Alamos National Laboratory, and the Idaho National Laboratory.

As part of the Generation IV work on the SSTAR, it was proposed in 2003 that a lead cooled demonstration test reactor could be designed, constructed, and ready for operation by about 2015. It would be subsequently operated to support SSTAR commercial deployment in about 2025. There is considerable interest in a license-by-test approach that makes use of a demonstration test reactor.

Currently, funding is not available at a level sufficient to make feasible the design, construction, and initial operation of a demonstration test reactor within a 2015 timeframe; and there remains uncertainty as to what funding priorities the U.S. Department of Energy would place on this concept.

XXIII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

A demonstration test reactor is needed to perform the research and development and demonstration that is required to support commercially deployable LFR designs such as STAR-LM, STAR-H2, and SSTAR.

Two prototype land-based reactors utilizing LBE coolant were operated in Russia as prototypes for the LBE cooled submarine reactors. In the interest of reducing deployment times, full-size land-based prototype reactors were constructed. A progression of design, construction, and operation of LBE-cooled reactors of successively larger size was not followed. As a result, Russian submarines and their reactors became LBE reactor tests in the course of naval service. In light of that experience, the need for a demonstration test reactor of about 30 MW(e) power to support the development of the BREST-OD-300 300 MW(e) Pb cooled reactor has been identified in Russia [XXIII-14].

The STAR-LM design incorporates a number of specific attributes that justify demonstration in a reactor test facility including:

- Nitride fuel bonded by molten lead to ferritic-martensitic stainless steel cladding;
- Long core lifetime of 15 or 20 years;
- Open lattice core thermal-hydraulics. The velocities through the core from natural circulation are low (e.g. u_{mean}~0.7 m/s). The core design has a significant power spatial distribution ($P_{Peak}/P_{Avg.}$=1.63) that gives rise to significant temperature and velocity profiles across the core;
- Autonomous power driven natural circulation heat transport. The fast neutron spectrum core has strong reactivity feedbacks to changes in the coolant and fuel temperatures that enable autonomous changes in power in response to changes in load demand/heat removal (i.e. autonomous load following) and passive shutdown. The coolant flow rate as well as the temperature rise through the core (and, hence, the core coolant temperatures) directly depends upon the reactor power. Thus, a perturbation in coolant flow or temperature can cause a perturbation in the core power that, in turn, affects the fuel temperatures, coolant temperatures, and coolant flow rate. Crucial questions therefore concern the potential for development of oscillations in power, system temperatures, and flow, and whether small perturbations in power, temperature, or flow are unstable and can grow to significant magnitude. Demonstration of stable behaviour during operation is required;
- Autonomous load following and passive safety capabilities. Tests that demonstrate autonomous operation and passive shutdown are required. The core design might incorporate mechanical design features that enhance the negative reactivity feedback from core radial expansion/flowering to optimize the system temperatures during autonomous changes in power; in particular, the core outlet temperature may be maintained approximately unvarying during autonomous power changes. Such enhancement features are not needed for the current STAR-LM concept;
- Start-up strategy. Demonstration is required of the approach by which the reactor is taken from a deeply subcritical state with external heating sufficient to maintain the Pb coolant in a molten state to the nominal fission power level and primary coolant natural circulation flow;
- Refuelling of the core cassette/cartridge. Successful demonstration of refuelling operations involving the single assembly core cassette is required;

- Lead-to-S-CO_2 heat exchangers. Demonstration of the performance of the Pb-to-CO_2 heat exchangers is required;
- Removal of reactor afterheat. Tests are required that demonstrate passive natural circulation air cooling of the outside of the containment/guard vessel, removal of reactor afterheat, and transport to the atmosphere following termination of heat removal through the in-reactor Pb-to-CO_2 heat exchangers;
- Supercritical carbon dioxide Brayton cycle power converter. A demonstration of this new power conversion technology coupled to the autonomous LFR is required.

If the test demonstrator reactor is sufficiently representative of a commercially deployable design (e.g. SSTAR), then the demonstrator is needed for a license-by-test approach for the commercial reactor.

XXIII-1.11. List of other similar or relevant SMRs for which the design activities are on-going

Development of the STAR-LM at a power level of 400 MW(th) (175 MW(e)) has served as a starting point for the development of related LFR concepts including the 400 MW(th) STAR-H2 [XXIII-1, XXIII-15 to XXIII-18] for the production of hydrogen via a Ca-Br thermochemical water cracking cycle as well as fresh water production using reject heat, and the Small Secure Transportable Autonomous Reactor (SSTAR) [XXIII-19 to XXIII-23] which is a smaller-sized LFR having similar attributes as STAR-LM but at lower power rating in the 20 to 100 MW(e) range. The SSTAR concept is currently being developed at a power level of 20 MW(e) (45 MW(th)). Development of the SSTAR has been driven by interest in small modular reactors for deployment at remote sites such as potentially exist in the states of Alaska, Hawaii [XXIII-24], Ulung Island in the Republic of Korea, island nations of the Pacific Basin (e.g. Indonesia), and elsewhere.

XXIII-2. Design description and data for STAR-LM

XXIII-2.1. Description of the nuclear systems

Reactor core and fuel design

The STAR-LM cassette core design has been performed to meet the following requirements and constraints:

- Single batch fuelling with transuranic (TRU) nitride fuel enriched to 100% in ^{15}N;
- Core diameter small enough to meet the criterion for transportability by rail;
- Long fuel lifetime of 15 full power years;
- Coolant volume fraction large enough to enable natural circulation heat transport of more than the full core power;
- Minimization of burn-up reactivity swing ($k_{eff,\,max}$ - $k_{eff,\,min}$) during the cycle;
- Maximization of discharge burn-up;
- Peak fluence of fast neutrons less than or equal to 4×10^{23} 1/cm^2 for HT9 cladding.

The transuranic isotopic vector is assumed to be representative of fuel separated from used LWR fuel. The use of ^{15}N eliminates parasitic (n, p) reactions in ^{14}N and waste disposal problems that would be associated with ^{14}C production. In order to reduce the core peak-to-average power ratio, three TRU enrichment zones are employed.

Figure XXIII-4 shows the core map assumed in the core design analysis. The hexagonal geometry does not imply that the core is formed of individual hexagonal fuel assemblies or bundles. It merely reflects the assumed nodalization of the DID3D/REBUS-3 code package. The three enrichment zones are obvious as are assumed locations for shutdown and control rods. The fuel in the STAR-LM core consists of cylindrical fuel rods arranged on a triangular pitch. The nodalization is sufficient for analysis due to the long mean free paths of neutrons in lead cooled fast spectrum cores.

Surrounding the core is a radial reflector assumed to consist of a 50 volume % HT9-50 volume % Pb mixture. The steel shroud surrounding the core is also represented by this region. A reflector containing steel is necessary to reduce the fluence at the reactor vessel. Lead is a superior reflector but has a low effectiveness in shielding in-vessel structures. Downwards flowing lead in the downcomer between the shroud and the reactor vessel was also considered in the analysis.

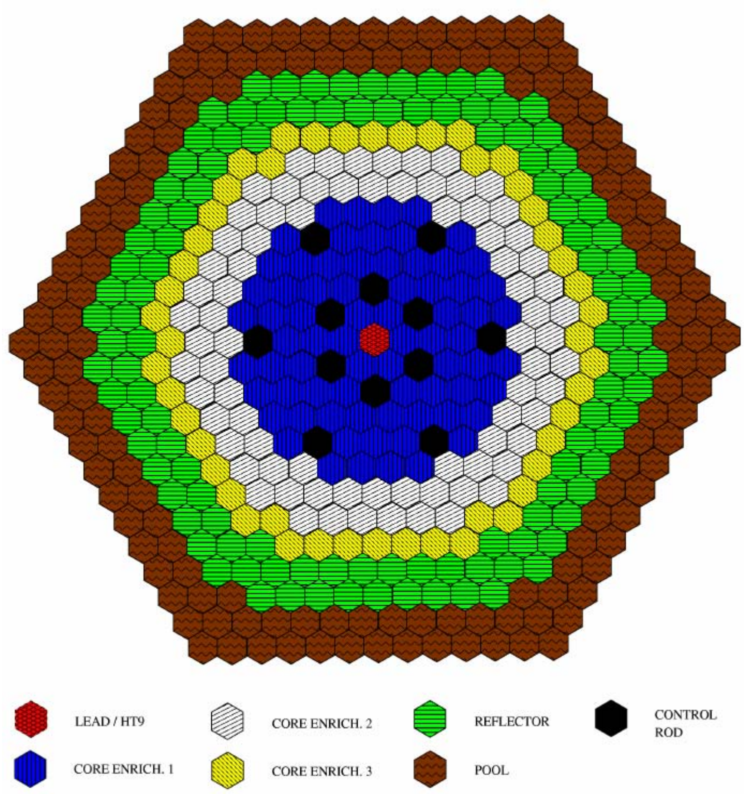


FIG. XXIII-4. Core map of STAR-LM.

Table XXIII-2 presents some details of the design conditions for the STAR-LM core; general core data is presented in Table XXIII-1 in the beginning of this description.

TABLE XXIII-2. SOME DESIGN CONDITIONS FOR 400 MW(th) STAR-LM FUEL AND CORE

<table>
<tr><th colspan="3">CHARACTERISTIC</th><th>VALUE</th></tr>
<tr><td colspan="3">Fuel pin diameter, cm</td><td>1.30</td></tr>
<tr><td colspan="3">Fuel pin pitch-to-diameter ratio</td><td>1.536</td></tr>
<tr><td colspan="3">Cladding thickness, cm</td><td>0.1</td></tr>
<tr><td colspan="3">Fuel pellet outer diameter, cm</td><td>0.971</td></tr>
<tr><td colspan="3">Fuel pellet-cladding bond</td><td>Pb</td></tr>
<tr><td colspan="3">Fuel smeared density, %</td><td>78.0</td></tr>
<tr><td rowspan="4">Volume fraction</td><td colspan="2">Fuel</td><td>0.2147</td></tr>
<tr><td colspan="2">Fuel-cladding bond</td><td>0.0606</td></tr>
<tr><td colspan="2">Cladding</td><td>0.0682</td></tr>
<tr><td colspan="2">Coolant</td><td>0.6566</td></tr>
<tr><td colspan="2" rowspan="3">BOC loading, MT (tons)</td><td>HM</td><td>25.56</td></tr>
<tr><td>U</td><td>21.47</td></tr>
<tr><td>TRU</td><td>4.09</td></tr>
<tr><td colspan="2" rowspan="2">Driver power density, W/cm^3</td><td>Average</td><td>43.58</td></tr>
<tr><td>Peak</td><td>70.15</td></tr>
<tr><td colspan="3">Peak linear heat rate, W/cm</td><td>590.48</td></tr>
</table>

Primary and secondary coolant systems

The design of the STAR-LM primary and secondary coolant systems has been carried out to meet the following requirements and constraints:

- Power level: 400 MW(th);
- Full transportability by rail;
- Natural circulation heat transport of primary coolant at power levels up to and exceeding 100% nominal;
- Core dimensions and fuel volume fraction from core design neutronics analyses;
- Peak cladding temperature equal to 650°C;
- Maximized S-CO_2 Brayton cycle efficiency;
- Fission gas plenum height above active core equal to 25% of active core height;
- Pb coolant channels about 1 cm or more in diameter to reduce potential for plugging by contaminants;
- Space for incorporation of cylindrical liner and annular gap escape path for CO_2 vapour/gas between in-vessel Pb-to-CO_2 heat exchangers and reactor vessel inner surface;
- Space for multi-plate thermal radiation heat shield between bottom of upper head/cover and Pb free surface;
- Heat removal from outside of guard/containment vessel to inexhaustible atmosphere heat sink by natural circulation of air.

Rail transportability imposes a size limitation upon the reactor vessel and guard vessel of 6.1 m in diameter and 18.9 m in height [XXIII-25]. The fission gas plenum height is based upon an assumed conservative gas release from nitride fuel of 2.5% per atom % of burn-up. The fuel volume fraction was held fixed in the thermal hydraulic design analyses at the value of 0.215 determined by the core design. The fuel rod outer diameter and pitch-to-diameter ratio were varied to determine an optimum combination. Figure XXIII-5 shows the relationship between pitch-to-diameter ratio and rod diameter for a triangular lattice with a fixed fuel volume fraction of 0.215 and a fixed fuel smeared density of 78%.

Assuming this relationship together with a fixed core inlet temperature and heat exchanger height, the peak cladding temperature exhibits a minimum at a fuel rod outer diameter of 1.3 cm (Fig. XXIII-6) such that the difference between the peak cladding temperature and the S-CO_2 heat exchanger outlet temperature is effectively minimized. Thus, if the inlet temperature is raised until the peak cladding temperature equals the 650°C criterion, the S-CO_2 temperature at the exit of exchangers is maximized, which maximizes the Brayton cycle efficiency. This optimal situation is obtained at an inlet temperature of 438°C. For a heat exchanger tube inner diameter of 1.0 cm, an optimal tube pitch-to-diameter ratio was found to be 1.75.

Some system design conditions for thermal-hydraulic calculations of the STAR-LM, including the heat exchanger (HX) data, are provided in Table XXIII-3; general data on core and thermohydraulics of the STAR-LM are presented in Table XXIII-1 in the beginning of this description.

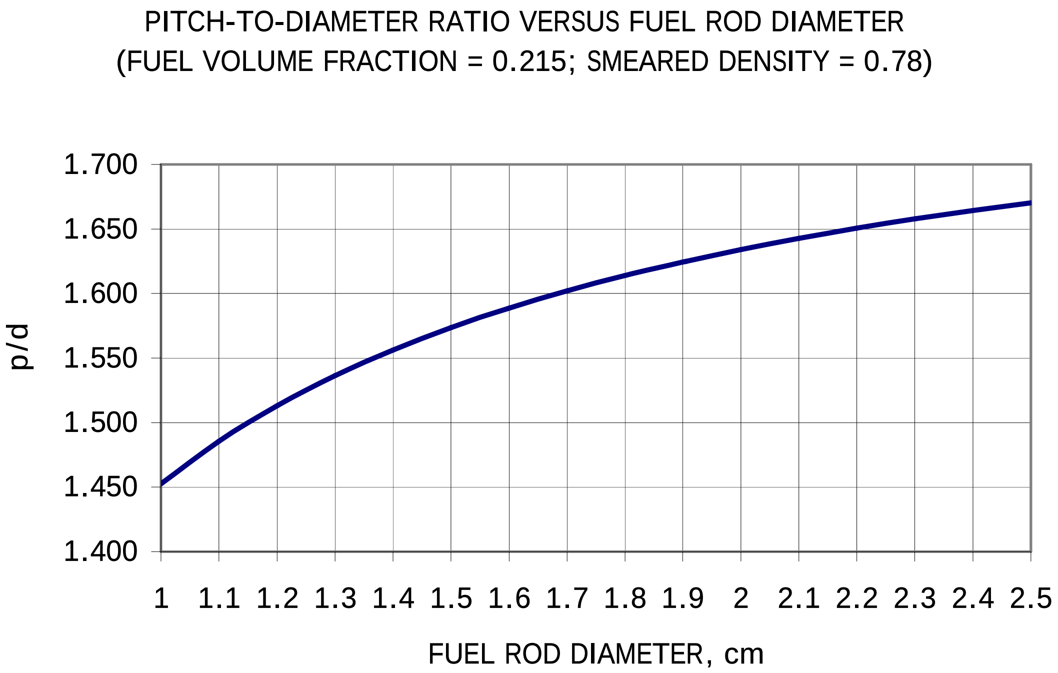


FIG. XXIII-5. Pitch-to-diameter ratio versus fuel rod outer diameter for triangular lattice with fuel volume fraction equal to 0.215 and fuel smeared density of 78%.

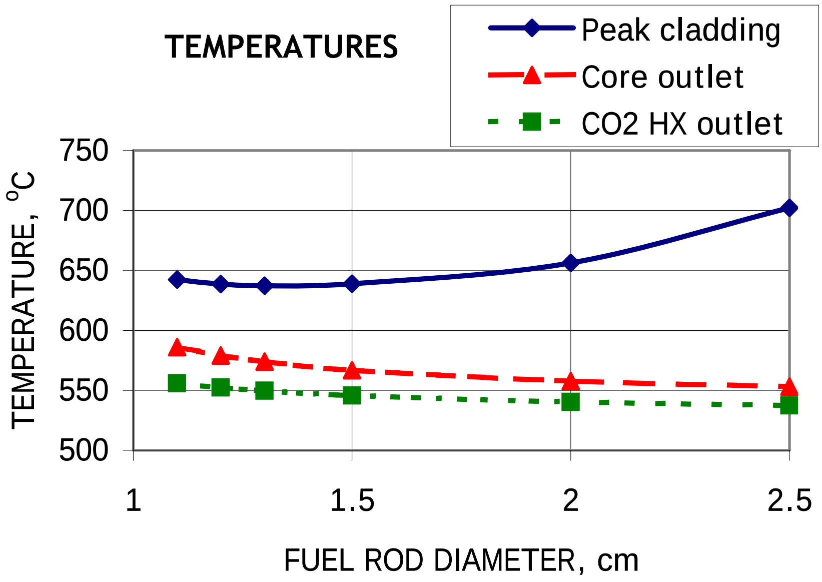


FIG. XXIII-6. Peak cladding temperature, core outlet temperature, and maximum S-CO_2 temperature versus fuel rod outer diameter.

Figure XXIII-7 shows the dependency of the S-CO_2 Brayton cycle efficiency upon the core inlet temperature. In the calculations, the heat exchanger tube height is chosen to satisfy the peak cladding temperature constraint of 650°C. It is confirmed that a fuel rod outer diameter of 1.30 cm and an inlet temperature of 438°C maximize the Brayton cycle efficiency.

TABLE XXIII-3. DESIGN CONDITIONS OF 400 MW(th) STAR-LM COUPLED TO A SUPERCRITICAL CARBON DIOXIDE BRAYTON CYCLE

CHARACTERISTIC	VALUE
Number of grid spacers	3
Height difference between heat exchanger (HX) and core thermal centres, m	6.25
Height difference between top of HX tubes and bottom of active core, m	12.25
HX tube height, m	10.0
HX inner diameter, m	3.16
HX outer diameter, m	5.23
HX tube inner diameter, cm	1.0
HX tube outer diameter, cm	1.4
HX tube triangular pitch-to-diameter ratio	1.75
HX primary coolant hydraulic diameter, cm	3.32
Total number of HX tubes in all HXs	21 900

Passive load follow capability

The strong reactivity feedbacks of the fast neutron spectrum core enable autonomous load following whereby the reactor power adjusts itself to match heat removal from the primary coolant solely as a consequence of inherent physical phenomena. Figure XXIII-8 shows the dependencies of the new peak cladding, core outlet, and core inlet temperatures upon the new steady state thermal power that is attained following an autonomous power change from the nominal (400 MW(th)) steady state. The results are dependent upon the value of the core radial expansion reactivity feedback coefficient. The specific value of -0.140 cents/°C represents thermal expansion-induced increase of the separation between the wide pitch-to-diameter ratio fuel rods as would be provided by simple grid spacers, similar to the approach utilized for LWR fuel bundles. It is observed that as the core power decreases, the core outlet temperature rises from 588°C to a maximum of 611°C at 20% nominal power and then decreases to 607°C. Significantly, the outlet temperature remains above the nominal full power value such that the S-CO_2 can be heated to a high temperature to maintain a high Brayton cycle efficiency. As the power level decreases, the peak cladding temperature also decreases such that the assumed 650°C temperature criterion is not exceeded. For power levels above 100%, the peak cladding temperature is approximately unvarying; the peak cladding temperature increases by only 1.6°C at 120% nominal power.

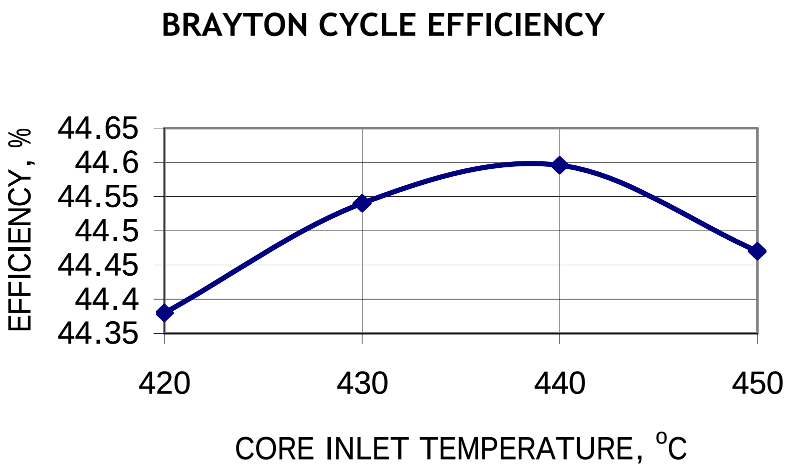


FIG. XXIII-7. Supercritical carbon dioxide Brayton cycle efficiency versus Pb core inlet temperature subject to peak cladding temperature constraint of 650°C.

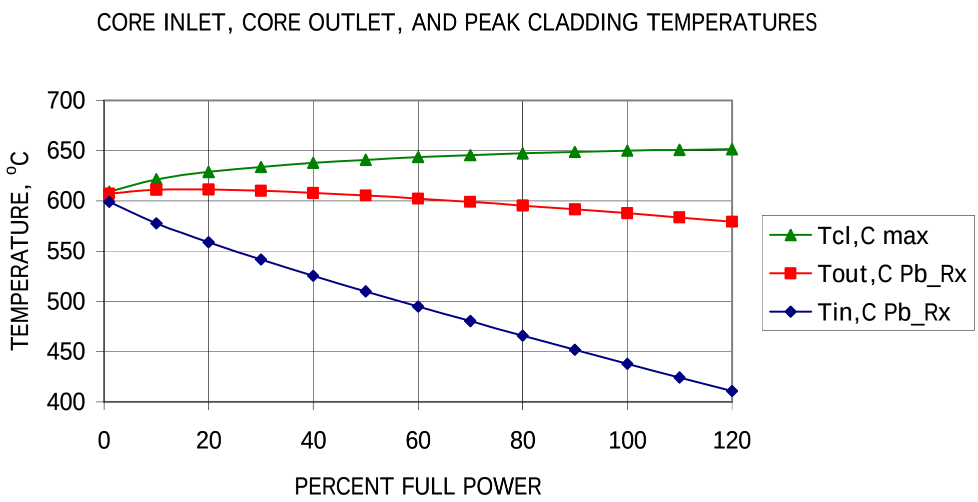


FIG. XXIII-8. STAR-LM characteristic core temperatures versus new steady state thermal power following an autonomous power change from nominal operating conditions.

Main heat transport system

Main heat transport system of the STAR-LM, indicating heat removal path in normal operation and in accidents, is given in Fig. XXIII-9.

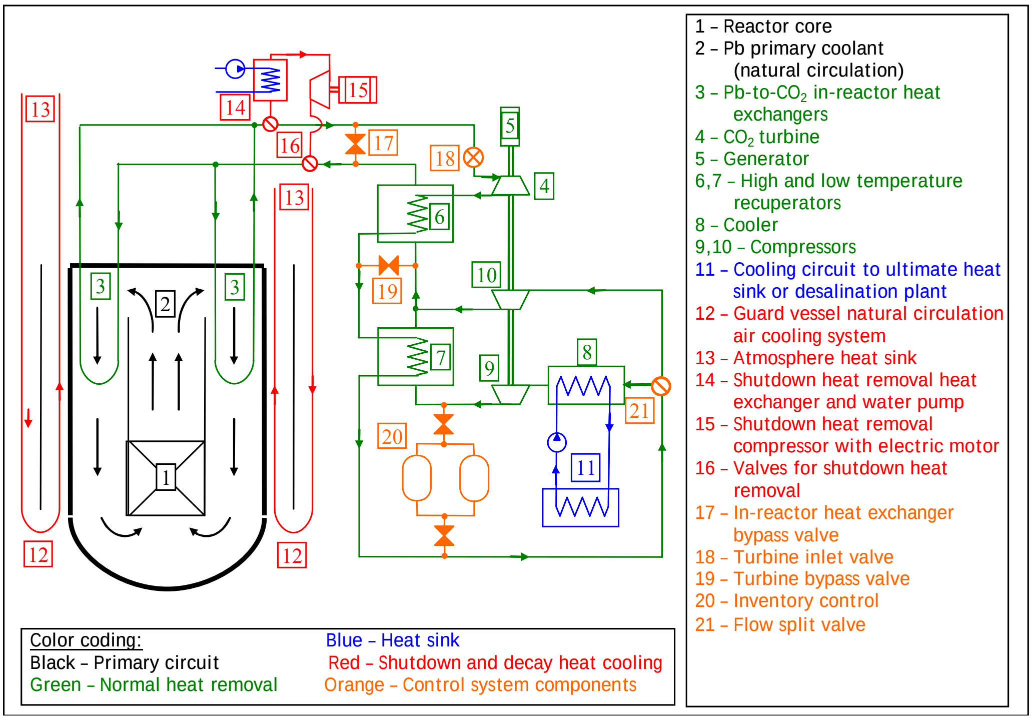


FIG.XXIII- 9. Main heat transport system of STAR-LM coupled to S-CO_2 Brayton cycle.

XXIII-2.2. Description of the turbine generator plant and systems

Figure XXIII-9 above shows a schematic illustration of the STAR-LM coupled to its S-CO_2 gas turbine Brayton cycle. A cycle efficiency of 45% is calculated. A key contributor to the high efficiency is the low amount of work (PDV work) to compress S-CO_2 immediately above the critical temperature due to the high S-CO_2 density. Table XXII-6 in ANNEX XXII compares the densities of S-CO_2 at cycle conditions versus those for helium in the Eskom Pebble Bed Modular Reactor as well as typical liquid coolants; the S-CO_2 density is more like that of an ordinary liquid. Thus, the S-CO_2 temperature and pressure at the low end of the cycle are close to but slightly greater than the critical temperature (30.98°C) and pressure (7.373 MPa). The pressure at the high end is taken to be 20 MPa (2900 psi). Further increase in pressure is calculated to bring diminishing returns in cycle efficiency gain. The expanded S-CO_2 from the turbine passes through two recuperators (regenerative heat exchangers) to preheat the S-CO_2 before it is returned to the in-reactor heat exchangers that are immersed in the lead coolant. This further contributes to an increase in cycle efficiency.

The specific heat of S-CO_2 strongly depends upon pressure; the value at 20 MPa is significantly greater than that at 7.4 MPa. To preheat the CO_2 effectively, it is necessary to split the flow such that only a portion, optimally 65%, passes through the cooler where heat rejection from the cycle occurs. The cooled S-CO_2 is partially heated in the low temperature recuperator. The remainder of the flow is directly compressed and merged with the compressed cooler flow stream; for this reason, the cycle is referred to as a recompression cycle.

Figure XXIII-10 shows the dependency of the S-CO_2 specific heat near the critical temperature.

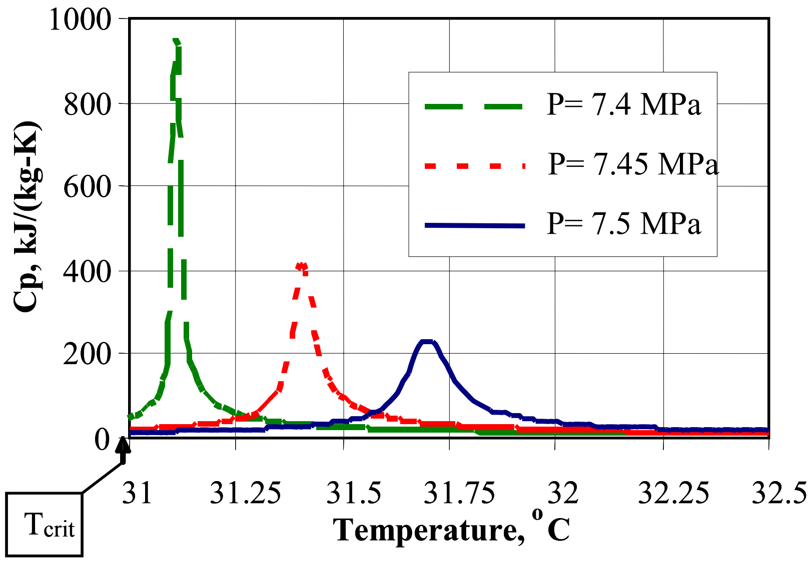


FIG. XXIII-10. Dependency of S-CO_2 specific heat near the critical temperature.

The spike in C_p results in a behaviour that is similar to a phase transition whereby a reduction in specific enthalpy produces virtually no decrease in temperature. The STAR-LM makes use of this behaviour to reduce the S-CO_2 temperature in the cooler to a value that is very close to but still above the critical temperature. Thus, the cooler channel length is chosen such that heat transfer to the heat sink fluid (e.g. water or air) reduces the S-CO_2 temperature to a value within the spike (e.g. to about 31.25 C). Dependency of the cycle efficiency upon the cooler outlet temperature is shown in Fig. XXIII-11.

There is a strong incentive to operate as closely as possible to the critical temperature to raise the cycle efficiency. The dependency of the S-CO_2 Brayton cycle efficiency upon the turbine inlet temperature is presented in Fig. XXIII-12.

The S-CO_2 turbine and compressors that are optimally designed for the 400 MW(th) cycle conditions have a remarkably small size due to the high S-CO_2 density. For the small blade sizes, bending and vibration stresses have been found to be greater than centrifugal stresses. For the turbine, the ratio of bending plus vibration stress to centrifugal stress varies from eight for the first stage of the turbine to 3.8 for the last stage. This is in contrast to traditional ideal gas turbines for which centrifugal stresses determine the blade tip diameter of the final stage. Results of the design analyses are presented in Table XXIII-5.

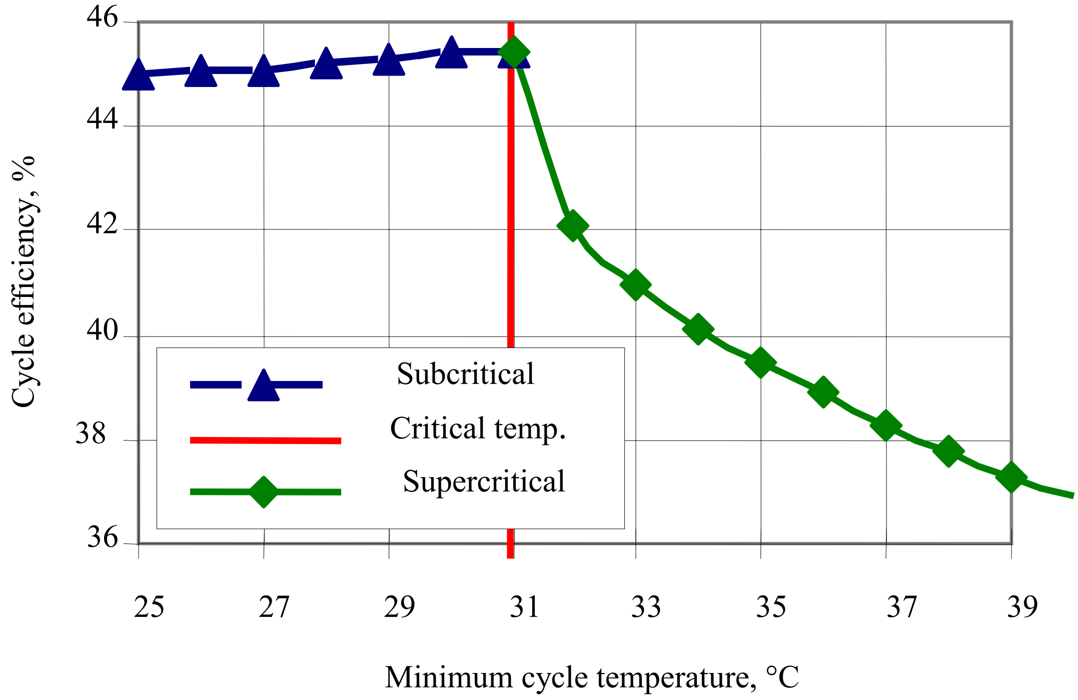


FIG. XXIII-11. Dependency of S-CO_2 Brayton cycle efficiency upon cooler outlet temperature.

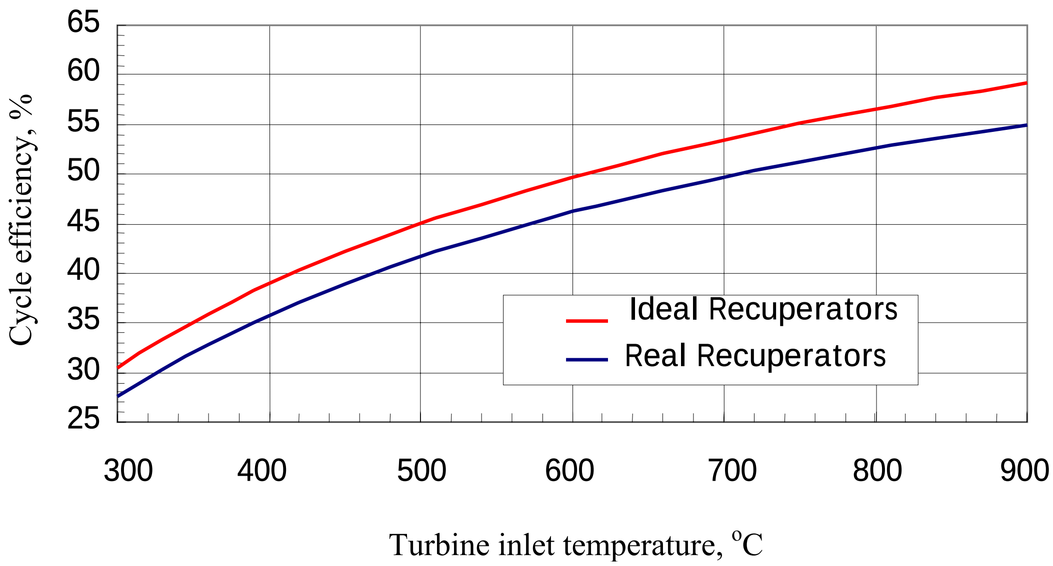


FIG. XXIII-12. Dependency of S-CO_2 Brayton cycle efficiency upon turbine inlet temperature.

The volume taken up by the recuperators can also be radically reduced through the utilization of printed circuit heat exchangers (PCHEs) of the type manufactured by Heatric Division of Meggitt (U.K.), Ltd [XXIII-26]. The PCHEs are fabricated from flat metal plates into which semicircular fluid flow channels are chemically milled. The plates are stacked and diffusion bonded together to form a strong metal block containing rows of many small heat exchange channels; hot and cold fluid flows in counterflow in alternating rows. Nozzles and headers for the hot and cold fluids are welded onto the block. Recuperator performance was analyzed with steady state, one-dimensional, multi-cell, finite difference heat transfer calculations. The calculations account for local variations in thermophysical properties especially for S-CO_2. For the assumption of 1 mm semicircular heat exchange channels, a 3-meter recuperator channel path length is calculated.

TABLE XXIII-5. RESULTS OF TURBINE AND COMPRESSORDESIGN ANALYSES FOR 400 MW(th) STAR-LM

ITEMS	TURBINE	COMPRESSOR 1	COMPRESSOR 2
Number of stages	4	4	4
Length (without casing), m	0.8	0.5	0.3
Maximum diameter (without casing), m	1.25	0.5	0.7
Efficiency without secondary losses, %	97.4	96.2	95.3
Assumed secondary losses, %	5	5	5
Net efficiency, %	92.4	91.2	90.3

For each recuperator, a block cross-sectional area (normal to the length) of 36 m^2 provides suitably high cycle efficiency. The small volumes calculated for the PCHEs represent a significant reduction of 70% compared to the volume of 10 meter long finned shell-and-tube recuperators incorporating 1 cm inner diameter tubes that would otherwise be required.

Coupling of a LFR to the S-CO_2 Brayton cycle is dependent upon the thermal hydraulic design of in-reactor lead-to-CO_2 heat exchangers (HXs). The Pb-to-CO_2 HXs must meet several requirements and constraints:

- The HXs must fit within the available volume inside the reactor vessel;
- The HXs must heat the S-CO_2 to a high turbine inlet temperature, to achieve high Brayton cycle efficiency;
- The HXs must not provide too large of a frictional pressure drop for Pb flow thereby impeding natural circulation heat transport;
- The HX channels must be large enough to resist potential plugging by contaminants in the Pb melt; and
- The HXs must be factory fabricated, modular, and removable.

The four modular HXs provide for counter-current heat exchange. The S-CO_2 flows upwards through the inside of 1 cm inner diameter tubes while the Pb flows downwards over the exterior of the tubes (Fig. XXIII-13). The tube inner diameter of 1.0 cm was chosen to preclude occlusion of the lead flow channels due to growth upon the structure of protective oxide layers or plugging by contaminants. A triangular tube array having a pitch-to-diameter ratio of 1.75 was determined to be optimal.

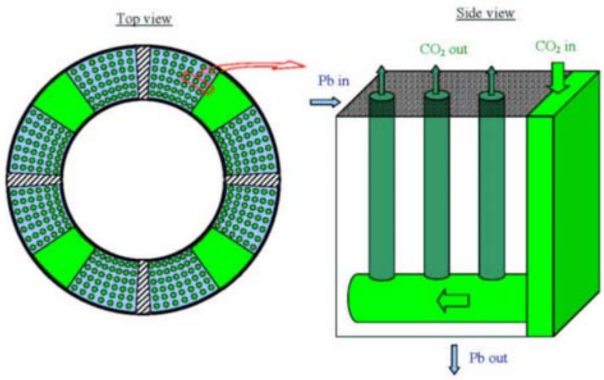


FIG. XXIII-13. Illustration of possible configuration of in-reactor Pb-to-CO_2 heat exchanger.

Although Fig. XXIII-13 depicts a possible downcomer header for S-CO_2 flow, the actual configuration could involve a central bundle of tubes for downwards S-CO_2 flow in a region from which Pb flow is largely excluded; a bend at the bottom of each tube would redirect the S-CO_2 flow inside the tube upwards in a triangular lattice.

XXIII-2.3. Systems for non-electric applications

A desalination plant can be optionally added between the low temperature recuperator and the cooler.

XXIII-2.4. Plant layout

The plant layout is currently under development. The following are considerations in the general philosophy that governs development of the plant layout:

- Seismic isolation using seismic isolators and a seismic island. Given the small size of the S-CO_2 Brayton cycle secondary side components, an objective is to attempt to locate the reactor, containment, and secondary side all on a common seismic island. This approach would reduce the effects of differential motion between S-CO_2 secondary side components and the S-CO_2 piping to and from the reactor;
- Resistance against an aircraft crash and malevolent human-induced acts;
- Provision for core cassette refuelling using refuelling equipment including a heavy lift crane that is brought to the site only at the end of the core lifetime and is removed from the site following refuelling;
- Modular assembly of components at the site to reduce construction/assembly costs and time. It is planned to use 4^+ D CAD/CAM visualization for development and planning of modularization and site assembly sequences in the development of modules and a plant layout; and
- A layout that facilitates expansion of the number of reactor units at the site in response to growing customer needs.

REFERENCES

[XXIII-1] WADE, D., DOCTOR, R., PEDDICORD, L. STAR-H2: A battery-type lead cooled fast reactor for hydrogen manufacture in a sustainable hierarchical hub spoke energy infrastructure (Proc. Int. Conf. Kyoto, Sep. 15–19, 2003), GENES4/ANP 2003, paper 1189 (2003).

[XXIII-2] GROMOV, B.F. et al., Heavy liquid metal coolants in nuclear technology (Proc. of Int. Conf., Obninsk, October 5–9, 1998), Vols. 1 and 2 (1999).

[XXIII-3] TOSHINSKY, G., A LEIPUNSKY, Nuclear power installations with lead-bismuth liquid metal coolant for submarines, Yadernaya Energetica Centenary of Leipunsky, A., Supplement to No. 4, 13 (2003).

[XXIII-4] SIENICKI, J.J. et al., The STAR-LM lead-cooled closed fuel cycle fast reactor coupled to a supercritical carbon dioxide Brayton cycle advanced power converter, GLOBAL 2003 (Proc. Int. Meeting, New Orleans, LA, USA, November 16–20, 2003), ANS/ENS.

[XXIII-5] SIENICKI, J.J., et al., The STAR-LM lead-cooled closed fuel cycle fast reactor with a supercritical carbon dioxide Brayton cycle advanced power converter, Russian Forum for Science and Technology FAST NEUTRON REACTORS, HLMC-2003 (Paper presented at the 2nd Int. Conf. on Heavy Liquid Metal Coolants in Nuclear Technologies, Obninsk, Russia, December 8–12, 2003), paper 2105.

[XXIII-6] MOISSEYTSEV, A.V., SIENICKI, J.J., WADE, D.C. Cycle analysis of supercritical carbon dioxide gas turbine Brayton cycle power conversion system for liquid metal-cooled fast reactors, ICONE-11 (Proc. 11th Int. Conf. on Nuclear Engineering, Tokyo, April 20-23, 2003), ICONE 11-36023.

[XXIII-7] MOISSEYTSEV, A.V., SIENICKI, J.J., WADE, D.C., Turbine design for a supercritical carbon dioxide gas turbine Brayton cycle, ICAPP'03 (Proc. Int. Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4–7, 2003) paper 3064 (2003).

[XXIII-8] NISHI, Y., et al., A new concept of the 4S reactor and thermal hydraulic characteristics, ICONE-12 (Proc. 12th Int. Conf. on Nuclear Engineering, Arlington, April 25–29, 2004), ICONE12-49257.

[XXIII-9] SIENICKI, J.J. Safety and licensing aspects of the LFR/SSTAR, Transactions of the American Nuclear Society, American Nuclear Society 2004 Nuclear Installations Safety Division Session on Safety Issues for Advanced Nuclear Energy and Transmutation Systems (Paper presented at the Annual Meeting, Pittsburgh, June 13–17, 2004), 90, 24.

[XXIII-10] ILINČEV, G. Research results on the corrosion effects of liquid heavy metal Pb, Bi and Pb-Bi on structural materials with and without corrosion inhibitors, Nuclear Engineering and Design, 217 (2002), 167–177.

[XXIII-11] The Experimental Breeder Reactor-II inherent safety demonstration, EBR-II Division, Argonne National Laboratory, April 1986, Nuclear Engineering and Design, Special Issue, ed. S. H. Fistedis, 101 (1987), 1–91.

[XXIII-12] HANNUM, W.H., WADE, D.C., MCFARLANE, H.F., HILL, R.N. Non-proliferation and safeguards aspects of the IFR, Progress in Nuclear Energy, Special Issue, 31, No. 1/2 (1997), 203–217.

[XXIII-13] RUNGE, J.M., LEIBOWITZ, L., BARNES, L.A., RARAZ, A.G. MCDEAVITT, S.M., Performance of structural materials in lead-based reactor coolants at temperatures up to 800 EC, GLOBAL 2003 (Proc. Int. Meeting, New Orleans, LA, USA, November 16–20,2003), ANS/ENS.

[XXIII-14] ZRODNIKOV, V., et al., Research fast reactor with lead coolant BREST-30, Russian Forum for Science and Technology FAST NEUTRON REACTORS, HLMC-2003 (Paper presented at the 2nd Int. Conf. on Heavy Liquid Metal Coolants in Nuclear Technologies, Obninsk, Russia, December 8–12, 2003).
[XXIII-15] WADE, D.C., et al., STAR-H2: A 400 MWth lead-cooled, long refuelling interval reactor for hydrogen production, ICONE-11, (Proc. 11th Int. Conf. on Nuclear Engineering, Tokyo, April 20–23, 2003) ICONE 11-36576.
[XXIII-16] WADE, D., MOISSEYTSEV, A. SIENICKI, J. STAR-H2: Passive load follow/passive safety coupling to water cracking balance of plant, MIT Symposium on Nuclear Energy and The Hydrogen Economy (Paper presented at Int. Conf., Cambridge, MA, USA, September 23–24, 2003).
[XXIII-17] DOCTOR, R.D., et al., STAR-H2 with a calcium-bromine cycle: Delivering hydrogen, electricity and water from a modular reactor preliminary design for a 5-MW test system, American Institute of Chemical Engineering 2004 Spring National Meeting, Session 14004, Novel Hydrogen Generation Processes and Energy Applications II (Paper presented at Int. Conf., New Orleans, April 25–29, 2004).
[XXIII-18] WADE, D.C., et al., STAR-H2: A long-refuelling interval battery reactor for hydrogen and water supply to cities of developing countries, 5th International Conference on Nuclear Options in Countries with Small and Medium Electricity Grids (Paper presented at Int. Conf., Dubrovnik, Croatia, May 16–20, 2004).
[XXIII-19] SIENICKI, J.J., et al., STAR performer, Nuclear Engineering International (July 2005) 24.
[XXIII-20] SIENICKI, J.J., MOISSEYTSEV, A.V., SSTAR lead-cooled, small modular fast reactor for deployment at remote sites – system thermal hydraulic development, 2005 International Congress on Advances in Nuclear Power Plants (ICAPP 2005) (Paper presented at Int. Conf., Seoul, May 15–19, 2005) paper 5426.
[XXIII-21] WANG, W.S., et al., lead-cooled, long-life fast reactor concepts for remote deployment, 2005 International Congress on Advances in Nuclear Power Plants (ICAPP 2005) (Paper presented at Int. Conf., Seoul, May 15–19, 2005) paper 5102.
[XXIII-22] SIENICKI, J.J., et al., SSTAR lead-cooled, small modular fast reactor with nitride fuel, OECD Nuclear Energy Agency Workshop on Advanced Reactors with Innovative Fuels (ARWIF-2005) (Paper presented at Int. Conf., Oak Ridge, February 16–18, 2005) paper 2.03.
[XXIII-23] SIENICKI, J.J., WADE, D.C., Nonproliferation features of the small secure transportable autonomous reactor (SSTAR) for worldwide sustainable nuclear energy supply, American Nuclear Society 2005 Winter Meeting 2005 International Congress on Advances in Nuclear Power Plants (ICAPP 2005) (Paper presented at the Winter Meeting, Washington, November 13–17, 2005).
[XXIII-24] Report to Congress on Small Modular Nuclear Reactors, U.S. Department of Energy, Office of Nuclear Energy, Science and Technology (May 2001).
[XXIII-25] GYOREY, G.L., HARDY, R.W., MAGEE, P.M., Safety design for the advanced liquid-metal-cooled reactor, Nuclear Safety 31 (1990), 323.
[XXIII-26] DEWSON, S.J., THONON, B., The development of high efficiency heat exchangers for helium gas cooled reactors, ICAPP'03 (Proc. Int. Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4–7, 2003) paper 3213. See also: www.heatric.com.

SECURE TRANSPORTABLE AUTONOMOUS REACTOR (STAR-H2)

ANL, Oregon State University, Texas A&M University, and Ohio State University, United States of America

XXIV-1. General information, technical features and operating characteristics

XXIV-1.1. Introduction

STAR[1] concept development is being conducted for a portfolio of reactor and balance of plant designs to enable an incremental market penetration that is time-phased according to the degree of R&D required. Secure Transportable Autonomous Reactor – Liquid Metal (STAR-LM) [XXIV-1], described in ANNEX XXIII, is a Pb-cooled, 400 MW(th), natural circulation reactor of 565°C core outlet temperature driving a supercritical CO_2 Brayton cycle for electricity production. It draws on many proven technologies and will be ready for market in 15–20 years. The STAR-H2 [XXIV-2], presented in this annex, raises the Pb outlet temperature to 800°C to drive a thermochemical water cracking cycle and will require additional R&D. It is targeted for deployment by 2030. The SSTAR [XXIV-3], described in Annex XXII, takes the STAR-LM design features down to ~25 to 50 MW(th) to provide for secure energy supply to remote small villages. It is targeted for early prototyping of the technology and institutional features of the STAR concept.

All STAR concepts are designed for 15 to 20-year refuelling interval and rely on outsourcing the associated fuel cycle and waste management services to proposed regional fuel cycle centres. All employ desalination (or alterative) bottoming cycles to extend their scope of energy services and to minimize their environmental footprint. The small sizing and outsourced fuel cycle and waste management configuration allows for plant deployment at modest initial capital outlay by the customer. Turnkey plants are transported to the customer's site and rapidly connected to a pre-constructed non-nuclear safety grade balance of plant to achieve a rapid start of the revenue stream.

The R&D for the STAR-H2 concept is being performed in Argonne National Laboratory (ANL, USA). Institutions involved in STAR-H2 research and development together with ANL are Oregon State University, Texas A&M University, and Ohio State University.

XXIV-1.2. Applications

STAR-H2 is a member of the STAR reactor and fuel cycle concept, which is meant to attain the Generation IV goals by responding to foreseen mid century needs and market conditions. STAR-H2 is a long-term concept targeted to fill primary energy and potable water needs for urban centres in developing countries and is designed to fit within a hierarchical hub-spoke energy architecture based on regional fuel cycle centres, using nuclear fuel and hydrogen as the long distance energy carriers – with distributed electricity generation as the local carrier. STAR-H2 plants could be sited near cities for the manufacture of hydrogen from water and manufacture of potable water from seawater (see Fig. XXIV-1).

---

[1] STAR = Secure, Transportable, Autonomous Reactor. The STAR reactors are referred to as "Batteries" because they store 20 years worth of heat and they load follow by passive means – delivering heat when it is requested by the balance of plant and passively shutting off when the request stops.

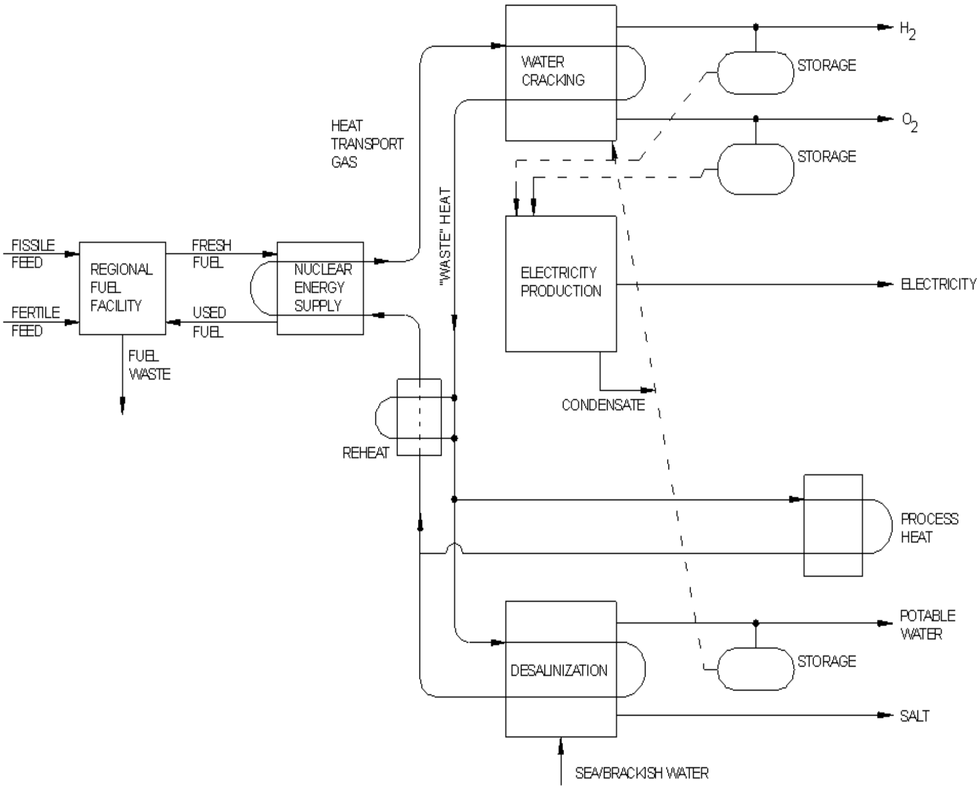


FIG. XXIV-1. STAR-H2 primary energy supply for cities.

The STAR-H2 plant would mesh with existing and planned urban energy distribution infrastructures using grid delivery of hydrogen, electricity, potable water, and communications (and sewage return) through a common grid of easements.

STAR-H2 is intended to meet the needs of two categories of customers – (1) utilities seeking to provide all primary energy and potable water needs of cities in developing countries including those in the early stages of economic development and having limited infrastructure; and (2) independent power producer (IPP) customers in developed countries who wish to enter emerging markets for hydrogen and/or potable water production.[2]

Both categories of customers desire limited capital outlay, rapid site assembly and early initiation of a revenue stream, outsourcing of the front end and back end fuel cycle and waste management services, reduced operational staffing requirements, near-urban siting based on unprecedented levels of safety and robustness with respect to equipment malfunction and/or human error, and a non-nuclear safety grade balance of plant.

Developing country customers may additionally seek the energy security afforded by very long refuelling interval and of legally binding assurances of access to fuel cycle and waste services from the regional fuel cycle centre. Finally, developing country customers would welcome the job creation and economic growth opportunities, which could derive from the

[2] Independent power producers (IPPs) are merchant generation companies who operate outside the regulatory framework of regulated utilities and sell their product on a competitive market (i.e., they receive no guarantee of profitability in exchange for a guarantee of providing service to consumers.)

STAR-H2 design approach for a non-safety grade balance of plant, which can be constructed and operated by local companies to local standards and using local labour.

The 400 MW(th) STAR-H2 is being designed to achieve 44% conversion of heat to lower heating value (LHV) of H_2 in a thermochemical water cracking cycle [XXIV-4] – making 160 MW(th) days/day of H_2 (LHV). It uses a supercritical CO_2 Brayton cycle to generate electricity for on-site needs [XXIV-5], and a feed-forward multi-effect distillation bottoming cycle [XXIV-6] to manufacture 8000 m^3/day of water – enough to support primary energy and potable water needs of a city of 25 000 using primary energy at 4 toe/capita/year – the level of use in Western Europe. Overall 85% of the reactor's 400 MW(th) is to be converted to energy products; 15% would be rejected in the form of heated brine.

XXIV-1.3. Special features

STAR-H2 is designed to fuel a sustainable mid 21st century global nuclear-driven hydrogen economy operating in hierarchical hub-spoke energy architecture. STAR plants are targeted for worldwide deployment and especially for urban centres in developing countries – using nuclear fuel and hydrogen as the long distance energy carriers – and supporting distributed electricity generation as the local energy carrier.

To break the energy security / non-proliferation dilemma, STAR-H2 plants are designed with 20 year refuelling interval to fit within a proposed hierarchical hub-spoke energy supply architecture using 20-year whole core refuelling cassettes as the energy carrier from regional fuel cycle centres [XXIV-7]. The regional centres are provided to handle both front and back end fuel cycle services, including waste management. They are assumed to be under the operational control of consortia of regional customers and could operate under international non-proliferation oversight. Whole core cassette refuelling operations on a 20-year refuelling interval would conducted by regional centre personnel using relocateable refuelling equipment, which they bring to the STAR site to conduct refuelling operations and then remove and take away with the used cassette.

Figure XXIV-2 illustrates the proposed hierarchical energy delivery infrastructure at an abstract level. The "hubs" represent conversion equipment where one energy carrier (nuclear fuel, hydrogen, electricity) is converted into the successive energy carrier along the supply chain – a carrier better suited to the required function. The "spokes" represent the transmission channels of the energy carrier from its source point to its point of use. The ordered sequence of energy carriers (nuclear fuel shipped from the regional centres to the battery type nuclear power plants sited near a city's perimeter; hydrogen and water piped from the STAR nuclear plants to the district load centres; and electricity wired from distributed fuel cell and/or micro-turbine electricity production centre to end use) are organized sequentially (hierarchically) in the order of their energy density and their associated power carrying capacity through practical-sized conduits (e.g. ships/trains; pipelines/trucks; wires, respectively). The widths of the spokes in Fig. XXIV-2 suggest the power carrying capacity of practical conduits for each energy carrier; the fractal-type expansion of the architecture as it progresses from the uranium ore energy resource to the point of end energy use reflects the diminishing energy carrying capacity and corresponding multiplicity of carrier conduits in the hierarchical sequence of energy carriers.

For example, a two-week voyage to deliver a single 400 MW (th) whole core fuel cassette good for 20 years (at a capacity factor of 0.9) in a STAR power plant represents a 188 GW(th) power transmission conduit. A single ship carrying ten cassettes on an itinerant one month delivery voyage from a regional centre could supply nearly 1000 GW(th) (1 terawatt years/year) to its service region.

A fleet of 10 ships could provide 10 terawatt (thermal) years/year, which rivals the current entire world primary energy use of ~12 terawatt (thermal) years/year.

Marchetti has observed [XXIV-8] that the economical scale of equipment sited at the "hubs" would expand to match the energy demand in the geographical area circumscribed by the spokes. Because of the enormous energy density of nuclear fuel contained in the refuelling cassettes, the "reach" of the nuclear fuel supply "spokes" through practical sized transport conduits (ships), i.e. the energy demand met in the area circumscribed by the spokes, can be thousands of miles and, as a result, the fuel cycle facilities at the regional centres should be sized for economy of scale to service the very large demand arising from a significant global region. Even if providing for a plausible world demand (~50 terawatt (thermal) years/year) by mid century, no more than a dozen such fuel cycle centres could meet the world's entire primary energy needs. In that sense they could be viewed as the 21st century analogue to the oil fields of the twentieth century.

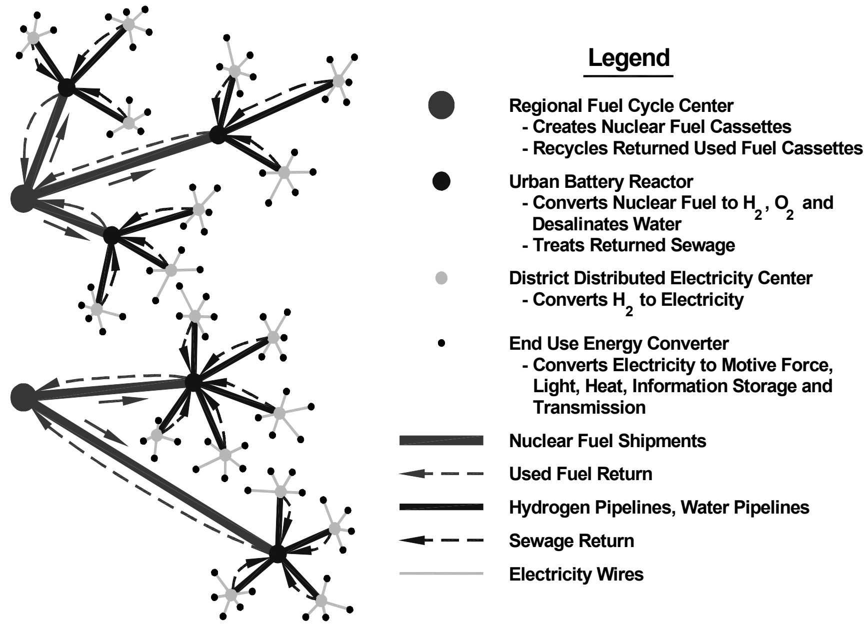


FIG. XXIV-2. Hierarchical hub/spoke energy architecture.

The "reach" of the next link in the supply chain – the hydrogen pipeline "spokes" – would reflect their several GW carrying capacity [XXIV-9] and would service regions of several hundred mile dimension though pipeline grids such as are currently used to distribute natural gas to load centres. Pipeline grids would carry hydrogen (and water) to district centres scattered throughout the city and its surrounding population region. At a primary energy use rate of 4 toe/capita/year [3] (i.e. ~5.5 kW(th)/person continuously for a year), a 5.5 GW hydrogen pipeline could service a city and its environs with a population of a million people.

After manufacture at a STAR-H2 power plant located at the margins of the city, the hydrogen and water will be piped or trucked to city districts through a grid of distribution conduits.

---

[3] Four tons of oil equivalent per capita-year (toe/capita year) is the average current primary energy use rate in Europe.

At district level distribution hubs, the hydrogen would be partitioned to meet society's energy service needs; currently in developed countries, primary energy is allocated roughly equally among transportation, heating of homes and industry, and applications to electricity generation. In this way:

— A third could be dispensed for hydrogen-fuelled transportation services;
— A third could be distributed by pipe throughout the district for heating homes, apartments, offices, and factories; and
— A third could be converted in fuel cells and/or micro-turbines to electricity for distribution throughout the district.

The "reach" of the electricity distribution wires starting at district micro-turbine or fuel cell converters of hydrogen to electricity and taking the electricity to final use in lighting, motors, and information management would be of the scale of city districts and skyscrapers, as is the current usage. This last stage of distribution would use the existing electrical and water distribution network (where it already exists) and would thereby make the conversion to the new energy architecture nearly transparent to the end user of energy services.

By mid century, district-level conversion of hydrogen to electricity – as opposed to conversion of heat to electricity at the STAR reactors sited at the city perimeter – is envisioned for several reasons. The first – and the one, which is already driving a transition – is supply reliability. Micro-turbines and (imminently) fuel cells could provide secure electricity at a district level, even if the broader grid suffers a shutdown, because they run on a storable supply – currently natural gas, but eventually hydrogen. Some planners believe that distributed generators will, in fact, eventually drive the grid.

The second driver is that the hot water produced as the "waste" from conversion of hydrogen to electricity at district hubs can be used in support of the city's hot water needs. This will increase marketable product for the owner of the conversion equipment but more importantly, it would reduce the water vapour and thermal plume ecological footprint of the conversion step. This sets the scale of electricity production at a district level because of the limited "reach" of hot water distribution spokes.

The overall conversion efficiency of nuclear heat to district-level re-conversion of hydrogen to electricity, [fission heat → hydrogen → electricity] would be about $0.45 \times 0.80 = 0.36$ which is already better than that of current LWRs. The overall conversion efficiency of nuclear heat to district level energy products [fission heat → hydrogen → electricity + hot water] would be about 0.45. When potable water manufacture from the STAR-H2 process plant is included, the overall conversion [nuclear heat → energy services] could reach 85%.

The proposed hub/spoke energy architecture optimized for nuclear energy systems thus envisions a worldwide total of a dozen or less regional fuel cycle centres each servicing thousands of long refuelling interval STAR battery heat source reactors, which individually or in clusters service cities and their surrounding regions with hydrogen and potable water. Hydrogen substitutes for fossil fuel in the transportation and heating sectors. Electricity and hot water are produced from hydrogen at distributed district centres and electricity reaches its final point of use through wires.

Carrier conduit cross-connections of the user hubs to multiple supplier hubs (not shown in Fig. XXIV-2) and energy storage buffers provided by the storable nuclear fuel and hydrogen energy carriers would provide for robustness of energy security at both the national and the individual user levels, and for protection against monopolistic pricing.

Over time, in a transition lasting of the order of a century, the hydrogen could gradually displace oil, gas, and coal and the new sustainable, nuclear-based architecture would gradually replace the current fossil-based world energy supply infrastructure. The resulting fission based energy supply architecture could provide centuries of energy on the known plus speculative ore base recoverable at ≤ $130/kg U identified in the “Redbook” [XXIV-10].

The new architecture could mesh seamlessly with existing and imminent urban energy distribution infrastructures using grid delivery of electricity, hydrogen, potable water, and communications and sewage return through a common grid of easements. This would facilitate incremental market penetration.

XXIV-1.4. Summary of major design and operating characteristics

Table XXIV-1 summarizes the STAR-H2 design and operating characteristics

TABLE XXIV-1. SUMMARY TABLE OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

CHARACTERISTIC	VALUE
Installed capacity	400 MW(th)
Mode of operation	Autonomous load follow based on passive feedbacks
Load factor/availability	The targeted capacity factor (CF) is 90 % for the operation in (base load → storable product) mode, with refuelling once every 20 years. Possible need to regenerate the coolant every N years.
Type of fuel	Uranium/Transuranic (TRU) nitride in clad cylindrical fuel rods. Enriched in nitrogen isotope ^{15}N.
Fuel enrichment (TRU/HM) (Fast neutron spectrum, internal conversion ratio = 1.0)	Reference core design: radially heterogeneous core layout; ductless assemblies. 3 enrichment zones: 13.14 % inner and mid zones; 1.4×13.14 % outer zone.
Coolant	Lead (T_{Inlet} = 663.7°C; T_{Outlet} = 793.4°C)
Moderator	None
Core structural materials	Unknown; calculations assume SiC/SiC composite
In-vessel structural materials	Unknown; calculations assume SiC/SiC composite
Core (Fuel assembly is shown in Fig. XXIV-3) Assembly flat to flat size Active core height Above core fission gas plenum height Below core axial reflector height Open-lattice of cylindrical fuel rods on a triangular pitch (optional square pitch) lattice. Cladding outer diameter Cladding thickness Coolant volume fraction Fuel volume fraction	- 9 rows of fuel (or blanket) assemblies; - 2 rows of reflector assemblies. 16.24 cm 2 m 2.0 m 0.25 m 1.905 cm (driver and internal blanket) 1 mm 0.667 0.248

CHARACTERISTIC	VALUE
Fuel pellet-cladding bond Reflector	Pb 50 volume % ferritic-martensitic stainless steel and 50 volume % Pb
Reactor vessel Outer diameter Height Thickness Design lifetime	Cylinder with hemispherical lower head. 5.5 m 16.9 m Unknown 60 years
Cycle type	Indirect cycle: intermediate loop of forced circulation molten salt. Ca-Br thermochemical water cracking cycle/ supercritical CO_2 Brayton cycle with a feed-forward multi-effect distillation bottoming cycle
Number of circuits	2 Primary circuit: natural circulation, ambient pressure Pb; intermediate circuit: forced circulation ambient pressure flibe molten salt
Neutron physical characteristics Refuelling cycle length Number of batches Burn-up reactivity swing Peaking factor Power flattening	 20 full power years 1; whole core cassette refuelling K_{eff} = 1.00 (BOL); 1.013 EOL ~1.5 except ~1.8 near BOL and EOL Three-zone radial enrichment zoning and internal blankets
Reactivity control mechanism	- Shutdown rod for start-up and shutdown. - During operation, reactor power autonomously adjusts to load by means of inherent physical processes without the need for any motion of control rods or any operator actions. - System temperatures change corresponding to reactivity feedbacks from fuel Doppler, fuel and cladding axial expansion, core radial expansion, and coolant density effects. - Control rods for possible fine reactivity compensation during cycle (tentative). (Control rods would also provide for diverse and independent shut down.)
Energy conversion cycle type (See Fig. XXIV-4) Conversion efficiency (LHV of H_2)/(reactor heat) Bottoming cycle	The reactor heat drives a Ca-Br thermochemical water cracking cycle and a supercritical CO_2 Brayton cycle (sized for on site needs). ~ 44% Feed forward multi-effect distillation desalination.

CHARACTERISTIC	VALUE
Thermal hydraulic characteristics	Primary coolant based on natural circulation of lead. No primary coolant pumps.
Core inlet temperature Core outlet temperature Primary coolant flow rate Primary coolant cover gas pressure Temperature limit for cladding Average fuel temperature Average cladding inner surface temperature Maximum fuel temperature Maximum cladding inner surface temperature during normal operation	663.7°C 793.4°C 21 770 kg/s Slightly below 1 atmosphere ~ 950°C (tentative) 970°C 803°C 995°C (hot channel) 878°C (hot channel)
Maximum/average discharge burn-up of fuel	Average = 82 MWd/kg (drivers); 28 (internal blankets) Peak = 126 MWd/kg (driver).
Fuel lifetime/period between refuellings	20 full power years
Mass balances/flows of fuel: Initial loading Initial TRU loading Internal conversion ratio	 29 600 kg of heavy metal 1700 kg TRU ~ 1.0 Best estimate calculation using DIF3D and REBUS-3 computer codes.
Design basis lifetime: Core refuelling cassette Reactor vessel In-vessel structures	 20 years. 60 years (tentative) 60 years (tentative).
Design and operating characteristics of systems for non-electric applications	STAR-H2 is dedicated to H_2 production with desalinated water production using reject heat
Economics	To be determined

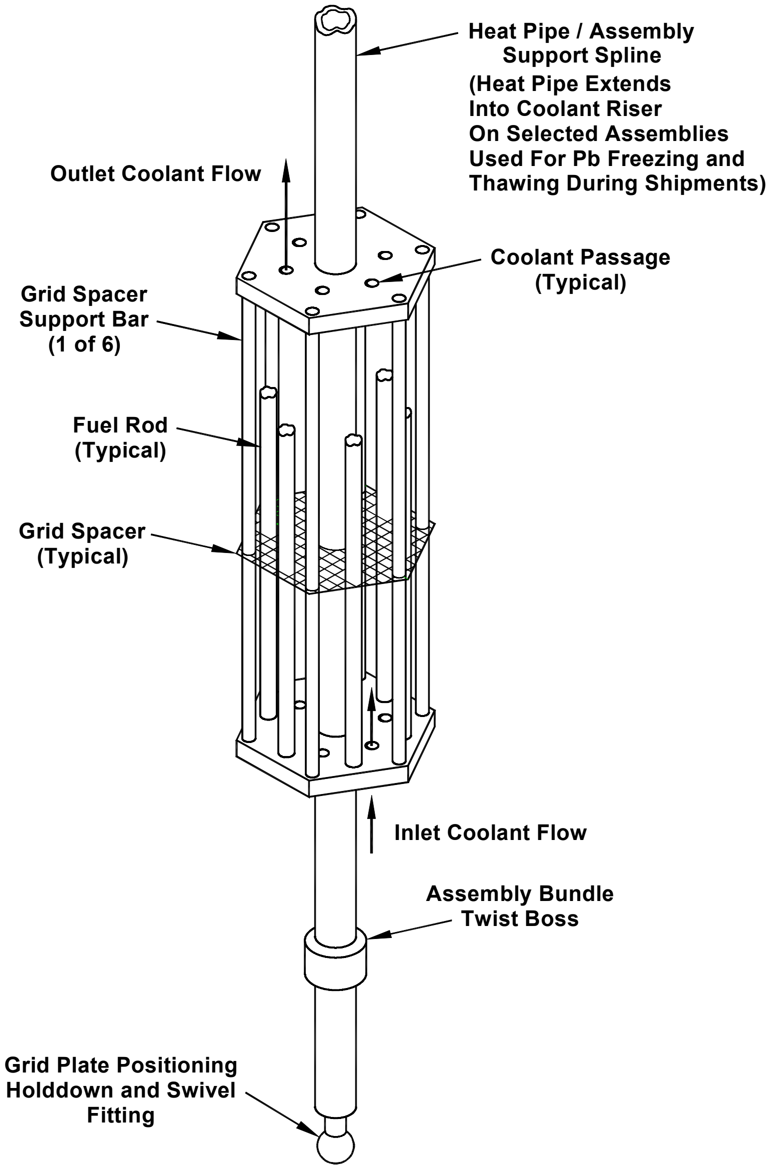


FIG. XXIV-3. Open sided fuel assembly with local grid spacer, mounted on a central heat pipe support spline.

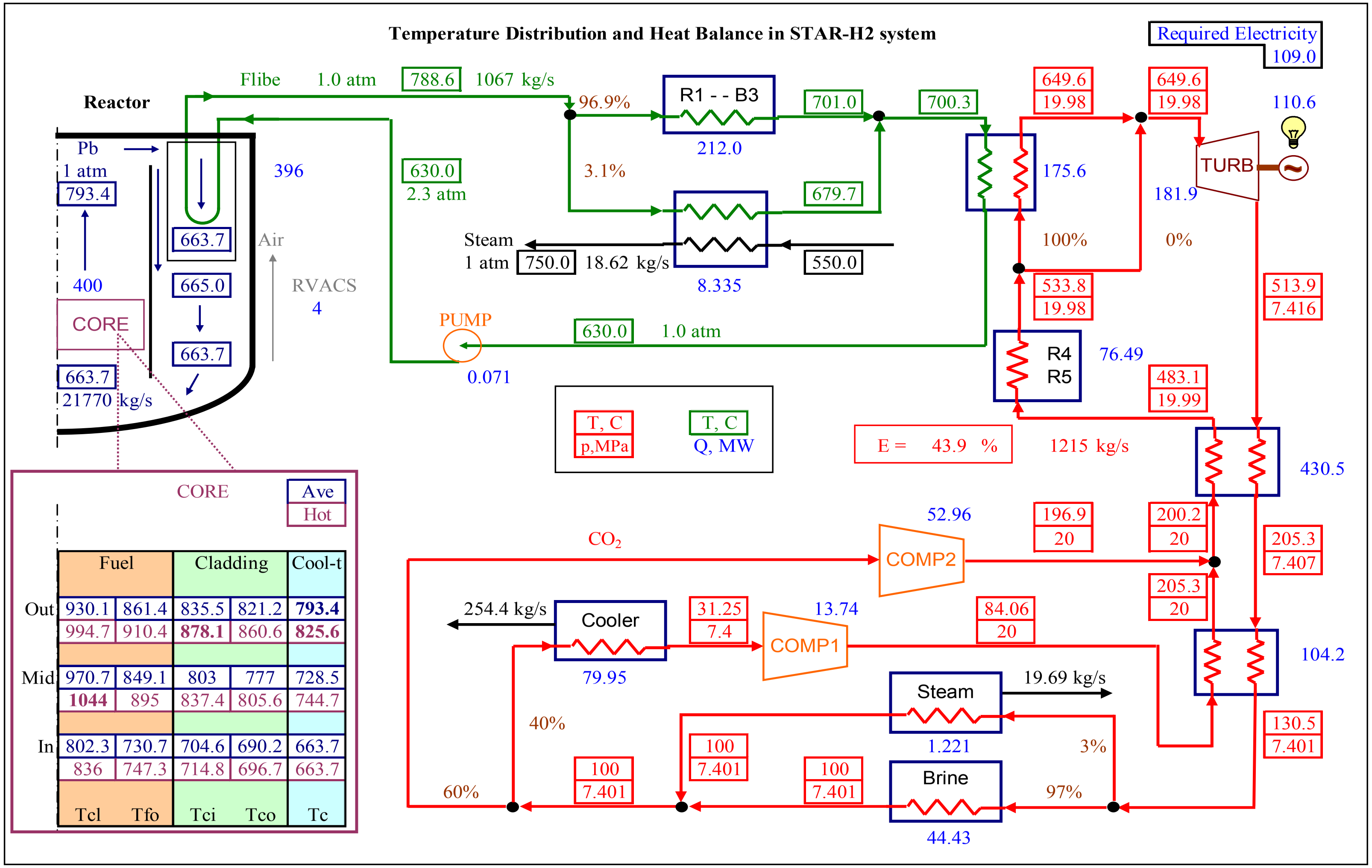


FIG. XXIV-4. Simplified schematic diagram of the STAR-H2 plant, illustrating the cascade of processes: water cracking → Brayton cycle → desalination.

XXIV-1.5. Outline of fuel cycle options

STAR-H2 employs a closed fuel cycle. The reactor is fuelled with uranium/transuranic (TRU) nitride fuel enriched in ^{15}N and it operates on a 20-year whole core cassette-refuelling interval; it is fissile self sufficient with an internal core conversion ratio of one.

All fuel cycle services and waste management is assumed to be outsourced to a regional fuel cycle centre, which is owned and operated under control of consortia of its customers while under international non-proliferation oversight.

Current thinking is that pyro recycle and vibropack remote refabrication technology would be developed and used for the TRU/U based nitride fuel.

The recycle technology is assumed to produce a commixed stream of all transuranics and achieve incomplete fission product removal such that the transuranic materials during processing and during fresh and used cassette shipping are always at least as unattractive for weapon use as is LWR spent fuel. All fuel cassette shipments and used cassette returns would be conducted by regional centre personnel who bring the refuelling equipment with them and take it away with the spent cassette. No refuelling equipment would remain at the site. No spent fuel would be stored at the site for cooling.

The fuel cycle feedstock will be natural or depleted uranium, and multi recycle through sequential cassette reload cycles could achieve total fission consumption of the ^{238}U feedstock; only fission product waste forms (and trace losses of transuranics) would go to a geologic repository operated by the regional centre.

Once deployed, each STAR will be fissile self-sufficient. Initially, fuel for STAR-H2 new deployments could come from transuranics recovered from spent LWR fuel. Later, when that source is exhausted, fast breeder reactors could be sited at the regional fuel cycle centres. Their function will be to manufacture excess fissile material so as to fuel the initial working inventories of new STAR deployments in a growing economy. The heat from their operation could be converted to hydrogen for shipment to regional consumers.

XXIV-1.6. Technical features and technological approaches that are definitive for STAR-H2 performance in particular areas

Two major challenges for deployment of the hub/spoke architecture are (1) to achieve affordability of the STAR-H2 plant itself by producing designs and deployment approaches which can replace the economy of scale paradigm with an economy of mass production approach; then (2) to attract industrial interest for developing a STAR-H2 supplier business and a regional fuel cycle centre business by finding ways to overcome the financial barrier raised by the large up-front investment required of the suppliers.

Historically, the nuclear power plant business has been based on an economy of scale approach for capping capital cost per unit of power rating ($/kW(e)) combined with low fuel cost. This approach requires large initial capital outlay by the customer but it has succeeded in regulated utility markets because low risk attendant to the regulatory compact caps the utility’s cost of capital. Moreover, in developed countries the grid can accept large incremental additions to capacity. The regulated utility carries most of the financial risk but can garner equity and debt because of a regulatory guarantee for return on capital. The supplier works on an essentially cost plus contract to build a customized plant.

The business strategy for small battery type plants might be totally reversed from that used in the past. For small battery reactors the business risk could be transferred predominately to the supplier who would spread its cost over many hundreds of replicate units. The customer will purchase a standard design commodity nuclear power plant, - already license certified - which he can bring on-line with a very short on-site installation and checkout period. This would allow to start a revenue stream shortly after taking on his financing loans and equity. Moreover, the small heat rating of the plant would lower his overall capital outlay and provide just in time capacity additions into a small grid.

The abovementioned battery plant business strategy is designed to reduce customer financial risk for the market conditions found in developing countries and/or for merchant plants operating outside the regulatory compact in developed countries. But it would require the supplier to take on the increase in risk attendant to emplacement of economy of scale reactor production factories and regional fuel cycle centres.

To give more details of the proposed business strategy for small battery type plants, section XXIV-1.6.1 outlines strategies to achieve market penetration by meeting customer needs for the targeted categories of customers. Then, section XXIV-1.7 outlines potential supplier business strategies to deal with the radical realignment of risk in the approach to STAR nuclear deployments operating in a hub/spoke nuclear architecture.

XXIV-1.6.1. Economics and maintainability

Market penetration potential is strongly influenced by price of the service offered in comparison to alternatives that the buyer has available to him at the time and place of his placing his order. However, his selection decision is nearly always made in light of many additional considerations – not all of which are financial ones.

Starting with price, the Generation IV economics goals call for competitiveness with respect to available alternatives. As metrics, the Generation IV targets were pegged to the current US energy market where a modern coal-fired power plant costs US $1.50 per watt to build and 1 cent[4] to 1.5 cents per kilowatt-hour (kWh) to operate; a small gas fired aero-derivative turbine plant costs 60 cents per watt to build and 3.8 cents per kWh to operate; and a large modern LWR nuclear plant costs between US $1.50 to US $2.00 per watt to build and around 1cent per kWh to operate.

However, given a global marketplace it is evident that at any given time, the price for the same energy service varies dramatically both geographically and by customer type. Of energy resources, only oil has a global price, whereas wood, coal, gas, and uranium have regional prices. Moreover, the price of an energy service to the consumer reflects not only resource cost but also conversion and delivery costs which vary according to labour markets, infrastructure availability, weather, government interventions (e.g. taxes or incentives) and many other variables. Finally, price available to the consumer reflects local market conditions including such things as the amplitude and stability of each consumer’s demand and his time dependence of demand – factors, which may give him favoured status with the supplier.

Thus, as is reported by such agencies as the US EIA and the OECD/IEA, the price of the same energy service to the consumer can vary geographically by well over a factor of five at any given time. As an example, Fig. XXIV-5 compares the consumer price for electricity in various developed countries. Similar diversity can be seen in the price of gasoline. Clearly, what is considered an economically competitive price for market entry of STAR-H2 depends on what alternative options the customer has available to him, at the time; current US prices may have no relevance in targeted segments of the future global energy marketplace.

[4] Cents are with reference to the US$.

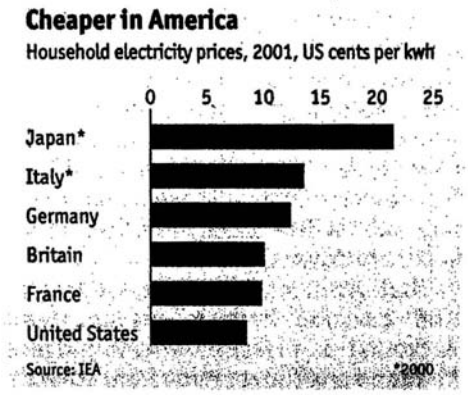


FIG. XXIV-5. Household electricity prices.

Therefore, STAR-H2 market penetration strategies could rely on:

(a) First, segmenting the market to the targeted customer base in developing countries and to merchant plant customers in developed countries; and then meeting perceived customer needs by:

(b) Reducing customer risk and associated cost of capital;

(c) Providing for customer's non-monetary needs; and

(d) Containing the customer's costs through:
   (i) Economies of plant simplification;
   (ii) Economies of mass production; and
   (iii) Operation and maintenance (O&M) cost reductions.

These business strategies directed at the customer are summarized below. Then, in Section XXIV-1.7, business strategies to attract supplier interest are described.

Segmenting the market

Large monolithic plants will continue to hold electric market share, and clearly the battery plant concept is only one slice of the overall future market share that nuclear energy concepts would occupy. The battery plant offering favours customer situations where availability of capital is dear and/or where financing costs are high and payback periods must be short. This might be the situation in developing countries and for merchant plants in developed countries.

From the buyer's point of view the attractiveness of a battery plant rests on his acceptance of receiving a standard-design, pre-licensed commodity product and outsourcing responsibility for both the power plant fabrication/installation and the fuel cycle services – as an alternative to the cost of developing an indigenous front-to-back fuel cycle industry.

Such a market segment is a different segment from the one serviced by large, economy of scale plants and, therefore, a head-to-head capital cost comparison is meaningless. For targeted niche applications of the STAR-H2 such as remote areas and developing countries, the energy supply alternatives are constrained such that prices for services are far from the US average prices used as metrics in the Generation IV rating of concepts. As a relevant example, Table XXIV-2 shows the price currently paid for electricity in remote villages in northern Alaska and Hawaii [XXIV-11] – potential market locations indicative of where STAR-H2 might expect to attract customers. These prices are ten times higher than the US averages and moreover they illustrate the additional price spread in the differentiated market between industrial and domestic users.

Future market research will be required as STAR-H2 becomes further developed, but the point is that the price targets for STAR-H2 market penetration will depend strongly on the specific market segment to be served. That is not to say that strong efforts at cost reduction are not needed, and these are discussed in later sections of the chapter.

TABLE XXIV-2. COST OF ELECTRICITY TO USERS IN SELECTED LOCATIONS

Cost of Electricity to Users in Selected Locations
(1999 Financial & Production Data[3])

State	Utility	Location	Cost cents/kWh (Industrial or Commercial)	Cost cents/kWh (Residential)	Total Number of Customers	Net Generation MWh	Purchased Power MWh	Peak Power MWe
Alaska	Golden Valley Electric Assoc.	Fairbank	5.9	9.3	35,945	611,227	438,528	175
	Chugach Electric Assoc.	Anchorage	6.5	10.3	68,862	2,091,897	232,789	412
	Alaska Electric Light & Power	Juneau	7.2	9.6	14,623	66,533	246,896	59
	Sitka Municipal Utilities	Sitka	7.6	9.3	4,533	94,583	0	18
	Ketchikan Public Utilities	Ketchikan	7.7	9.5	7,090	90,863	68,135	28
	Anchorage Municipal L&P	Anchorage	7.8	9.9	29,567	704,704	215,090	151
	Nome Joint Utility Systems	Nome	13.4	20.8	1,781	29,298	0	5
	Bethel Utilities Corp	Anchorage	18.5	23.7	2,219	37,152	0	7
	Alaska Village Electric Coop Inc.	Anchorage	36.0	45.0	6,371	53,940	0	12
Hawaii	Hawaiian Electric Co.	Honolulu	8.8	13.0	273,968	4,391,007	2,965,718	1161
	Maui Electric Co.	Kahului	12.7	14.5	55,786	1,062,099	69,973	180
	Hawaii Electric Light Co.*	Hilo	15.1	20.6	59,744	599,875	380,603	167

*1997 data for Hawaii Electric Light Co.
Note: Costs have been escalated to year 2000 dollars.

Reducing customer risk and cost of capital

Strategies are being specifically designed to reduce customer financial risk when purchasing a STAR-H2 plant.

First, the smaller heat rating of a battery plant lowers his overall capital outlay from several billion to several hundred million dollars and thereby reduces customer's financial exposure. The customer won't get as much power as from an economy of scale plant but he may not need as much.

The customer could purchase a commodity nuclear power plant, already license certified, and he would bring it on-line with a very short on-site installation and checkout period so as to start a revenue stream shortly after taking on his financing loans and equity. In this way, the licensing uncertainty would be eliminated and the interest during construction could be reduced.

Strategies for addressing the significant cost of the fuel cassette are discussed in the end of the section.

Providing non-monetary value

Price isn't everything; energy security and economic growth are major considerations in decisions on energy asset acquisition. For example, limited access to fuel supply or a fuel delivery route which is under the control of regional adversaries or even an unreliable delivery schedule due to weather conditions are all important energy security considerations. The 20-year assured fuel supply offered in the STAR-H2 refuelling cassette might offer unprecedented energy security advantages.

Given that an energy service provider is building a (non safety grade) balance of plant in any case and assuming that he has long-term contracts for the products he will produce, then he wants to select a heat source having long-term predictable cost of heat. Since the balance of plant assets have multi decade lifetimes, the perceived likelihood that future fuel costs will remain capped over the projected multi decade plant lifetime is a consideration. When competing with a coal or gas fired heat source to drive his balance of plant, a customer might assign value to a nuclear battery heat source because nuclear fuel cost is not only lower but is likely to be stable over multi decade balance of plant and asset lifetimes, whereas oil and gas prices are already rising and coal is facing emissions mitigation cost escalation.

Over and above the generally upward trending in fossil fuel prices is their volatility. As resource bases and/or distribution infrastructure becomes stressed, temporary supply shortages occur locally and/or globally – giving rise to price volatility – especially in deregulated energy markets. Figure XXIV-6 shows an example of electricity price volatility in the western Canadian market during the year 2002 [XXIV-12]. Spikes of factors of 20 are seen, and a trend of increasing volatility is seen in the last half of 2002.

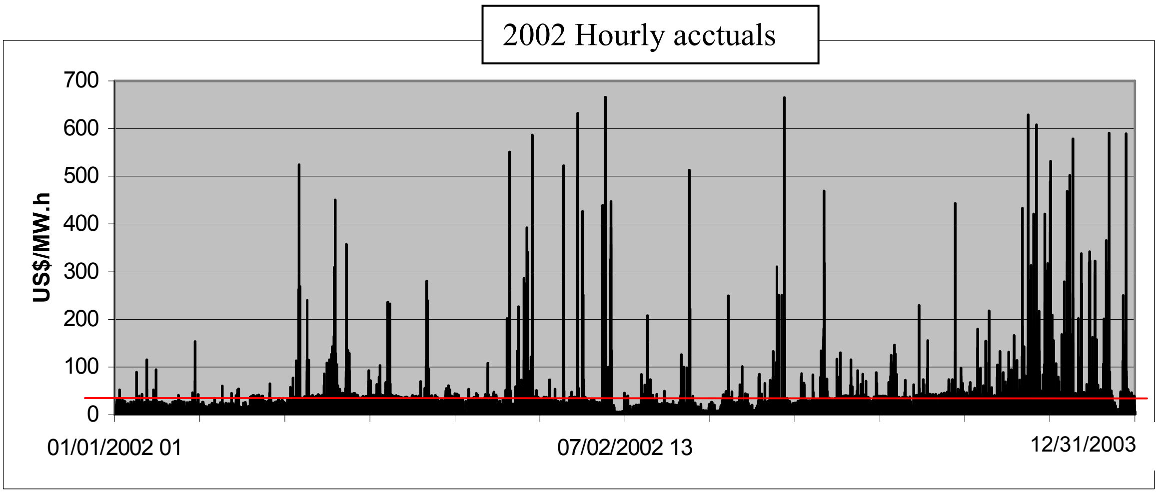


FIG. XXIV-6. Alberta Pool electricity price (US$/MW h).

As another example, a volatility-exacerbating trend was building up in US electricity markets in the mid 1990s as producers increased their dependence on gas fired turbine plants – not only for peaking demand but incipiently for base load as well. While temporarily halted in the last few years, this trend may be revised as the economy recovers and if so, this will place stress on gas markets which are already strained by needs for home heating and inadequate pipeline and storage infrastructure. If this trend resumes, the US electricity prices can be expected to become ever more volatile.

Long term fuel cost trending and price volatility are considerations which affect a utility's purchase decision; an operator with multiple units might be prepared to pay a premium to assure that at least one component of his energy service supply portfolio will be based on a stable fuel cost and will function to dampen the volatility of his overall cost of energy product. Avoidance of fuel price volatility risk could be yet another non-monetary reason to favour the stable energy costs afforded from long refuelling interval battery type reactors such as STAR-H2.

Local job creation

A desire for creation of jobs and business for local companies, which retains national wealth within the nation, is a strong consideration in developing economies. STAR-H2 has purposely been designed so that the balance of plant is non-nuclear-safety grade and could be built by local companies to local building standards using local labour. While the STAR-H2 heat source reactor itself might be purchased from an outside supplier, the BOP, at least, could benefit the local economy directly through job creation in both construction and operation and financing denominated in local currency.

Achieving economy of simplification and mass production

Coming now to the design approaches to contain capital costs; in general the customer's capital outlay for small battery type reactors would be reduced because of small heat rating (size) and this could be a business advantage in markets where capital is dear and interest rates are high.

But the capital cost per unit of nameplate energy (\$/kW(e)) is likely to increase – perhaps dramatically – by foregoing the economy of scale benefit. The STAR-H2 design strategy attempts to at least partially overcome this by simplification, factory fabrication, short construction interval and more efficient and complete conversion of heat to marketable products.

The most important strategy for capital cost containment is the safety approach. By designing STAR-H2 for passive safety, two crucial cost saving benefits might be accrued. First, the balance of plant could be built to non-safety grade standards using standard components which are factors of up to 3 or 5 times less expensive than nuclear safety grade.

Second, a classical containment building could be avoided. For LWRs and for classical oxide fuelled liquid metal fast breeder reactors (LMFBRs), under a hypothetical severe accident the containment pressure goes inversely with volume whereas the containment building cost goes directly with containment surface area. Since surface to volume goes inversely with diameter, it has been a traditional design strategy to minimize the large fixed cost component of containment building per unit power by designing for a large power rating, i.e. to employ an economy of scale strategy.

It is, therefore, clear that the first crucial issue for small battery type reactors is to develop a passive safety approach that would eliminate the need for a conventional containment and a nuclear safety grade balance of plant. If this is not achieved, then there is no hope for economy with small reactors that have foregone economy of scale at the outset.

The additional strategies to counteract the capital cost (\$/kW(e)) escalation for small battery reactors is based on simpler reactor control and refuelling systems, factory mass fabrication, modular construction and rapid start-up of a pre-licensed plant. For example, the STAR-H2 design eliminates primary pumps and on-site refuelling equipment.

Multiple unit factory production runs can be expected to accelerate learning curve benefits and yield better pricing from component suppliers.

Achieving O&M cost reductions through simplification

O&M cost reduction strategies for the STAR-H2 depend first on increasing marketable product per unit of heat released by fission. The Ca-Br hydrogen production process suggested for the STAR-H2 may achieve ~44% conversion of heat to lower heating value (LHV) of H_2. Then, the 66% remaining "waste heat" could be converted to marketable potable water. For STAR-H2, eighty-five percent of fission heat overall is converted to marketable products – hydrogen and potable water – and only 15% of the STAR-H2 heat is rejected to ambient; strategies are being considered to reduce this still further to near zero by exploiting the low temperature waste heat in biological converter processes.

Further operating cost ($/kW(e) h) reduction strategies depend on very high capacity factor due to few refuelling shutdowns, fewer safety grade systems to maintain, use of passive load follow with increased automation of controls – all leading to consequent possible reduction in staffing. Moreover, remote monitoring and dispatch of specialized itinerant maintenance support teams provided by the supplier could yield reductions in the number of highly trained staff required to be stationed permanently at the site (similar strategy is already used for heavy mining and construction equipment on a global basis).

Finally, a more advanced heat engine is proposed for the STAR-H2. Traditionally, a superheated steam Rankine cycle has been used for liquid metal cooled reactors having a core outlet temperature in the 500°C range. However, the Rankine cycle BOP constitutes a major contributor to the cost disadvantage of nuclear systems vis-à-vis combustion gas turbine plants. Brayton cycles have benefited from five decades of additional R&D in comparison to Rankine steam cycle designs; Brayton cycle BOPs could be both simpler to operate and cheaper to build. For these reasons the STAR-H2 design provides for coupling to a Brayton cycle for production of electricity needed for on-site operations. The supercritical CO_2 Brayton cycle (called the Feher cycle) is proposed for the STAR-H2 [XXIV-4], which features extremely small component sizes (even compared to helium cycle equipment) and which can achieve thermal efficiencies exceeding 40% at the inlet temperature of 600-650°C available from Ca-Br water cracking cycle heat rejection.

Fuel leasing arrangements

The specific power, (kW/kg of TRU fuel), of long refuelling interval battery plants has been reduced by a factor of 5 or 6 relative to traditional fast neutron spectrum reactors. This allows for a 5 or 6 times increase in fuel residence time per refuelling interval (from 3 years to 15 to 20 years) within a given maximum achievable fuel discharge burn-up capability:

$$\left.\frac{MW\ days}{kg}\right|_{\substack{discharge \\ burnup}} \equiv \left(\frac{MW}{kg}\right)\left(\frac{days}{reload}\right) \tag{1}$$

i.e. with the attainable discharge burn-up (left member of the identity) fixed, then the first factor on the right must decrease in order that the second can increase. This strategy will always be the essence of a battery reactor design until higher burn-up fuel has been developed.

Derating of the specific fuel power has significant financial implications. The same amount (and fabrication cost) of fuel for the battery, if placed instead in a conventional fast reactor, could generate revenue from power sales in only three years instead it is put in a battery and generates its revenue in 20 years. All things being equal, the battery reactor user would have to pay an upfront premium for the fuel concomitant to his stretched out revenue stream, because the net present value of his revenue is derived over a longer period[5].

The interesting feature of the battery plant fuel cassette, however, is that the fissile content of the cassette remains constant. In order that the burn-up reactivity loss will be zero, the core is designed to breed as much new transuranics as it burns. There is as much fissile transuranic fuel left in the cassette at the end of 20 years as there was at the start; it need only to be

[5] This penalty is mitigated to some extent by two factors; first the TRU/heavy metal ratio is reduced in STAR designs compared to conventional breeders; second, the required out of reactor to in-reactor working inventory ratio for a 20-year cycle is much less than for a 1-year cycle 3 batch core.

reprocessed to separate out the fission products; then with the addition of (inexpensive) ^{238}U the core cassette can be refabricated ready for another 20 years of power production.

A third party long-term investor might think of a reload cassette as a 20-year "bond" because there is no loss of principal (transuranic mass). In this way, a third party long-term investor who wants the security of retaining his principal and the payoff of receiving a fair return from "loaning" the principal (core cassette) might purchase the cassette and lease it to the owner of the battery power plant in return for monthly payments. From the leasor's viewpoint, his risk is limited because he can always repossess the cassette. From the leasee's viewpoint, he can avoid the upfront capital purchase price of the fuel cassette, and can instead pay a monthly expense tied to his use of it – partially offsetting his revenue from power sales. He "pays on time".

The fuel cassette leasing idea is a business plan like automobile leasing; therefore, all the various options developed for that business could be applied as well to cassette leasing – lease with option to buy; lease with option to upgrade; lease with option for accelerated or decelerated payments upon refinancing, etc.

Reactor leasing arrangements

Since not only the refuelling cassette but even the battery heat source reactors are themselves transportable, the same ideas regarding leasing could be applied to the battery modules as well – invoking all the business strategies employed in the used car, used truck, and used airplane secondary market industry. While the balance of plant energy converters are owned by the customer and they remain grounded at the customer's site; the STAR nuclear battery heat sources that drive the balance of plant are designed to be transportable, replaceable, and upgradeable. A third party might wish to own and lease a fleet of them, or a third party "dealer" might buy and sell them – a used STAR reactor module in good condition might represent an upgrade relative to a customer's worn out or out-moded one.

XXIV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The hierarchical energy architecture utilizing the STAR-H2 concept is being devised to meet both the energy resource sustainability and the environmental compatibility tenants of sustainable development [XXIV-13].

As to resource sustainability, the known plus speculative economically recoverable ore of ~15 million tons of U, when fully fissioned, could supply the world's entire primary energy needs for a millennium.

The application of nuclear heat to produce hydrogen as a replacement for fossil fuels might achieve an essentially greenhouse gas free energy supply chain extending from resource to end use and it allows nuclear to move beyond electricity to service all sectors of primary energy usage.

Processes convert one energy carrier to another at the hubs of the architecture illustrated in Fig. XXIV-2 (e.g. nuclear fuel to hydrogen; hydrogen to electricity) or to energy services (nuclear heat to potable water, electricity to motive force, etc.). These conversion processes generate wastes. The proposed architecture provides for an ecologically neutral closure of the entire energy supply enterprise through recycle of these wastes as illustrated in Fig. XXIV-7. Referring to Fig. XXIV-7, closure is obtained on electricity production and use by electron return through ground. Closure is obtained on thermochemical water cracking hydrogen production and its use in fuel cells or micro-turbines by nature's oxygen and water cycles.

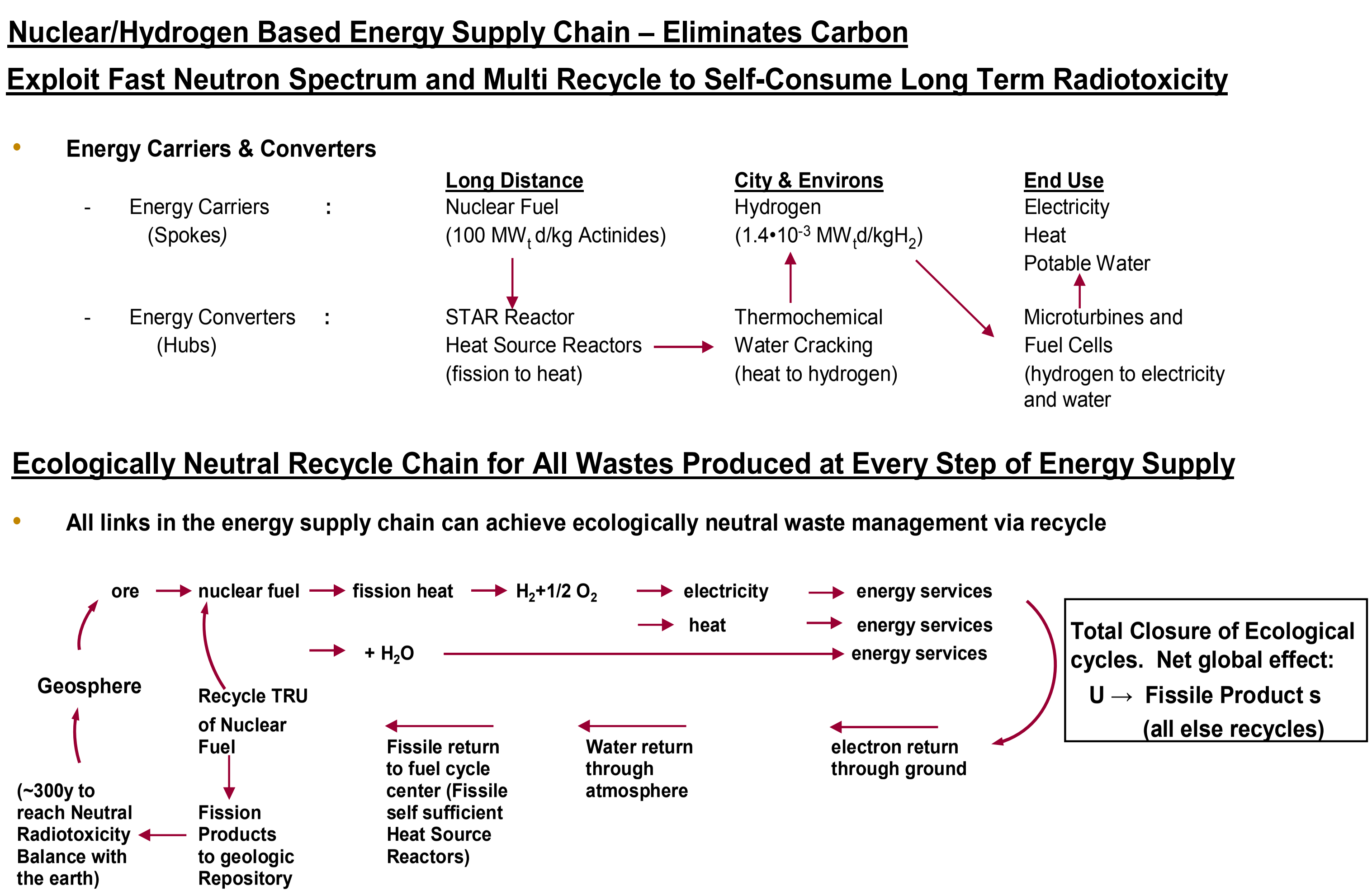


FIG. XXIV-7. Ecologically neutral energy supply chain.

Potable water closure could be obtained by pipeline return of sewage from city districts to the battery power plants where the O_2 by-product of hydrogen production would be put to productive use for sewage treatment. Closure of the nuclear fuel cycle could be obtained by designing the battery reactors for fissile self-sufficient operation and transuranic recovery and recycle at the regional fuel cycle centres. With only fission products consigned to waste, the radiotoxicity sent to waste would decay to a value no larger than the radiotoxicity of the original uranium ore within two and a half centuries [XXIV-14]. With the use of reject heat from the STAR plant to desalinate water and the use of the hot water produced by fuel cell or micro-turbine conversion of hydrogen to electricity, even the thermal plume "waste" in this architecture could be minimized.

In an ideal sense, the net ecological effect of the entire energy supply chain would be the consumption of uranium ore and the creation of waste fission products; whereas everything else in all links of the energy supply chain could be recycled. The envisioned hierarchical hub/spoke nuclear-based global energy architecture is intended to be ecologically neutral and fully sustainable in both its resource availability and waste management aspects.

XXIV-1.6.3. Safety and reliability

Safety concept and design philosophy

The STAR-H2 reactor employs the passive safety strategy developed for the Integral Fast Reactor [XXIV-15]. A low pressure system and double walled vessel eliminates loss of coolant vulnerabilities. A large thermal inertia, large margins between operating and damage temperatures and passive decay heat removal channels eliminate loss of decay heat removal vulnerabilities. Passive self-regulation, which innately matches heat production to heat removal via reactivity feedbacks, – when coupled with design for zero burn-up reactivity loss – eliminates reactivity insertion and station blackout vulnerabilities. Natural circulation cooling at full power eliminates pumping loss vulnerabilities.

The balance of plant is assigned no nuclear safety function, and passive load following is achieved via reactivity feedbacks in response to heat demand communicated only by means of the molten salt intermediate loop. These two features could help totally decouple reactor safety performance from equipment failures and plant operator or plant maintenance personnel mistakes in the balance of plant.

Open pitch of the fuel pin lattice helps avoid blockages. Chemical compatibility of fuel and coolant allows for run beyond cladding breach. Dramatic margin exists between coolant operating point (800°C) and boiling point (~1700°C). Seismic isolation is being considered to reduce inertial loads, which might affect operation of thermo-structural reactivity feedbacks. Disrupted fuel floats in the coolant and disperses radially at the coolant/cover gas interface, avoiding recriticality.

Provisions for simplicity and robustness of the design

Improved reliability

Long refuelling interval and production of storable products (H_2, O_2, water) facilitates base loading with infrequent power level adjustments; this also could allow for the achievement of very high capacity factors irrespective of hydrogen delivery schedules. Infrequent load changes, large safety margins and plant simplification (no primary pumps, no refuelling equipment) along with passive load following (simplified control system) and no safety functions assigned to the balance of plant could help reduce complexity and scale of overall plant operations. Modern energy converters (fuel cells, Brayton cycles and/or H_2/O_2

combustion turbo-generators) reduce scale and complexity of balance of plant operations. Remote monitoring via satellite and timely dispatch of specialty support teams from the regional fuel cycle support centre could improve cost effectiveness and reliability of specialty maintenance tasks.

Investment protection

The plant is designed at low power density and large operating margins with natural circulation cooling, innate load following, and passive safety for high reliability and forgiving robustness with respect to balance of plant failures or operator/maintenance personnel mistakes.

The reactor is a low pressure vessel filled with a low chemical potential coolant – explosions and fire hazards are low for the reactor itself. The chemical plant, where explosive chemicals are handled, and industrial hazards exist is decoupled from the reactor by distance and by the molten salt intermediate heat transport circuit operating at ambient pressure; since the reactor can remain within a safe operating regime while innately adjusting its power production to any heat demand communicated through the intermediate circuit, – intended or spurious – events in the industrial chemical plant would not influence reactor safety performance.

The modular sizing puts a cap on capital outlay. The long refuelling interval provides long term fuel supply security.

Structure of the defence in depth

The reactor is designed for a near-zero reactivity burn-up swing such that the safety rod system is vested with minimal positive reactivity at Beginning of Life (BOL) full power. A safety rod scram system provides a first line of defence for reactivity initiators. Moreover, passive reactivity feedbacks and passive self adjustment of natural circulation flow could maintain reactor power to flow ratio in a safe operating range even with failure to scram; this safe passive response applies for all out-of-reactor vessel initiated events, i.e., for any and all events communicated to the reactor through the flibe intermediate loop. Periodic in situ measurements would be made to confirm the operability of these passive feedbacks.

A decay heat removal path is provided through the heat transport system through the balance of plant and ultimately to a seawater ultimate heat sink. Additionally, a passive decay heat removal channel operates continuously carrying ~ 1 % of full power from the pin lattice to the ambient air, using passive natural circulation, conduction, and radiation heat transport links. This passive path may be periodically tested in situ to assure its operability. The thermal inertia of the primary circuit coolant is sufficient to safely absorb the initial decay heat transient, which exceeds the 1% capacity of the passive heat removal channel.

The fuel, coolant, and internal structural materials are chemically compatible such that clad/coolant chemical interaction is avoided with control of coolant chemistry and such that run beyond clad breach due to manufacturing flaws would not lead to autocatalytic degradation – even for the very long duration of refuelling operations. In situ monitoring of coolant and cover gas conditions would be used to confirm normality of conditions.

The first line of containment defence is the fuel cladding; the second line of defence is the reactor vessel wall and head cover and the intermediate heat exchanger (IHX) tube walls. The third line of containment defence is the guard vessel and its cover and perhaps quick acting valves on the flibe intermediate loop piping. Since there is no credible high pressure hazard within the reactor vessel or the intermediate heat transport loop, the guard vessel and its cover is a low volume (high surface/volume ratio) containment made of thin-wall steel. The reactor building has no containment function.

Active and passive systems and inherent safety features

Passive load follow capability

The reactor is connected to the BOP through the flibe heat transport loop and only through the flibe heat transport loop. The heat demand from the BOP is made known to the reactor through the flibe flow rate and the flibe return temperature. The flibe loop delivers heat from the reactor to three heat exchangers in the BOP: to the $CaBr_2$ bed in the water cracking vessel; to the last stage steam super-heater, and to the last stage SC-CO_2 heater. The flibe is then recirculated back to the nuclear reactor heat exchanger, and the rest of the BOP runs on heat cascaded down from these processes, see Fig. XXIV-1.

The goal for passive load flow design is to use the intermediate loop flow and temperature information and only this information to cause the reactor to self adjust its power level such as to exactly match that heat demand communicated through the flibe loop – and to do so within a safe operating envelope.

Since the Ca-Br water splitting chemical reaction requires a heat source at ~725ºC to cause the $CaBr_2 + H_2O \leftrightarrow CaO + 2HBr$ reaction to take place, that temperature must be maintained even if the water cracking plant is operating at only partial load. Therefore, for STAR-H2, another design constraint was imposed – the core outlet temperature of 800ºC at full load must also be maintained at all levels of partial load.

The basic character of the STAR-H2 passive load follow design strategy is as follows: the Pb coolant outlet temperature should remain constant vs. fraction of full load whereas coolant temperature rise across the core should increase vs. fraction of full power – implying that core inlet temperature T_{Inlet} decreases with increasing fraction of full load. This, in turn, increases the buoyancy driving head for natural circulation, which depends on differences in temperature of coolant exiting the core and the heat exchanger.

The core average coolant temperature and fuel temperature - the two temperatures which dominate contribution to reactivity feedbacks – should behave as follows: as fraction of full power is increased, the fuel temperature rise above the coolant will increase, which adds negative reactivity due to Doppler and fuel axial expansion. To the contrary, and so as to offset the negative reactivity of fuel temperature rise, the coolant average temperature should decrease with increasing fraction of full power, adding positive reactivity. This can happen – given that core outlet temperature remains fixed – by causing coolant inlet temperature to decrease as heat demand increases. The design challenge is to create a core design and a reactor natural circulation cooling circuit such that the two reactivities cancel at every value of partial load and the flow readjusts to produce the required core temperature rise ΔT vs. power level.

The coupled neutronics/thermo-hydraulic/thermo-structural reactivity feedback design approach for the STAR-H2 reactor has achieved the proper ratio between that reactivity which is vested in the coolant temperature rise relative to inlet temperature vis-à-vis that reactivity which is vested in the fuel temperature rise above the coolant, and at the same time in having designed an overall coolant flow circuit pressure drop tailored to cause coolant flow rate to adjust properly to changes in pressure driving head caused by source/sink temperature difference. A non-conventional open-pitch ductless fuel assembly structural design coupled with a non-conventional core support approach (the assemblies tend to neutral buoyancy in the dense Pb coolant) has been proposed to simultaneously provide low pressure drop, structural reliability of grid spacers, and an appropriate value for coolant power/flow reactivity temperature coefficient.

The resulting STAR-H2 partial load schedule is shown in Fig. XXIV-8. The interplay of coolant ΔT and natural circulation coolant flow rate is shown in Figure XXIV-9.

Coupled neutronics/thermal hydraulics stability analyses of the STAR reactor at these plant equilibrium states at full and partial load will be required. Such analyses have been conducted already for the STAR-LM which shares the neutronics and thermal hydraulics properties of STAR-H2 reactor, – and stability has been demonstrated.

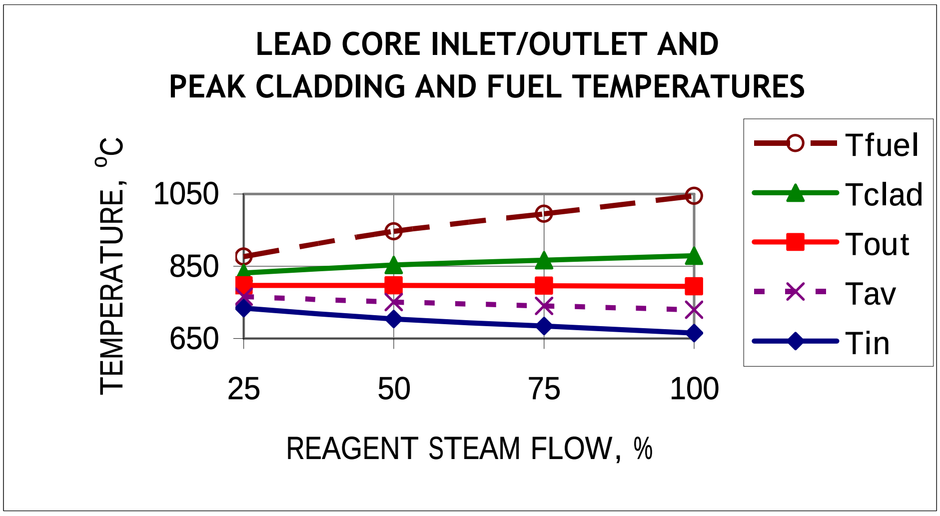


FIG. XXIV-8. Reactor partial load schedule.

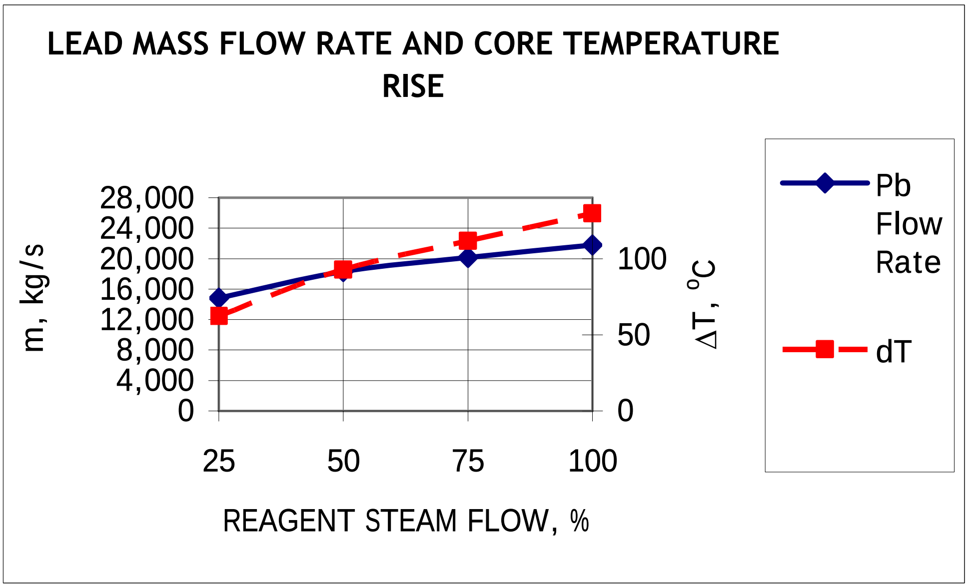


FIG. XXIV-9. Reactor coolant flow rate and temperature rise vs. percent (%) of full load.

Passive safety response

Given that the STAR-H2 reactor will passively self adjust its power level to meet the heat request from the BOP communicated via the flibe flow rate and return temperature (when the BOP is under purposeful control), it is still necessary to show that the reactor would not self adjust itself to damaging power or power/flow levels when upset conditions exist in the BOP.

The STAR-H2 reactor has a central safety rod and an active scram circuit. It also will have a decay heat removal path through the flibe loop. However, in order to achieve the levels of safety required for deployments of tens of thousands of STAR-H2 plants worldwide – sited near cities, it will be essential to avoid core damage even in the absence of a scram. The technology to achieve passive safety response to anticipated transients without scram (ATWS) events was well developed for the IFR concept [XXIV-15] and was famously demonstrated in tests conducted at the EBR-II reactor [XXIV-16].

The quasi-static reactivity balance theory of designing for passive safety response to ATWS initiators has been applied in the design of STAR-H2. The efficacy of this design approach is confirmed by the evaluation results to be presented next.

Design basis accidents and beyond design basis accidents

Bounding BOP and control room events

Since the only communication channel from the BOP to the reactor is through the flibe loop, it might be possible to bound all possible BOP conditions – whether intentional or spurious – via limiting flibe conditions at the reactor IHS. These off normal conditions could arise from BOP equipment failure or from maintenance errors or from operator error. The point is that the reactor "sees" the external world through only one widow – the flibe loop. It could thereby be possible to deterministically span the space of all possible externally initiated accident events that the reactor will face and to determine whether the reactor's passive response (without scram) will hold the reactor in a safe condition. The flibe flow rate can change; but it can't decrease to less than zero nor can it increase to more than that which cavitates the pump – taken here to be 115% of full flow. The flibe return temperature can change, but it can't increase to above the Pb delivery temperature at the reactor IHX nor can it decrease to below the lowest temperature it encounters in the heat exchangers it passes through in the BOP.

Finally, the flibe pressure can change, but it can't go below ambient (which is the normal condition) nor can it go above the pressure of fluids, which it encounters in heat exchangers (should a tube rupture occur). Since all processes except the Brayton cycle operate at ambient pressure, this upper pressure bound is 20 MPa – should a tube in the flibe to SC-CO_2 heat exchanger rupture.

All physically feasible conditions of the flibe communication channel from the external world to the reactor are bounded by the physically limited extremes listed in Table XXIV-3. These are innate physical bounds – they span the space of all possible conditions communicated to the reactor from outside the vessel.

TABLE XXIV-3. PHYSICAL BOUNDS ON FLIBE LOOP PARAMETERS

CATEGORY OF DISRUPTION	PHYSICAL LIMITS	
	Lower limit	*Upper limit*
Flibe flow rate disruptions	Zero	Pump cavitation
Flibe return temperature disruptions	Lowest temperature encountered in the three heat exchangers in BOP: - Reagent steam super-heater; - Water cracking heat exchanger; - CO_2 heat exchanger.	Pb temperature in reactor heat exchanger (HX)
Flibe pressure disruptions	Ambient (normal conditions)	CO_2 Pressure (Flibe/CO_2 HX tube rupture)

Reactor passive safety response to bounding BOP events

The resulting reactor power level and the coolant, cladding, and fuel temperatures which result from the ensemble of the bounding conditions in the flibe loop have been calculated using the quasi-static methodology used previously for passive load follow analysis. The calculations were made in response to the specific BOP conditions enumerated in Table XXIV-4. These BOP conditions represent plausible off normal events in the categories of off normal flibe flow disruptions (overcooling and under-cooling) and off normal flibe return temperature disruptions – under-cooling and overcooling events. Table XXIV-4 defines a mnemonic name for each off normal Anticipated Transient With Scram (ATWS) event analyzed – for shorthand use in Fig. XXIV-10 showing the results.

TABLE XXIV-4. ATWS EVENTS, WHICH SPAN THE SPACE OF BOP INITIATED ACCIDENTS

CATEGORY		NAME	FLIBE LOOP CONDITION			DESCRIPTION
			Flibe flow	Flibe return temperature	Flibe pressure	
Base case		Nominal 100% Power	Nominal	~ 630ºC Nominal	Ambient	Normal 100% power condition
Pump disruptions		LOHS	0		Ambient	Loss of heat sink; flibe flow stops
Pump disruptions		POS	115% of normal	Nominal	Ambient	Pump over-speed; flibe pump over-speeds to cavitation (assumed 115%)
Return temperature disruptions	Under-cooling	LOCP	Nominal	>630ºC	Ambient	Loss of chemical plant heat sink; Brayton cycle continues to run
Return temperature disruptions	Under-cooling	LOBC	Nominal	>630ºC	Ambient	Loss of Brayton cycle heat sink; chemical plant heat sink continues to run (assumes off-site or emergency electricity source)
Return temperature disruptions	Overcooling	COS	Nominal	<630ºC	Ambient	CO_2 Compressor over-speed (assumes CO_2 compressors at 155% of normal flow rate)
Return temperature disruptions	Overcooling	SBD	Nominal	<630ºC	Ambient	Reagent steam line blow down (taken to be 300% normal flow through flibe to steam heat exchanger)
Pressure disruption			Nominal	Nominal	Overpressure	Flibe to CO_2 heat exchanger tube rupture

Loss of heat sink

Considering the flibe pump speed disruptions to the limits of their physical bound, the first case (see Table XXIV-4) is the reactor loss of heat sink (LOHS) case where the flibe flow rate stops. As shown in Fig. XXIV-10a, the reactor power level is driven to decay heat level by action of the negative temperature coefficient operating on a reactor coolant inlet temperature increase. Although fuel temperature decrease adds positive reactivity, negative reactivity from coolant temperature increase dominates and leads to net reactivity decrease. The Pb outlet temperature decreases slightly and, as the power is zero, the coolant temperature rise collapses to essentially zero. Clad and fuel temperatures each drop (see Fig. XXIV-10c), and the reactor becomes essentially isothermal at about 790ºC. There is no core damage and the reactor's asymptotic equilibrium state is delayed critical at decay heat level with the RVACS passively removing decay heat.

Flibe pump over-speed

Second, the overcooling accident (POS) caused by flibe pump speed increase to cavitation assumed at 115% of full flow leads to an asymptotic state with power and power to flow ratio at about 101% of nominal (Figs. XXIV-10a and XXIV-10b). Since the power/flow remains essentially unchanged, the coolant average temperature raises only slightly; clad and fuel temperatures remain very near nominal.

In summary, the two flibe flow disruption accidents – which bound all other flibe flow disruptions – cause the reactor to self adjust asymptotically to a safe operating state – even without scram.

Off-normal flibe return temperature – too high

Looking next at the flibe return temperature disruptions, it is clear that the flibe return temperature can never exceed the reactor Pb outlet temperature (which would comprise a LOHS case already discussed) nor can it lie below the lowest temperature it encounters in the BOP heat exchangers it passes through. It passes through three heat exchangers: the reagent steam super-heater, the heat exchanger to the Ca Br_2 + H_2O water cracking bed, and the CO_2 heat exchanger. An off-normal condition in any heat exchanger could arise from too little heat removal from the flibe (partial loss of load) or from too much heat removal (overcooling accident).

The first category (flibe return temperature increase) results from partial loss of heat sink; two plausible "partial loss of heat sink" cases have been examined – loss of chemical plant heat sink (LOCP) and loss of Brayton cycle heat sink (LOBC). The resulting clad and fuel temperatures are shown in Figure XXIV-10c. When the flibe return temperature increases due to partial loss of load, the reactor power self adjusts downward to match the decreased load under action of the coolant average reactivity coefficient. Clad and fuel temperatures decrease and the reactor passively adjusts to a safe operating state.

Off-normal flibe return temperature – too low

The second category (flibe return temperature decrease) results when too much heat is removed in one of the three heat exchangers – reagent steam, water cracking, or CO_2 heat exchangers. In the case of the water cracking $CaBr_2$ bed heat exchangers, the $CaBr_2$ bed is a fixed solid bed at atmospheric pressure, and the only plausible overcooling would come from excess reagent steam flow. Thus, this case is bounded by the analysis of overcooling in the reagent steam heat exchanger. Overcooling by reagent steam flow could occur from feedwater pump over-speed, from loss of function of the regenerative HBr to reagent steam heat exchanger or from steam line blow down following a pipe rupture. However, as the reagent steam is at atmospheric pressure and lacking a driving force, the “blow down” would be benign. Detailed studies of such a blow down or of loss of the HBr to reagent steam regenerative heating have not yet been completed and so a conservative case (SBD) of 300% nominal reagent steam flow has been used[6]. The results of this case are shown in Figure XXIV-10. Due to a decrease in flibe return temperature, the positive reactivity change from lowering reactor inlet temperature causes power to rise from 400 to 405.3 MW(th). Power to flow ratio increases by only 1%; thus negligible increase in clad and fuel temperatures occur. The reactor passively accommodates this accident sequence without damage, even without scram.

The other possibility for flibe overcooling is in the flibe to CO_2 heat exchanger. That case has been examined assuming the CO_2 compressors both over-speed to 115% of nominal capacity (COS). The resulting reactor power and temperatures are shown in Figure XXIV-10. Here power increase is substantial – from 400 MW(th) to 424.7 MW(th). However, again natural circulation passively increases to match increased power; power to flow ratio increases by only 6% and the clad and fuel temperature increases are small. Again the reactor passively adjusts to a safe operating state – even without scram.

Flibe loop over-pressurization – SC-CO_2 HX tube rupture

The last category of a bounding off normal BOP condition affecting the reactor is flibe loop over-pressurization. A tube rupture in the flibe to SC-CO_2 heat exchanger would subject the flibe, which is normally at atmospheric pressure, to a ~3000 psi pressure source. Absent some intervention, a compression wave would travel at the speed of sound in flibe through the flibe loop piping to the in-vessel flibe to Pb intermediate heat exchanger (IHX) where the thin walled tubes designed for ambient-to-ambient pressure heat transfer would likely rupture and expose the reactor vessel to an abrupt pressure increase. This highly undesirable scenario has been faced and handled for Na reactors in the case of an intermediate Na loop/steam generator tube rupture. The design solution is to put a large diameter rupture disk in the intermediate loop. A small overpressure will rupture the disk – allowing the pressure to release to the atmosphere. This terminates the over-pressurization transient and leads to the loss of heat sink (LOHS) case, which has been shown above to lead to a benign passive shutdown. Alternately, a more advanced design for the flibe to SC-CO_2 heat exchanger itself could be considered, with flibe and SC-CO_2 tubes both immersed in a common pool of high conductivity fluid, such as flibe or Pb. A SC-CO_2 tube rupture would cause CO_2 venting to ambient through the fluid in the tank, and the flibe tubes would remain intact.

Summary of ATWS event passive response

To summarize, the asymptotic (bounding event) analyses performed have shown all bounding cases of ATWS initiators originating from a BOP disruption to be passively accommodated within safe asymptotic temperature conditions. Decay heat removal was assumed to rely on passive reactor vessel air cooling system (RVACS). Both, the innate thermo-structural reactivity feedbacks and the innate decay heat removal pathway to ambient could be non-intrusively monitored to assure their continued capability to provide safe response. Thus, no matter what happens in the BOP, the reactor might self adjust itself to a safe asymptotic

[6] This will also bound a feedwater pump over-speed to cavitation – estimated to occur at 115% of normal flow.

condition. This could make it possible to construct and operate the BOP to ordinary industrial standards.

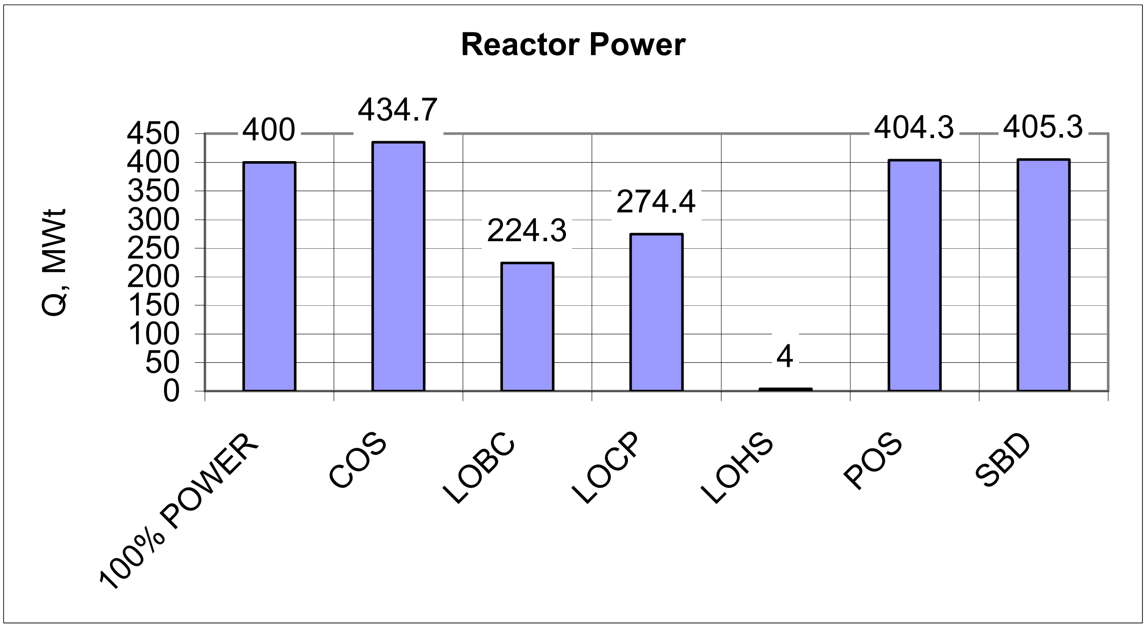


FIG. XXIV-10a. ATWS event asymptotic power levels.

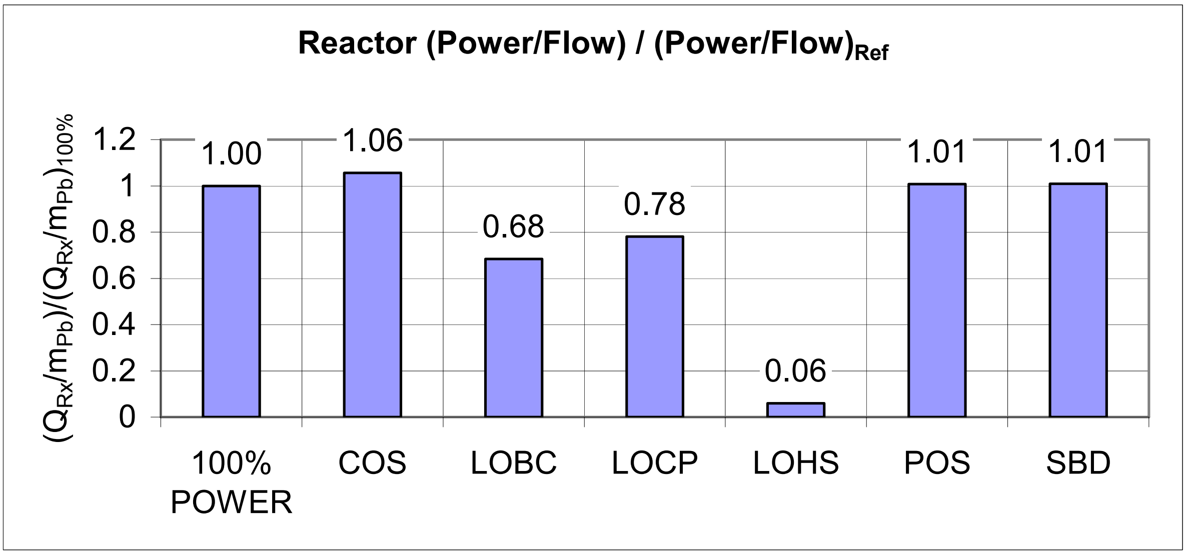


FIG. XXIV-10b. ATWS event asymptotic power/flow ratios.

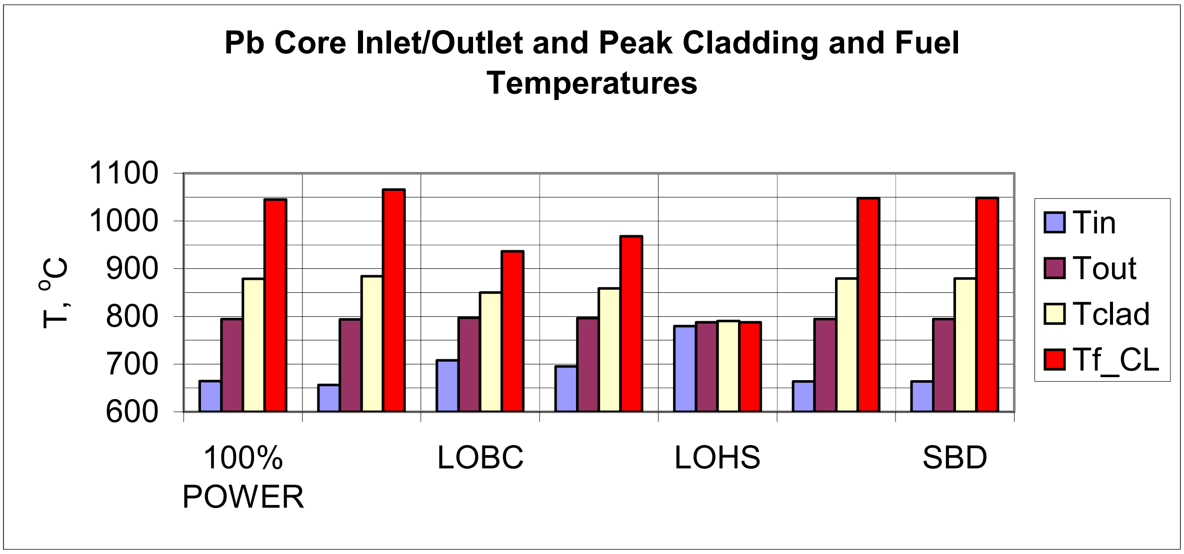


FIG. XXIV-10c. ATWS event asymptotic temperature levels.

Given that the asymptotic states in response to ATWS initiators are safe, it remains to show that the dynamic transition to the asymptotic state will not engender damaging conditions on the in-core structures. A plant dynamic code, which can model the STAR-H2 balance of plant, was not available at the time when this report was prepared. Such a code is being developed first for the STAR-LM, which has a simpler (SC-CO2 Brayton cycle) balance of plant. In the future, after further refinement of the Ca-Br water cracking cycle, that dynamics code will be modified for applicability to the STAR-H2.

Moreover, coupled neutronics/thermal-hydraulics stability analyses would be required for the ending equilibrium states from the passive accommodation of ATWS initiators. Work for the STAR-LM suggests that these states are indeed stable ones.

Beyond design base events and elimination of need for off-site emergency response

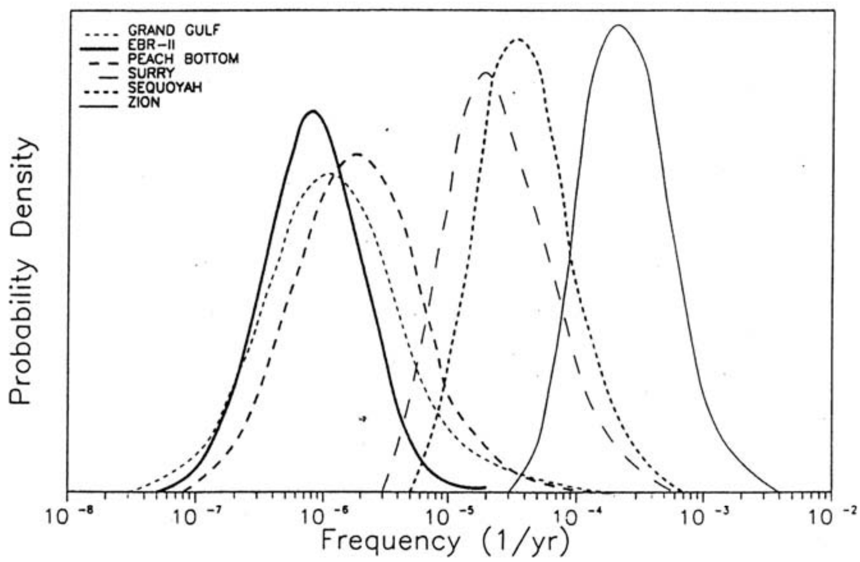
As shown in the previous section, the designed in and always operating passive reactivity feedback response and passive decay heat removal pathways might be capable to close off all conceivable pathways to core disruption using innate, in-situ testable processes. These innate responses apply to any event originating outside the reactor vessel.

Accident initiators, which might originate inside the vessel, could only have come from initial manufacturing or assembly flaws during construction or from long-term neglect of coolant chemistry control. Such initiators would (in future) be addressed by probabilistic risk assessment (PRA) methods and are expected (based on similar PRAs performed for sodium cooled systems) to represent triple-fault events of such low probability as to lie in the beyond design basis range. Moreover, as there are no credible mechanisms for high ramp rate, many such non-credible events would be gradual and could be annunciated early by coolant and cover gas monitoring before gross fuel pin disruption occurs.

Even in the event of fuel pin disruption, since the specific gravity of the nitride fuel and the Pb coolant are nearly identical, one could expect fuel particle dispersal and dilution in the vast Pb inventory, thus precluding re-criticality concerns.

Probability of unacceptable radioactivity release beyond the plant boundaries

The STAR-H2 safety strategy is adapted from that used for the IFR, which was demonstrated in full scale tests at the EBR-II sixty-two MW(th) power plant in 1986 tests [XXIV-16]; loss of heat sink without scram (LOHSWS) and loss of flow without scram (LOFS) both from full power as well as run beyond cladding breach were all demonstrated to yield benign results. The Level 1 PRA conducted for the EBR-II [XXIV-17] showed that probability of technical specification violation with marginal loss of fuel pin lifetime came in at a slightly lower frequency ($\sim 10^{-6}$/year) than the probability for core disruption and overall loss of the reactor for the PWR PRAs reported in NUREG-1150, see Figure XXIV-11.



"Damage" for EBR-II defined as overheating (vis-à-vis technical specifications) of "aggressive" test pins

"Damage" for LWRs defined as core disruption

FIG. XXIV-11. Comparison of EBR-II damage frequency with core damage frequency at commercial LWRs (LWR data from NUREG-1150).

XXIV-1.6.4. Proliferation resistance

The hierarchical energy architecture based on a closed nuclear fuel cycle using regional fuel cycle centres supporting long refuelling interval STAR-H2 heat source reactors has been devised to break the energy security/non-proliferation dilemma, which might arise for a massive deployment of nuclear energy systems as the world's principal future primary energy source.

Under the proposed architecture, the basic bargain of the Non-Proliferation Treaty (NPT) would be augmented by an additional bargain in which a nation would agree to forego the emplacement of an indigenous fuel cycle infrastructure in exchange for services of a regional fuel cycle centre, guaranteed by international law. In this way, bulk fissile handling can be confined to less than a dozen centres worldwide, allowing to cost-effectively focus safeguards oversight as compared to a situation where all countries deploy an indigenous fuel cycle infrastructure.

The idea of regional fuel cycle centres is not new; it was evaluated during the 1979–1981 INFCE activity [XXIV-18]. However, what are newly available are first, recycle/refabrication technologies, which maintain all transuranics in a commixed product containing residual fission products - which is unsuitable for weapons use [XXIV-19], and second battery plant designs for 20 year refuelling interval, which could provide for two decades of energy security and thereby assuage a nation's concern over foregoing the emplacement of indigenous front-to-back fuel cycle infrastructures including enrichment and reprocessing capabilities.

The recycle based on electrometallurgical recycle and remote vibropack refabrication technology produces a commixed stream of all transuranics and can achieve incomplete fission product removal such that the transuranic materials during processing at the centre and during fresh and used cassette shipping would be always at least as unattractive for military use as is LWR spent fuel [XXIV-19]. No transuranics (except trace recycle loses) is assumed to go to the waste repository.

It is expected that all fuel cassette shipments and used cassette returns would be conducted by itinerate regional centre refuelling teams who will bring the replacement cassette and the refuelling equipment with them, perform the refuelling operations, and take the refuelling equipment away with the spent cassette. No refuelling equipment would, therefore, remain at the battery heat source plant site (see Fig. XXIV-12).

The IAEA Director General Mohammed ElBaradei has recently highlighted [XXIV-20] the need for consideration of revised institutional strategies for achieving non-proliferation assurances in response to changing conditions including the potential expansion of nuclear energy deployments. M. ElBaradei has called for a nuclear energy architecture having three parts:

- "First, it is time to limit the processing of weapon-usable material (separated plutonium and high-enriched uranium) in civilian nuclear programmes, as well as the production of new material through reprocessing and enrichment, by agreeing to restrict these operations exclusively to facilities under multinational control."
- "Second, nuclear-energy systems should be deployed that, by design, avoid the use of materials that may be applied directly to making nuclear weapons."
- "Third, we should consider multinational approaches to the management and disposal of spent fuel and radioactive waste."

The proposed hub-spoke architecture for STAR deployment could meet the abovementioned three criteria for a new approach to non-proliferation. At the same time, by placing each regional fuel cycle centre's operational control under the governance of the customer countries themselves – as secured by international law – it might remove the current asymmetry of supplier vs. customer state and provide for every country's energy security.

The events of the past decade have shown that one of the most severe proliferation hazards can occur as a result of regime change when sovereign custody of widely dispersed nuclear assets suddenly evaporates or when national policy regarding nuclear weapons is reversed by a new regime and the NPT is abrogated openly or secretly. The architecture proposed here might reduce this vulnerability – even in a situation of widely extended deployment – by centralizing bulk fissile handling operations to less than a dozen worldwide sites under international oversight and by distributing fissile material – which is anyway unusable for weapons – contained exclusively in 20 year refuelling cassettes. These cassettes are immensely heavy; both fresh and used cassettes are radiation self-protecting "packets"; and while numbering in the tens of thousands, they could be nonetheless subject to remote global positioning system (GPS) monitoring and item accountancy procedures.

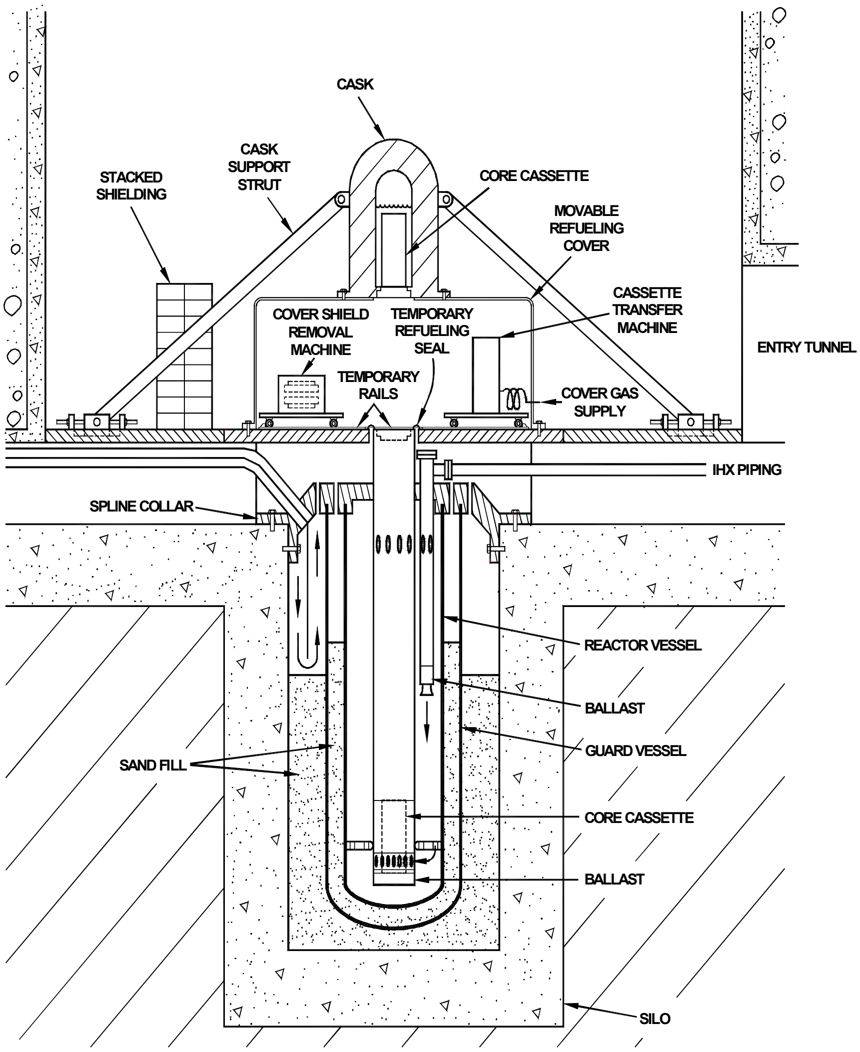


FIG. XXIV-12. Relocateable refuelling equipment.

XXIV-1.6.5 Technical features and technological approaches used to facilitate physical protection of STAR-H2

First off, the balance of plant (BOP) would have no nuclear safety function. Moreover, the STAR-H2 heat source reactor is being designed not only for passive safety response to Anticipated transients without scram (ATWS) initiators but also for passive load follow. The only information flow path from the BOP to the reactor would be the fused salt intermediate heat transport loop, which will convey the BOP heat request to the reactor by means of its flow rate and return temperature (see Fig. XXIV-3). In this way, the reactor could passively adjust its power to match heat demand while remaining in a safe operating regime. The safety implication of passive load follow is that the reactor would safety respond to all possible combinations and timing of ATWS initiators taken more than one at a time; it would also safety respond to all conceivable human errors of the maintenance crew and the operator. In summary, all faults exterior to the reactor vessel might be safely accommodated on the basis of passive thermo-structural feedbacks.

The passive safety design, the ambient pressure primary; and absence of internal chemical potential hazards would permit use of a high surface to volume containment comprised of a close-filling guard vessel, which in turn would enable the passive decay heat removal pathway across the guard vessel to ambient air. In summary, nothing that happens (planned or spurious; equipment or human related) in the BOP would lead to reactor damage because all pathways to core damage could intercepted and terminated by innate passive termination mechanisms.

The STAR-H2 reactor is assumed to be sited in a silo underneath an earthen berm. Figure XXIV-13 shows a side view of the concrete reactor building, its earthen mound covering the building and the reactor vessel emplaced below grade in a concrete silo within the reactor building.

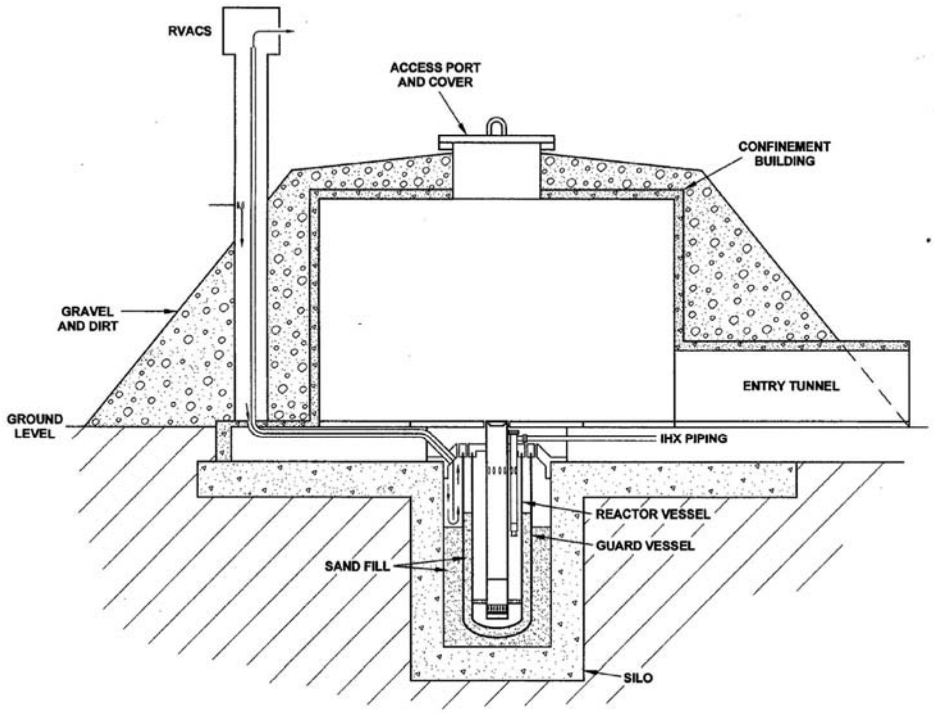


FIG. XXIV-13. Protection of the STAR-H2 reactor from external hazards.

The reactor building's function is to protect the reactor silo and reactor head from the natural elements, to contain ancillary reactor support equipment and to provide operating floor space for refuelling operations. It has no containment function.

The building has a large-diameter access port in the ceiling – large enough for open top reactor construction and whole core cassette shipping cask entry for refuelling operations.

The earthen/gravel mound over the reactor building is a low-cost means to protect the reactor and its containment structure from external hazards, which may include:

— Chemical plant/tank farm explosions and/or fires;
— Natural hazard missiles generated by high winds, tornadoes, hurricanes, or typhoons;
— Forest fires and scrub fires;
— Ocean swells or tsunami; and
— Deliberate attacks such as tank shells, airplane crashes, etc.

Chemical plant explosion hazards are addressed by providing separation distances and berms. The oxygen tank farm is placed furthest from the reactor and the BOP, and is separated from the hydrogen tank farm by the water tanks. Earthen berms are used to separate the hydrogen and oxygen tank farms, each from the other, and from the BOP. The BOP in turn is separated from the reactor by a berm and a canal.

XXIV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of STAR-H2

Assuming that development of the design strategies described above for the STAR-H2 concept would lead to a product judged to have market potential in its targeted customer segments, then it will still be necessary for suppliers to position themselves to proffer the STAR-H2 product. Emplacing the proposed architecture will require suppliers of STAR plants; regional fuel cycle facility builders and operators; and reactor and cassette transport and logistics companies. Moreover, numerous enabling institutional changes would be required. Strategies to emplace all these are discussed in this section.

STAR plant supplier: market entry strategies

The battery plant architecture rests on an assumption that the battery plant market size and durable growth (hundreds to thousands of plants) will create a sufficient demand to induce supplier strategies based on factory fabrication/rapid site installation of plant modules. Consider the range of projections cited previously in Section XXIV-1.3. In order to grow from 1 terawatt thermal (350 GW(e)) deployed today to 10 terawatt deployed in 2050, then the market potential for 400 MW(th) STAR-H2 plants starting in 2030 lies in the range of an average of 500 to 750 new plants delivered each year over the 20 years from 2030 to 2050. This potential market size could justify supplier interest.

The measures described in Section XXIV-1.6.1 are intended to reduce the risk premium on cost of capital for the customer. For small battery type reactors the business risk would be transferred predominately to the supplier who must initially emplace a large factory for economy of mass production fabrication. Uncertainty in durability of the market may raise the supplier's cost of capital, but unlike a customer, his intent will be to spread business risk cost over many hundreds of replicate units.

A supplier would have to foresee a sufficient market to invest in factories large enough to achieve economy of mass production from production runs of many hundreds of turnkey plants. While this is a change from historical patterns for the nuclear business, markets having such characteristics are already widespread in many other industries of large capitalization – for example in the airplane, automobile, construction equipment, military equipment, combustion gas turbine power plant and many other industries. In fact, in most industries the customer desires the immediate benefit from deploying a commodity product and receiving outsourced support services; in order to avoid development costs and delays incurred for a one of a kind custom product; the customer is prepared to pay incrementally for what others have already developed and proffered for sale at a profit. Business strategies to address the STAR plant suppliers initial risk are likely no different than encountered elsewhere, except for the licensing uncertainty risk.

Major financing challenges can be foreseen when attempting to create a battery plant supply business. Because the strategy to hold capital cost down is to rely on economy of mass production, a potential supplier must invest in a large-throughput factory, and the risk premium on capital borrowed to build the factory will be high absent a full order book to show to the bankers. But customers want to see both a fully licensed prototype and an evident capacity to deliver before they place an order. The module supplier faces a chicken & egg dilemma in preparing to exploit economy of mass production, because of need to reduce the risk of building a high volume factory, having no orders in hand, while at the same time, customers want to see evidence of official license approval and supplier commitment evidenced by palpable capacity to deliver before they place an order. And bankers also want to see such licensing assurance plus orders in hand before reducing the risk premium they will require on factory construction loans. Means to address these financing challenges could be drawn from analogy to other industries.

One way this chicken and egg dilemma might be broken is by creating a "virtual" battery supplier company by assembling a consortium of component suppliers – the vessel; the heat exchangers; the reload fuel cassette, etc. The battery supplier company would then handle assembly and marketing while each component supplier would supply specific components that they are already making anyway in their existing businesses. The battery supplier company could avoid much of the risk associated with fixed costs of upfront factory construction because the only factory that would have to be built afresh would be the modularized component assembly factory, and it could be deferred until the marketing arm had generated a sufficient volume of orders.

The component suppliers pick up much of the fixed cost, but by joining the consortium at limited buy-in cost, could benefit from the potential to increase sales of a product they are already making anyway. Figure XXIV-14 illustrates the success of such a strategy as applied in the specialty vehicle construction business.

Ford unveils FASTM bus

The FASTM bus, conceived in 90 days by Ford Motor Company and four of its leading suppliers, is a hybrid electric medium duty shuttle for airports.

In March, Ford Motor Company met with four of its leading suppliers to conceive and design a new bus developed specifically for airport shuttle and mass transit use – the working prototype was unveiled June 19.

"The goal was to start from a clean sheet of paper to quickly design and build a completely new hybrid vehicle capable of running in a zero emissions mode," said Paul Szilagyi, CEO of the Denver-based TransTeq firm.

Called the FASTM Bus (F for Ford, which supplied the engine, fueling system and E450 chassis; A for AAM which supplied the driveline modifications; S for Supreme Industries, which supplied the body and interior of the bus; T for TransTeq, which supplied hybrid

FIG. XXIV-14. Ford unveils FASTM Bus after 4 months from initial conception.

A prototype will be needed for any new reactor concept; and for the STAR mass production business strategy, it is proposed to use it for a "design certification by test" licensing process. In this proposed process the prototype would be built and subjected to a pre-agreed set of ATWS and other accident initiators. By demonstrating safety based on passive response, on the prototype, the licensing authority might be able to certify the design, permitting the manufacture of many tens (or hundreds) of replicate plants to the set of prints and design specifications used for the prototype. In order to assure that aging effects do not degrade the passive safety features of deployed plants, the licensing authority could prescribe the performance of periodic in situ tests on the plant to confirm continued presence of reactivity feedbacks in the required range and of passive decay heat removal (continuously) operating at the required rate.

The notion of building a prototype in order to gain the safety licensing authority's "design certification by test" also provides a vehicle needed to fill the order book. Potential customers could be brought to tour the prototype and to witness replays of the passive safety demonstration tests. Operations crews could be offered training at the prototype as an inducement to become familiar with the product and to place an order.

The battery supply company would assembly the battery heat source reactor at his factory; then he must transport it to the customer's site, install it next to a pre-built, non-safety grade balance of plant and bring it to power over a short start-up period. The supplier consortium, therefore, needs to include a large-scale construction and logistic company. Such companies exist in the ocean oil rig business to build ocean oil rigs in shipyard-like facilities, tow them by sea to the installation site, and using cranes of many hundred ton capacity, install and start up the complex rigs in a matter of months (see Fig. XXIV-15).

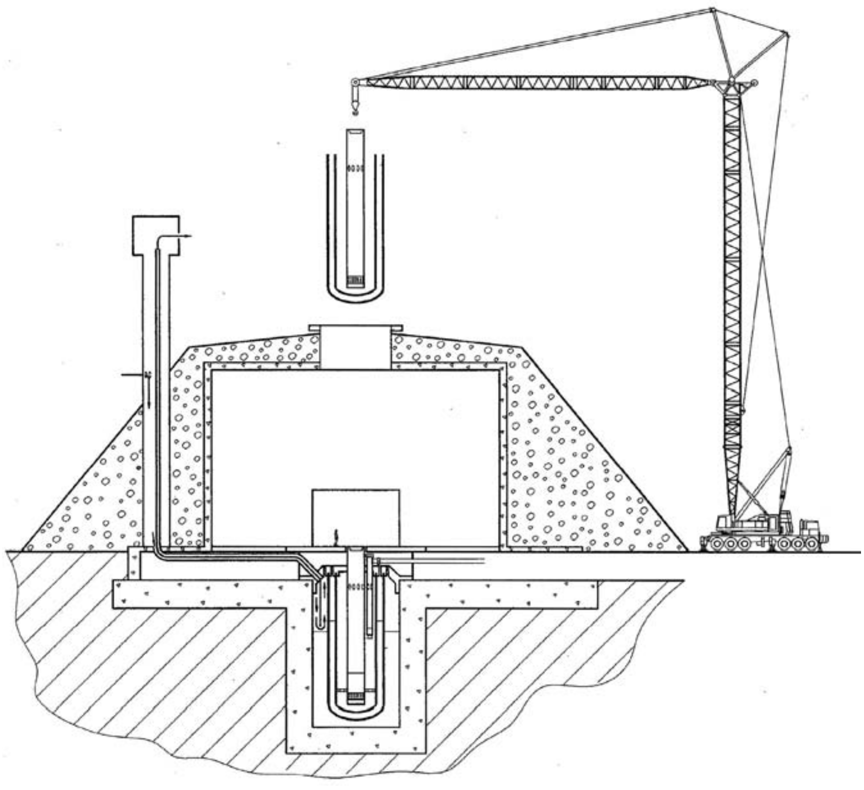


FIG. XXIV-15. Rapid site assembly of the STAR-H2 reactor.

Fuel cycle service operators: market entry strategies

The battery plant idea won't work unless business entities are prepared to invest in the creation of regional fuel cycle centres capable of supporting front and back end fuel cycle services for hundreds to thousands of battery type plants in the region. Each such regional centre must be very large in order to handle a significant fraction of the world's energy supply needs. They could benefit from economy of scale in bulk fissile handling (reprocessing, fuel fabrication, and waste management) and in centralized fuel and hydrogen manufacture in large economy of scale breeders, but concomitantly they will require several tens of billions of dollars of upfront investment in infrastructure. Each would require facility and infrastructure investments on the scale of developing an oilfield, or of building pipelines, seaports or transcontinental railroads.

The energy business already involves investments on the scale of those envisioned for a regional fuel cycle centre; they are routinely made by petroleum companies for exploration and infrastructure emplacement in their ongoing efforts to replenish petroleum reserves in concert with reserve depletion. The battery type plants are envisioned to come online in large numbers during the decades from 2030 to 2060. It is during those decades that oil reserves are expected to go through a significant decline (see Fig. XXIV-16).

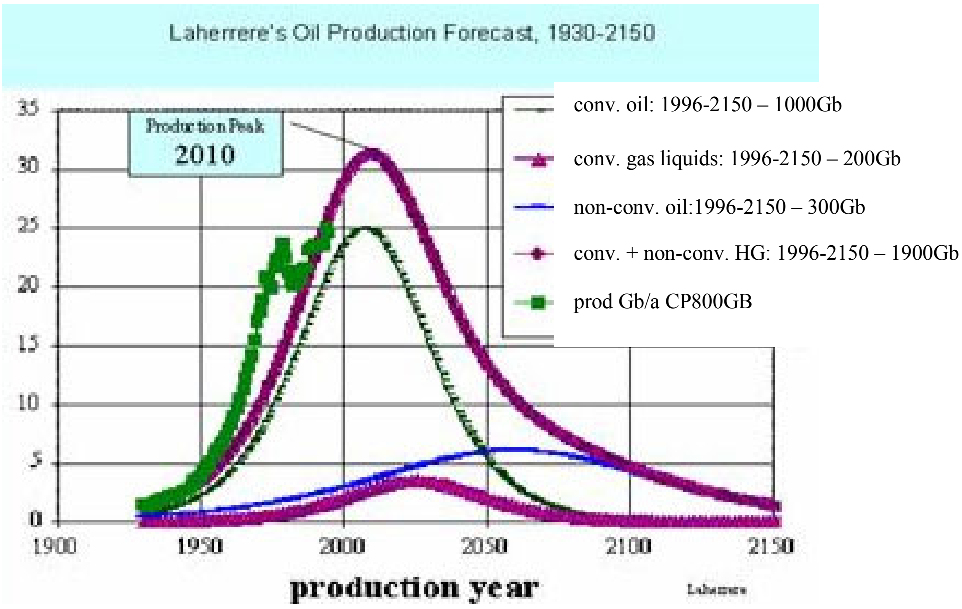


FIG. XXIV-16. Illustration of the finite fossil resource base.

One might imagine a multinational petroleum company going into the regional fuel cycle centre business under contract to the consortium of customers in order to remain a viable energy resource supplier even as his oil reserves diminish. The consortium of customers desiring to create a regional centre could pool economic resources and could offer long-term contracts for services to a builder/operator company competing for the contract to build and operate the centre. The petroleum company desiring to enter this business could share risk and profit with the consortium and would become a hydrogen company and a nuclear fuel supply company. The business plan could rest on the following logic:

(a) Nuclear energy and nuclear hydrogen production would be growing in energy market share even as petroleum is decreasing;

(b) LWR discharge fuel from earlier nuclear deployments would be shipped to and reprocessed at regional centres as a waste management strategy – to avoid growing repository deployments;

(c) The recovered transuranics and some of the recovered ^{238}U from the LWR spent fuel management recycle operations would be used by the regional fuel cycle centre owner to provide working inventory for onsite deployment of large fast neutron spectrum reactors. In the early years, these fast reactors might be configured as net burners (conversion ratio <1) and used to reduce the LWR waste inventory destined for geologic repositories in two ways: by tying it up in fast reactor working inventory and by incinerating it through fission;

(d) The heat from the fast burner reactors would be used to manufacture hydrogen as a synthetic chemical energy carrier – allowing for continued use of the petroleum company's sunk cost infrastructure in pipelines, filling stations, transport tankers, etc. The petroleum company would eventually become a hydrogen company;

(e) Some of the transuranics recovered from the LWR spent fuel and some of the ^{238}U would be used to manufacture fuel cassettes to sell to third party fuel leasing companies, which in turn would support a growing fleet of battery type heat source reactors distributed at customers' sites throughout the service region. The petroleum company would become a nuclear fuel supply company also;

(f) The recycle facilities would be used to reprocess fuel from the on-site fast burner reactors and from the off-site battery reactor cassette returns as a revenue generating business;

(g) After some time, as virgin uranium ore became scarce and expensive, the core layout of the on-site fast reactors would be changed to make them breeders. Thereafter, the excess transuranics bred in the breeders would gradually replace virgin uranium reserves. The on-site fast reactors would become nuclear fuel factories for driving a growing global economy – drawing from the huge inventory of ^{238}U feedstock;

(h) Over several decades the petroleum company would transition to a nuclear fuel supplier – exploiting a thousand year supply of ^{238}U reserves accumulated during their earlier LWR spent fuel waste management business;

(i) Gradually, the petroleum company would transition away from petroleum and take up a nuclear fuel and hydrogen energy resource business – a business that might be good for many centuries.

The conditions for this transition from oil to hydrogen and uranium are foreseen by energy planners to occur no sooner than 25 to 60 years from now. That may be beyond the planning horizon of petroleum executives. The reason that an international petroleum company might develop an incentive to consider entering into the regional fuel cycle centre business much earlier, i.e. within the next few decades and before their oil reserves become exhausted, is to provide themselves with a hedge against shifting customer preference (for H_2 vs. gasoline) or as a hedge against potential carbon taxes.

Another energy resource industry that might be facing diminishing resource availability for their core business over the next century is the uranium mining and enrichment industry. As their virgin uranium ore and enrichment products become overpriced due to degraded assay of remaining ore deposits, uranium mining and enriching companies might see value in joining into the regional fuel cycle centre business in order to be positioned to shift over their product from enriched ^{235}U to bred fissile transuranic material. The regional centre configuration and business plan could be the same as discussed above.

Configuring the institutional arrangements for breaking the energy security / non-proliferation dilemma

Developing institutional and business innovations, which could enable risk and profit sharing among a multi-national consortium of customers and a contracted operator of the regional fuel cycle centre, would be a challenge for the business and political communities – going well way beyond technology alone.

In order to provide energy security to its customers, the governance of the regional fuel cycle centres must be under control of the nations being served. Otherwise the energy security provided by this arrangement is no better than the current situation based on centralized control of oil reserves, oil refining and delivery. This highlights the urgent need for institutional innovations designed to simultaneously create a fuel cycle centre operating under free market conditions but regulated under multi national treaties, laws, and norms.

In the proposed hub/spoke nuclear architecture it is assumed that a new "non-proliferation compact" would augment the current one. A country will be encouraged to agree to eschew the emplacement of an indigenous fuel cycle infrastructure in exchange for guaranteed access to fuel cycle and waste management services from a regional centre, which operates under control of a consortium of customers with oversight by international non-proliferation agencies (the IAEA).

At the same time, as discussed just above, free market mechanisms would be channelled into raising the substantial financing for emplacing the regional fuel cycle centres. And, finally, once STARs and breeders dominate the nuclear market share of global energy supply, free market profit mechanisms of supply and demand might be relied on to self-regulate the global inventory of fissile material such that fissile material remains entirely tied up in working inventories, avoiding growing inventories in interim storage or in permanent (waste) storage. Market forces might be capable to do so innately without need for governmental intervention.

In such architecture there could be an interesting arrangement wherein free market profit mechanisms are employed while at the same time the governance of the regional centre must be multi-national. Clearly, institutional innovations in the form of treaties and international law must be emplaced before these competing imperatives for national energy security, non-proliferation assurances and market force mediation of fissile inventories could be brought into play.

First, the customers of a regional fuel cycle centre must be guaranteed a dominant role in the governance of the regional centre as a central tenant of their national energy security posture. For example, a nation's suppliers of energy services cannot be denied access to fuel cycle services on the basis of internal or external political pressure, regional disputes, or international sanctions without "due process". Otherwise the "non-proliferation compact" to eschew emplacing an indigenous fuel cycle infrastructure in exchange for guaranteed access to services from the regional centre will hold no appeal.

At the same time, free market rather than centrally planned pricing of fissile material is the only realistic method to hold fissile supply and demand in balance such that no excess inventories build up. And also the balance between centrally located breeders producing hydrogen and distributed STARs producing hydrogen could be best optimized by the relative profit margin between centrally vs. distributed hydrogen production – mediated by the "value" that each customer nation will assign to the energy security offered by having a STAR with a 20-year fuel supply sited on his sovereign territory vis-à-vis buying hydrogen delivered by thousands of shipments from a central location thousands of miles away (as in the case of oil currently).

To meet these needs, it is proposed that regional or international treaties could form the basis for establishing each regional centre under international law:

(a) The centres would be governed as a consortium by the nations for which the regional centre provides services;

(b) Ratification of the supplementary non-proliferation compact to forego emplacement of indigenous fuel cycle infrastructure in exchange for guaranteed access to services from the regional centre (as administered by, e.g. the IAEA) would be a requirement for national membership in the consortium;
(c) Arrangements for accepting new members to the consortium and for leaving the consortium would be worked out in the articles creating the consortium;
(d) Denial of services to a country by the consortium could not be executed without "due process" pre-specified in the articles creating the consortium;
(e) As a precondition for receiving services from the regional centre, a nation would be obliged to declare a legal commitment to regulate its nuclear power plants and nuclear activities in compliance with a set of international norms on:
    - Safety;
    - Safeguards;
    - Radiological standards;
    - Indemnification;
    - Early notification of accidents;
    - Mutual assistance;
    - Shipping norm;
    - etc.
(f) Arrangements for sharing financial risk and sharing profits arising from the consortium would be pre-specified in the articles of creation of the centre; and
(g) Legal arrangements for siting an international centre on a host nation's sovereign territory will raise liability, access, security, and other issues – such as are faced for embassies, the UN, the EU, and others.

Some of the elements enumerated in (e) are simply extensions of institutional arrangements, which have already been built up over 45 years of the nuclear era and are already administered by organizations such as the IAEA, OECD, EC, and others [XXIV-21]. Others would require innovation which is more revolutionary even than the technical innovations themselves – because a nation joining the regional fuel cycle consortium will be giving up important elements of national sovereignty in exchange for achieving energy security.

The regulation among the several regional centres would also have to be worked out in international law in order that a world price for fissile material will exist – reducing the likelihood of monopolistic pricing to regional customers, reducing opportunity for arbitrage, and increasing robustness and reliability of the energy supply offered to all customers from the global energy supply architecture as a whole.

Institutional innovations needed for safety and licensing

The battery type heat source reactor is intended to be of a standard design – replicated to license-certified specifications and blueprints – and delivered as a turnkey plant to customers worldwide. There would of course be a diversity of such standard pre-licensed designs proffered by competing supplier companies. In order that the product applies internationally, its safety licensing validity must be recognized internationally.

The need for licensing reciprocity across national boundaries raises the need for further institutional innovations:

(a) Mutual reciprocity agreements among national licensing authorities to "accept" a license certified design from some other nation's licensing authority will be needed. This in turn may lead to a set of international norms on safety principles and to creation of multinational safety review teams (such as have already been initiated by the IAEA and the EU) whose advice to responsible national authorities lends support to their licensing decisions;

(b) A "design certification by test" approach for licensing battery heat source reactors of standardized design could benefit cost effectiveness. The small battery plants are intended to be replicates produced in mass. And they might be relatively inexpensive so that a prototype reactor could be built at limited cost, installed and subjected to a battery of licensing authority witnessed tests which measure passive response to accident initiators.

Given the prototype's response meets safety criteria, then the design might be certified and each replicate reactor produced in mass could thereafter require minimal further licensing effort.

This would lead to need for factory inspectors to certify that each replicate was indeed in conformance with the certified design and lead to the need for start-up inspectors to certify that a reactor's transport and installation had incurred neither damage nor modification. Again, multinational reciprocity agreements would be needed.

(c) Mechanisms would be needed to avoid discouragement of improvements in an already-certified standard design so that lessons learned can be exploited without incurring an unnecessarily costly re-certification procedure nor unnecessary retrofitting of already approved and operating plants.

(d) A way will have to be devised to conduct safety licensing and regulation for the fuel cycle and waste disposition facilities at the regional fuel cycle centres because it must meet the safety requirements of all nations who are members of the consortium.

This international normalization of safety licensing and regulation of nuclear facilities – be they reactors or fuel cycle facilities – might require lengthy bilateral and multinational debate and negotiation. Again, the central issue is loss of a fraction of national sovereignty in exchange for energy security and economic benefit from reduced duplication of licensing effort and by avoiding "customization" of otherwise standard designs.

As a start, one can look to the airliner industry for some guidance on how this might be done.

The need to start now to emplace enabling institutional innovations

Recalling Hafële's analogy of the Industrial Revolution [XXIV-22], the replacement of water wheels and animal power with coal-fired steam engines bought to bear a factor of 10^6 in energy density for the service of society. But this technological innovation by itself was not enough to dramatically alter the economic development that England and America experienced in the late 18th and the 19th centuries. It was also necessary to dramatically re-engineer the architecture of production to exploit the factor of 10^6 – and this in turn could not be accomplished absent a (dramatic) change in the institutions of societal organization.

Now the goal might be to move the world as a whole beyond the bimodal coexistence of the industrialized West and the undeveloped "South" in a new revolution called "sustainable economic development". Fortuitously and in analogy to coal, a new nuclear technological innovation for energy supply became available in the mid 20th century; it offers an additional factor of 10^6 in energy density and its resource base and ecological footprint can be configured to meet the requirements articulated for sustainable development. To exploit it would require to substantially re-engineer the world's energy supply architecture – transitioning from an architecture optimized for fossil to one optimized for nuclear. To do that will require the institutional changes sketched above.

Specifically, the nuclear-based sustainable global energy architecture proposed here cannot be implemented absent enabling institutional innovations in international law to create a supplementary non-proliferation compact which would augment the current one, and to facilitate international harmonization and mutual national reciprocity arrangements of numerous safety licensing and operational standards.

These necessary institutional innovations are not so radical, nor extensive as those of the Industrial Revolution and Scottish Enlightenment. But, they won't be easy because, in general, in exchange for increased energy security and economic efficiency they may reduce national sovereignty.

Working out the enabling international treaties and laws will take several decades. Work must start now and must progress in parallel with the R&D on the technology in order that both can come to fruition in the 2030 time period when the transition might start in earnest.

XXIV-1.8. List of enabling technologies relevant to STAR-H2 and status of their development

Crucial strategies for breaking the economy of scale paradigm

The STAR concept eschews the economy of scale approach, which has been the hallmark of the light water reactor (LWR) industry. Before listing specific enabling technologies it is useful to elucidate several fundamental departures of the STAR approach from the current LWR nuclear energy approach – and their importance to meeting economic goals – which are the key enabler for market penetration.

First, a principal driver historically forcing the LWR economy of scale strategy is the robust containment building to mitigate severe accidents – the containment pressure goes inversely with the containment volume while the containment cost goes with the containment surface area. The ratio of surface to volume goes inversely with diameter, so in order to minimize the large fixed cost component of containment per unit of power, a large containment/large power strategy is used for LWRs. Since the STAR concept is based on small heat rating to match market needs in developing countries, it is clear that the first crucial requirement is to develop a safety approach, which would eliminate the need for a conventional containment, or to at least minimize pressure requirements placed on it. If that is not achieved, then there is no hope for containing capital cost for small liquid metal reactors that have given up economy of scale at the outset. The STAR reactors are being designed to eliminate the need for a conventional containment by using a coolant at ambient pressure; no internal sources for chemical explosion hazards; and a passive safety design which by use of innate processes might close off all pathways to core damage. In light of no internal pressure hazard for STAR-H2, a close-fitting (small volume) guard vessel and top cover could suffice as containment.

Second, the business strategy for the STARs is entirely reversed from that used for LWRs in the past where the utility customer held most of the financial risk for a custom built plant while the supplier Architect/Engineering (A-E) firm held very little risk. To the contrary, for STARs the business risk would be transferred predominately to the supplier who will have no other choice but to spread its cost over many hundreds of replicate units. The customer, unlike in the past, would be offered to purchase a commodity nuclear power plant, – already license certified – and will be able bring it on-line with a very short on-site installation and checkout period so as to start a revenue stream shortly after taking on his financing loans. Additionally, the smaller heat rating lowers his overall capital outlay. All these strategies are designed to reduce customer financial risk. STAR reactors approach this need for customer risk reduction by simplification and elimination of components; by factory fabrication; by transportability of the modular components; and by placing no safety requirements on the BOP so that it could be pre-built by local labour to local standards. Just as with the elimination of a classical containment, this totally new business strategy is crucial to success of small liquid metal reactors and it must be kept in mind during every design decision so as to not jeopardize the a-priori licensing, delivery logistics, and rapid start-up features upon which the STAR business plan depends.

Third, the fuel cycle of the STAR concept, which is targeted to fuel sustainable development, must be closed for three reasons. 1st, to exploit the entire energy potential of the earth's endowment of uranium ore – allowing to provide a millennium of world energy supply. 2nd, to recycle all transuranics for self consumption; in so doing the energy supply architecture will send only fission products to waste and would assure that 300 years of sequestration causes a neutral radiotoxicity exchange between ore withdrawals and waste emplacement in the earth's crust. And third, to avoid ever-growing inventories of fissile material as a consequence of nuclear power deployments. By holding all fissile material in active working inventory and fully consuming all transuranics via recycle one can avoid build-up of stores of weapons relevant material – in temporary storage or in waste disposal sites. The STAR concept approaches these needs by employing new, electrometallurgical recycle technology, remote vibrocompaction refabrication technology of ^{15}N enriched transuranic mono-nitride fuel, and ceramic and metal alloy waste forms for the chemically active and noble metal fission products, respectively.

These three crucial departures in approach from the historical open cycle, economy of scale, LWR-form of nuclear energy permeate all elements of the STAR reactor and fuel cycle concept.

Status of technology and technology needs for the STAR concept

The STAR-H2 concept is at the conceptual design stage of development. However, its safety strategy and its fuel cycle and waste management strategies have been adapted from the ten years of development for the Integral Fast Reactor (IFR) [XXIV-15]. Furthermore, the reactor structural, refuelling, neutronics, and thermal hydraulics design approaches have been adapted from the STAR-LM project [XXIV-1] (see also ANNEX XXII), which has a several year head start in design effort relative to STAR-H2.

The salient new reactor design features of STAR-H2 relative to STAR-LM are materials related – choice and qualification of cladding and structural materials for 800°C service conditions in Pb. Also, the fabrication technologies for low-cost serial factory fabrication of reactor modules and of refuelling cassettes using the new structural materials. Materials screening tests have been conducted and among the materials tested or to be tested in the corrosion/mass transport convection harps are composites including SiC and ZrC and refractory metal alloys. If the ceramic composites prove out, it may be possible to bring aerospace fabrication technologies to bear on STAR manufacture.

STAR-H2 relies on passive safety accommodation of ATWS initiators and passive decay heat removal – technologies, which are already well developed in the IFR programme. Several safety-relevant issues require more work, however. Potential for degradation of cooling capacity by sludge build-up in the event of loss of control of coolant chemistry and/or by coolant solidification in the event of local system cooldown (327°C Pb freezing temperature) need to be addressed. The phenomenology and consequences of nitride fuel dissociation under high temperature accident conditions must be understood; on the one hand it may provide a fuel dispersal/HCDA[7] quenching mechanism; on the other it might produce significant reactor tank over-pressurization.

The non-aqueous recycle technology for nitride fuel, while under development in Japan and in the Russian Federation, is not as well advanced as for metallic alloy, nor is its Russian developed vibropack remote fabrication technology as available outside the Russian Federation as is the case for the Argonne developed remote casting fabrication used for metallic alloy. Development and prototype testing of these technologies and resulting waste forms is needed. A major nitride fuel irradiation test program is required, and a fast spectrum fuel irradiation test facility is needed to conduct it.

The development of the Ca-Br thermochemical water cracking process must be taken beyond the bench scale, which has been achieved in Japan. Reliable thermodynamic data (Gibbs free energies) are available for all reactions. However, reaction kinetics data on prototypic reaction bed configurations are lacking and are being researched. Significant and cost effective proposed modifications of the flow sheet based on plasma chemistry must be researched starting at the bench scale.

The supercritical (SC) CO_2 Brayton cycle has been optimized on paper and control strategies are under current study. Testing is being initiated on printed circuit heat exchanger components, which hold potential for recuperator size and cost reduction. An entire SC-CO_2 Brayton cycle prototype will have to be built and tested to bring the technology to a state of commercial availability.

The desalination bottoming cycle uses commercial technology and off-the-shelf components.

A capital cost containment strategy based on simplification, component elimination, serial factory fabrication and rapid site assembly and non-nuclear safety grade balance of plant has been devised. And an operating cost containment strategy based on ultra high capacity factor, high energy conversion efficiency, product diversification and operating staff reductions based on simplification, passive load following, and passive safety and elimination of components has been devised. Whether or not these strategies can overcome the economic penalty of derating power density to achieve 20 year refuelling interval and reducing plant rating to lower initial capital outlay is not yet known. Capital cost estimates will not be determinable until after substantial further engineering refinement of the concept is completed.

XXIV-1.9. Status of R&D and planned schedule

Development and design of the STAR-H2 concept was funded by a US-DOE NERI grant from years 2000 through 2003. STAR-LM development at Argonne National Laboratory is ongoing – supported by U.S. Department of Energy research and development funds; its main source of support has been a U.S. Department of Energy Nuclear Energy Research Initiative project. Institutions involved in STAR research and development together with ANL are Oregon State University, Texas, A&M University, and Ohio State University. Research and development and design of the Generation IV lead fast reactor (LFR) are also carried out under the U.S. Department of Energy Generation IV Nuclear Energy Systems Initiative. Development of the SSTAR small modular fast reactor under Generation IV LFR funding also involves LFR-related funding at Lawrence Livermore National Laboratory and Los Alamos National Laboratory.

---

[7] HCDA is for hypothetical core disruptive accident.

As part of the Generation IV work on the SSTAR, it has been proposed that a lead-cooled demonstration test reactor could be designed, constructed, and ready for operation by about 2015. It would be subsequently operated to support SSTAR commercial deployment in about 2020 to 2025. There is considerable interest in a license-by-test approach that makes use of a demonstration test reactor. This developer-proposed schedule has not been embraced as a formal element of the Generation IV programme. Currently, funding is not available at a level sufficient to make feasible the design, construction, and initial operation of a demonstration test reactor within a 2015 timeframe; and uncertainty remains as to what funding priorities the U.S. Department of Energy would place on this concept.

XXIV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

A central business and licensing strategy for the STAR concept is:

- Licensing design certification by test of a prototype STAR heat source reactor;
- Factory production of hundreds of replicate heat source reactors;
- Transport to the customer's site for rapid site assembly, connection to a non-safety grade balance of plant already emplaced; and
- Prompt generation of a revenue stream to minimize out-of-pocket interest during construction.

This strategy requires institutional innovations, which include:

- Emplacement of a licensing design certification by test regime with the safety regulator of the supplier country;
- Licensing reciprocity arrangements between the regulators of the supplier and customer countries;
- Factory inspections by the supplier-country regulator to assure compliance with the certified design;
- Site inspections by the customer-country regulator to assure compliance with the certified design assembly sequence;
- Arrangements to facilitate amendments to a certified design so as to incorporate design improvements in a cost effective manner.

A prototype is an essential element in the execution of this business and licensing strategy. Satisfactory outcomes of safety performance tests on the prototype – as observed by the regulator – might comprise the basis for the granting of a licensed design certification. Such a licensed design certification could serve as the basis for fabrication of dozens or hundreds of replicate heat source reactors.

Moreover, the prototype, once built and licensed, could serve additional marketing functions such as:

- Demonstration of operational and safety performance to potential customers;
- Demonstration of operational and safety performance to regulators from countries of potential customers;
- Training for customer operational crews.

The STAR portfolio of heat source reactors, including SSTAR, STAR-LM and STAR-H2, facilitates a time evolution of both technical and institutional innovations. SSTAR – being the smallest STAR concept and the earliest one to achieve technical readiness for market entry – allows for early exercise of the STAR business and licensing strategy. STAR-LM for electricity and potable water production could be deployed in cities of developing nations prior to widespread adoption of the hydrogen economy. Finally, STAR-H2, being the latest to achieve technical readiness for market entry, might support a sustainable hydrogen economy in the middle decades of the century and beyond. Figure XXIV-17 illustrates how numerous of the technical features and especially the institutional and business innovations required for STAR-LM and STAR-H2 could be implemented and exercised by earlier construction and licensing of a SSTAR prototype.

APPLICABLE INSTITUTIONAL INNOVATIONS		
Na	Pb-Bi	Pb

- Small size for low buy-in cost
- Non safety grade BOP built/operated to normal industrial standards
- Design certification of reactor by licensing test
- Factory mass production/ rapid site assembly
- Long refuelling interval/outsourced whole core cassette refuelling
- Remote monitoring/ outsourced specialty maintenance
- Financing innovations – leasing

Pb, Pb-Bi vs. Na

- Cassettes transported in frozen coolant

Pb vs. Pb-Bi

- Eliminate demand on scarce Bi
- Reduced long term activation D&D products (vs. Pb-Bi but not Na) - eliminate polonium hazard

APPLICABLE TECHNOLOGICAL INNOVATIONS		
Pb	Pb-Bi	Na

- Fissile self-sufficient core
- Power density derating for long refuelling interval
- Passive decay heat removal/ low pressure – close coupled containment
- Passive safety response
- Passive load follow

Supercritical CO_2 Brayton cycle (optional for Na, Pb-Bi) (~ required for Pb)

Pb, Pb-Bi vs. Na

- Eliminate intermediate loop
- Natural circulation – eliminate primary pumps

Pb cooled STAR-H2

- High outlet temperature for hydrogen production

FIG. XXIV-17. Applicability of enabling technological and institutional innovations vs. coolant choice.

XXIV-1.11. List of other similar or relevant SMRs for which design activities are ongoing

The STAR concept was inspired by the work in the early 1990s of G.I. Toshinsky and S. Hattori. They envisioned small transportable reactors of long refuelling interval based on existing Pb-Bi (Toshinsky – SVBR-75/100, see ANNEX XVIII) and Na (Hattori – 4S, see ANNEXES XIII and XIV) reactor technology.

The STAR portfolio of designs was initiated in the 1997 time frame and is comprised of SSTAR, STAR-LM and STAR-H2. They employ technological innovations of increasing lead-time to deployment (see Fig. XXIV-18). The STAR portfolio share many features with the Encapsulated Nuclear Heat Source (ENHS, see ANNEX XIX) and the 4S concepts, see Fig. XXIV-18. The Pb technology and nitride fuel technology are in common with the BREST designs.

The Star Portfolio

Portfolio Member	Power	Coolant	Tout	Converter	Products	Client*	Deployment Target
SSTAR S=small	20-50 MW_{th}	Pb Nat'l Circ.	550°C	Rankine Steam – or SC-CO_2 Brayton Desalination	Electricity +potable water or potable water	Electricity for Remote town of ~ 6,500	**~2015** Potential 1st Prototype
Star-LM LM=Liquid Metal	400 MW_{th}	Pb Nat'l Circ.	550°C - 580°C	SC-CO_2 Brayton Desalination	Electricity +potable water	Electricity for city of ~ 115,000	**~2020**
STAR-H2 H2 = Hydrogen	400 MW_{th}	Pb Nat'l Circ.	800°C	Ca-Br Thermochemical Cycle SC-CO_2 Brayton Desalination	H_2 +potable water	All primary energy and potable water for city of 25,000	**~2030**

* Assume 4 toe/capita year primary energy = 12 kw th year/person year
Assumes 1/3 of primary energy converted to electricity

Options for Early Prototype
(to Exercise Institutional Innovations)

Related Concepts

Portfolio Member	Power	Coolant	Tout	Converter	Products	Client	Deployment Target
ENHS Ehud Greenspan U of California – Berkeley	125 MW_{th}	Pb or PB-Bi Nat'l Circ.	550°C	Rankine Cycle	Electricity	Electricity for City of 32,000	**~2015 - 2020**
4S Minato CRIEPI	135 MW_{th}	Na pumped	500°C	Rankine Cycle	Electricity or potable water	Electricity for City of 32,000	**~2010 - 2020**

FIG. XXIV-18. STAR portfolio and related concepts.

XXIV-2. Design description and data for STAR-H2

XXIV-2.1. Description of the nuclear systems

Reactor core and fuel design

The STAR-H2 reactor uses a radially heterogeneous core layout of ductless assemblies. The pin lattice is open with a large coolant volume fraction.

Neutronic design

Table XXIV-5 shows the neutronics design parameters and performance results while Fig. XXIV-19 and XXIV-20 show radial power density profiles at beginning and end of cycle.

TABLE XXIV-5. NEUTRONIC DESIGN PARAMETERS AND CALCULATED PERFORMANCE RESULTS FOR THE REFERENCE HETEROGENEOUS CORE LAYOUT (INTERNAL BLANKET; CLADDING MATERIAL SiC; FUEL RESIDENCE TIME 15 YEARS; CAPACITY FACTOR 90%)

CHARACTERISTIC	VALUE
Design parameters	
Enrichment Pu/HM, %	13.14
- Inner core	× 1.0
- Middle core	× 1.0
- Outer core	× 1.4
Driver fuel pins	
- Fuel pin diameter (clad), cm	1.905
- Fuel volume fraction	0.247657
- Cladding volume fraction	0.078858
- Coolant volume fraction	0.666785
Blanket fuel pins	
- Fuel pin diameter (clad), cm	1.905
- Fuel volume fraction	0.247657
- Cladding volume fraction	0.078858
- Coolant volume fraction	0.666785
Number of fuel driver pins in the core	4638
Number of blanket pins in the core	2301
Number of Inner core driver assemblies	1
Number of medium core driver assemblies	48
Number of outer core driver assemblies	84
Number of blanket assemblies	66
Number of control rods locations	12
Number of reflector locations	54
Number of core barrel locations	60
Calculation results	
K_{eff}; Beginning of cycle (BEOC)	1.000
K_{eff}; End of cycle (EOEC)	1.013
Peaking factor, BOEC	1.77

CHARACTERISTIC	VALUE	
Peaking factor, EOEC	1.84	
Power split BOEC	94.21/4.59	
Power split EOEC	75.14 / 23.96	
Average discharge burn-up, MW day/kg	82.17 / 28	
Peak discharge burn-up, MW day/kg; <150	126.0	
Peak fast fluence, 10^{23}n/cm^2; <4.0	2.70	
Breeding ratio	1.0078	
Reactivity swing, %Dk	-1.27	
Maximum temperature at the centre of driver fuel pin, (BOEC/EOEC), ^{o}C	1362.9	1259.0
Maximum temperature of the driver pin cladding, (BOEC/EOEC), ^{o}C	951.4	914.6

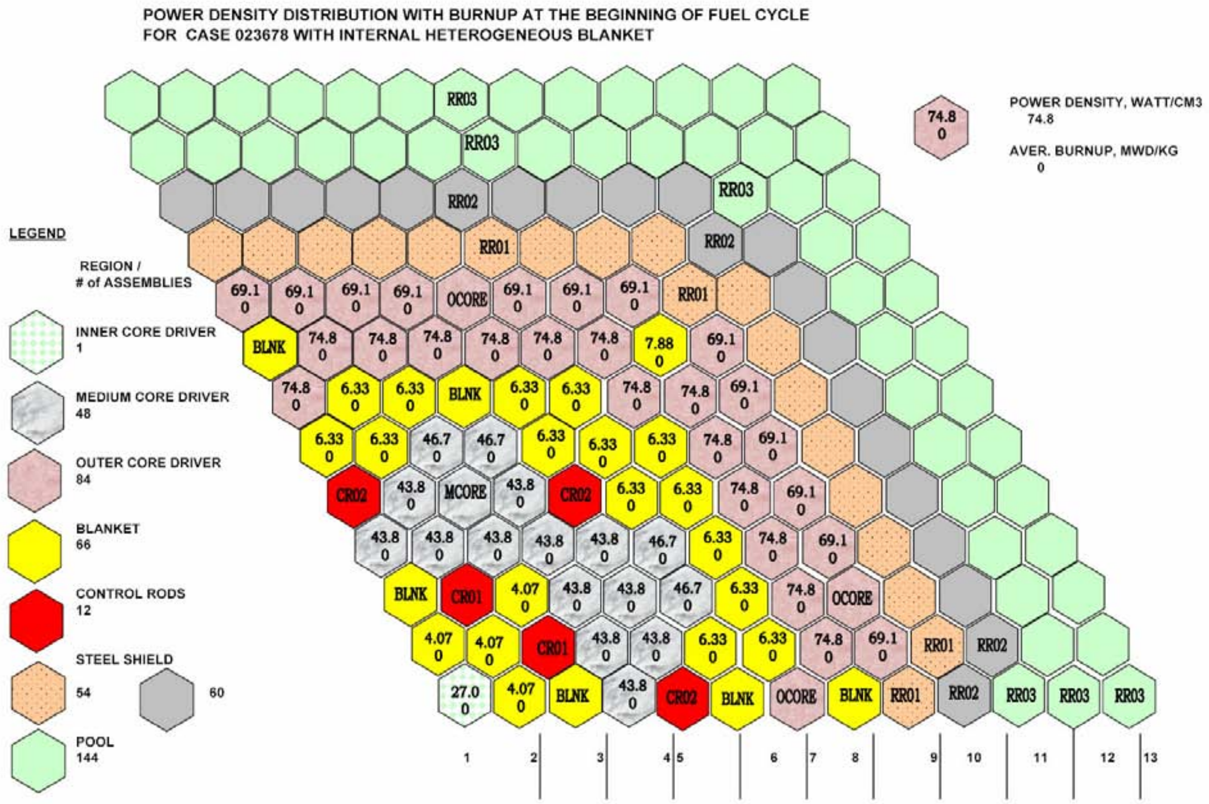


FIG. XXIV-19. Power density distribution with burn-up in the core for reference core configuration (Table XXIV-5) at the beginning of cycle.

The radially heterogeneous core layout (i.e. with fertile material blanket assemblies interspersed in the core itself) is used to flatten radial power profile and to enhance internal breeding so as to reduce burn-up reactivity loss.

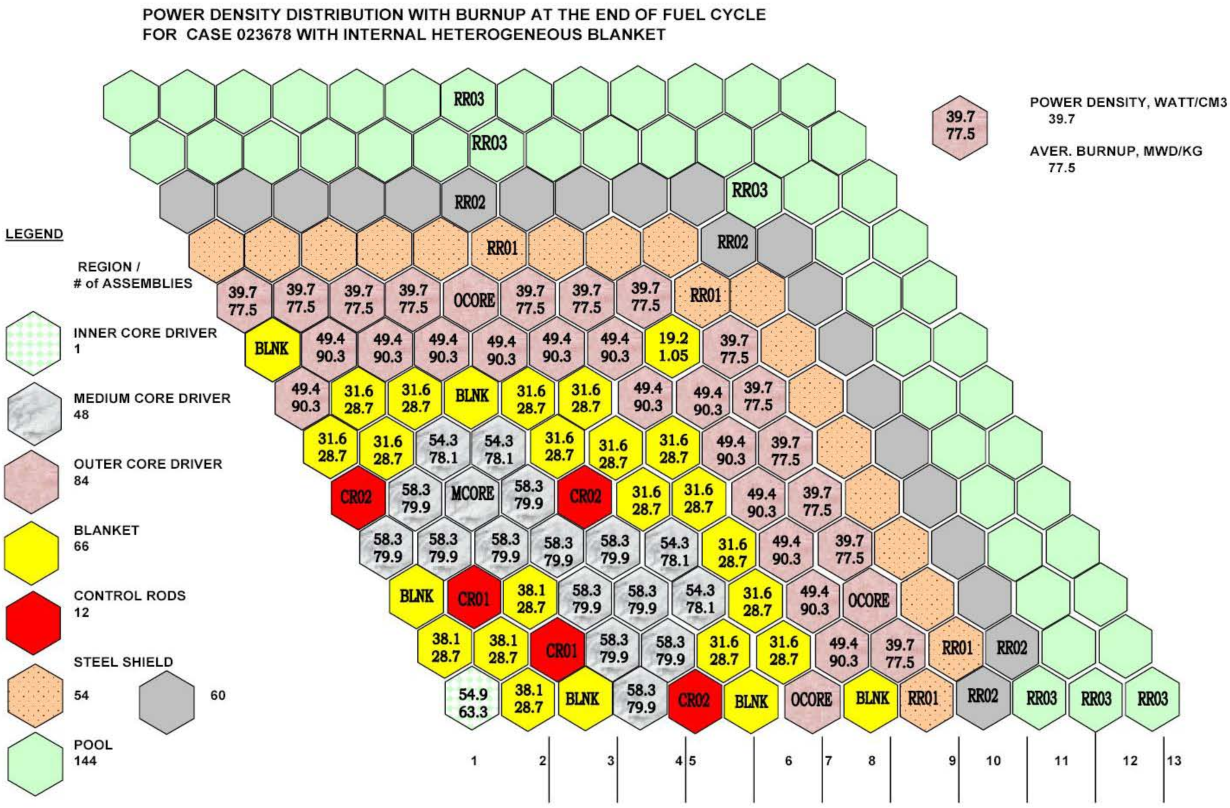


FIG. XXIV-20. Power density distribution with burn-up in the core for reference core configuration (Table XXIV-5) at the end of cycle.

Thermal-hydraulic design

The STAR-H2 reactor is cooled by natural circulation of the Pb primary coolant; Table XXIV-6 gives the relevant design data and Table XXIV-7 gives the full power results for thermal-hydraulics.

TABLE XXIV-6. STAR-H2 (400 MW(th)) DESIGN CONDITIONS FOR THERMAL-HYDRAULIC ANALYSES*

Core thermal power, MW	400
Coolant	Pb
Core diameter, m	2.5
Core active (heated) zone height, m	2.00
Fission gas plenum height, m	2.00
Total core (frictional) height, m	4.00
Fuel rod / cladding outer diameter, cm	1.905
Fuel rod triangular pitch-to-diameter ratio	1.50
Cladding thickness, cm	0.10
Fuel material	(92%U-8%Pu)N
Fuel smeared density	0.78
Fuel porosity	0

TABLE XXIV-6 (continued)

Fuel pellet diameter, cm	1.51
Cladding-fuel pellet gap thickness, cm	0.0996
Gap bond material	Lead
Core hydraulic diameter, cm	2.82
Number of spacer grids in core	3
Core-wide fuel volume fraction	0.252
Core-wide cladding volume fraction	0.0802
Core-wide bond volume fraction	0.0710
Core-wide coolant volume fraction	0.597
Core fuel mass, kg	29 600
Core uranium mass, kg	27 900
Core flow area, m^2	2.93
Number of fuel rods	6940
Number of support and flow distributor plates below core	2
Plate open area fraction	0.6
Core coolant-to-fuel rod volume ratio	1.48
Core specific power of uranium, KW/kg	12.0
Core power per volume, MW/litre	0.044
Core mean heat flux, MW/m^2	0.520

*Owing to the iterative process of design, the volume fractions used for neutronic and for thermal-hydraulic calculations differ by a few percent at the conceptual design stage

TABLE XXIV-7. THERMAL-HYDRAULIC RESULTS CALCULATED WITH NATURAL CIRCULATION MODEL

Mean temperature rise across core, ºC	129
Core outlet temperature, ºC	793
Core inlet temperature, ºC	664
Total core coolant flow rate, kg/s	21 770
Coolant velocity in IHX tubes, m/s	0.369
Coolant Reynolds number in IHX tubes	58 000

Intermediate heat transport circuit

A forced circulation, ambient pressure fused salt (flibe) intermediate heat transport loop carries the heat from the in-vessel intermediate heat exchanger (IHX) to the balance of plant (BOP). Figure XXIV-4 shows the overall heat flow for the reactor and BOP at full power of 400 MW(th).

Main heat transport system

Figure XXIV-21 illustrates the heat removal pathways provided for normal and upset (decay heat removal) conditions.

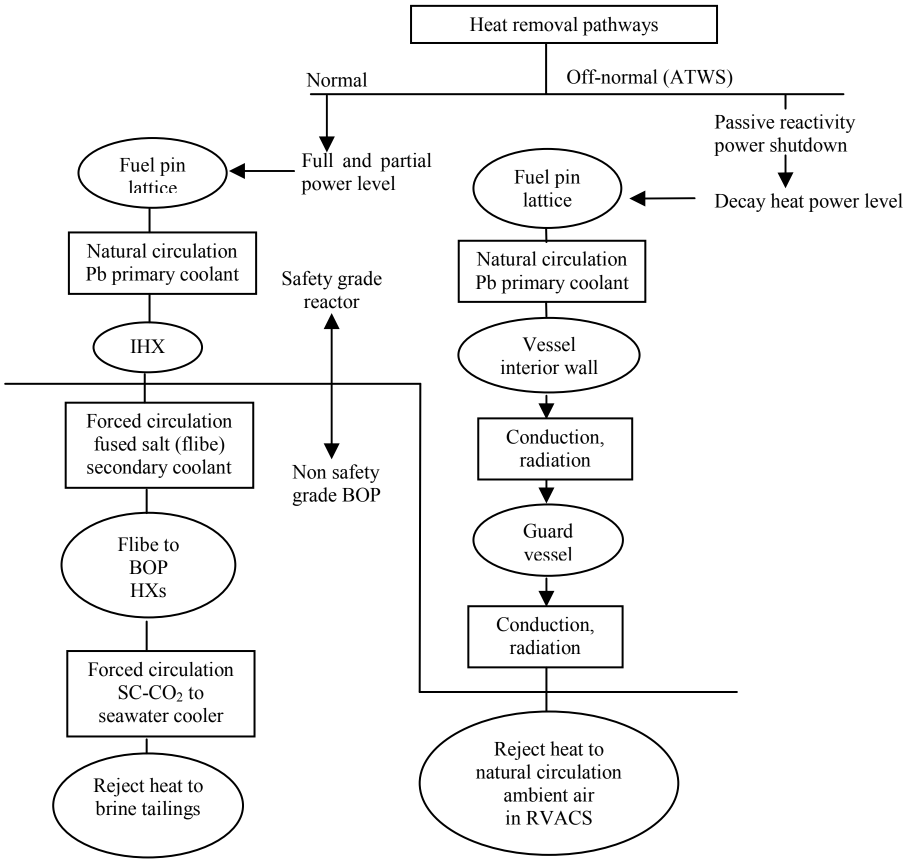


FIG. XXIV-21. Heat removal pathways.

XXIV-2.2. Description of the balance of plant and systems

The STAR-H2 reactor heat source drives a balance of plant through an ambient pressure, forced circulation, fused salt (flibe) intermediate heat transport loop.

The STAR-H2 balance of plant (BOP) is comprised of three cascaded cycles (water cracking; Brayton cycle; and desalination) operating at successively lower temperatures – and with the heat rejected from each cycle used to drive the succeeding cycle, see Fig. XXIV-4. The reactor supplies 400 MW(th) of heat between 800 and ~ 650ºC. The strategy for BOP plant design is to use that heat as follows:

(a) Use as much of the heat as possible to maximize hydrogen production – consistent with (b);
(b) Use only as much of the heat to make electricity in the Brayton cycle as is required to run the BOP (i.e., neither off-site electricity sales nor reliance on off-site power is required);
(c) Use whatever heat is finally left over to desalinate water – first to supply distilled water feedstock to the water cracking plant and second, to use the excess for off-site sales of potable water;
(d) Finally, heat at a temperature near ambient is rejected from the plant in the desalination brine tailings. The plant is designed to minimize this lost heat – both for efficiency and to minimize ecological impacts of heat rejection.

Figure XXIV-22 shows the cascaded cycles in the BOP.

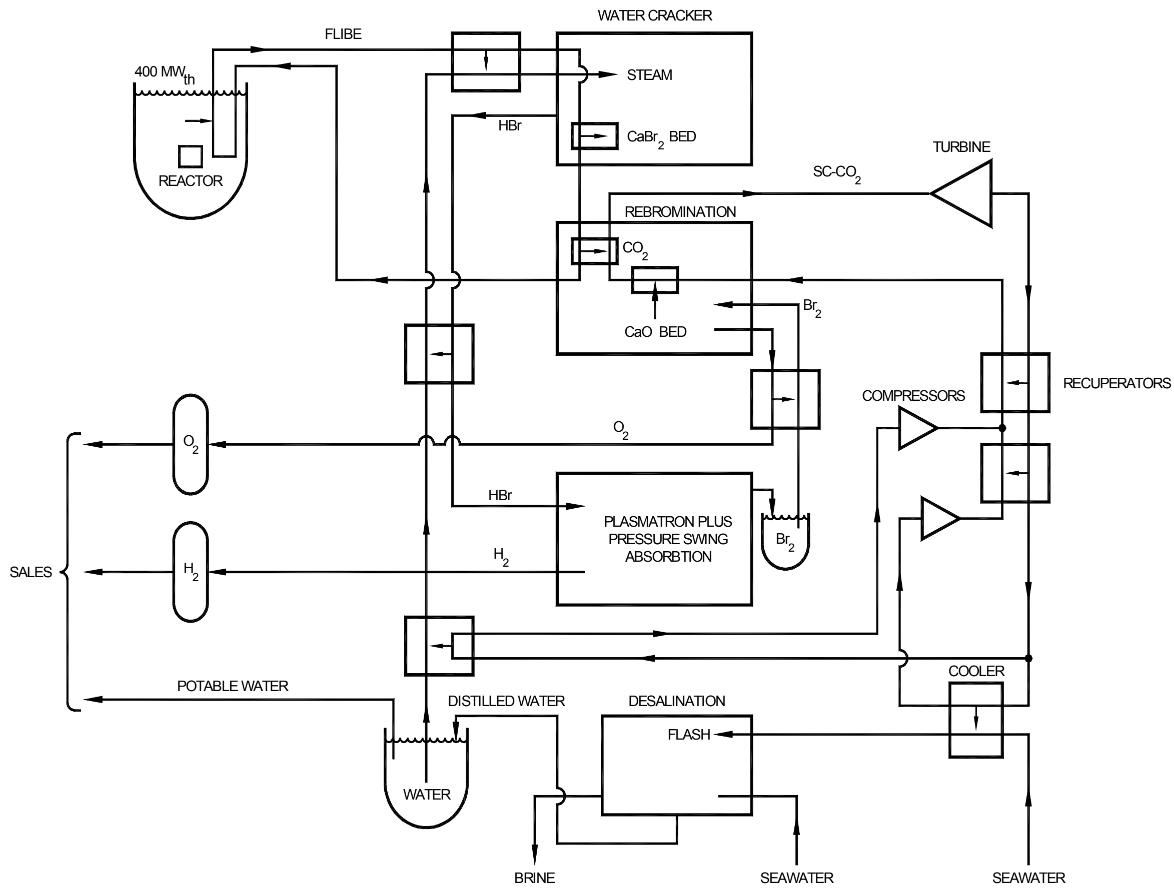


FIG. XXIV-22. Process heat cascade: water cracking → Brayton cycle → desalination.

The Ca-Br water cracking cycle has three main segments: an endothermic "water cracking" segment where $CaBr_2$ and steam react at 700 to 750°C to produce HBr and CaO; an exothermic Ca rebromination segment at 600°C where CaO and bromine react to regenerate $CaBr_2$ for recycle and release heat and oxygen; and a plasma chemistry HBr cracking segment at room temperature where electrical driven (RF frequency) energy cracks HBr to regenerate bromine for recycle and to release hydrogen. The plasmatron is followed by a pressure swing absorption cascade, which cleans and pressurizes the hydrogen to meet pipeline delivery specifications.

The high temperature endothermic water-cracking segment receives heat from the reactor through the flibe (molten salt) intermediate heat transport loop. Heat is delivered to the water cracking plant over the range ~750 to ~700ºC via two heat exchangers – one to complete the superheating of steam to 750ºC; the other to maintain the $CaBr_2$ beds at 725ºC. It must remain at those temperatures – even at partial load – in order to drive the chemical reaction. The final stage of steam superheat includes allowance for heat to overcome thermal inertia of the calcium titanate support for the calcium reagent as it cycles from one reaction segment at 600ºC to the other at 725ºC.

In the segment for regeneration of $CaBr_2$ from CaO, heat at 600ºC is rejected. It is used to help drive the SC-CO_2 Brayton cycle. However, because that heat supply is not sufficient, the Brayton cycle also receives heat from the flibe loop. The flibe then returns to the reactor intermediate heat exchanger (IHX) at ~650ºC.

After expansion in the Brayton cycle turbine, the SC-CO_2 passes through a high temperature and a low temperature recuperator. It exits the low temperature recuperator at 125ºC and the heat liberated from cooling it further to 100ºC is used to provide vaporization and slight superheat to the distilled water feedstock destined for the water cracking cycle. Further heat is rejected from the SC-CO_2 in the Brayton cycle cooler, which cools the SC-CO_2 to 31ºC in preparation for its compression. Seawater provides the cooling fluid for the cooler, and the resulting 100ºC seawater, which exits the Brayton cycle cooler, then delivers heat and seawater feedstock to the desalination plant. Finally, as little as is feasible of heat at temperature above ambient exits the plant in the form of heated brine tailings from the desalination process.

Regenerative heating is used in the water cracking plant to increase efficiency. The O_2 coming off the CaBr2 regeneration step at 600ºC is cooled to room temperature by heating up bromine, which had been recovered in the plasmatron. The bromine is taken back up to 600ºC for driving the Ca rebrominization reaction. The HBr from the water cracking reaction at 700ºC is cooled to room temperature for introduction into the plasmatron, and its rejected heat is used to superheat the distilled water feedstock to near reaction temperature.

The Brayton cycle turbo-generator is sized to meet onsite demands; it is not intended for electricity sales. The electricity generated by the SC-CO_2 Brayton cycle drives the flibe pump, the plasmatron, reagent pumps, pressure swing absorption compressors and the desalination plant brine pumps. The Brayton cycle is run at constant temperature and pressure at the turbine inlet and at constant temperature and pressure at the cooler outlet. For partial load, its power output is adjusted via SC-CO_2 mass flow rate.

The desalination plant to produce potable water from seawater feedstock (assumed at 25ºC) is a feed forward Multi-Effect-Distillation (MED) design, which is driven by the heat recovered in cooling the SC-CO_2 from 100ºC down to the SC-CO_2 critical temperature of 31ºC. The brine from the desalination plant is rejected at 35ºC.

Two reagent buffers are used so that mass flows throughout the BOP need not always be in perfect quasi-equilibrium. After regeneration in the plasmatron, the bromine is stored as liquid in an ambient temperature and pressure buffer tank from which it can be withdrawn as needed to feed the $CaBr_2$ regeneration step. Similarly, the distilled water produced by the desalination plant – which is substantially in excess of requirements for water cracking feedstock needs – goes to an atmospheric pressure, 35ºC holding tank for off-site sales of potable water. The distilled water feedstock for the water-cracking segment of the Ca-Br cycle is drawn from this buffer tank as needed.

Many alternative options can be considered for productive use of reject heat at any of the three temperatures. Moreover, the passive safety / passive load-follow design of the reactor facilitates siting it in industrial parks near urban areas and / or close to cities – facilitating cogeneration opportunities.

Several options for yet further extraction of marketable product from the brine tailings have been identified for study in the future. Similarly, several alternative bottoming cycles for use at landlocked sites have been identified for consideration in the future.

XXIV-2.4. Plant layout

General philosophy governing plant layout

Key features for the STAR-H2 concept relevant to plant and reactor layout include:

— A heat source reactor driving a non nuclear safety grade BOP through a fused salt intermediate loop at ~800°C operating temperature;

— A BOP which itself presents an explosion and fire hazard to the reactor;
— Siting near the ocean for a desalination mission;
— A strategy of factory fabrication and rapid site assembly of the reactor;
— Construction of the BOP by indigenous industry and labour – in parallel or preceding reactor installation;
— A strategy of whole core cassette refuelling after very short cooling time.

These features would strongly affect the plant site layout and civil engineering, the reactor structural design and the reactor building design; and they are all interconnected through the logistics of site assembly. For example, the site layout should (1) include provision for heavy lift crane and crawler access for reactor assembly and for (later) refuelling operations; (2) must provide separations and hardening to account for mutually induced safety issues when coupling a nuclear and chemical plant; and (3) must be configured to facilitate flows of the feed stocks and products entering and exiting the site.

The reactor itself presents unique design challenges and opportunities in that the core internals and fuel tend to float in the coolant – essentially reversing gravity for the internals – while at the same time the density of the Pb coolant presents a heavy load on the vessel and its supports. While the coolant is not chemically reactive with air, the trace Po content requires that cover gas control must be maintained (as in Na reactor operations) while at the same time a very large diameter hole through the vessel cover is needed so as to pass the whole core refuelling cassette during refuelling.

Many of the design and interface issues, which arise, are unique – not having been encountered for classical nuclear power plant designs. As a result, a number of the design solutions suggested here are non-conventional and qualitative. This first cut at an overall conceptual approach has been undertaken to help identify the issues and to clarify the tradeoffs among the coupled individual systems.

Figure XXIV-23 shows one segment of a potential layout for a multi STAR-H2 site on a location adjacent to an ocean.

The overall layout provides for ocean-going ship or barge access to an array of man-made peninsulas, each of which sites twin STAR-H2 plants. The canals provide shipping access for the initial delivery of both reactor and BOP equipment, for ship or barge access for the cassette refuelling operations, and for optional ship transport of product hydrogen and oxygen. They also provide flow channels for seawater feedstock delivery to the desalination plants and for brine discharge; the brine discharge pathway provides opportunities for further cooling of discharge heat from the cascaded thermodynamic cycles of the BOP – for example in greenhouses.

The site is intended to be expandable, so that as the customer city grows, capacity can be added incrementally by extending the canal system and man-made peninsulas for plant siting.

The chemical plant and tank farms present a fire hazard; the canal layout will provide two egress routes for ships and two access routes for fireboats; all peninsulas can be accessed by water from two sides.

The site civil construction and the non-nuclear safety grade BOP could be constructed and operated to local industrial standards using local labour and local construction companies. Only the reactor building itself houses nuclear material and contains nuclear safety grade equipment and construction. Thus, only the reactor compound would operate behind a safeguards fence to nuclear safeguards and nuclear safety standards. The ocean access avenues to the BOP and separately to the reactor segments of the site would facilitate separate construction activities performed under separate administrative control to go on in parallel.

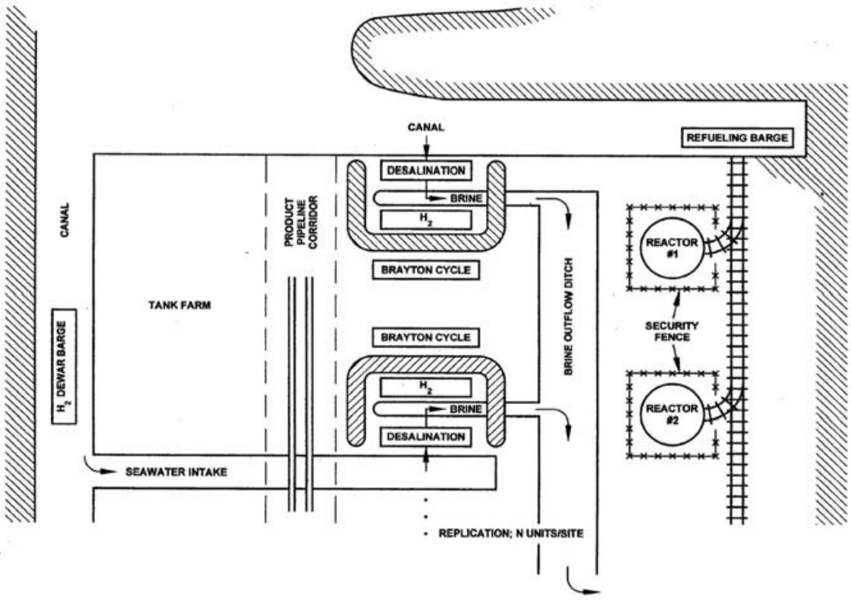


FIG. XXIV-23. A multi-plant site layout on an ocean setting.

The topological considerations for transporting the numerous mass fluxes entering and leaving the site:

- Product streams:
  - Potable water;
  - Hydrogen;
  - Oxygen;
  - Electricity (optional);
- Feedstock streams:
  - Seawater;
  - Fresh fuel cassettes;
  - Electricity (optional); and
- Effluent streams:
  - Brine;
  - Used fuel cassettes;

combined with the safety separation considerations for the volume-occupying assets of the site:

- Conversion assets:
  - Reactor;
  - Water cracking plant;
  - Desalination plant; and
- Inventories:
  - Water;
  - Hydrogen;
  - Oxygen;

provide a challenging set of layout issues.

Figure XXIV-23 illustrates some of the proposed approaches to those topological issues. Twin plants are laid out as mirror images on each man-made peninsula – water on three sides and land on one side. The facilities are laid out linearly on the peninsula; when starting on the water side tip of the peninsula and moving toward the right, one first encounters the oxygen storage tanks; then the potable water tanks; then the hydrogen tanks; then the close-coupled Ca-Br/Brayton cycle/desalination BOP equipment – which are aligned in a heat cascade sequence running crosswise of the peninsula; and finally the reactor.

Figure XXIV-23 shows the routing of input, output and internal mass fluxes. Fresh seawater enters the site from the open ocean into the man-made harbour and then moves down the canals separating the peninsulas. These canals feed the desalination plants; brine exits to the right into a brine collection canal running between the BOPs and reactors. This brine discharge canal circles around the reactors. This circuitous route is taken to avoid the need for multiple bridges under the heavy capacity railroad in Fig. XXIV-23. The brine is rejected from the desalination plant at 10°C above the seawater inlet temperature – a normal practice for desalination plants – and the extended travel time of the brine from BOP back to the ocean allows opportunity for further cooldown of the brine before return to the sea. Alternately, that heat and brine may be put to profitable use such as heating acres of greenhouses or perhaps shellfish beds.

A heavy capacity rail line (running vertically at the right of the Fig. XXIV-23) connects the refuelling ship dock to each of the reactors. This rail line is used in support of the original site installation of the reactor and is later used for cassette refuelling operations.

A pipeline corridor (running vertically in Fig. XXIV-23) carries products (hydrogen, oxygen, and water) inland to the customer city – passing between the tank farms and the BOP.

The hydrogen tank farms and the ship access to them permits the site to be used as a hydrogen-shipping terminal. If the STAR-H2 facilities are undersized for the city, transport ships may arrive carrying supplemental hydrogen generated by the breeder reactors located at the regional fuel cycle centres, for unloading into the tank farm from which it can be fed into the hydrogen pipeline. Alternately, if the STAR-H2 facilities are oversized for the city, surplus hydrogen can be loaded onto ships for delivery to a sea or river-accessible hinterland.[8]

The brine exhaust canal flows between the reactor and the BOP, and as a result the flibe intermediate heat transport loop carrying heat from the reactor to the BOP must cross this canal. The flibe pipes tunnel under the earthen/gravel mound, which covers the reactor building through an underground concrete pipe tunnel. Then, two choices exist – they can continue to the BOP moving under the canal through a concrete tunnel, or they can emerge from the tunnel, rise into the air, and pass over the canal. The latter choice would provide for a piping loop to accommodate thermal expansion in a natural way; while on the other hand, the tunnel route would provide for an insulation function in a natural way.

Neither choice would present incremental nuclear safety vulnerability – because the reactor is designed to passively accommodate a loss of heat sink accident and, therefore, a flibe pipe break or freeze-up is expected to present no hazard to the reactor. A decision is yet to be made between the two approaches.

Two choices were available for the linear layout of assets along the lengths of the peninsulas – the reactors could be positioned inland and serviced by a heavy capacity rail line (as shown in Fig. XXIV-23), or the layout could be reversed, placing the reactors at the tip of the peninsulas where they could be serviced more directly by the reload ship. The reactor-inland orientation was selected for several reasons:

---

[8] Like it might be the case in countries like Indonesia.

(1) Once installed, the reactors are refuelled only once in 20 years so even with a string of ten reactors lined up, the use of the railroad will be infrequent (i.e. the fuel cassette mass flux on the railroad is small);

(2) To the contrary, product water, hydrogen, and oxygen is delivered continuously and may optionally be delivered from the STAR-H2 plant to consumers by ship as well as through pipelines, so shipping access to the tank farms may be needed frequently (the mass flux of product on the canals might be high);

(3) Additionally, the tank farms may be used as receiving terminals for hydrogen produced by central breeders and shipped from the regional fuel cycle centres to augment the production conducted at the STAR-H2 plant;

(4) The reactor itself is designed for a 60-year lifetime before replacement whereas chemical plants tend to be designed for 30 years and the high temperature tanks and pipes containing corrosive reagent may require replacement at 10 years. Some of the chemical plant vessels are as big or bigger than the reactor vessel itself, so direct barge access to the chemical plant is likely to be more valuable than direct barge access to the reactor plant;

(5) The inland location for the reactor is likely to be better located in the water table for emplacing the below ground silo into which the reactor is placed;

(6) The inland location for the reactor provides somewhat better shelter from ocean swells and tsunami – leaving the non-nuclear-safety grade BOP and tank farms to take the brunt of nature's furies;

(7) Finally, placing the tank farms on the tips of the peninsulas places them within operating range of fire ships. That was the consideration for providing two water routes to each peninsula – so that product transport ships would never get trapped by a single tank farm fire, and so that fireboats would always have access to both sides of a peninsula to better fight a fire.

Reactor building and containment layout

Figure XXIV-13 shows a side view of the concrete reactor building, its earthen mound cover and the reactor vessel emplaced below grade in a concrete silo within the reactor building. The reactor building's function is to protect the reactor silo and reactor head from the natural elements, to contain ancillary reactor support equipment and to provide operating floor space for refuelling operations. It has no containment function.

The building has a large-diameter access port in the ceiling – large enough for open top reactor construction and whole core cassette shipping cask entry for refuelling operations (see Fig. XXIV-15).

The reactor building is protected from outside influences by the earthen mound, which covers it. If necessary, the earthen mound could be truncated on one side to allow for close-up approach of a heavy-lift crawler crane for the assembly and refuelling operations.

The containment function is not vested in the reactor building or the earthen mound or the flibe heat transport pipes to the BOP. The first containment boundary is the fuel pin cladding. The second containment boundary is comprised of the reactor vessel, the refuelling cap on the reactor head, and the IHX tube walls. The third containment is the guard vessel and an extension of the vessel head – and (maybe) fast acting valves on the flibe loop pipes.

The earthen/gravel mound over the reactor building is a low-cost means to protect the reactor and its containment structure from external hazards, which may include:

— Chemical plant/tank farm explosions and/or fires;
— Natural hazard missiles generated by high winds, tornados, hurricanes, or typhoons;
— Forest fires and scrub fires;
— Ocean swells or tsunami; and
— Deliberate attacks such as tank shells, airplane crashes, etc.

REFERENCES

[XXIV-1] SIENICKI, J.J., et al., The STAR-LM lead-cooled closed fuel cycle fast reactor coupled to a supercritical carbon dioxide Brayton cycle advanced power converter, GLOBAL 2003 (Paper presented at Int. Winter Meeting, New Orleans, November 16–20, 2003) ANS/ENS.

[XXIV-2] WADE, D., et al, STAR-H2: A 400 MW(th) lead-cooled, long refuelling interval reactor for hydrogen production, ICONE-11 (Proc. of Int. Conf. on Nuclear Energy, Tokyo April 20–23, 2003), paper ICONE11-376576, ASME.

[XXIV-3] SIENICKI, J., et al, A small secure transportable autonomous lead cooled fast reactor for deployment at remote sites, Americas Nuclear Energy Symposium 2004 (Paper presented at Symposium, Miami Beach, Florida, Oct. 3–6, 2004).

[XXIV-4] DOCTOR, R., MATONIS, D., WADE, D., Hydrogen generation using a calcium-bromine thermochemical water-splitting cycle, OECD-NEA Second Information Exchange Meeting on Nuclear Production of Hydrogen, Argonne National Laboratory, Oct. 2–3, 2003 (Proceedings) OECD NEA No. 5308 (2004).

[XXIV-5] MOISSEYTSEV, A., SIENICKI, J., WADE, D., Cycle analysis of supercritical carbon dioxide gas turbine Brayton cycle power conversion system for liquid metal-cooled fast reactors, ICONE-11 (Proc. of Int. Conf. on Nuclear Energy, Tokyo April 20–23, 2003), paper ICONE 11-36023, ASME.

[XXIV-6] FAIBISH, R., Seawater desalination as a bottoming cycle for a hydrogen producing nuclear plant complex, International Youth Nuclear Conference (paper presented at Int. Conf., Toronto Canada, May 9–13, 2004).

[XXIV-7] WADE, D.C., et al., STAR-H2: A long-refuelling interval battery reactor for hydrogen and water supply to cities of developing countries, 5th International Conference on Nuclear Option in Countries with Small and Medium Electricity Grids (paper presented at Int. Conf., Dubrovnik, Croatia, May 16–20, 2004).

[XXIV-8] MARCHETTI, C., Energy Islands in the final configuration of the hydrogen economy, Plenary lecture at the World Hydrogen Conference, 21–26 June 1998.

[XXIV-9] NITSCH, J., VOIGT, C., Hydrogen in a future energy supply system, Chapter 11 of Hydrogen As An Energy Carrier, C. Winter and J. Nitsch (eds), Springer Verlag (1988).

[XXIV-10] Redbook: Uranium resources, production and demand, the OECD NEA and IAEA, 1999. See also http://www/uilondon.org.

[XXIV-11] US-DOE-NE Report to Congress, "Small modular nuclear reactors", May 2002, available at DOE web site, (http://www.doe.gov).

[XXIV-12] MILLER, A., DUFFEY, R., POEHNELL, T., Signports to a nuclear + hydrogen (N+H2) future, American Nuclear Society (Proc. of Int. Winter Meeting, Teno, Nevada, Nov 11–15, 2001).

[XXIV-13] BRUNTLAND, G., World Commission on Environment and Development, Our Common Future (The Brundtland report), Oxford University Press, Oxford, United Kingdom (1987).

[XXIV-14] KOCH, L., Formation and recycling of minor actinides in nuclear power stations, Chapter 9 of the Handbook on the Physics and Chemistry of the Actinides, A. J. Freeman and C. Keller (eds), Elsevier Science Publishers BV, North Holland Publishers, Amsterdam, Oxford, New York, Tokyo (1986).
[XXIV-15] HANNUM, W., guest editor, The technology of the integral fast reactor and its associated fuel cycle, Special issue, Vol. 31. No. 1/2 of Progress in Nuclear Energy, ISSN 0149-1970 (1997).
[XXIV-16] FISTEDIS, S.H. The Experimental Breeder Reactor-II Inherent Safety Demonstration, Elsevier Science Publishers B. V., North Holland Publishers, Amsterdam, Oxford, New York, Tokyo (1987).
[XXIV-17] HILL, D., RAGLAND, W., ROGLANS, J., The EBR-II probabilistic risk assessment: Lessons learned regarding passive safety, reliability engineering and system safety **62**, p. 43–50, Elsevier Science Limited (1998).
[XXIV-18] INTERNATIONAL ATOMIC ENERGY AGENCY, International Nuclear Fuel Cycle Evaluation, Report, INFCE/PC/2/9, Vienna (1980).
[XXIV-19] HANNUM, et al, Non-proliferation and safeguards aspects of the IFR, Chapter 12 of The Technology of The Integral Fast Reactor and Its Associated Fuel Cycle, Special Issue of Progress in Nuclear Energy, W. Hannum, guest ed. Vol. 31, No. 1/2, ISSN 0149-1970 (1997).
[XXIV-20] MOHAMED ELBARADEI, Towards a safer world, The Economist, (Oct. 18, 2003 issue).
[XXIV-21] Nuclear Energy in a Sustainable Development Perspective, published by OECD-NEA, ISBN92-64-18278-0, Paris (2000).
[XXIV-22] HÄFELE, W., On the nature of nuclear power and its future, Plenary Speech, pp. 3–17, GLOBAL'93, Future Nuclear Systems: Emerging Fuel Cycles and Waste Disposal Options (Proc. Int. Conf. Seattle, Washington, Sept. 12–17, 1993).

LEAD-BISMUTH EUTECTICS COOLED LONG-LIFE SAFE SIMPLE SMALL PORTABLE PROLIFERATION RESISTANT REACTOR (LSPR)

Research Laboratory for Nuclear Reactors, Tokyo Institute of Technology (RLNR TITech),
Japan

XXV-1. General information, technical features and operating characteristics

XXV-1.1. Introduction

The design philosophy behind the LSPR concept is as follows.

Past trend of nuclear power development and associated issues

Conventional nuclear power reactors have almost reached a power size limit after pursuing the economies of scale by building larger plants. Future directions in which nuclear power can be effectively developed are unlikely to be toward larger sizes. Currently it is almost impossible to find new sites for larger plants in the developed countries; larger plants also pose a large economic risk that might be unbearable even for large companies or governments.

Small reactors [XXV-1]

Small reactors can be built on less than ideal land such as small and less stable areas; therefore, it is much easier to find a proper site for a small reactor.

Small reactors can also be utilized for several purposes other than electricity production, such as heat generation, seawater desalination, etc. The transport of heat and pure water for long distances requires high costs and encounters energy and material losses. For such purposes, a local reactor is appropriate; the power required of a local reactor is small and, therefore, the reactor should be small.

Potential power-plant customers generally hesitate to build large reactors due to a high investment risk. Even a delay in construction may incur a considerable economic penalty; therefore, a smaller reactor may be preferable, if economically feasible.

Innovative nuclear reactors

Innovative nuclear reactors are expected to solve future problems such as global warming and other environmental issues, resource shortages, proliferation and security concerns, etc.

Innovative reactors are also required for purposes other than electricity production, such as fuel breeding, hydrogen production and high temperature process heat applications, motive power, etc. Small reactors provide an attractive domain for the innovations needed to address the abovementioned problems.

Economical performance

The disadvantage of scale is a considerable factor degrading the economic performance of small reactors. However, there are many factors pertaining to small reactors that might be used to improve economic performance. These are discussed in section XXV-1.6.1.

Small reactors for developing countries

Smaller reactors pose less radiological hazard because the total quantity of contained radioactive material is smaller. Furthermore, certain small reactors can incorporate inherent safety characteristics, where safety function relies more on natural phenomena and less on human actions or mechanical devices.

If a reactor is small enough to be transportable and has a sufficiently long core lifetime, it can be built in a factory and shipped to a site. When such reactor is designed to be sealed, the discharge of any fuel becomes impossible outside the factory, and this would enhance its proliferation resistance.

In the 21st century, global warming caused by the carbon dioxide emissions might become an urgent problem. The carbon dioxide emissions from developing countries would be especially important, and nuclear reactors could be deployed to minimize their scope.

However, the infrastructure and technical skills in developing countries are often insufficient to realize a full-scale nuclear power development programme. Furthermore, some developing countries are politically unstable. The energy demand is in many cases local and small. As mentioned before, a small reactor could be made simple and easy to operate and maintain; it may incorporate inherent safety features and enhanced proliferation resistance. Therefore, small reactors may have a good potential to solve global warming problems resulting from carbon emissions in developing countries.

Targeted features and their interrelation

Based on the abovementioned considerations, the features targeted for a small reactor include long-life core, design simplicity (resulting in easy maintenance and operation), small size and transportability, strong reliance on inherent safety features, and enhanced proliferation resistance achieved through the operation with a sealed reactor vessel.

However, some of these features are tightly interrelated. For example, transportability requires a small size and, therefore, these two features are essentially similar. By examining all these characteristics from a viewpoint of similarity, it appears that only long core lifetime and small reactor size are basic characteristics and all others can be derived from these two.

With these two features being achieved, a scenario is that a long-life small reactor is built at a factory in a developed country, shipped to a site in a developing country, installed there and operated over a certain period without reloading and shuffling of fuel in the core, and replaced with a new reactor after its operational life is completed. A barge-mounted reactor is an alternative; it can be shipped to a site and operated as a power plant in an appropriate port.

In these scenarios, the procedures requiring high technical skills, such as fuel replacement, are not required. The maintenance of such reactor becomes simple; and heat produced in accidents escapes easily from the core surface due to a favourable core surface-to-volume ratio; in addition to this, the power shape is more stable. For fast reactors, it is especially important that void coefficients in small reactors are shifted toward the negative side.

From the abovementioned considerations, it appears that both long core life and small reactor size are tightly related to enhanced safety and design simplicity.

In the scenarios outlined, the reactor is always sealed during transportation and operation and measures are taken to prevent the access to fuel located inside the reactor vessel. Therefore, it could be assumed that such reactors incorporate enhanced proliferation resistance.

Small reactor with a long-life core

From the arguments presented above it appears that the features of long-life core and small reactor size are basic to realize the concepts of long-life, safe, simple, small, portable, and proliferation-resistant reactors. However, these two basic features are generally in conflict because small sized reactors usually show poor neutron economy, also resulting in the impossibility to achieve a high fuel burn-up. The neutron economy or, in other words, the requirement of reactor criticality, provides restrictions for both the size and lifetime of the reactor core.

This discussion leads to a conclusion that a long-life safe simple small portable proliferation-resistant reactor requires excellent neutron economy. It is well known that fast reactors show much better neutron economy than thermal or epithermal ones.

Lead-bismuth eutectics cooled fast reactor

The abovementioned arguments suggest that small fast reactors should be investigated.

At present, sodium is considered the best coolant for fast reactors due to its superior cooling ability, which can help to increase the core power density and shorten the doubling time. Short doubling time was an indispensable requirement in the early phases of development and construction of fast breeder reactors from 1960s through 1980s. It is reported that for safety reasons, the lead-bismuth eutectic (LBE) cooled fast reactor was originally considered [XXV-2].

As previously mentioned, the neutron economy is very important to realize long-life small reactors. For these, it is expected that LBE coolant has better performance in neutron economy than sodium coolant because of a larger elastic and smaller inelastic scattering cross section. It is reported that the LBE cooled long-life small fast reactor shows better performance for neutron economy, burn-up reactivity swing and void coefficient [XXV-3].

However, in the Western world for a long time it has been considered that LBE cannot be used as a reactor coolant due to negative experimental results on corrosion. Opposite to this, in the Russian Federation this problem has been solved by control of the oxygen concentration and LBE was employed as a submarine reactor coolant. It is reported that 8 nuclear submarines with LBE coolant were constructed and operated for about 80 reactor-years [XXV-2]. After the Russian research results have been opened, many research works targeting corrosion experiments were started worldwide. The corrosion problem is considered solvable by choosing proper materials, temperature, and fluid velocity and oxygen concentration.

Characteristics of LBE

The most important merit of LBE compared to sodium is chemical inertness; the LBE does not react violently with water or air.

The boiling temperature of sodium is 1156 K and it is not easy to prevent boiling in severe accidents. If the void coefficient is positive, sodium boiling may lead to a core destruction accident. By contrast, the boiling temperature of LBE is 1943 K with which the possibility of boiling is negligible. Furthermore, as mentioned before, the void coefficient for LBE is more negative than for sodium.

The density of LBE is about 12 times the sodium density; the viscosity of LBE is large and the pressure drop is expected to be large; the Prandtl number is about 3 times the sodium value. These characteristics lead to a poor cooling ability of the LBE; therefore, the power density of a LBE cooled reactor should be lower and, for corrosion protection, the flow rate must be lower too.

Since the power density of a small reactor is usually restricted by the requirement of criticality preservation under fuel burn-up [XXV-4], the power density of some very small fast reactors, even with sodium coolant, is very low. Therefore, the poor cooling ability of LBE may be not so important for long-life small reactors.

For natural circulation capability, LBE-cooled reactors can offer better potential through larger equivalent hydraulic diameter of the core [XXV-5]; it also improves the reactor response in accidents.

As mentioned before, the LBE cooled long-life small fast reactor shows better performance for neutron economy, burn-up reactivity swing and void coefficient due to a larger elastic scattering cross section. The LBE also exhibits a better shielding effect against neutrons and gamma rays, which facilitates a reduction of the total reactor size.

The radioactive materials produced in the coolant during operation are also important. For sodium, ^{24}Na should be considered with the half-life of 15 hours, which emits high-energy gamma rays (2.8 MeV and 1.4 MeV). Therefore, the primary loop of a sodium cooled reactor shows a very high dose rate. On the other hand, LBE does not produce so many gamma ray emitters, although polonium, an alpha ray emitter, is produced. Altogether, the expected dose rate around the primary loop of a LBE cooled reactor is much lower than for sodium cooled reactor.

LSPR

To achieve a long-life safe simple small portable proliferation-resistant reactor, a lead-bismuth-eutectic (LBE) coolant was selected as the best candidate. The original concept of a long-life small LBE cooled fast reactor was proposed more than 10 years ago [XXV-3], which was the world's first trial of this kind. The name of this reactor, the LBE cooled long-life safe simple small portable proliferation-resistant reactor (LSPR) distinguishes it from similar reactors proposed by other institutes.

The LSPR concept is being developed at the Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology (RLNR TITech, Japan).

XXV-1.2. Applications

The LSPR is a long-life small reactor, in which the thermal power output is 150 MW and the coolant output temperature is nearly 800 K. It can be used for many applications. Current plans are to use it for electricity generation and co-generation including district heating, seawater desalination, hydrogen production, process steam production, or a combination thereof.

XXV-1.3. Special features

As it was already mentioned, the LSPR has a long-life core, incorporates many inherent safety features, has a simple design with easy maintenance and operation, has a small size core; is transportable and designed to operate with a sealed reactor vessel.

The LSPR is a factory fabricated and fuelled reactor designed to operate without on-site refuelling, i.e. without reloading or shuffling of the fuel during the whole reactor lifetime.

XXV-1.4. Summary of major design and operating characteristics

A summary of major design and operating characteristics of the LSPR is given in Table XXV-1.

Figure XXV-1 gives a general view of the reactor vessel and internals. An integral type reactor design is employed in which steam generators are installed within the reactor vessel, which is possible since severe reaction between the LBE reactor coolant and steam generator water coolant is not anticipated. Nitride fuel with a high thermal conductivity is chosen as a principal fuel candidate because of compatibility with the LBE coolant. The application of metal fuel, which has a potential for higher performance, is left for future studies to be concerned with material compatibility issues. A simplified schematic diagram of the LSPR plant is shown in Fig. XXV-2. The integral type primary circuit incorporates 2 steam generator sets; a mechanical pump; a coolant-purifying unit and an oxygen concentration control unit. Serpentine tube type steam generators are employed for the reason of compactness.

Selection of the driving devices for the heavy metal coolant is one of the key issues in the reactor design. In this design, the mechanical centrifugal pumps were selected assuming that further studies would address long-life cores, possibly with a higher pressure drop, and because of versatility in the ability to assure adequate pump coastdown times. The natural circulation potential of the primary circuit is arranged to constitute from 30% to 40% of the nominal primary flow at the nominal heat balance level.

TABLE XXV-1. SUMMARY TABLE OF MAJOR DESIGN AND OPERATING CHARACTERISTICS OF LSPR

ITEMS	SPECIFICATIONS
General characteristics	
Installed capacity: - Thermal; - Electric.	 150 MW(th) 53 MW(e)
Load factor (target)	95%
Major design characteristics	
Type of fuel	Nitride
Fuel enrichment	10 – 12.5%
Type of coolant	Lead-bismuth eutectics (LBE)
Type of structural materials	HT-9
Core type / characteristic dimensions: - Core diameter; - Core height.	 1.65 m 2.0 m
Reactor vessel type / characteristic dimensions: - Reactor vessel diameter; - Reactor vessel height.	 5.2 m 15.2 m
Number of circuits	2
Thermodynamic cycle efficiency	35%
Neutron-physical characteristics	
Void reactivity effect	< - 0.8% δk/k (total void)
Burn-up reactivity swing	< - 0.1% δk/k
Power peaking factor	1.64

Reactivity control	
Reactivity control mechanism	Secondary coolant flow rate
Number of independent active reactor control and protection systems	2
Thermal-hydraulic characteristics	
Circulation type	Forced
Pump type	Centrifugal
Core coolant temperatures: - Inlet; - Outlet.	 360°C 510°C
Primary coolant flow rate	12 300 t/hour
LBE velocity in the core	0.9 m/s
Pressure in primary circuit: - Vessel bottom static pressure; - Total pressure drop.	 15 kg/cm^2 0.7 kg/ cm^2
Secondary coolant system: - Feedwater temperature; - Feedwater flow rate; - Steam temperature at steam generator (SG) outlet; - Steam pressure at SG outlet.	 210°C 294 t/hour 280°C 6.47 MPa
Temperature limits: - Fuel; - Cladding; - Coolant (boiling point).	 2780°C 700°C 1670°C
Maximum fuel temperature	630°C
Maximum cladding temperature	510°C
Fuel burn-up characteristics	
Maximum burn-up of discharged fuel	9% FIMA
Fuel lifetime / period between refuellings	4 200 effective full power days
Design basis lifetime of reactor vessel and structures	
Reactor core	12 years
Reactor vessel	> 12 years
Structures	> 12 years

As shown in Fig. XXV-2, re-circulation in the secondary system is arranged with a free surface in the water drum (in power operation); which is to ensure passive heat removal by natural circulation through the steam generators. The steam generator auxiliary heat removal system (SGAHRS) is adopted, in which the decay heat is removed by natural convection through the steam generators to air coolers, without auxiliary cooling systems in the reactor vessel. In this, the reactor vessel auxiliary cooling system (RVACS) is installed as a back-up system to the SGAHRS. To control the thermal conductivity and enhance the function of the RVACS, the reactor wall is designed to stay at a cold leg temperature in power operation but to encounter the hot leg temperature brought by the coolant overflow in accident conditions.

Natural uranium or depleted uranium based fuel assemblies are placed in the centre of the core as an inner blanket, whereas plutonium fuel assemblies are settled outside of the inner blanket.

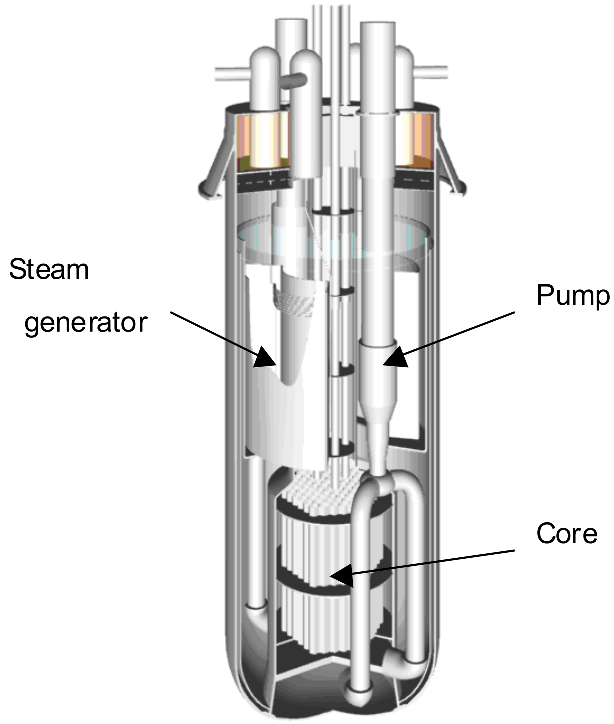


FIG. XXV-1. General view of the LSPR vessel and internals.

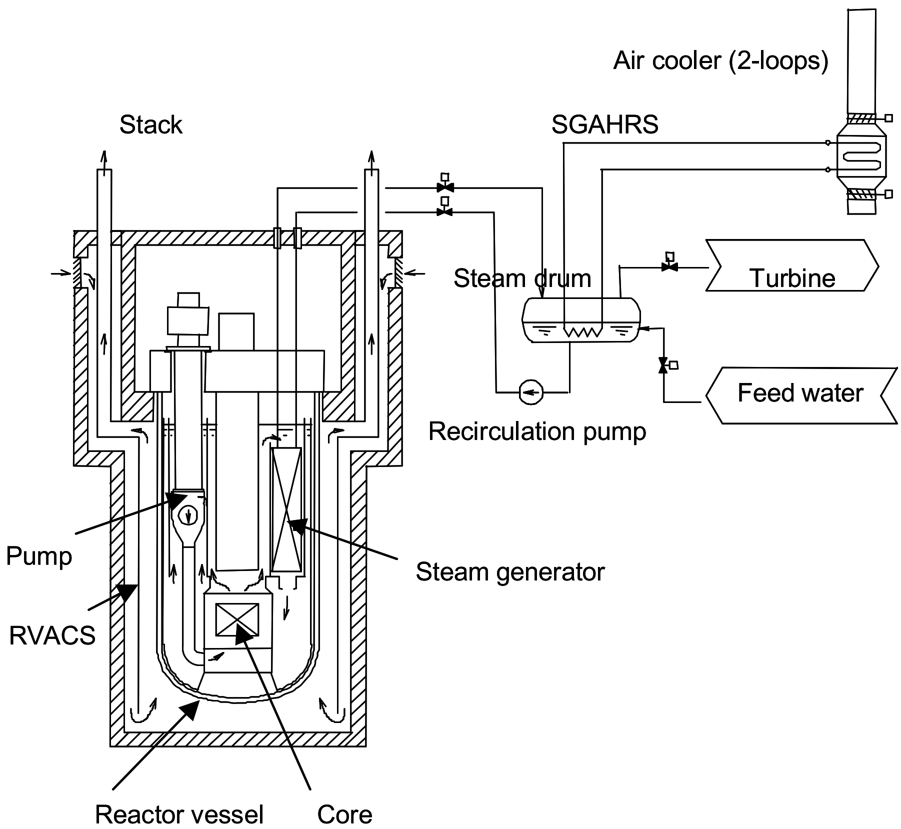


FIG. XXV-2. Simplified schematic diagram of LSPR.

In such core composition, the burn-up of fuel progresses from the outer core into the inner blanket region, which is beneficial for sustaining the reactivity for long-term burn-up with a small reactivity swing [XXV-6]. For the reactor lifetime of 12 years the expected burn-up reactivity swing is around 0.1% (see Fig. XXV-3) and, therefore, the possibility of a prompt criticality is eliminated.

In a long-life core it is not easy to achieve high power density but the LSPR design provides that of 60 MW/m^3, which is reasonably acceptable compared with about 100 MW/m^3 averaged over the core and blankets in typical fast reactors. Three control rods are placed within the core region. An option not to use these control rods for power regulation, except for the reactor start-up and shutdown, is being examined, to take advantage of very small excess reactivity.

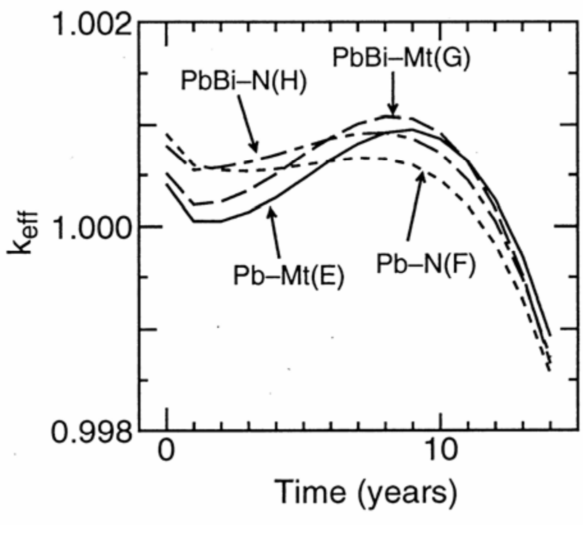
The changes in coolant void coefficient for the whole core during reactor lifetime are shown in Fig. XXV-4; the coefficients for lead coolant and metal fuel are also shown for comparison [XXV-6]. In all cases, the coolant void coefficient remains negative. It could be noted that it becomes positive, if the coolant is changed to sodium.

XXV-1.5. Outline of fuel cycle options

The fuel cycle concept of LSPR is shown in Fig. XXV-5.

All operations with fuel are performed in a centralized way within the nuclear park. A closed nuclear fuel cycle where separation and transmutation are performed to ensure an acceptable balance between the inflow and the outflow of radiotoxicity is applied, see Fig. XXV-5.

The LSPR is a factory fabricated and fuelled reactor designed for operation without on-site refuelling. Therefore, there are no operations with fuel on the utility site and during transportation.



Mt – metallic fuel; N – nitride fuel

FIG. XXV-3. Effective neutron multiplication factor versus fuel burn-up with Pb and Pb-Bi coolants being employed.

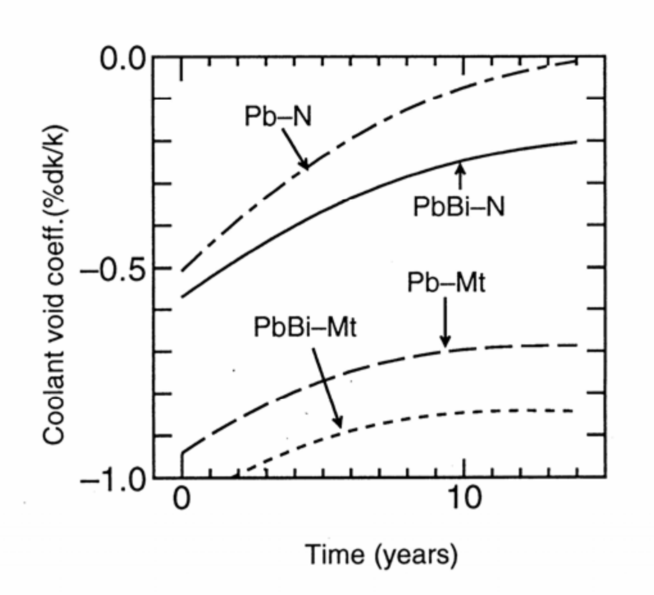
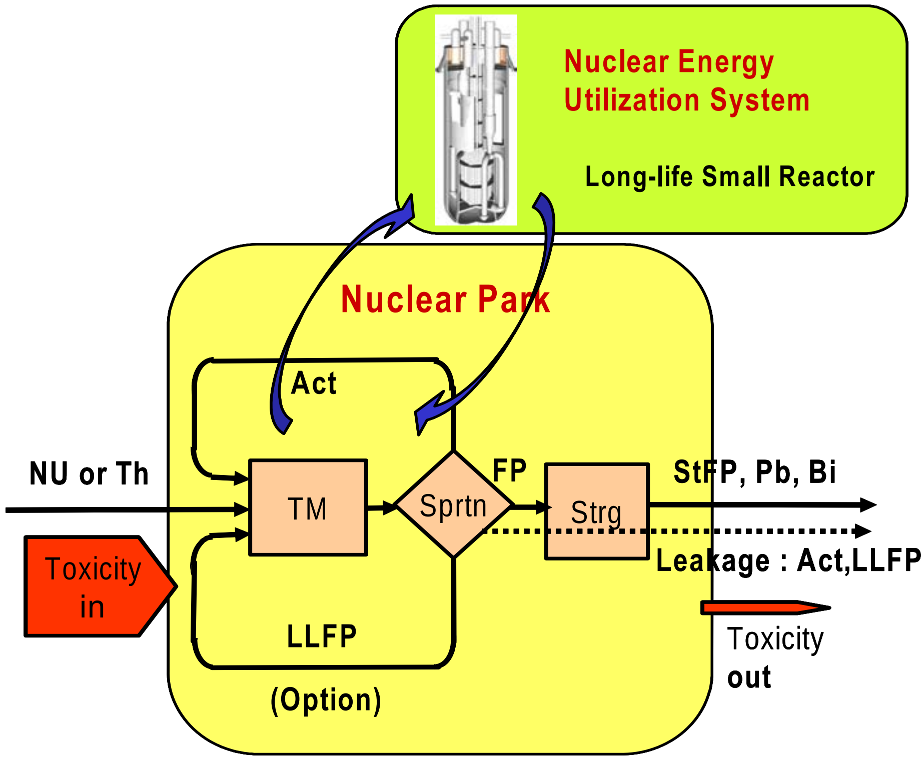


FIG. XXV-4. Coolant void coefficient versus fuel burn-up.



TM – transmutation	Sprtn – separation	Strg – storage
LLFP – long-lived fission products	StFP – stable fission products	Act – actinides
Toxicity in - inflow of radiotoxicity	Toxicity out - outflow of radiotoxicity	NU – natural uranium

FIG. XXV-5. Nuclear park concept.

XXV-1.6. Technical features and technological approaches that are definitive for LSPR performance in particular areas

XXV-1.6.1. Economics and maintainability

Small reactors do not benefit from economy of scale; however, several approaches in design and construction might contribute to improving their economy.

The LSPR can be produced complete in a factory and, if it is produced in series, that could considerably reduce the reactor cost. For a given rated power, the number of small reactors is larger than the number of large reactors and, therefore, more experience can be gained from the construction and operation of small reactors. The terms for licensing and construction could be shorter for small reactors and the amount of interest on the investment would be also smaller. Small reactors can be used to build modular plants of larger capacity with predicted good economic performance.

Long-life reactor core is also associated with an economic disadvantage related to a higher upfront premium or a higher interest rate on fuel cost. For a core with very long lifetime the corresponding effect in cost increase could be quite substantial. Other approaches to improve economic characteristics of small reactors should be used to compensate for this disadvantage.

Refuelling systems installed in conventional reactors are eliminated in the LSPR, contributing to a reduction of maintenance costs. High fuel burn-up reduces the fuel cycle cost and contributes to increased plant availability.

XXV-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The LSPR is a transportable factory fabricated and fuelled reactor, providing for no handling of spent fuel or any other waste on the utility site.

The LSPR has a good neutron economy, resulting in a breeding ratio of about unity and an enhanced transmutation capability. When operated in a closed fuel cycle with a nuclear energy park (see Fig. XXV-5), the total system is fissile self-sustainable and ensures that the radiotoxicity of disposed waste is comparable to that of the extracted natural uranium. Moreover, the transmutation requirements to a nuclear energy park can be reduced because all actinides could be effectively recycled in the LSPR.

Polonium produced by neutron absorption on bismuth will not be hazardous because the reactor vessel is sealed and not opened for regular fuel handling as in conventional reactors. If the reactor vessel is opened soon after the shutdown, the high radioactivity is no doubt very hazardous but may be beneficial from the viewpoint of proliferation resistance.

XXV-1.6.3. Safety and reliability

Safety concept and design philosophy; provisions for simplicity and robustness of the design

The philosophy behind the LSPR safety concept is maximum reliance on the inherent and passive safety features incorporated in the original design concept, and reliance of passive systems for decay heat removal. One of the important advantages of LBE cooled reactors is the possibility to reduce the number and complexity of engineered safety systems by effectively mitigating the impact of a reactor coolant leakage accident with a simple guard vessel.

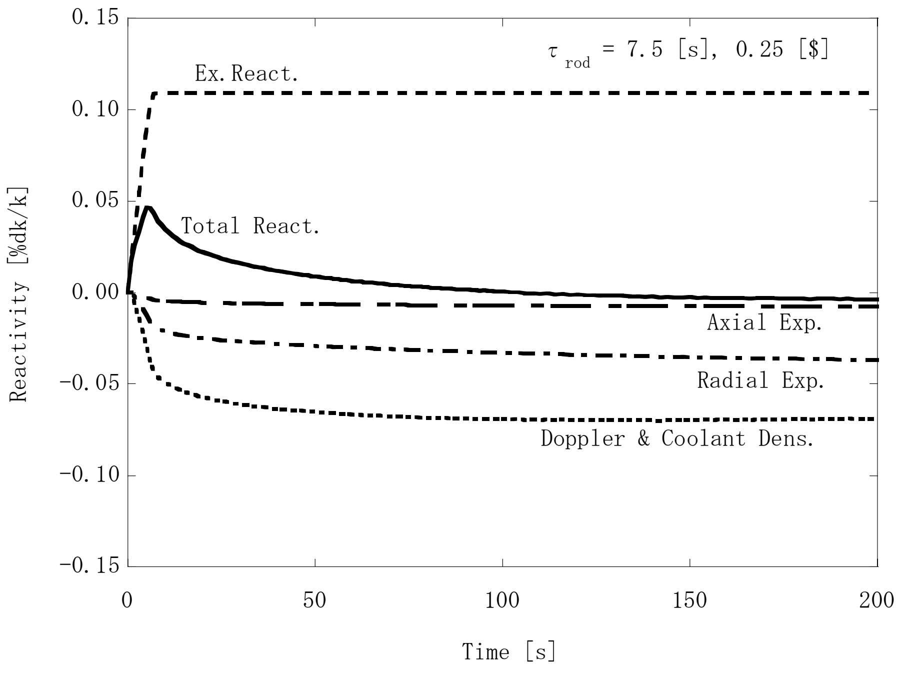
Active and passive systems and inherent safety features; design basis and beyond design basis accidents

In safety analysis of the LSPR, several uncontrolled transients and combinations thereof have been considered. The categorization of these transients into design basis and beyond design basis accidents has not been applied.

As a representative initiating event of an anticipated transient, the loss of external power is commonly postulated, in which a diesel generator is expected to start up and supply electricity for safety demands in a conventional design. Different from this, the LSPR incorporates a fully passive system of decay heat removal without diesel generators - the decay heat can be removed by steam generator auxiliary heat removal system (SGAHRS) through the SGs to the air coolers by natural circulation.

The transient overpower (TOP) due to a control rod withdrawal, the loss of primary flow (LOF) and the loss of heat sink (LOHS) due to a loss of the heat removal capability of the secondary system are commonly postulated as accident scenarios for power reactors. Even though the loss of external power is commonly superposed on these events, this does not lead to any serious problem if the reactor is safely tripped. Severe accidents, where the failure of a scram system is superposed on the abovementioned accidents, are surveyed below, for the LSPR. The analytical methods employed are described in [XXV-7].

In an uncontrolled transient overpower (UTOP), the power returns to a stable state without scram by virtue of a negative reactivity coefficient, since the maximum reactivity insertion is limited by 0.25 $ due to a very low burn-up reactivity swing. The changes of reactivity, normalized power and hot spot temperature in this scenario are shown in Figures XXV-6 to XXV-8. In all cases, maximum temperatures reached are much lower than the corresponding temperature limits.



Ex – external	Exp – expansion	Dens – density
React. – reactivity	0.25$ are inserted in 7.5 s	

FIG. XXV-6. Reactivity changes in UTOP.

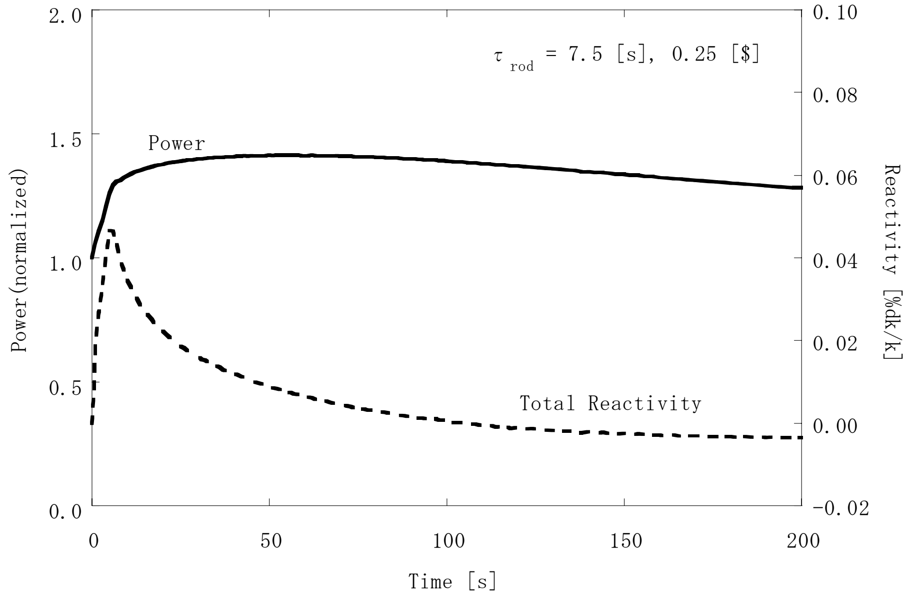


FIG. XXV-7. Normalized power and total reactivity changes in UTOP.

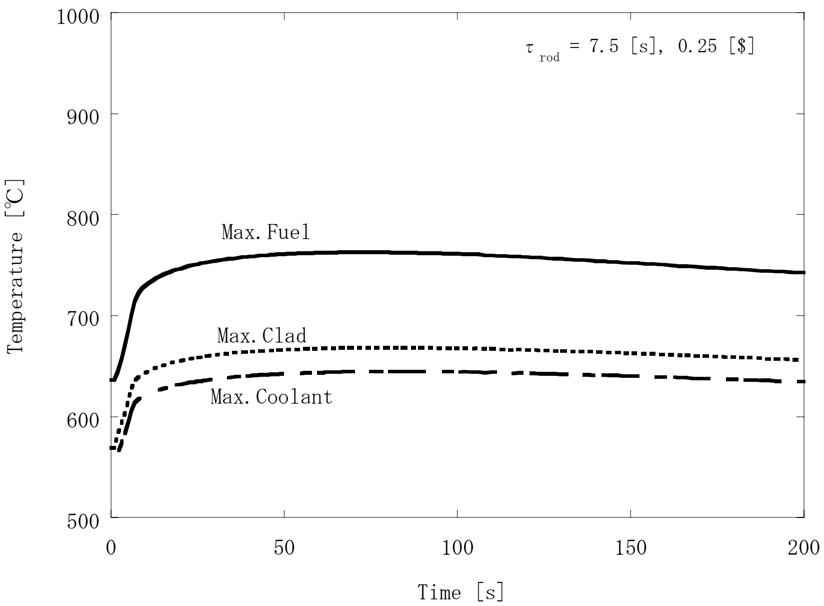


FIG. XXV-8. Hot spot temperature changes in UTOP.

In an uncontrolled loss of flow (ULOF), all primary pumps are postulated to stall without scram, and the total coolant mass flow rate along the core and SG changes as shown in Fig. XXV-9, where the coastdown half time of the primary pump is set to 6 sec. The changes of reactivity, normalized power and hot spot temperature for this scenario are shown in Figures XXV-10 to XXV-12. In all cases, maximum temperatures reached are much lower than the corresponding temperature limits.

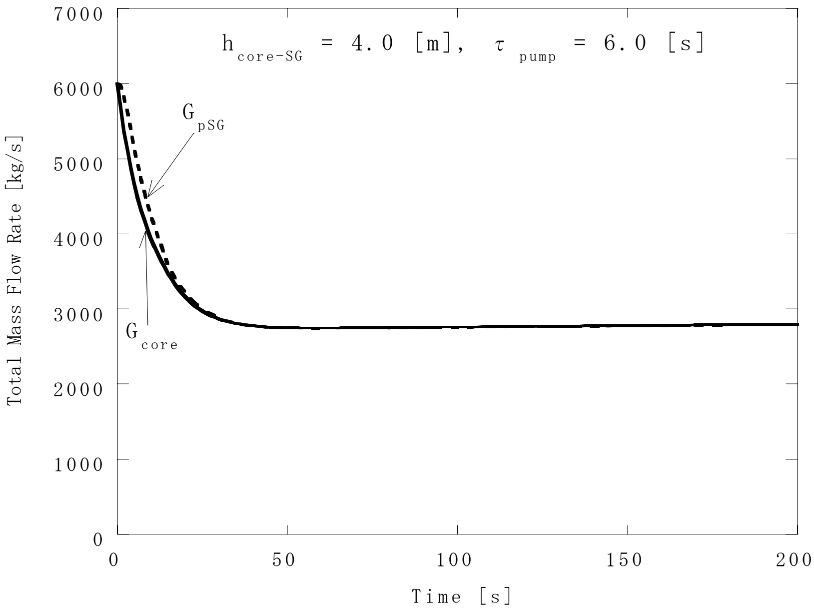
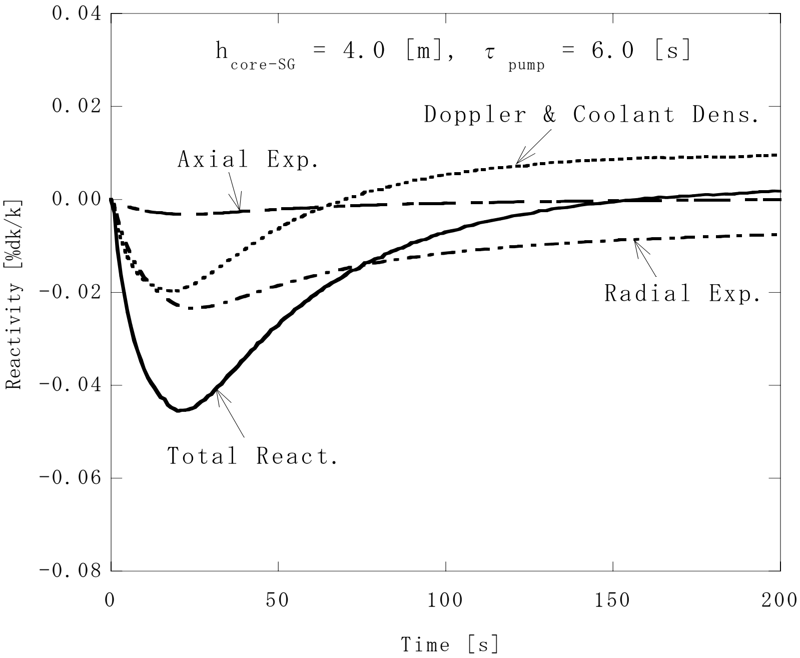


FIG. XXV-9. Change of total coolant mass flow rate along the core and SG in ULOF (thermal centre elevation difference between the core and the SG is 4.0 m).



Ex. – external; Exp. – expansion; Dens. – density; React. – reactivity

FIG. XXV-10. Reactivity change in ULOF.

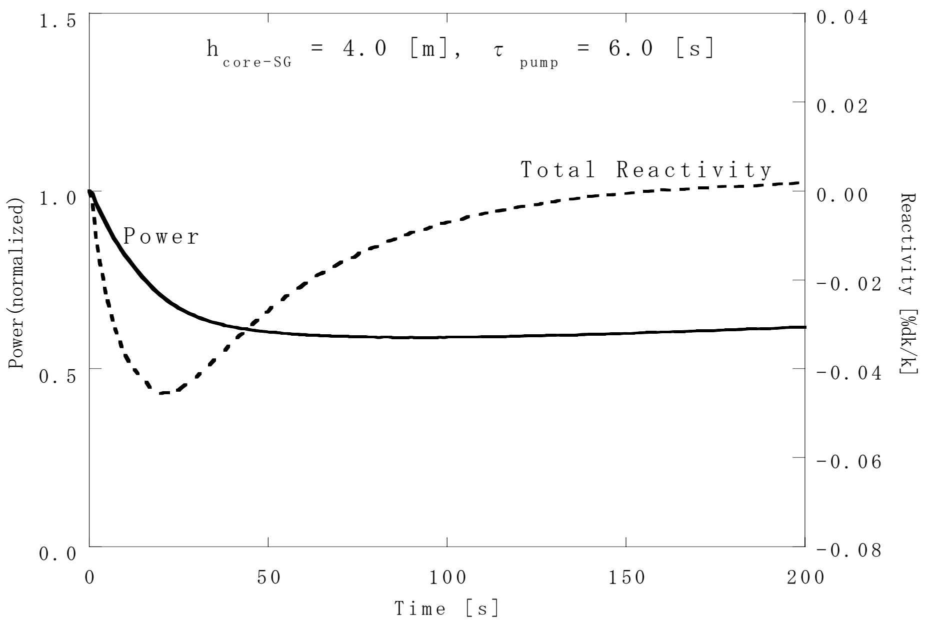


FIG. XXV-11. Normalized power and total reactivity change in ULOF.

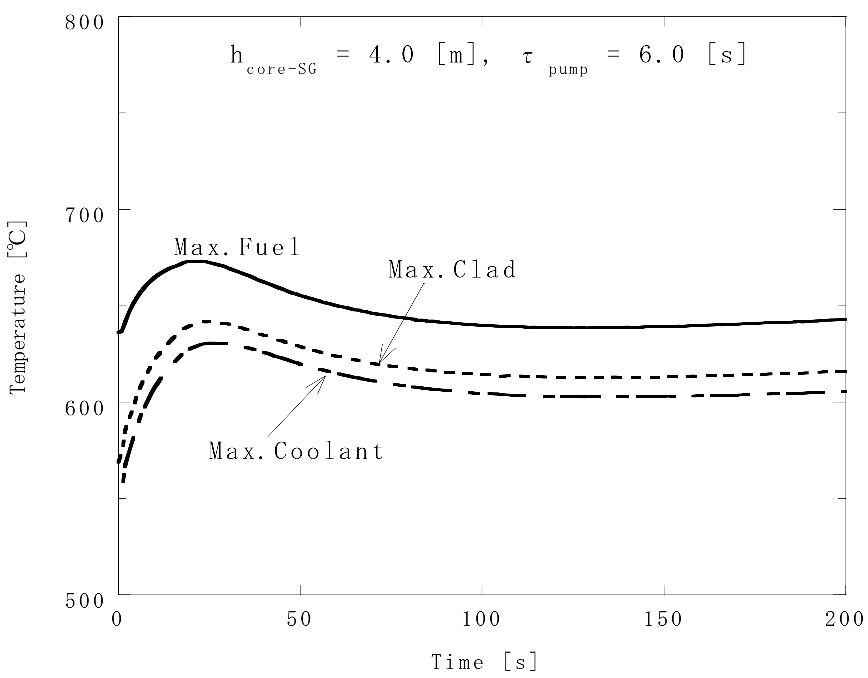
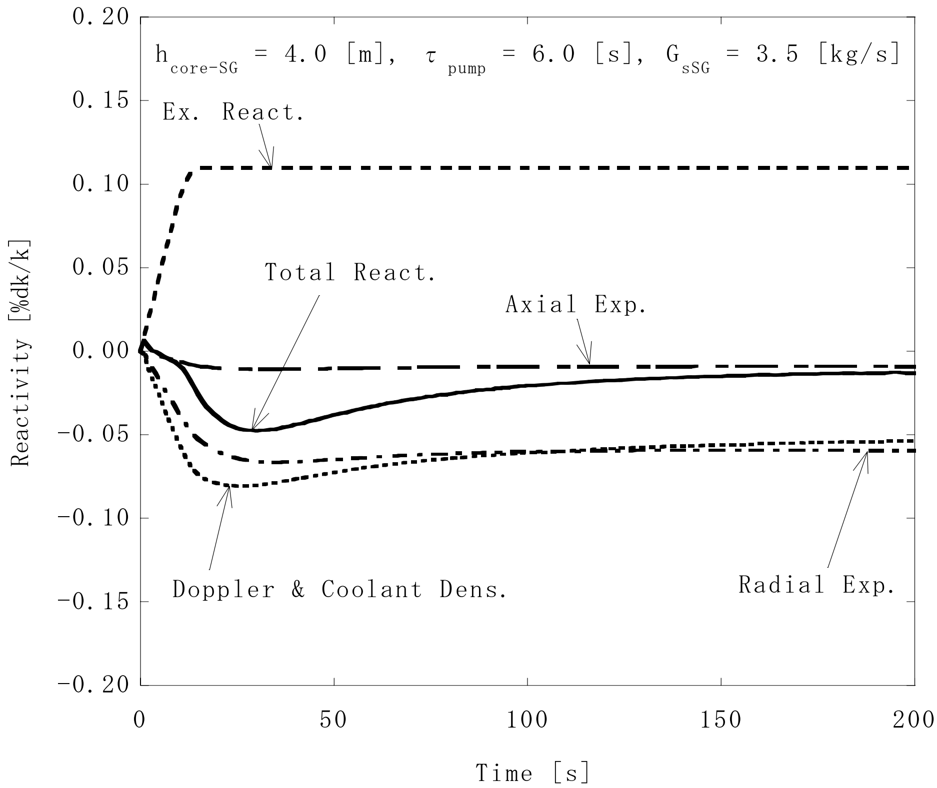


FIG. XXV-12. Hot spot temperature changes in ULOF.

The uncontrolled loss of heat sink also terminates safely if the SGAHRS holds the safety passive function. The reactor vessel auxiliary coolant system (RVACS), which removes heat through the reactor wall to the chimney, serves as a passive system to back up the function of the SGAHRS in accident conditions.

Since the results for each of the abovementioned accidents show considerable safety margins, investigation of a combined UTOP+ULOF+ULOHS accident is being performed, though it is the case not necessarily to be considered in conventional safety analysis. The changes in reactivity, normalized power and hot spot temperature in this scenario are shown in Figures XXV-13 to XXV-15. No fuel damage is anticipated though the cladding temperature is slightly over the safety temperature limit (700^{o}C) for a short period. It might be possible to restart the reactor even in these circumstances, if causes of the accidents can be removed.

The reactor performance can be improved by changing design parameters. The effects of changing the pump coastdown half-time and the thermal centres elevation difference on maximum cladding temperature are shown in Figures XXV-16 and XXV-17, respectively.



Ex. – external; Exp. – expansion, Dens. – density; React. – reactivity

FIG. XXV-13. Reactivity change in UTOP+ULOF+ULOHS.

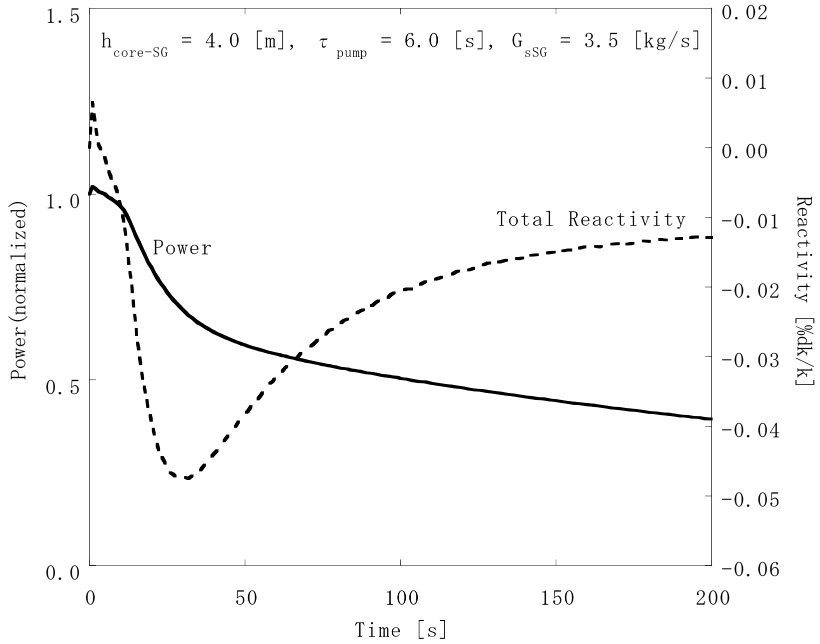


FIG. XXV-14. Normalized power and total reactivity change in UTOP+ULOF+ULOHS.

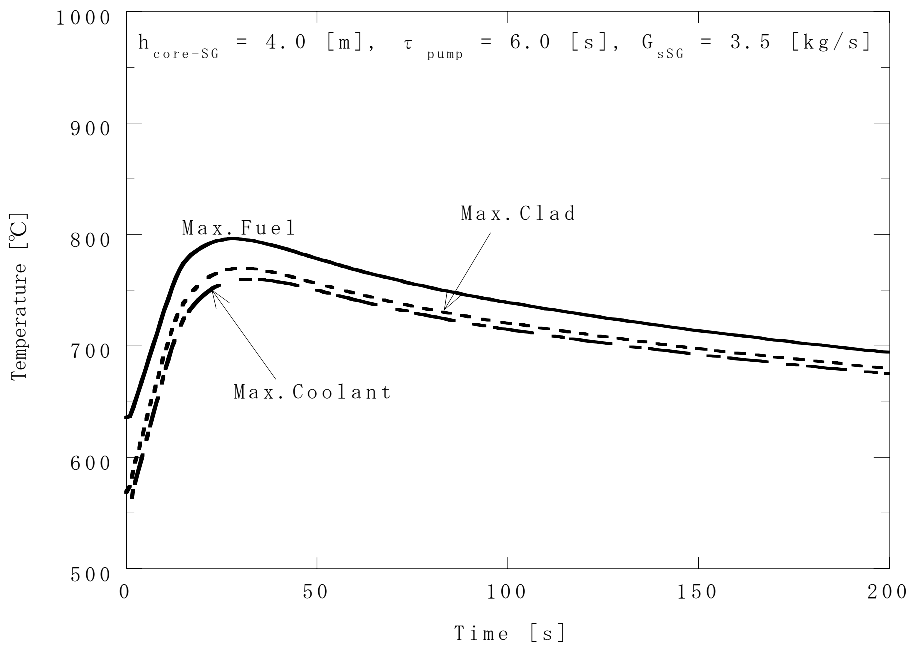


FIG. XXV-15. Hot spot temperature changes in UTOP+ULOF+ULOHS.

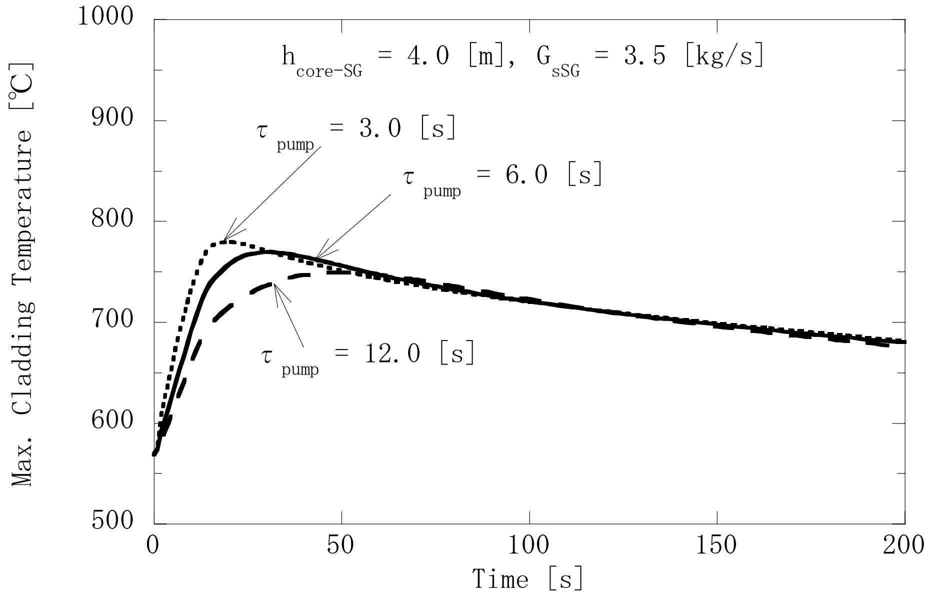


FIG. XXV-16. Maximum cladding temperature change in UTOP+ULOF+ULOHS for different pump coastdown half-times.

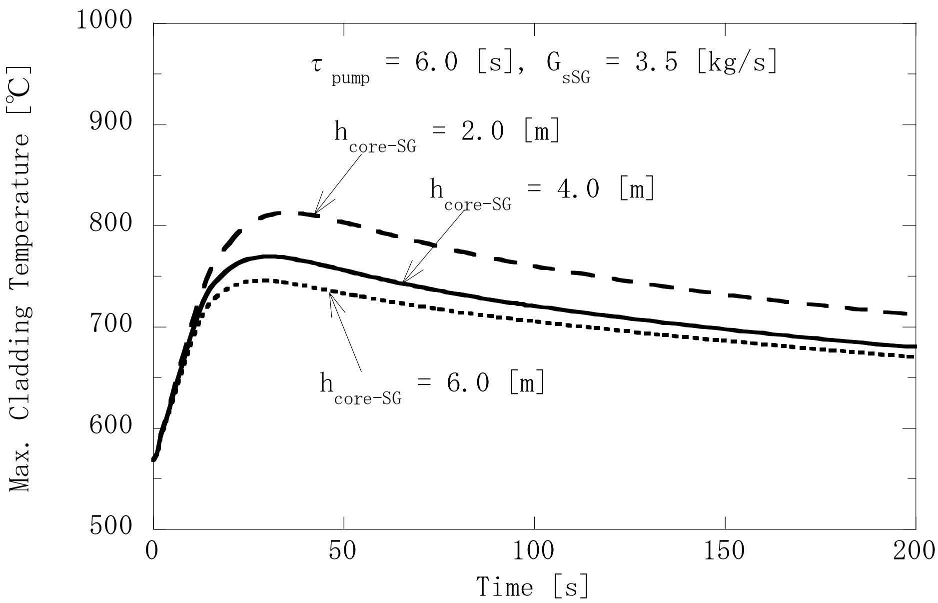


FIG. XXV-17. Maximum cladding temperature change in UTOP+ULOF+ULOHS for different elevation difference of thermal centres.

A steam generator (SG) tube rupture accident, which brings water of the steam generator tubes into the LBE coolant, is one of the most serious postulated events. The high pressure steam ejected in the reactor coolant is relieved through relief valves to a steam relief tank located above the reactor vessel, keeping the pressure in the reactor vessel below a specified level.

The resulting impact on the hydrodynamic behaviour of the primary coolant and the behaviour of steam bubbles due to a steam generator tube rupture as well as the possibility of production of the oxides of lead and bismuth must be fully investigated.

XXV-1.6.4. Proliferation resistance

The LSPR is a factory fabricated and fuelled reactor designed for operation without on-site refuelling. During the whole period of reactor operation and transportation to and from the site, the reactor vessel is always closed (sealed) and the fuel is confined in the vessel. Because of very small operation reactivity margin in the core, the fuel inside the reactor vessel cannot be removed and fertile materials cannot be inserted in the reactor to produce fissile materials. No refuelling equipment is provided in the reactor or at the site during the whole period of reactor operation, including its transportation to and from the site.

XXV-1.6.5. Technical features and technological approaches used to facilitate physical protection of LSPR

The LSPR strongly relies on inherent and passive safety features to achieve a high level of safety in a variety of uncontrolled accidents or combinations thereof, which secures an enhanced level of protection against human actions of malevolent character.

Even the reactor is highjacked, it would not be easy to open the reactor vessel for a certain period after shutdown of the reactor because of the high polonium activity in the coolant.

XXV-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of LSPR

As it was mentioned before, the LSPR may be a good choice for developing countries with small electricity grids and insufficient infrastructure; in particular, through operation without on-site refuelling, it could facilitate making a decision to forego the development of the indigenous fuel cycle. To realize these potential benefits of the LSPR, an infrastructure framework for nuclear power plant leasing needs to be created.

In the 21st century, global warming caused by the carbon dioxide emissions may become an urgent problem and the carbon dioxide emissions from developing countries would then become important.

More countries joining the Kyoto protocol and other international conventions targeted at greenhouse effect prevention would objectively facilitate the progress of nuclear power and, specifically, increase the deployment opportunities for small reactors without on-site refuelling, such as the LSPR.

XXV-1.8. List of enabling technologies relevant to LSPR and status of their development

The enabling technologies, relevant for the LSPR, that require further development are the following:

(1) Structural materials compatible with lead-bismuth coolant; the major trends of further research and development (R&D) are:

- Development of instrumentation and control techniques and equipment to effectively control oxygen concentration in the LBE coolant;
- Development of new material to increase coolant output temperature and velocity;

(2) Polonium treatment technology;

(3) Countermeasures for accidents with a SG tube break;

(4) Cost reduction methods;

(5) Design approaches to reduce in-vessel coolant inventory (as a anti-seismic design measure).

Based on the LSPR concept fundamentals, several concepts of innovative small LBE cooled reactors are under development in RLNR TITech, incorporating:

(a) A lift-up pump concept, described in [XXV-8];

(b) A "Constant Axial shape of neutron flux, Nuclide densities, and power shape During Life of Energy producing reactor" (CANDLE) burn-up concept, described in [XXV-9, XXV-10].

XXV-1.9. Status of R&D and planned schedule

In Japan, the R&D for technology development in areas specified in section XXV-1.8 are underway in the Tokyo Institute of Technology (TITech), being funded by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) in Japan.

Sufficient data to support implementation of the LBE coolant technology is expected to be available within the next 10 years and, then, another 10 years may be necessary to design and construct a prototype reactor.

The presented conceptual design study of the LSPR has been carried out conservatively and the primary coolant velocity was selected low compared with the other designs, in consideration of the ambiguity of material corrosion data with respect to the LBE coolant. The height of the reactor vessel was chosen to be high enough to give a sufficient natural circulation head with ambient margins.

One option of further R&D option might be to consider mounting a reactor compartment on a barge, which might facilitate installation and dismantling operations at a site.

Alternatives to the present centrifugal mechanical pumps, such as natural circulation without pumps, or application of the lift-up pumps providing the introduction of gas bubbles into the coolant to increase the buoyancy force, would be further investigated in the future. The present development, however, makes an emphasis on a simple and feasible reactor concept with the requested functions being performed only with the use of conventional and reliable devices.

The plans for future R&D, targeted at further improvement of the LSPR safety and economy, include studies of the core design incorporating the CANDLE burn-up concept [XXV-9, XXV-10], simplification of passive decay heat removal systems, identification of measures to cope with a steam generator tube rupture, and development of simplified maintenance techniques for in-vessel devices.

XXV-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

There is no experience in commissioning and operation of small lead or lead-bismuth cooled reactors with long-life cores in civil nuclear power and, therefore, a prototype plant would be required to test and demonstrate the innovative technologies of the LSPR outlined in sections XXV-1.8 and XXV-1.9.

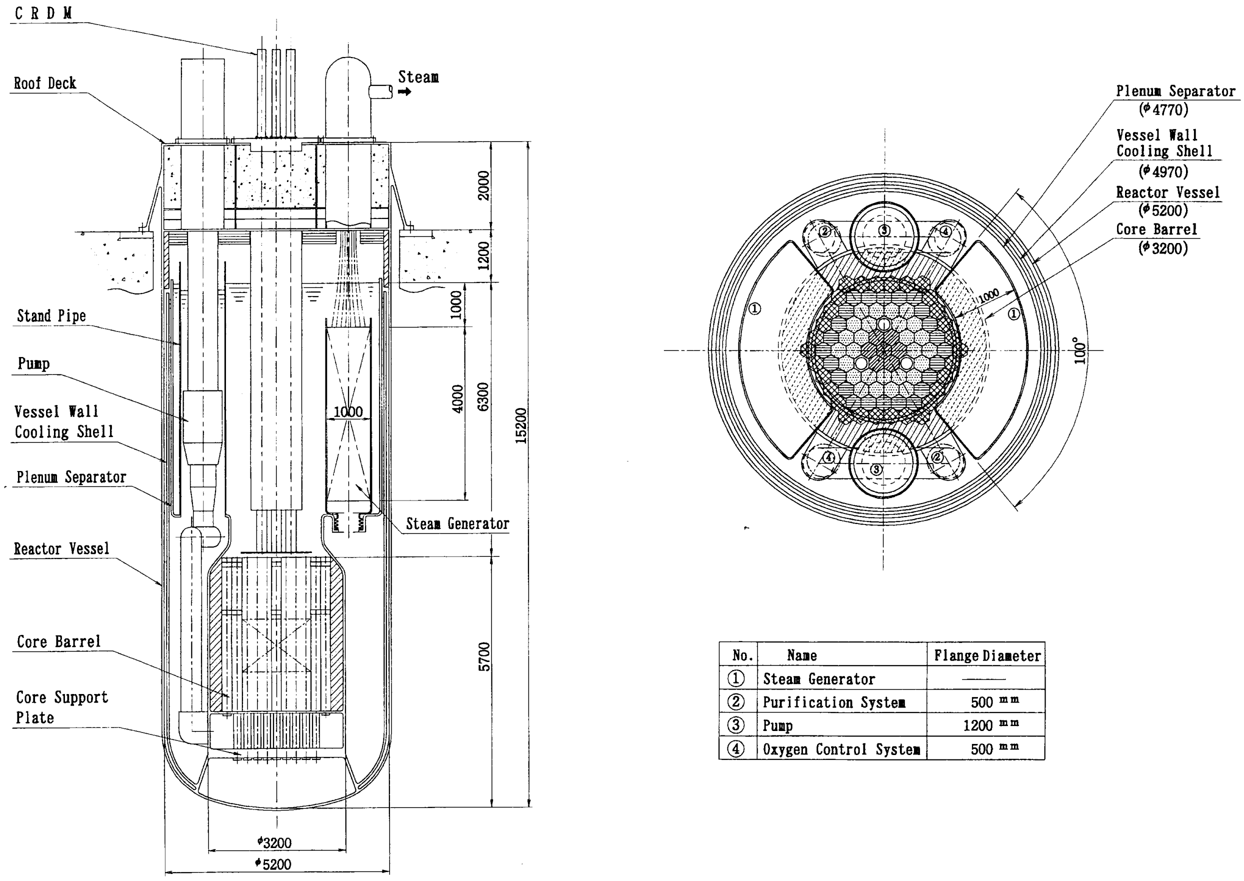
XXV-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

Several similar design studies for small lead or lead-bismuth cooled reactors with long-life cores are being performed at Experimental Design Organization (EDO) "Gidropress" and Institute of Physics and Power Engineering (IPPE) in the Russian Federation [XXV-11 to XXV-13]; at the Bandung Institute of Technology (ITB) in Indonesia [XXV-14]; at the University of California in Berkeley and Argonne National Laboratory (ANL) in the USA [XXV-15, XXV-16]; and at the TITech and JNC in Japan [XXV-8].

XXV-2. Design description and data for LSPR

XXV-2.1. Description of the nuclear systems

A vertical and a horizontal cross-section of the LSPR core arrangement is shown in Fig. XXV-18.



(a) Vertical cross section (b) Horizontal cross section

FIG. XXV-18. LSPR core arrangement in reactor vessel.

Summary description of the core and fuel design and primary circuit design is provided in section XXV-1.4; summary of the specifications is in Table XXV-1; and the outline of heat removal paths is given in section XXV-1.6.3. No further details were provided.

XXV-2.2. Description of the turbine generator plant and systems

No information was provided.

XXV-2.3. Systems for non-electric applications

No information was provided

XXV-2.4. Plant layout

The plant layout of a LSPR plant is shown in Figures XXV-19 and XXV-20. The conventional fuel handling system is not available for a long life core. However, maintenance handling machines and maintenance spaces are accommodated and the pullout space necessary for mechanical pump impellers and purifying units is provided. When the reactor vessel lifetime expires, the current assumption is to comply with the need to exchange the reactor vessel with a new one.

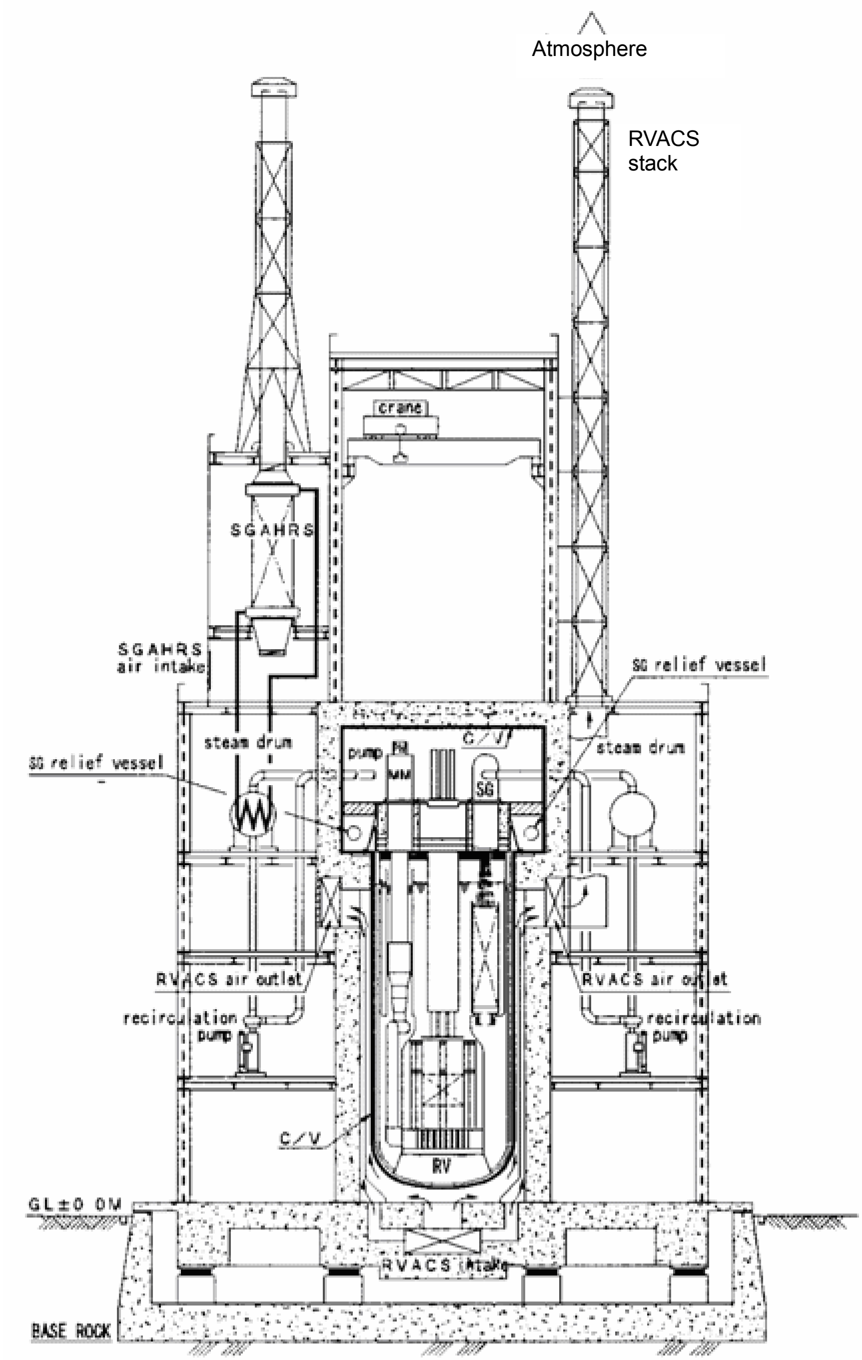


FIG. XXV-19. LSPR plant layout (vertical section).

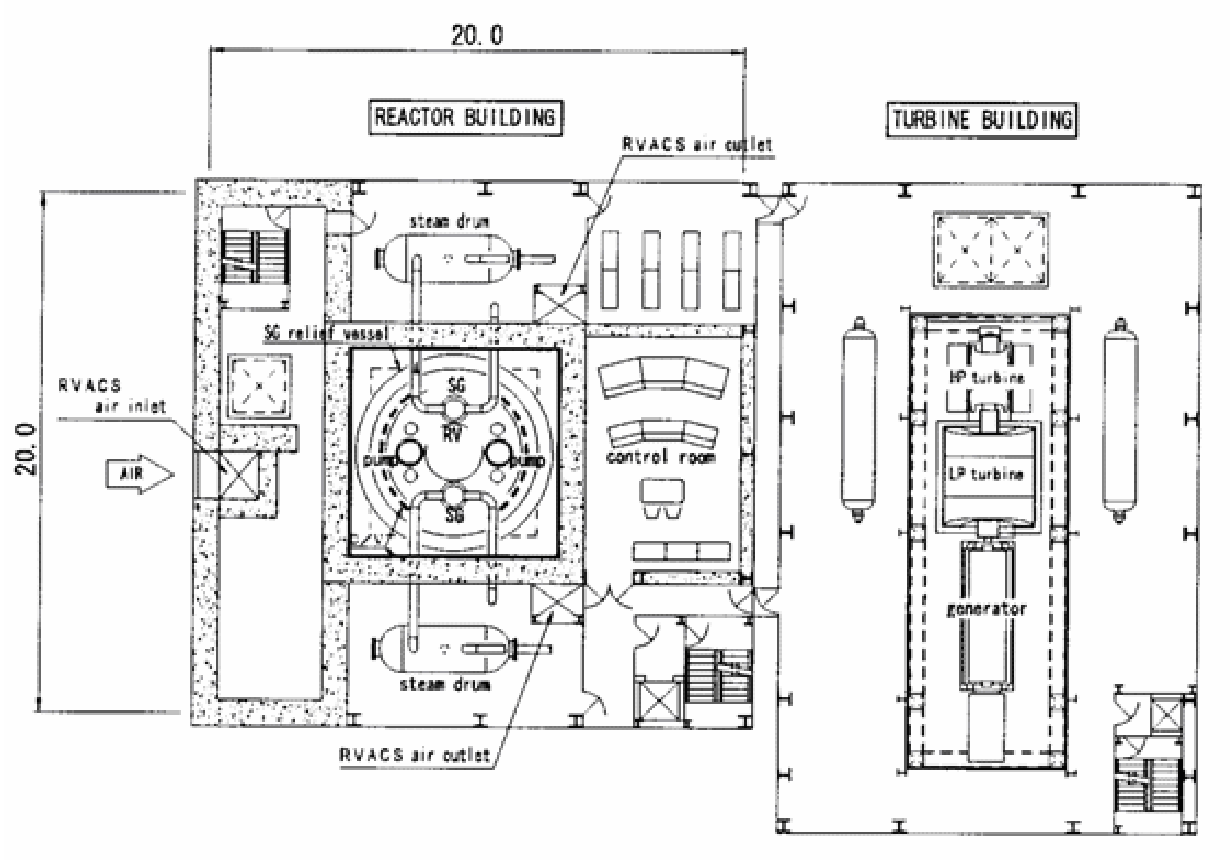


FIG. XXV-20. LSPR plant layout (horizontal section).

REFERENCES

[XXV-1] SEKIMOTO, H. (editor), Potential of small nuclear reactors for future clean and safe energy sources, Elsevier, Amsterdam, The Netherlands (1992).

[XXV-2] GROMOV, B.F., et al., Use of lead-bismuth coolant in nuclear reactors and accelerator-driven systems, Nuclear Engineering and Design, **173**, pp. 207–217 (1997).

[XXV-3] ZAKI, S., SEKIMOTO, H., A Concept of long-life small safe reactor, Potential of Small Nuclear Reactors for Future Clean and Safe Energy Sources, Elsevier, Amsterdam, The Netherlands, pp. 225–234 (1992).

[XXV-4] SEKIMOTO, H., Several features and applications of small reactors, potential of small nuclear reactors for future clean and safe energy sources, Elsevier, Amsterdam, The Netherlands, pp. 23–32 (1992).

[XXV-5] BUONGIORNO, J., TODREAS, N.E., KAZIMI, M., Thermal design of lead-bismuth cooled reactors for actinide burning and power production, MIT-ANP-TR-066, Massachusetts Institute of Technology, Cambridge Massachusetts (1999).

[XXV-6] SEKIMOTO, H., ZAKI S., Design study of lead- and lead-bismuth-cooled small long-life nuclear power reactors using metallic and nitride fuel, Nuclear Technologies, **109**, pp. 307–313 (1995).

[XXV-7] ZAKI, S., SEKIMOTO, H., Accident Analysis of lead or lead-bismuth cooled small safe long-life fast reactor using metallic and nitride fuel, Nuclear Engineering and Design, **162**, pp. 205–222 (1996).

[XXV-8] TAKAHASHI, M., et al., Design and experimental study for development of Pb-Bi cooled direct contact boiling water small fast reactor (PBWFR), ICAPP ’04 (Proc. Int. Congress, Pittsburgh, PA, USA, June 2004).

[XXV-9] SEKIMOTO, H., RYU, K., YOSHIMURA, Y., CANDLE: The New Burnup Strategy, Nuclear Science and Engineering, **139** vol. 3, pp. 306-317 (2001).

[XXV-10] SEKIMOTO, H., TANAKA, T., Application of CANDLE Burnup Strategy to Small Reactors, Trans. American Nuclear Society, **87** (CD), (2002).

[XXV-11] ZRODNIKOV A.V., et al., NPP on the Basis of SVBR-75/100 reactor modules, Atomnaya Energia, **91**, vol. 6 (2001).

[XXV-12] DRAGUNOV, YU.G., et al., Project of SVBR-75/100 reactor plant with improved safety for nuclear sources of small and medium power, 5th International Conference on Nuclear Option in Countries with Small and Medium Electricity Grids (paper presented at Int. Conf., Dubrovnik, Croatia, May 2004).

[XXV-13] TOSHINSKY, V., HAYASHI, H., Feasibility study on small long-life Pb-Bi cooled reactor with capability of load following by flow rate adjustment, PHYSOE2002 (Paper presented at Conf. on the New Frontiers of Nuclear Technology, Seoul, Korea, 2002).

[XXV-14] ZAKI SU’UD, Comparative study on safety performance of nitride fueled Lead-Bismuth Cooled Fast Reactor with Various Power Level”, Progress of Nuclear Energy, Vol.**32**, No. 3/4, pp. 571~577 (1998).

[XXV-15] GREENSPAN, E., et al., The encapsulated nuclear heat source reactor – A Generation IV reactor, Proc. of GLOBAL-2001, Paris, France, ANS/ENS.

[XXV-16] SIENICKI, J.J., SPENCER, B.W., Power optimization in the Star-LM modular natural convection reactor system, ICONE-10 (Proc. 10th Int. Conf. on Nuclear Engineering, Arlington, Virginia, USA, April 2002) ASME.

SMALL AND VERY SMALL LEAD-BISMUTH COOLED NUCLEAR POWER REACTORS WITHOUT ON-SITE REFUELLING (SPINNOR / VSPINNOR)

Bandung Institute of Technology (ITB),
Indonesia

XXVI-1. General information, technical features and operating characteristics

XXVI-1.1. Introduction

The SPINNOR (Small Power Reactor, Indonesia, No On-site Refuelling) and the VSPINNOR (Very Small Power Reactor, Indonesia, No On-site Refuelling) are concepts of small lead-bismuth cooled nuclear power reactors with fast neutron spectrum that could be operated for more than 15 years without on-site refuelling. They are based on the concept of a long-life core reactor developed in Indonesia since early 1990s in collaboration with the Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology (RLNR TITech). The reactor cores are designed to have near zero (less then one effective delayed neutron fraction) burn-up reactivity swing during the whole cycle of their operation to avoid a possibility of prompt criticality accident. The basic design concept is that central region of the reactor core is filled with fertile (blanket) material. During the reactor operation fissile material accumulates in this central region, which helps to compensate fissile material loss in the peripheral core region and also contributes to negative coolant loss reactivity effect. A concept of high fuel volume fraction in the core is applied to achieve smaller size of a critical reactor.

XXVI-1.2. Applications

The SPINNOR and VSPINNOR are mainly developed to meet the electrical and other energy demand in Indonesia, especially in the areas outside the Java-Bali Islands that are characterized by the absence of well-established network of electricity grids and relatively low local energy demand (only a few tens of MW(e)). In general, any relatively small island or other remote area is also favourable for the application of reactors of this type.

In addition to electricity generation, the SPINNOR and VSPINNOR could also be used for potable water production through seawater desalination or could combine these two applications.

XXVI-1.3. Special features

The SPINNOR and VSPINNOR are assumed to be factory fabricated and tested, then transported to the site by a cargo ship or railroad, provide long-life core operation of more than 15 years without reloading and shuffling of fuel, and after that transported back to the factory. The reactor vessel is assumed to be sealed and should not be opened in any situation to reduce the risk of nuclear proliferation.

XXVI-1.4. Summary of major design and operating characteristics

Table XXVI-1 shows major design and operating characteristics of the reactors of three types: SPINNOR A (20 MW(e)), SPINNOR B (10 MW(e)) and VSPINNOR (~6 MW(e)); the reactor arrangement is presented in Fig. XXVI-1. The reactors use two circuit systems without intermediate heat transport system. Steam generator is located inside the reactor vessel; two dedicated pipes connect it with the power circuit for the feedwater supply and steam removal.

The plutonium from pressurized water reactor (PWR) spent fuel after 25 years of cooling is used as a make-up fuel for the SPINNOR/VSPINNOR. The consideration is that, if PWR spent fuel is used without cooling, then it contains a relatively high portion of ^{241}Pu, which decays to ^{241}Am. If the reactors are designed for the use of plutonium from PWR spent fuel without cooling, and there is a delay in the deployment or reprocessing, or temporary suspension of operation for a relatively long period (e.g. half year), then the decay of ^{241}Pu will significantly affect the neutronic balance in their cores. When plutonium from PWR spent fuel is used after long-time cooling, the fraction of ^{241}Pu is significantly reduced and the core design is less sensitive to the delays in reprocessing, suspensions in operation, etc.

TABLE XXVI-1. MAIN DESIGN AND OPERATING CHARACTERISTICS OF SPINNOR AND VSPINNOR

CHARACTERISTIC	VALUE/DESCRIPTION		
	SPINNOR A	SPINNOR B	VSPINNOR
Installed capacity	55 MW(th) / 20 MW(e)	27.5 MW(th)/ 10 MW(e)	17.5 MW(th)/ 6.25 MW(e)
Period of operation without reloading and shuffling of fuel	15 years	25 years	35 years
Mode of operation	Base load/load follow (selectable)		
Load factor	Not less than 95% *		
Summary of major design characteristics:			
- Type of fuel	UN-PuN**	UN-PuN**	UN-PuN**
- Fuel enrichment	10 – 12.5%	10 – 12.5%	10 – 12.5%
- Type of coolant/moderator	Pb-Bi eutectic	Pb-Bi eutectic	Pb-Bi eutectic
- Type of structural material	Stainless steel	Stainless steel	Stainless steel

* The load factor is enhanced by easy maintenance due to operation without on-site refuelling.
** Plutonium from PWR after 25 years of cooling is used as initial load.

The SPINNOR and VSPINNOR plants adopt pool type Pb-Bi cooled fast reactors without intermediate heat exchanger. Centrifugal pump is adopted for primary circulation, which is, together with steam generator, placed inside the reactor vessel. Decay heat is removed with the use of a passive reactor vessel auxiliary cooling system (RVACS), as shown in Fig. XXVI-1.

Rankine cycle is used for the production of electricity, and the balance of plant can be adjusted for a cogeneration mode with potable water production. The cross-sectional view of one-fourth part of the core (symmetric in radial and axial directions) is shown in Fig. XXVI-2. B1 and B2 are the blankets, C1 and C2 are the inner and outer cores, R is the reflector, and S is the shielding.

Neutron-physical characteristics

Neutron physical characteristics of the SPINNOR and VSPINNOR are presented in Table XXVI-2.

Simplified schematic diagram

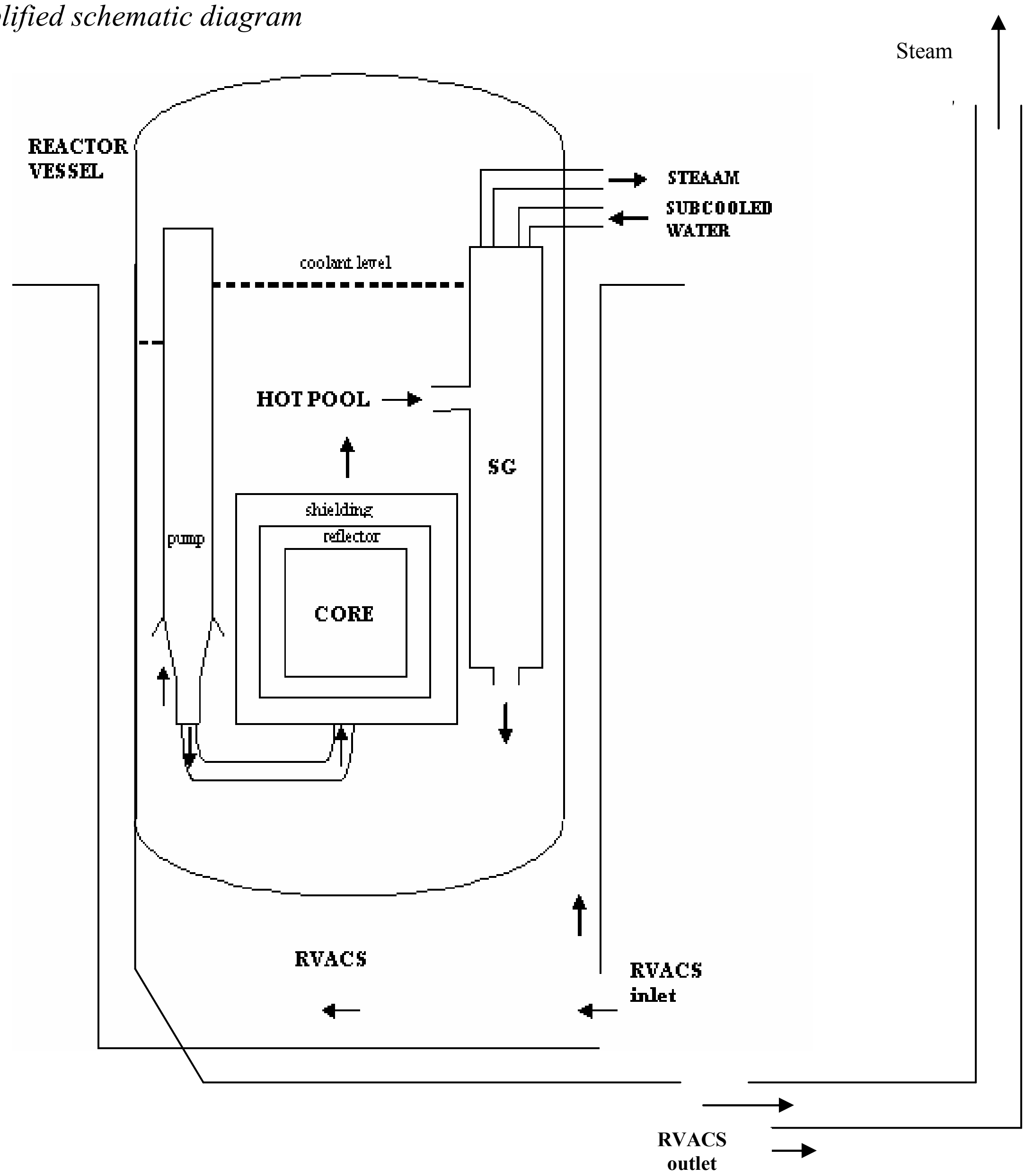


FIG. XXVI-1. Simplified schematic diagram of SPINNOR and VSPINNOR.

Reactivity control mechanism

Either control rods or operating temperature shift can be used for the reactor control. The burn-up reactivity swing during the whole period of operation is low, less than ~0.002 Δk/k for the SPINNOR and ~0.001Δκ/k for the VSPINNOR. Therefore, the operating temperature shift could be used to compensate it, especially for the VSPINNOR. The reactivity swing during reactor operation is minimized through the optimization of core configuration. In particular, an internal blanket is introduced to the central part of the core for this purpose.

Axial direction

S	S	S	S	S	S
R	R	R	R	R	S
C2	C2	C2	C2	R	S
C1	C1	C1	C2	R	S
B2	B2	C1	C2	R	S
B1	B2	C1	C2	R	S

Radial direction

FIG. XXVI-2. Core configuration.

Cycle type and thermodynamic efficiency

The SPINNOR and VSPINNOR are designed to operate within an indirect cycle without intermediate heat exchanger. Rankine cycle with superheated steam at about 7MPa is used. The thermodynamic efficiency is 35–37.5%.

Thermal-hydraulic characteristics

Major thermal-hydraulic characteristics of the SPINNOR and VSPINNOR are given in Table XXVI-3.

Discharge fuel burn-up

The average discharge burn-up of fuel for a ~ 6 MW(e) VSPINNOR and a 10 MW(e) SPINNOR is about 42.5 MW day/kg, while for a 20 MW(e) SPINNOR the corresponding value is about 54.5 MW day/kg.

TABLE XXVI-2. NEUTRON-PHYSICAL CHARACTERISTICS OF SPINNOR AND VSPINNOR

CHARACTERISTIC	VALUE/DESCRIPTION
a. Reactivity coefficients: - Doppler coefficient - Axial expansion coefficient - Radial expansion coefficient - Coolant density coefficient	 $-5 \sim -6\times10^{-6}$ $-1.5 \sim -1.8\times10^{-6}$ $-7.8 \sim -8.5\times10^{-6}$ $-1.4 \sim -1.65\times10^{-6}$
b. Burn-up reactivity swing	< 0.002 $\Delta k/k$
c. Void reactivity coefficient	~ -1.2 ~ - 2.0 %$\Delta k/k$*
d. Peaking factor	< 1.25
e. Delayed neutron fraction	0.003 ~ 0.004

*Calculated assuming that active part of the core and the axial reflector are simultaneously voided.

Fuel lifetime/period between refuellings

The period between refuellings is 32 years for a ~6 MW(e) VSPINNOR, 22 years for a 10 MW(e) SPINNOR, and 15 years for a 20 MW(e) SPINNOR. The refuelling is assumed to be conducted in a factory, not at the site.

Mass balances/flows of fuel and non-fuel materials

Mass balances of fuel and non-fuel materials for the SPINNOR and VSPINNOR are given in Table XXVI-4.

TABLE XXVI-3. MAIN THERMAL-HYDRAULIC CHARACTERISTICS OF SPINNOR AND VSPINNOR

CHARACTERISTIC	VALUE/DESCRIPTION
Circulation type	Forced circulation for normal operation and natural circulation for decay heat removal
Inlet coolant temperature	340 ~ 345°C
Average outlet coolant temperature	505 ~ 515°C
Flow rates	720 (A), 1125 (B), 2250 (C) kg/s
System pressure	2 ~ 3 bar
Coolant temperature limit	1600°C (boiling point)
Primary system temperature limit: Pb-Bi	~700°C *
Secondary system temperature limit: steam-water loop	~1200°C/800°C **

* Fuel element cladding degradation due to corrosion/erosion.
** Melting / restructuring temperature for SG tubes

TABLE XXVI-3. (continued)

CHARACTERISTIC	VALUE/DESCRIPTION
Fuel temperature limit	2500°C (melting point)
Maximum coolant temperature (normal operation)	~520°C
Maximum cladding temperature (normal operation)	~540°C
Maximum fuel temperature (normal operation)	~610°C

TABLE XXVI-4. ANNUAL FLOWS OF MATERIALS FOR SPINNOR AND VSPINNOR, KG/MW(e)

SPECIFIC MATERIAL CONSUMPTION	VALUE		
	SPINNOR A (20 MW(e))	SPINNOR B (10 MW(e))	VSPINNOR (6 MW(e))
Fuel: UN	18.4	20.9	24.1
Fuel: PuN	2.4	2.86	2.94
Stainless steel	2.9	3.67	3.7
Coolant	500	545	500

Design basis lifetime for reactor vessel and structures

The design basis lifetime for structural materials is 35 years for a 6 and a 10 MW(e) system, and about 20 years for a 20 MW(e) system. The design basis lifetime of reactor vessel it is estimated at 35 years.

Design and operating characteristics of systems for non-electrical applications

The system for seawater desalination, tentatively based on the reverse osmosis (RO) process, will be optimized using electric load change pattern.

Economics

The estimated capital cost is US$ 1500/kW(e) for the SPINNOR A of 20 MW(e); US$ 1750/kW(e) for the SPINNOR B of 10 MW(e); and US$ 2000/kW(e) for the VSPINNOR of 6.25 MW(e). It is projected that further reduction of these costs could be achieved through mass production of factory fabricated and fuelled reactor modules.

XXVI-1.5. Outline of fuel cycle options

The SPINNOR and VSPINNOR can operate both, in a once-through and in a closed fuel cycle. In case of a closed fuel cycle, all actinides could be used as fissile material in the core. After the fuel lifetime expires, the reactors are brought to the factory and the spent fuel is reprocessed to be reused in new plants.

XXVI-1.6. Technical features and technological approaches that are definitive for SPINNOR/VSPINNOR performance in particular areas

XXVI-1.6.1. Economics and maintainability

The reactor designs are optimized for use in remote areas with rather small population, with no access to centralized electricity grids, and with complicated transportation of fossil fuel. The operation without on-site refuelling could simplify operation and maintenance requirements for such reactors and also contribute to achieving higher load factor/availability.

XXVI-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The inherent safety features and passive operation capability of the SPINNOR and VSPINNOR are targeted to eliminate core meltdown and, therefore, to avoid adverse environmental impacts in accidents. Because their conversion ratio is about one, the fuel self-sustainable regime may be established, in which only fertile fuel material, e.g. depleted uranium, will be consumed to produce energy. If higher breeding ratios become necessary, they could be achieved just by placing an external blanket in the reflector position.

XXVI-1.6.3. Safety and reliability

The safety concept of the SPINNOR and VSPINNOR strongly relies on inherent safety features and passive operation capability provided by the design, with a focus on passive reactivity regulation and passive reactor shutdown. Long-life operation without refuelling and shuffling of fuel, along with nearly zero burn-up reactivity swing and a relatively high fraction of natural circulation in the core, secure an option for implementation of relatively simple safety systems.

During an unprotected transient overpower (UTOP), the core power will increase, causing the increase of coolant, cladding and fuel temperatures, and creating negative feedbacks from several components: Doppler, axial expansion, radial expansion, and coolant density. After the external reactivity is completely compensated by feedbacks, the reactor will reach new equilibrium state with higher power level as compared to the initial power before positive reactivity insertion. On the other hand, for an accident with the unprotected total loss of flow (ULOF), the loss of pumping power triggers lower coolant flow rate that consequently results in the increase of the coolant, cladding and fuel temperatures at the initial phase of the accident. In turn, this would cause negative reactivity feedbacks that would gradually suppress the reactor power.

The performance of the SPINNOR reactors under ULOF is shown in Fig. XXVI-3 to XXVI-6. Figure XXVI-3 indicates that following the loss of pumping power in the primary system, the flow rates in primary system and at the primary side of the steam generator (SG) decrease and progress toward the level of natural circulation. It causes the increase of temperatures as shown in Fig. 4, and results in the negative feedbacks including, in the order of importance, coolant density decrease, fuel axial expansion, Doppler effect, and core radial expansion, as shown in Fig. XXVI-5. These feedbacks assure the decrease of reactor power to match the new coolant flow rate, as shown in Fig. XXVI-6.

The reactor will then reach a new equilibrium state with the power level matching a natural circulation based flow rate, and the total reactivity feedback would become zero due to the corresponding increase of coolant temperature and decrease of fuel temperature in this new state.

Figures XXVI-7 to XXVI-12 provide more details on the results of UTOP and ULOF - UTOP simulation. Figures XXVI-7 to XXVI-9 show UTOP simulation results for a 20 MW(e) SPINNOR. In response to the ejection of a control rod, the reactor power increases causing the increase of the coolant, cladding and fuel pellet temperatures and resulting in negative reactivity feedbacks. The asymptotic state is a new, higher power level with higher temperatures of all components.

Figures XXVI-10 to XXVI-12 show the results of simulation performed for a combination of the ULOF and UTOP accidents (ULOF-UTOP) in a 10 MW(e) SPINNOR. The response includes the reduction of flow rate and the increase of reactor power at the initial stage, with the reactor power then going down toward the asymptotic level. The decrease of power is caused by negative reactivity feedbacks with major contributions coming from the core radial expansion and Doppler reactivity effect.

The inherent safety features incorporated through the reactor design optimization include:

- Reactivity feedback mechanism;
- Relatively high degree of natural circulation in the primary coolant system.

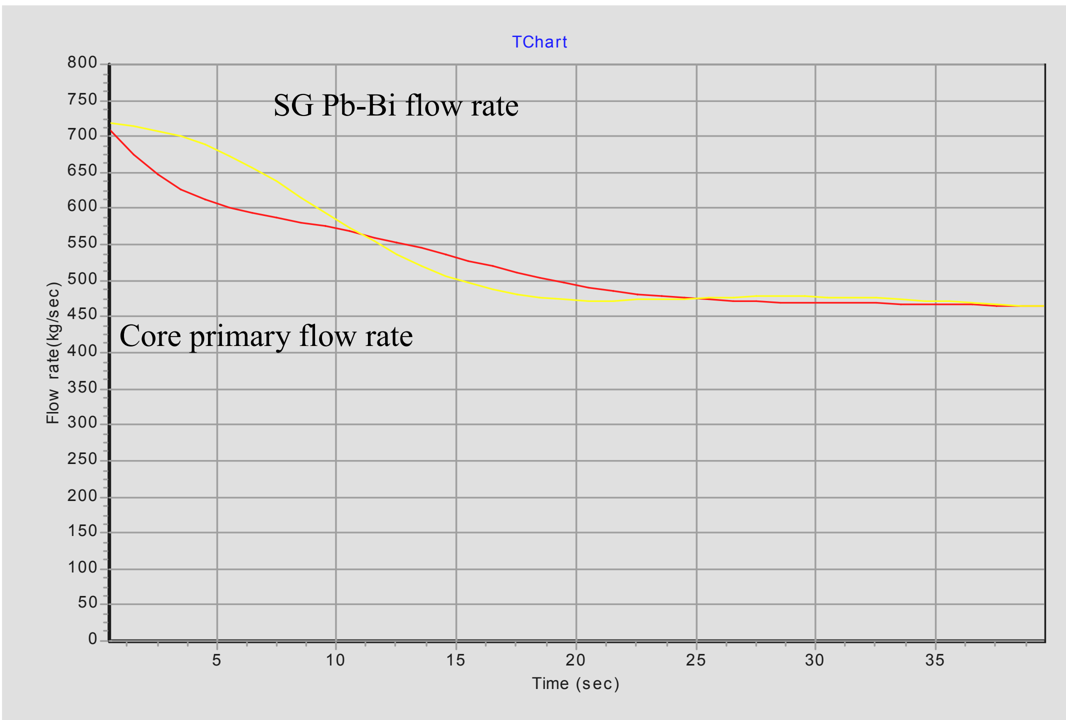


FIG. XXVI-3. Flow rate in the core and at the primary side of SG during ULOF accident in VSPINNOR.

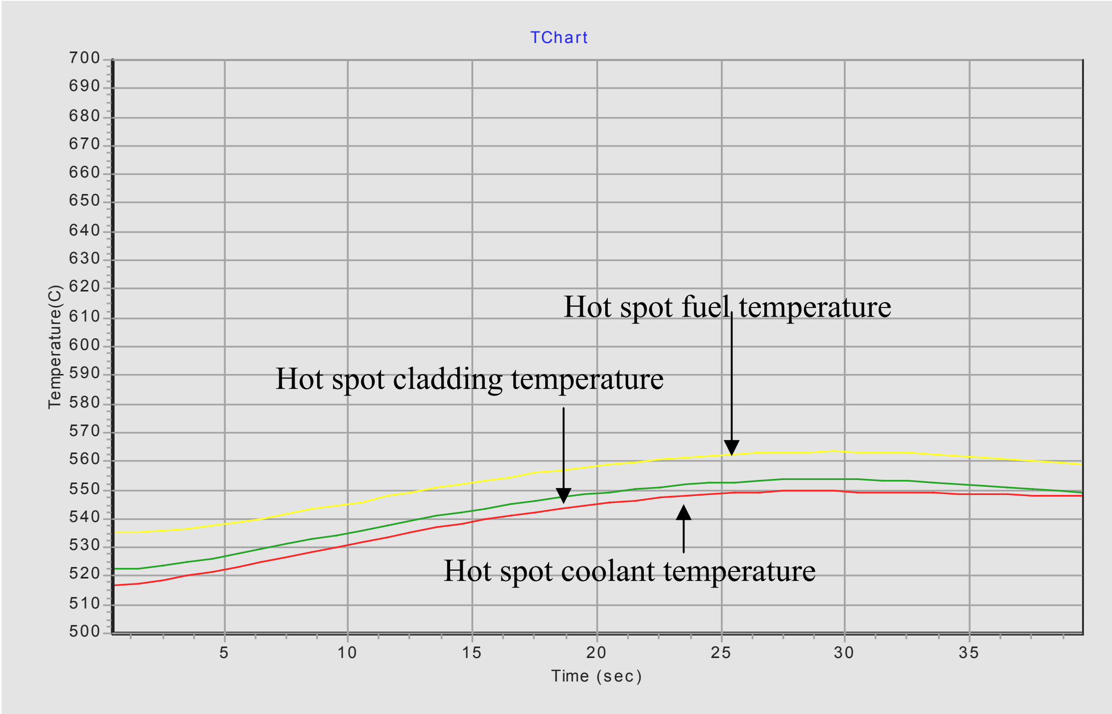


FIG. XXVI-4. Hot spot temperatures during ULOF accident in VSPINNOR.

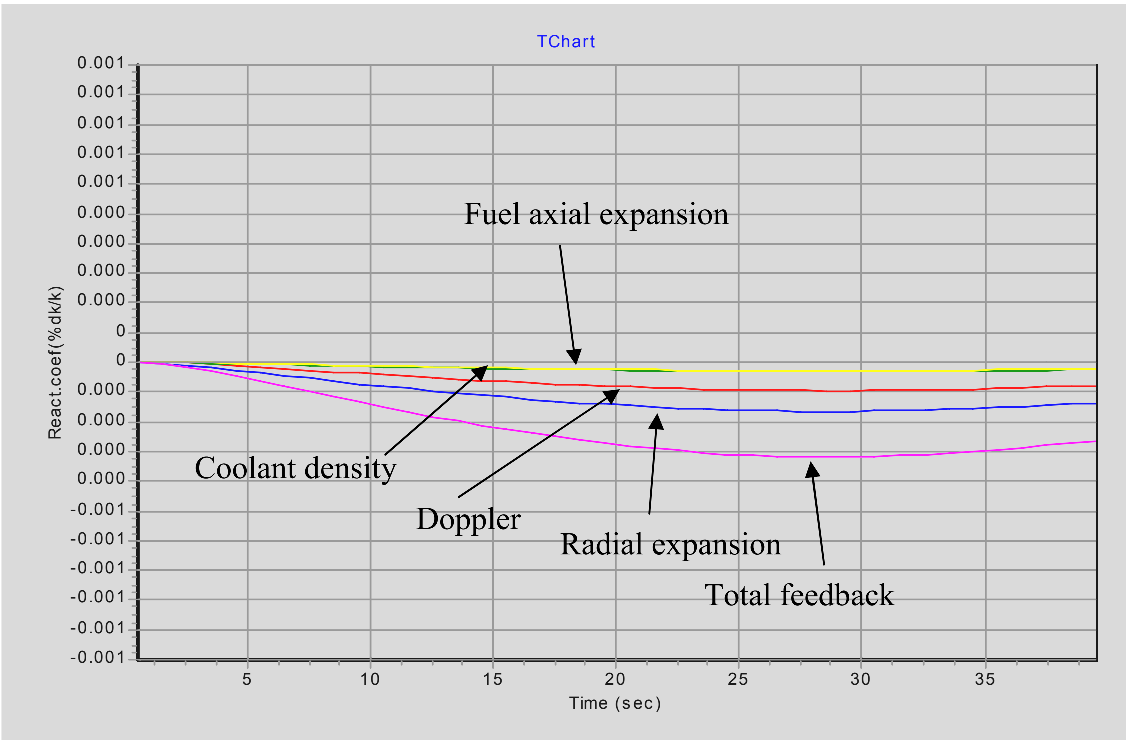


FIG. XXVI-5. Reactivity feedbacks during ULOF accident in VSPINNOR.

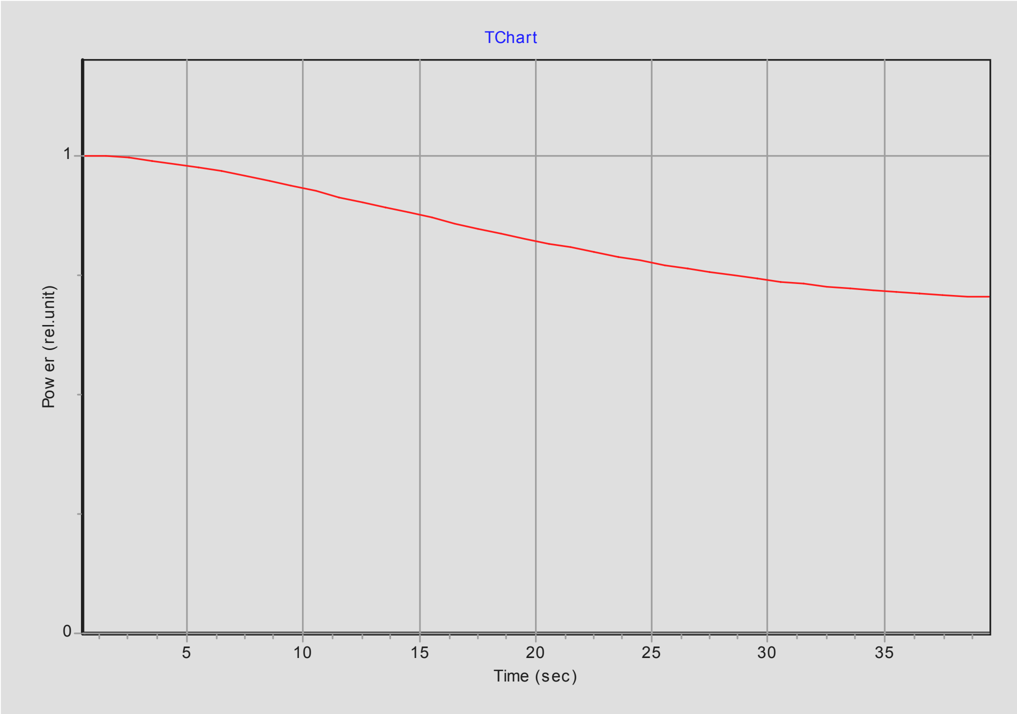


FIG. XXVI-6. Power changes during ULOF accident in VSPINNOR.

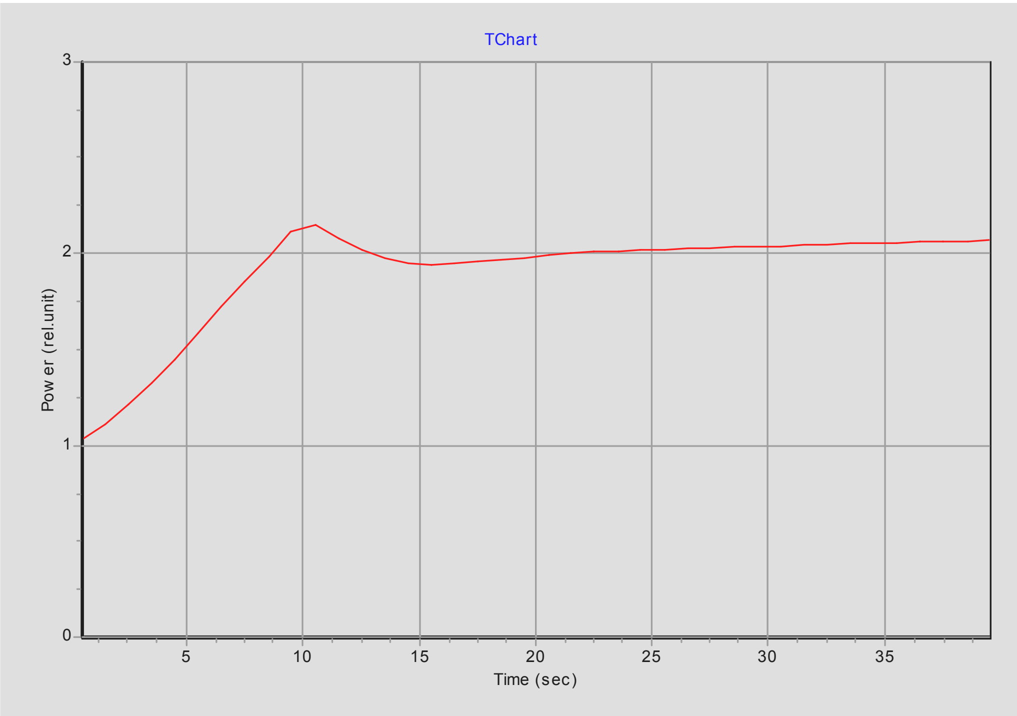


FIG. XXVI-7. Power changes during UTOP accident in a 20 MW(e) SPINNOR.

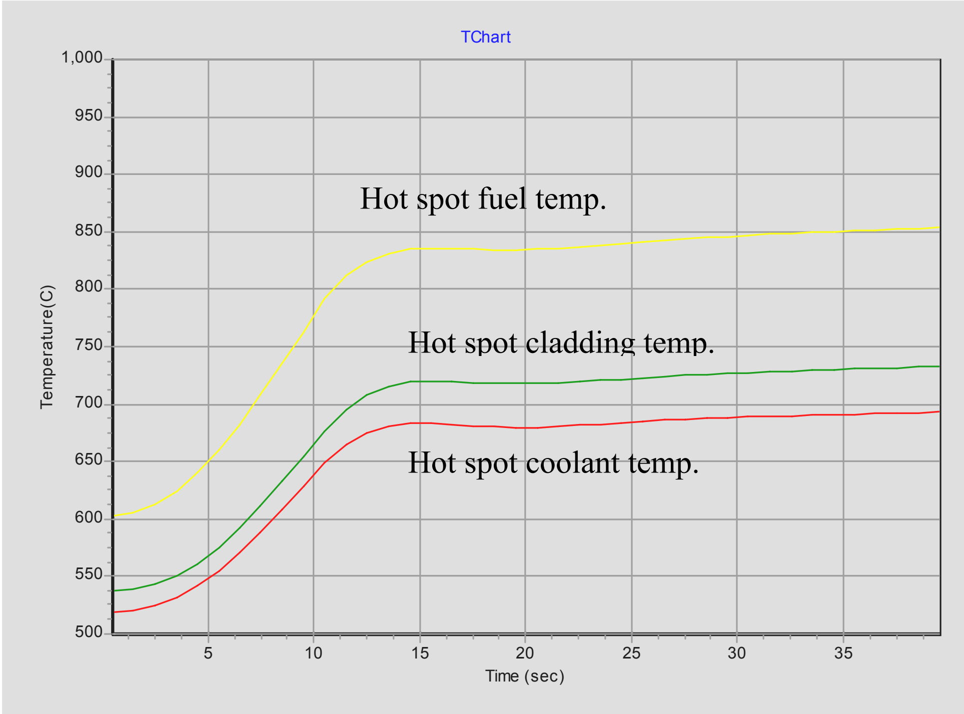


FIG. XXVI-8. Hot spot temperatures during UTOP accident in a 20 MW(e) SPINNOR.

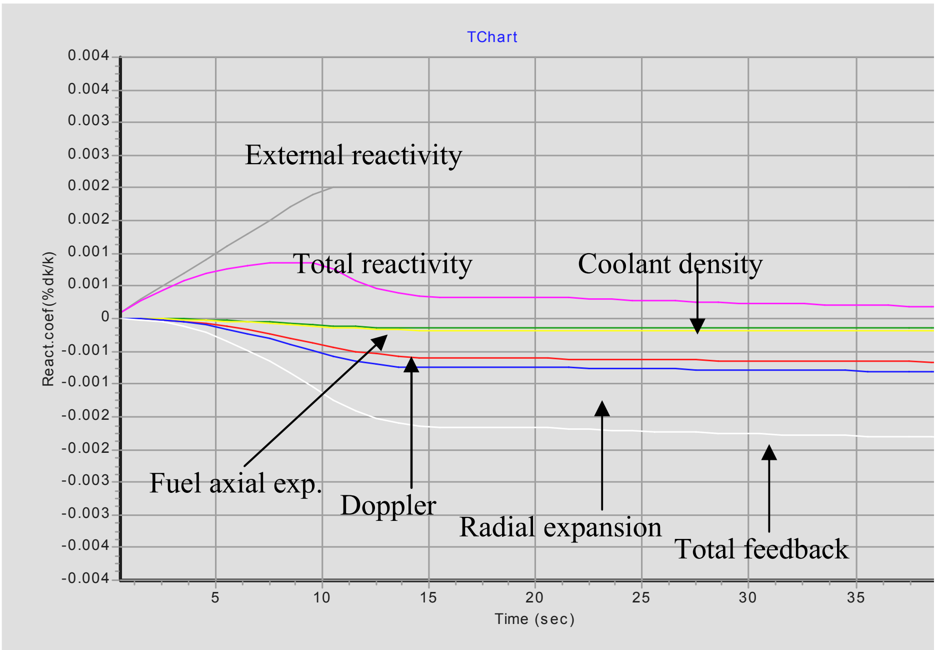


FIG. XXVI-9. Reactivity feedbacks during UTOP accident in a 20 MW(e) SPINNOR.

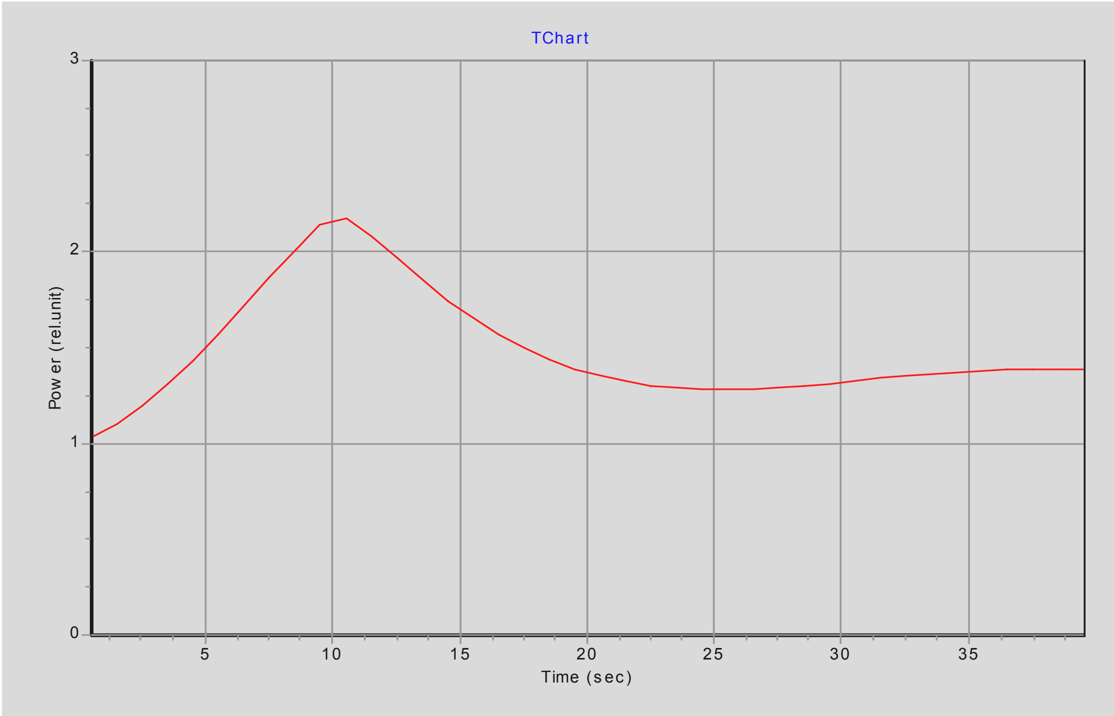


FIG. XXVI-10. Power changes during ULOF-UTOP accident in a 10 MW(e) SPINNOR.

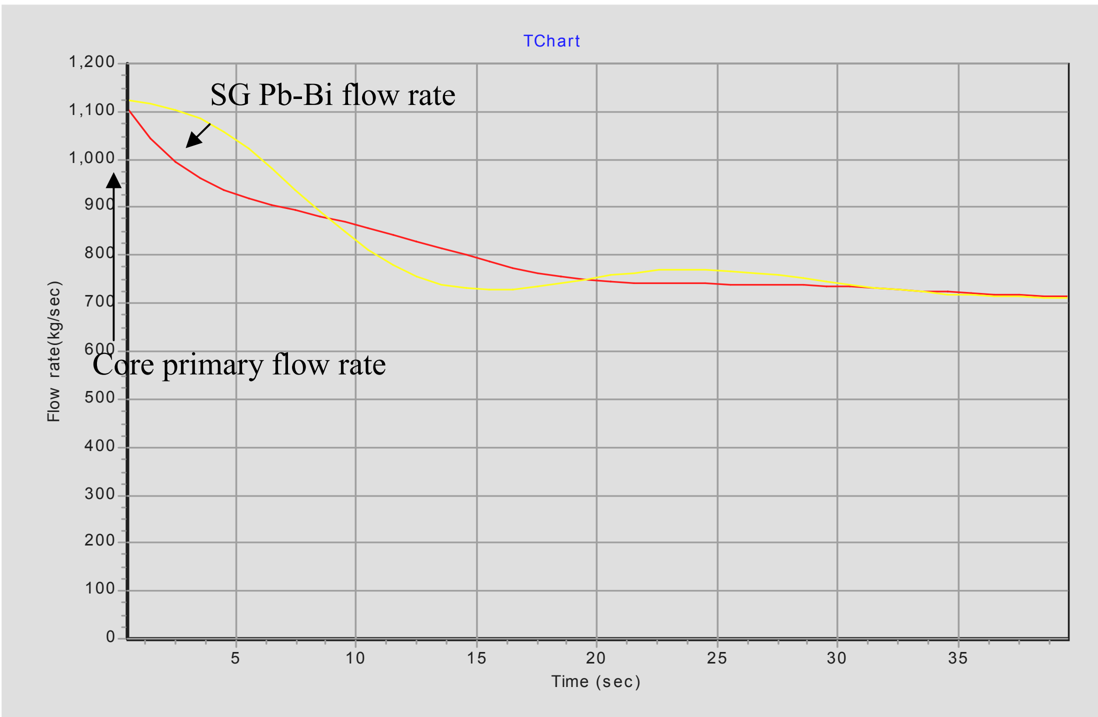


FIG. XXVI-11. Total flow rate of the primary coolant and flow rate through the Pb-Bi side of SG during ULOF-UTOP accident in a 10 MW(e) SPINNOR.

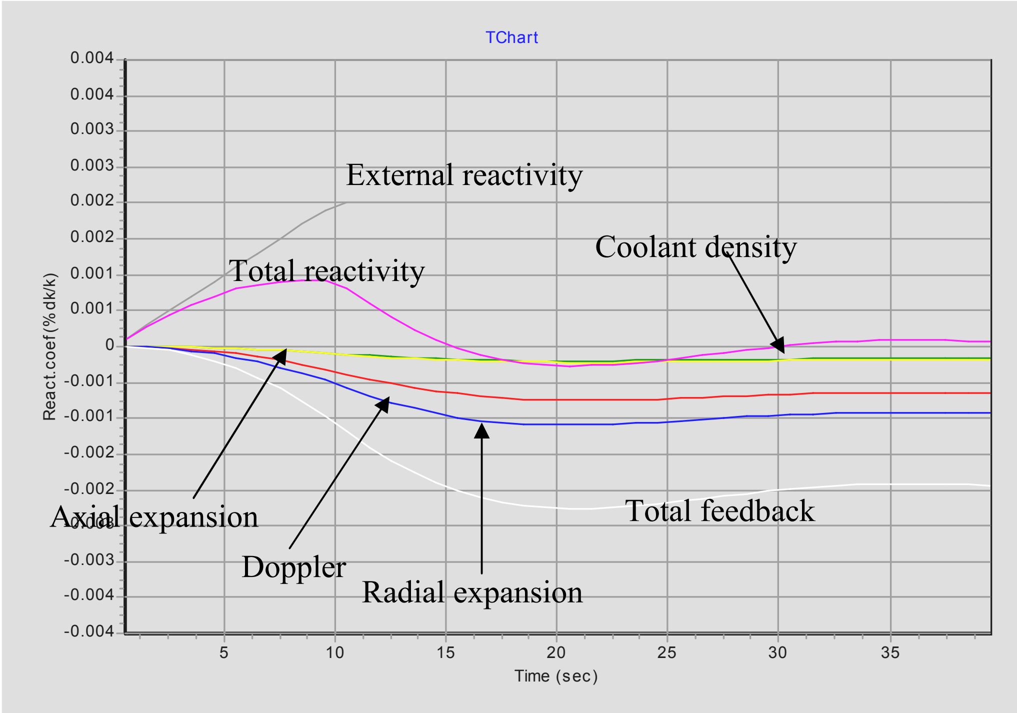


FIG. XXVI-12. Reactivity feedbacks during ULOF-UTOP in a 10 MW(e) SPINNOR.

The passive safety systems of the SPINNOR and VSPINNOR are:

- Gas expansion module (GEM);
- Self actuated shut down system (SASS);
- Reactor vessel auxiliary cooling system (RVACS).

The active safety systems include control rods and a scram system, but they are provided mostly for convenience of changing the reactor power and for performing reactor start-up and shutdown operations.

The following design basis accidents were considered for the SPINNOR and VSPINNOR reactors:

(1) Unprotected partial loss of primary pumping power;
(2) Unprotected partial withdrawal of control rods;
(3) Unprotected partial loss of secondary pumping power;
(4) Unprotected total loss of primary pumping power (ULOF);
(5) Unprotected withdrawal of all control rods (UTOP);
(6) Unprotected total loss of secondary pumping power (ULOHS).

The following combinations of accidents were considered as beyond design basis accidents for the SPINNOR and VSPINNOR:

(1) Simultaneous ULOF and ULOHS;
(2) Simultaneous ULOF and UTOP;
(3) Simultaneous UTOP and ULOHS;
(4) Simultaneous ULOF, UTOP and ULOHS;
(5) Local blockage accident.

Basically, nearly all severe accidents could be effectively prevented, controlled or mitigated by the inherent safety features, passive safety systems and through large provided margins to fuel melting and coolant boiling.

XXVI-1.6.4. Proliferation resistance

The SPINNOR and VSPINNOR provide some technical features to reduce the attractiveness of their nuclear materials for weapon programmes, to prevent the diversion of nuclear materials and undeclared production of direct use materials, and to facilitate nuclear material accounting and verification. These features are as follows:

(1) The plutonium composition in reactor core is unattractive for weapon purposes;

(2) The isotopic contents of fuel provide a radiation barrier that complicates fuel handling, therefore reducing the attractiveness of fuel thefts;

(3) Fuelling, refuelling and decommissioning of the SPINNOR and VSPINNOR are assumed to be performed at a factory;

(4) The reactor vessel is sealed and assumed never to be opened at the site. In addition to this, it is very difficult to open the vessel and then to remove the core, because it is covered by many components, such as steam generator, pump, and also the cooling pool filled with high temperature Pb-Bi eutectics. Due to decay heat, the fuel is at high temperature also;

(5) Because the vessel is sealed, there is no probability to use excess neutrons generated in the core to produce nuclear weapon materials.

XXVI-1.6.5. Technical features and technological approaches used to facilitate physical protection of SPINNOR and VSPINNOR

The SPINNOR and VSPINNOR are assumed to have their vessels sealed and never opened during operation at the site and during transportation from/to the factory. The reactor compartments are assumed to be housed within a reinforced containment to anticipate severe accidents caused by sabotage or massive bombing. Nearly all severe accidents could be effectively prevented, controlled and mitigated by the inherent safety features, passive safety systems and through large margins to fuel melting and coolant boiling.

XXVI-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of SPINNOR and VSPINNOR

No information was provided.

XXVI-1.8. Enabling technologies relevant for SPINNOR and VSPINNOR and status of their development

The key enabling technologies for the SPINNOR and VSPINNOR are:

(1) Technology to maintain certain oxygen regime of lead-bismuth coolant to eliminate corrosion and erosion of stainless steel claddings in coolant flow. This technology is available in the Russian Federation, which has an 80-year operation experience with small lead-bismuth cooled reactors for nuclear submarines.

(2) Advanced materials for high temperature lead-bismuth coolant service. For example, an effort is on-going to the Argonne National Laboratory (USA) to study the performance of ceramic materials, such as silicon carbide based composites.

(3) High accuracy nuclear data for extended burn-up calculations. The accuracy can also be improved by using integral experiments to adjust the nuclear data.

(4) Advanced materials for long-term maintenance of steam generator located inside the reactor vessel.

XXVI-1.9. Status of R&D and planned schedule

Conceptual designs of the SPINNOR and VSPINNOR are mainly developed by the Reactor Physics Laboratory of the Bandung Institute of Technology (ITB), Indonesia. There is cooperation with the National Atomic Energy Agency (BATAN-Indonesia), especially with its Advanced Reactor System Division to perform certain optimization studies.

The projects are funded under several Indonesian National Competitive Research Grants, including the Integrated Advanced Research Grant (RUT XI), the National Competitive Grant (Hibah Bersaing), and from some other sources. It is expected that further R&D will involve cooperation with some research centres and universities in Indonesia and abroad. One of the companies in Indonesia is currently interested to support further R&D for the SPINNOR series. It is anticipated that, under favourable conditions, the pilot project of SPINNOR/VSPINNOR could be established within the next 10 years.

XXVI-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

A reactor that could be operated without opening the vessel for 15 years or more needs many advanced materials for its core structures, coolant, fuel, steam generator, maintenance free pump, etc. Especially because of the use of Pb-Bi as coolant, the corrosion becomes an important issue to be resolved. There is a certain Russian experience for lead-bismuth cooled reactors of nuclear submarines, but they never operated for more than 7-8 years continuously. Therefore, development and validation of other innovative technologies may be required for lead-bismuth reactors targeted at 15 – 35-year operation without on-site refuelling.

Another important issue is that assessment of long-life cores requires accurate nuclear data and reactor system analysis codes. Here, some integral and differential experiments may be needed to provide a basis for the validation.

The choice of make-up fuel is also crucial. If spent fuel from light water reactors (LWRs) is used within a short period after its discharge, then any delay in operation of the SPINNOR/VSPINNOR may produce an unfavourable effect on the neutronic characteristics of the core. Therefore, special validation, including nuclear data adjustment based on integral experiments, is considered necessary.

The considerations outlined above point to a significant amount of R&D to be performed to demonstrate viability and reliability of the SPINNOR/VSPINNOR concepts and justify the possibility of a demonstration plant construction.

XXVI-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The designs of small lead-bismuth or lead cooled reactors relevant for or similar to the SPINNOR/VSPINNOR concept include LSPR of the RLNR TITech (Japan), SVBR 75/100 of the IPPE (Russian Federation), BREST series of the RDIPE (Russian Federation), and STAR-LM, STAR-H2, and SSTAR concepts from national laboratories and universities in the USA.

XXVI-2. Design description and data for SPINNOR/VSPINNOR

XXVI-1.2.1. Description of the nuclear systems

Reactor core and fuel design

The design philosophy behind the SPINNOR and VSPINNOR concepts is to place fissile material in outer part of the core while to fill its inner part with fertile material. In line with this, at the beginning of life, the core outer part will produce major contribution to overall criticality but will gradually loose fissile material with the progress of fuel burn-up. In turn, as burn-up progresses, more and more fissile material will be created in the central part of the core and, as comes to criticality, this fissile material will compensate for the fissile material loss in the core outer part. At higher burn-ups, central part of the core becomes the major contributor to overall criticality and, as newly produced fissile material in this region has higher importance compared to that lost in the peripheral part, it helps to reduce burn-up reactivity swing over the whole reactor lifetime, as shown in Figures XXVI-13 and XXVI-14.

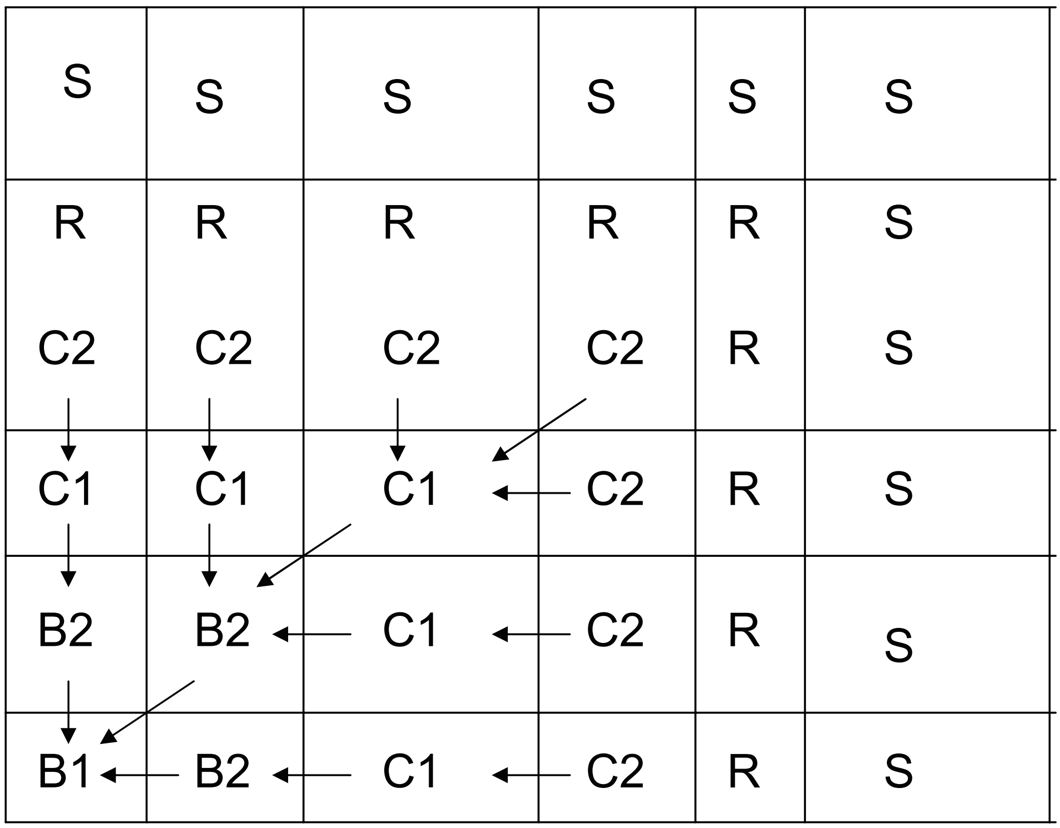
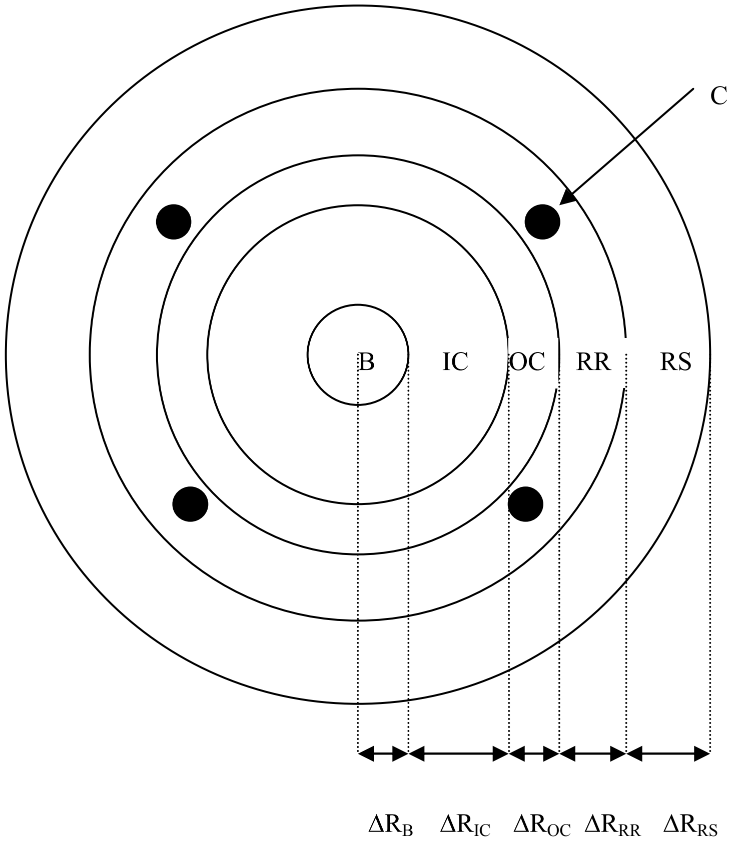


FIG. XXVI-13. Core configuration of SPINNOR/VSPINNOR reactors and the increase of importance of fissile materials generated in inner regions with progress of fuel burn-up.

Special adjustment of the parameters of each region is carried out at the design stage to minimize reactivity swing associated with fuel burn-up. The description of core parameters is given in Table XXVI-5.



B – blanket, IC – inner core, OC – outer core, RR – radial reflector, RS – reactor shielding

FIG. XXVI-14. Lateral arrangement of the SPINNOR/VSPINNOR core.

Both the SPINNOR and the VSPINNOR incorporate four control rods that are located outside the core, in radial reflector region.

TABLE XXVI-5. DESIGN PARAMETERS OF SPINNOR AND VSPINNOR CORES

PARAMETER	VALUE/DESCRIPTION		
	SPINNOR A	SPINNOR B	VSPINNOR
Dimensions* with reference to Fig. XXVI-14:			
Effective radial width of inner core (blanket)	10 cm	10 cm	5.5 cm
Inner core height	10 cm	10 cm	5.5 cm
Effective radial width of outer core	39 cm	38 cm	40 cm
Outer core height	39 cm	38 cm	40 cm
Width of radial reflector	20 cm	20 cm	30 cm
Radial reflector height	25 cm	25 cm	30 cm
Width of radial shielding	30 cm	30 cm	20 cm
Radial shielding height	25 cm	25 cm	20 cm

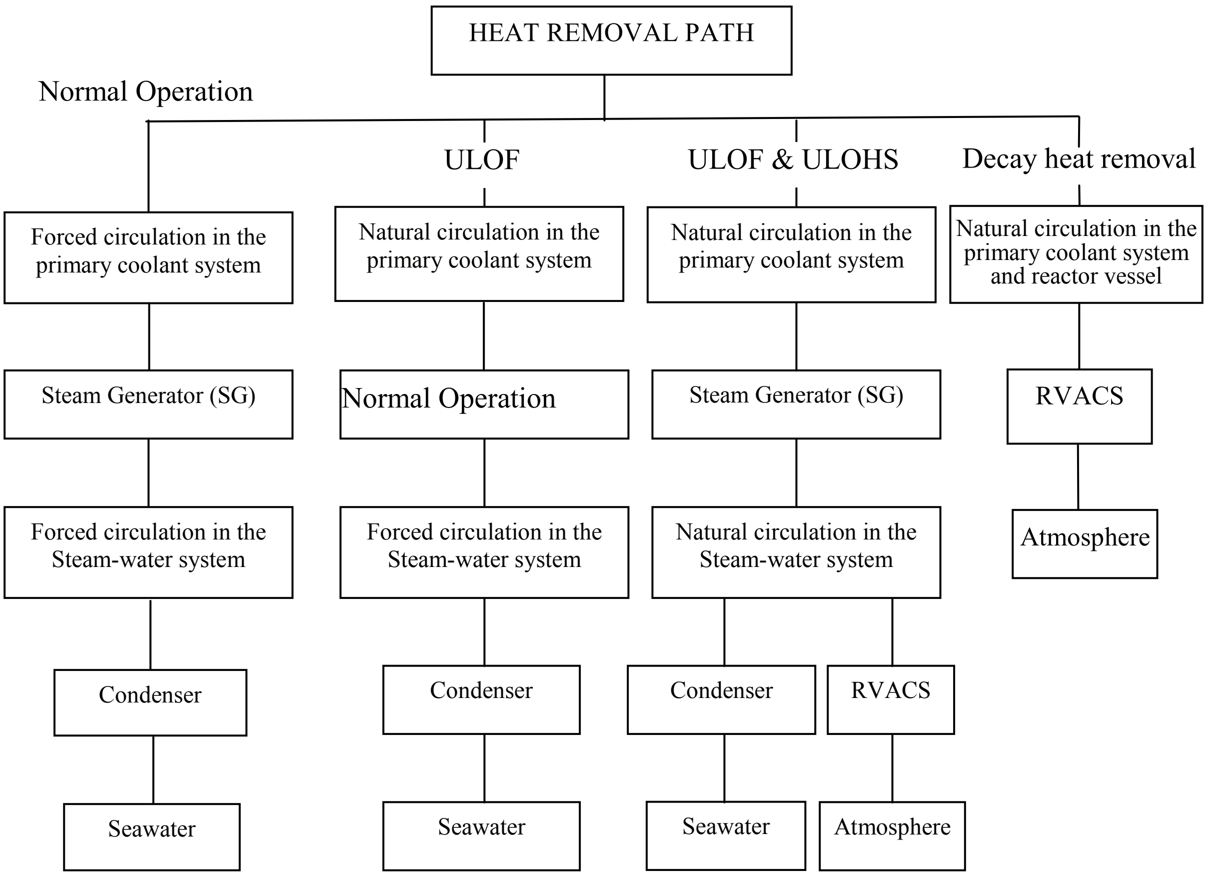
* Active core diameter is less than one meter in all cases. Together with the reflector and shielding, the core diameter never exceeds 2 m

Primary coolant system

The primary coolant system uses a steam generator located inside the reactor vessel with no intermediate heat exchangers. The heat from the core is transported by Pb-Bi primary coolant to a steam generator shown in Fig XXVI-1. In the steam generator, heat from the primary coolant is transferred to the secondary steam-water based circuit. Large fuel pin diameter of 1.0–1.2 cm and a relatively low core height contribute to the reduction of pressure drop in the core. Therefore, a relatively high degree of natural circulation is provided in the primary coolant system. In particular, the VSPINNOR, a very small reactor, could be operated on natural circulation alone by employing an appropriate design of the chimney.

The scheme of main heat transport system with indication of heat removal path in normal operation and in accidents is shown in Fig. XXVI-15.

Main heat transport system



RVACS - reactor vessel auxiliary cooling system; ULOF - unprotected loss of flow; ULOHS - unprotected loss of heat sink

FIG. XXVI-15. Heat removal paths for SPINNOR and VSPINNOR in normal operation and in accidents.

In the case of an ULOF accident, natural circulation in the primary system is sufficient to remove heat from the core to the secondary circuit, which operates in a normal regime. Seawater is an ultimate heat sink for this case. However, when ULOF coincides with ULOHS, the role of RVACS becomes more important, and atmospheric air shoulders the functions of additional heat sink for heat removal from the reactor core.

XXVI-2.2. Description of the turbine generator plant and systems

The turbine generator system is shown in Fig. XXVI-9. Standard Rankine cycle with a single feedwater reheater is used for generation / co-generation of electricity. The pressure of steam-water coolant in the secondary circuit is 7 MPa. Steam from the steam generator located inside the reactor vessel is directed to the turbine to produce electricity and can also be directed to a desalination plant for the production of potable water. Steam from the turbine is condensed to water by using seawater as an ultimate heat sink in normal operation. The water from the condenser is mixed with the feedwater to be pumped to the SG inlet through an orifice block.

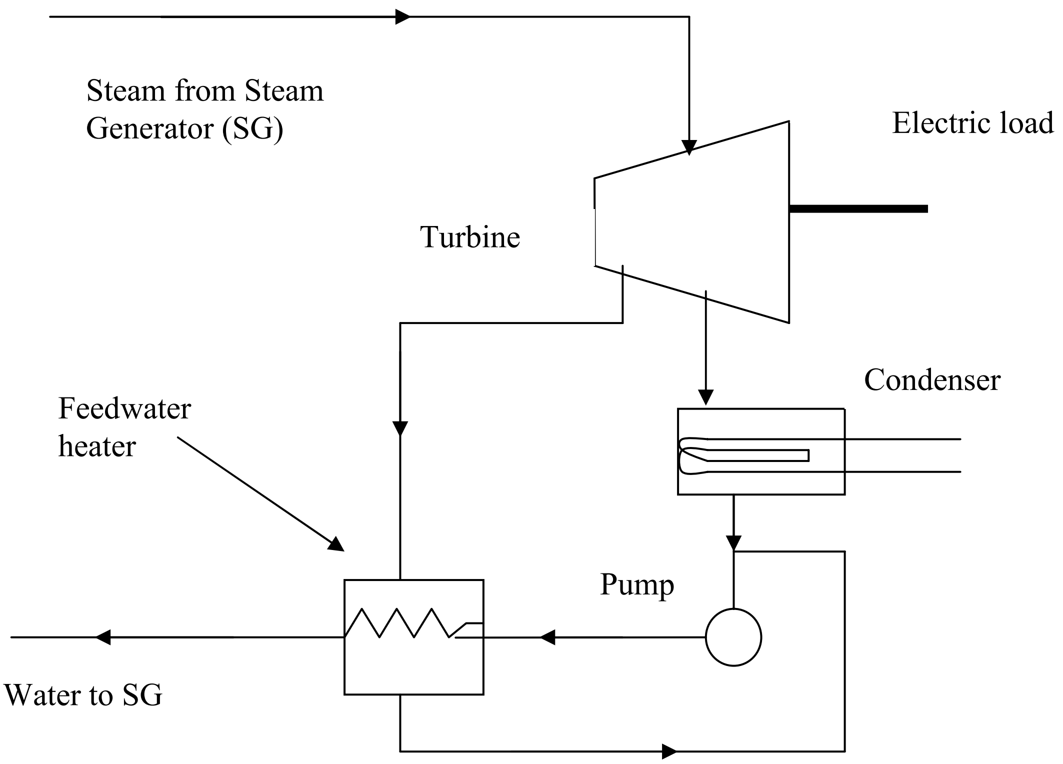


FIG. XXVI-16. Steam turbine system.

XXVI-2.3. Systems for non-electric applications

A desalination plant is assumed to be a standard option for the SPINNOR and VSPINNOR reactors. Seawater reverse osmosis (SWRO) system is targeted for use as described detail in [XXVI-8]. The specific energy consumption for potable water production is estimated at ~6 kWh/m^3. The pressure in the SWRO section is assumed to be 3.8~4.6 MPa.

XXVI-2.4. Plant layout

General philosophy governing plant layout

The plant layout for the SPINNOR is shown in Fig. XXVI-17. The turbine, the condenser and the desalination plant are assumed to be located near the seaside. The reactor building is directly linked to the turbine building, so that steam from the steam generator located inside the reactor vessel in the reactor building can be easily transported to the turbine and the desalination plant, while the sub-cooled water can be easily fed into the intake of the steam generator. The reactor service and maintenance building (SMB) is located near the reactor building to provide an easy access to the reactor system.

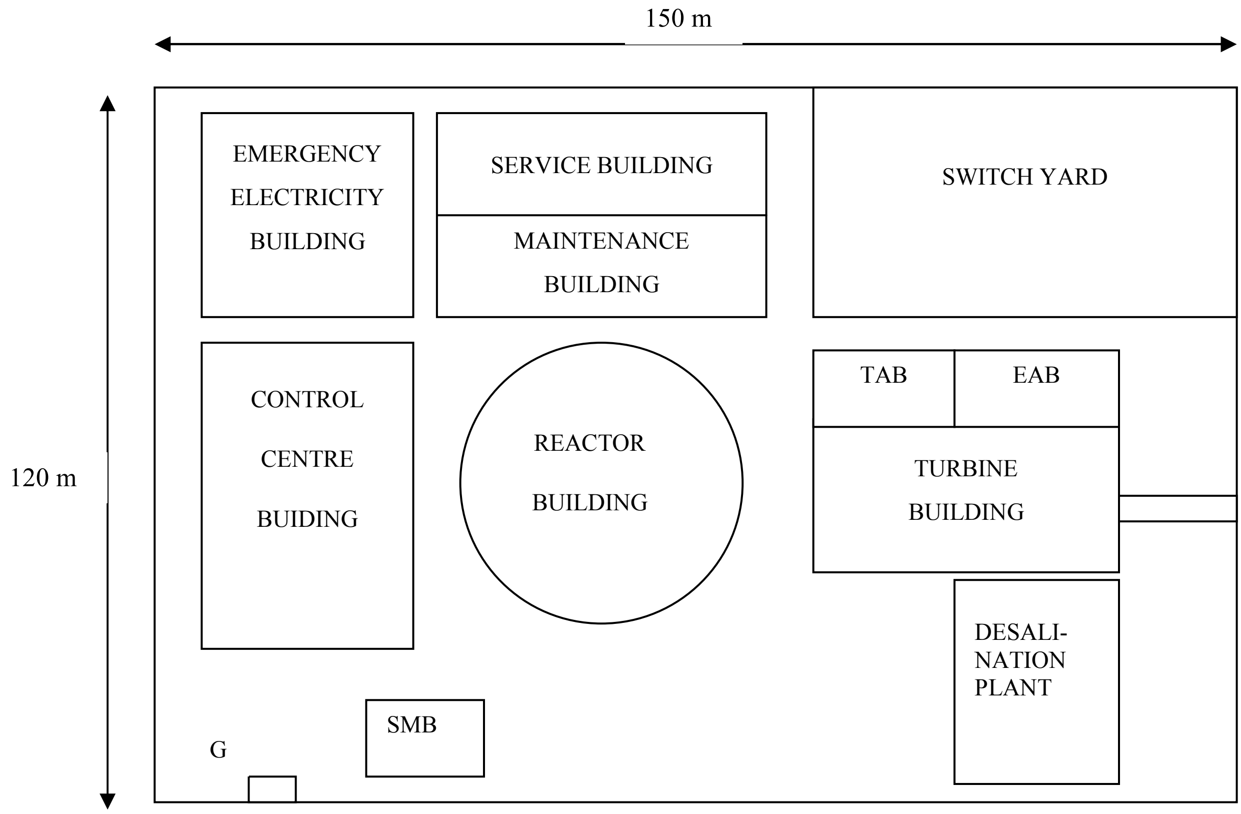


FIG. XXVI-17. Plant layout for SPINNOR.

The reactor main control room is set nearby the reactor core and also nearby the emergency electricity generator (G), so that in an emergency case it would be easier to switch on to the reserve power.

REFERENCES

[XXVI-1] SU'UD, Z., SEKIMOTO, H., Preliminary design study of the ultra long life fast reactor, Nuclear Engineering and Design 140 (1993), 251–260.

[XXVI-2] H. SEKIMOTO and ZAKI S, Design study of lead and lead-bismuth cooled small long-life nuclear power reactors using metallic and nitride fuel, Nuclear Technology, 105, No. 3 (1995).

[XXVI-3] SU'UD, Z. and H. SEKIMOTO, Safety Aspect of Long Life Small Safe Power Reactors, Analysis of Nuclear Energy (1995).

[XXVI-4] SU'UD, Z., SEKIMOTO, H., Design and safety aspect of lead and lead bismuth cooled long-life small-safe fast reactor for various core configuration, Journal of Nuclear Science and Technology 32/9 (1995).

[XXVI-5] SU'UD, Z. and H. SEKIMOTO, Accident analysis of lead or lead-bismuth cooled small safe long-life fast reactor using metallic or nitride fuel, Nuclear Engineering and Design 162 (1996) 205–222.

[XXVI-6] SU'UD, Z., Comparative study on safety performance of nitride fuelled lead-bismuth cooled fast reactor with various power level, Progress of Nuclear Energy, pp. 571~577, Vol.32, No. 3/4(1998).
[XXVI-7] SU'UD, Z. Design study on ship based nuclear power reactors, Global Environment and Nuclear Energy System (GENES) III, Tokyo Inst. of Technology, Tokyo, Japan, December 14–17 (1999).
[XXVI-8] KONISHI T. et al, Optimization of the coupling of nuclear reactors and desalination system, Global Environment and Advanced NPPs (Proc. Int. Conf. Kyoto, Sep. 15–19, 2003) GENES 4/ANP 2003.

ANNEX XXVII

LEAD-BISMUTH COOLED DIRECT CONTACT BOILING WATER SMALL REACTOR (PBWFR)

Research Laboratory for Nuclear Reactors, Tokyo Institute of Technology
Japan

Short description

XXVII-1. Basic summary

The Pb-Bi Cooled Direct Contact Boiling Water Small Reactor (PBWFR) is being developed in Japan by the Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology (RLNR TITech) in cooperation with Advanced Reactor Technology Co., Ltd., and Nuclear Development Corporation. The design and technology development for the PBWFR was supported by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan. Feasibility studies have been conducted and conceptual design of the PBWFR is being developed currently.

Reactor design

The PBWFR concept [XXVII-1] is an evolution of the concept of a direct contact Pb-Bi fast breeder reactor (PBWR) proposed in [XXVII-2]. It is a pressure vessel type reactor, in which sub-cooled water is fed into the hot Pb-Bi coolant above the core, resulting in a direct contact boiling, as shown in Fig. XXVII-1. Boiling bubbles rise due to buoyancy effect, which also works as a lift pump for Pb-Bi circulation. The generated steam passes through the separator and the dryer to remove Pb-Bi droplets, and then flows to the turbine-generator plant. The outlet steam is superheated by ~ 10^{o}C to avoid the accumulation of condensate on a free Pb-Bi surface in the reactor vessel.

The primary objective for employing direct-contact heat exchange scheme is to improve plant economy.

The PBWFR is a fast breeder reactor with a breeding ratio of ~1.1. The core is of homogeneous type and has 2 regions. With nitride fuel, the core lifetime of 15·years is achieved without reloading or shuffling of fuel.

Different from its predecessor [XXVII-2], the PBWFR employs a mechanical system of control rods that are inserted from the top. Such feature was adopted because it is much easier to seal steam at the top of the reactor vessel than to seal Pb-Bi at the bottom.

Plant design

The PBWFR is designed to generate electricity. The cycle type is direct and the system pressure is the same as in conventional boiling water reactors (BWRs), see Fig. XXVII-2. Steam is generated in the chimneys in direct contact with hot Pb-Bi coolant above the core. There are no steam generators and intermediate heat transport systems.

The balance of plant is similar to that used in conventional BWRs. As a difference, hydrogen is supplied in the feedwater to ensure adequate control of the oxygen potential in Pb-Bi coolant, to prevent the formation of PbO in Pb-Bi coolant.

Safety design

Main and auxiliary cooling systems of the PBWFR are driven by natural convection. The inherent safety features of the core are enhanced to avoid a core disruption accident even in anticipated transients without scram (ATWSs). Specifically, void reactivity for the case when the core, the axial blanket, and the plenum are totally voided is limited by 3 $ (design modifications are foreseen to make this effect negative). The burn-up reactivity swing during 15 years of operation without refuelling is minimized down to 1.5% $\Delta K/K$.

The design incorporates a mechanical reactivity control and shutdown system based on control rods with drives located atop the reactor vessel. In an emergency, the control rods can enter the core, driven by gravity. The PBWFR incorporates two primary reactor auxiliary cooling systems (PRACS) with two direct heat exchangers (DHX) installed in the downcomer. Each PRACS has two Pb-Bi loops, and the Pb-Bi of the secondary loop is cooled by natural circulation of air in the air cooler, as shown in Fig. XXVII-2. Both PRACS are passive systems.

The design incorporates the guard vessel, which is being cooled from outside by the passive reactor vessel air cooling system (RVACS), based on natural convection of atmospheric air.

XXVII-2. Major design and operating characteristics

Main characteristics of the reactor core are summarized in Table XXVII-1. Major characteristics of an NPP with the PBWFR are given in Table XXVII-2. The PBWFR concept and a general view of the reactor internals are given in Fig. XXVII-1. A simplified schematic diagram of the plant is in Fig. XXVII-2.

XXVII-3. List of enabling technologies and their development status

The enabling technologies of the PBWFR are listed in Table XXVII-3 with an indication of their development status.

TABLE XXVII-1. CORE CHARACTERISTICS

ITEMS	SPECIFICATIONS
Fuel type	Pu-U nitride (100% ^{15}N enriched)
Core type	Two-region; homogeneous zones
Type of fuel assembly	Hexagonal
Number of fuel assemblies (inner/outer core)	36 / 42
Number of fuel pins	58 per fuel assembly
Enrichment by Pu (inner/outer core)	11.5 / 15.8 weight %
Fuel burn-up	110 GW·day/t
Operation cycle length	15 years
Cladding outer diameter/lattice pitch	12 / 15.9 mm
Number of fuel pins per assembly	271
Pitch of fuel assemblies	275 mm

ITEMS	SPECIFICATIONS
Core effective diameter	267 cm
Core effective height	100 cm
Radial blankets	No
Axial blankets (lower/upper)	No / 30 cm
Maximum linear power	396 W/cm
Breeding ratio (BOL / EOL)	1.25 / 1.05
Void reactivity (core)	+7.4 $ (EOL)
Void reactivity (core, axial blanket and plenum)	+3 $ (EOL)
Burn-up reactivity swing	1.5 % Δk/k

TABLE XXVII-2. PLANT CHARACTERISTICS

ITEMS	SPECIFICATIONS
Reactor type	Pressure vessel type
Electric output	150 MW
Thermal output	450 MW
Primary coolant (Pb-Bi) temperature	307 / 457°C
Feedwater / steam temperature / pressure	220 / 296 °C/ 7 MPa
Plant efficiency	33 %
Pb-Bi circulation mode	Natural convection
Water – steam circulation mode	Forced convection
Steam generator	No; direct contact heat exchange between Pb-Bi and water-steam
Reactivity control / reactor shutdown	Mechanical control rods with drives mounted atop the reactor vessel; gravity-driven insertion in emergencies
Decay heat removal system	Two PRACS and one PRACS; all passive
Containment system	Reactor vessel, guard vessel, containment vessel
Estimated capital cost, US$/kW(e)	3300

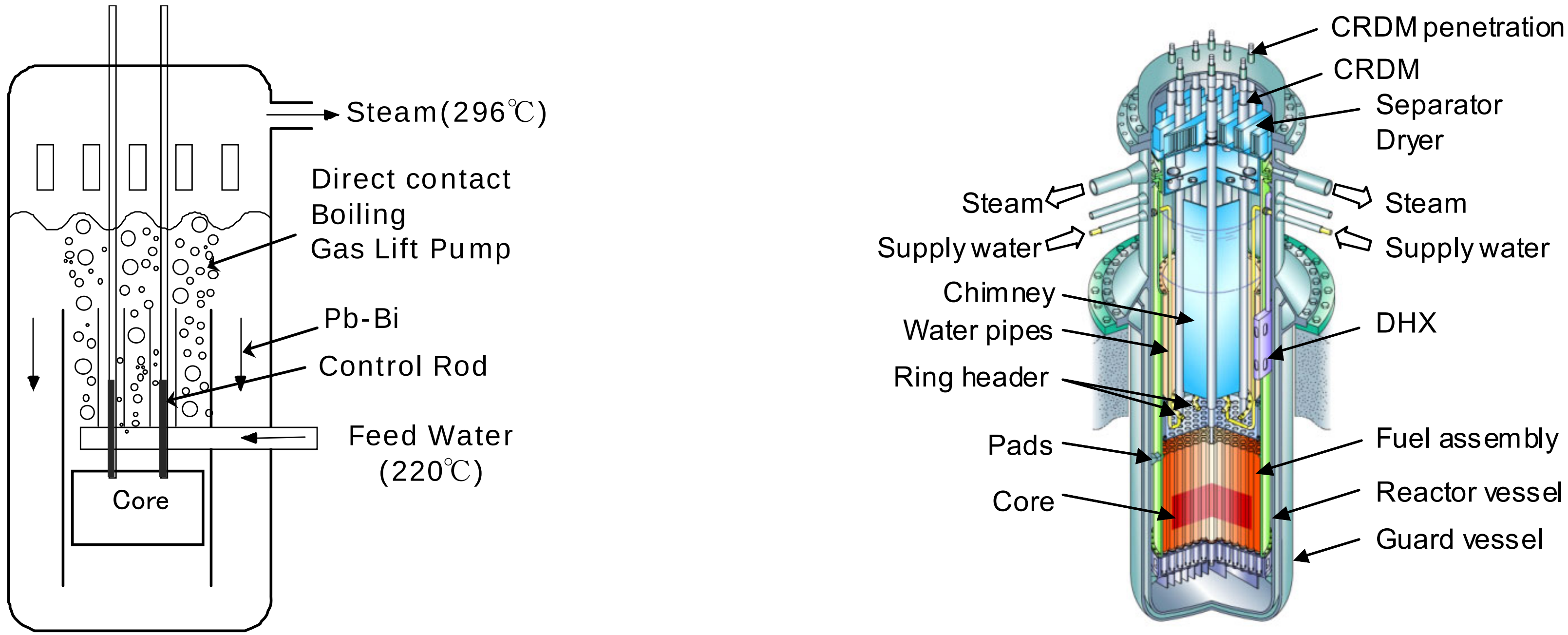
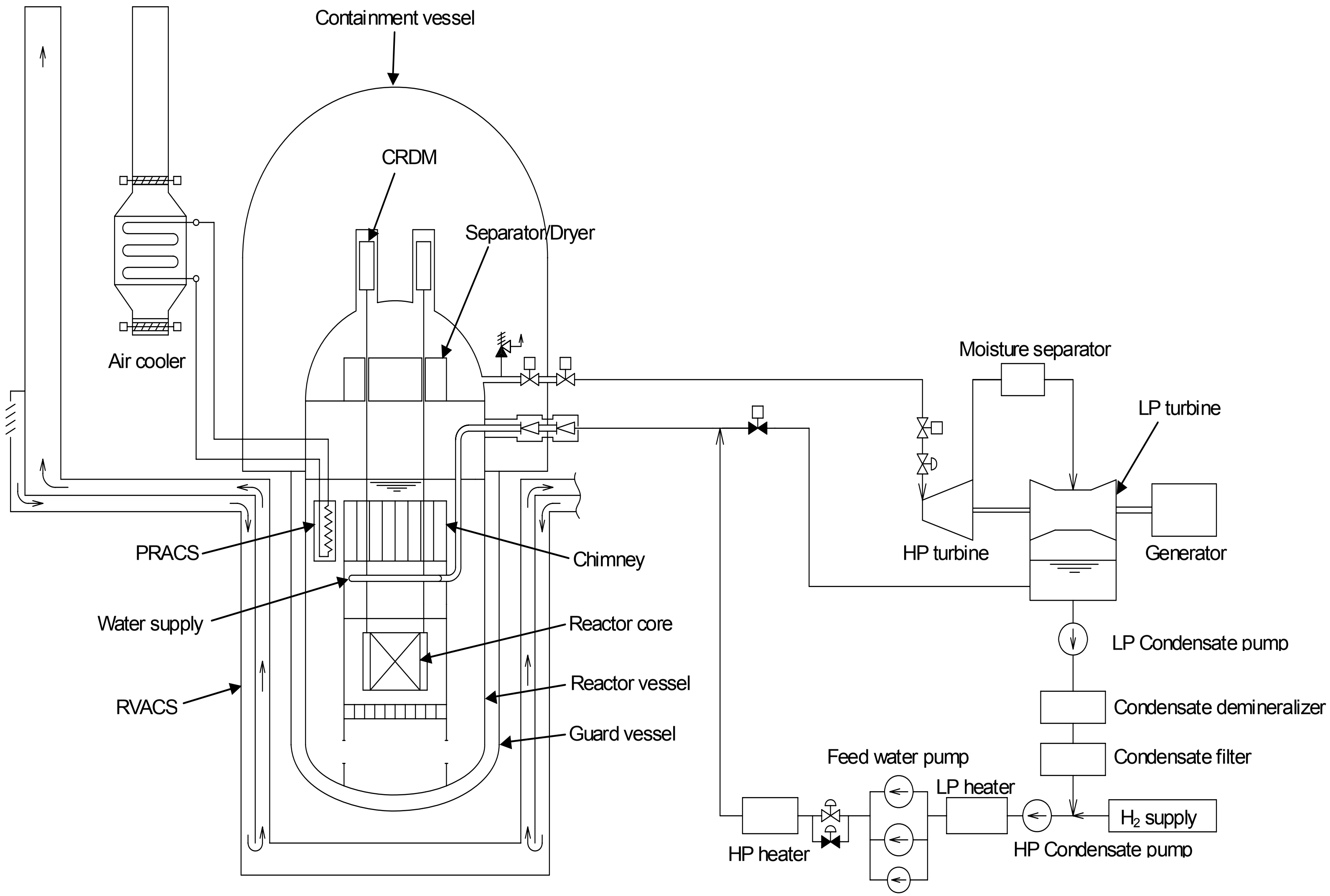


FIG. XXVII-1. PBWFR concept (left) and general view of the reactor internals (right).



LP – low pressure; HP – high pressure; RV – reactor vessel; GV – guard vessel; CRDM – control rod drive mechanism

FIG. XXVII-2. Simplified schematic diagram of PBWFR.

TABLE XXVII-3. LIST OF ENABLING TECHNOLOGIES FOR PBWFR

ENABLING TECHNOLOGY	DEVELOPMENT STATUS
Nitride fuel technology	Conceptual design
Direct contact heat exchange; natural convection of Pb-Bi, assisted by lift pump	Test programme ongoing in RLNL TITech
An oxygen potential control system to protect structural materials operating in Pb–Bi from corrosion	Controlled by adding hydrogen to the feedwater; test programme ongoing in RLNL TITech
Pb-Bi vapour and ^{210}Po control in the circuit	Test programme ongoing in RLNL TITech
Separator / dryer	Test programme ongoing in RLNL TITech
Structural design of reactor vessel and internals	The reactor vessel is supported at the gravity centre; conceptual design

REFERENCES

[XXVII-1] M. Takahashi, S. Uchida, K. Handa, et al. "Pb-Bi Direct Contact Boiling Water Small reactor", Progress in Nuclear Energy, vol. 47, No. 1–4, pp. 190, 201, 2005, Elsevier Ltd., UK.

[XXVII-2] Buongiorno, J., Todreas, N.E., Kazimi, M.S., et al. "Conceptual design of a Lead-Bismuth Cooled Fast Reactor with In-Vessel Direct-Contact Steam Generation", MIT-ANP-TR-078, Center for Advanced Nuclear Energy Systems, Massachusetts Institute of Technology, Boston, USA, 2001.

NON-CONVENTIONAL SMALL REACTORS

MICRO-PARTICLE FUEL AUTONOMOUS MOLTEN SALT COOLED REACTOR (MARS)

Russian Research Centre "Kurchatov Institute",
Russian Federation

XXVIII-1. General information, technical features and operating characteristics

XXVIII-1.1. Introduction

MARS is the Russian abbreviation for a micro-particle fuel autonomous molten salt cooled reactor.

This name reflects the basic technology of the concept, which incorporates micro-particle fuel and molten salt coolant, as well as its destination, which is to serve as an autonomous nuclear power plant (NPP) in remote areas with difficult access [XXVIII-1 to XXVIII-3].

The MARS concept originated from a demand for enhanced-safety power plants that might suit isolated areas with limited number of skilled labour. To facilitate its deployment, the concept is devised as a combination of approbated technical features, the combination that attributes the plant with the properties of a "nuclear cell".

Among possible combinations of nuclear fuel, coolant and energy conversion equipment for a NPP that is to operate in a remote and hard-to-access area, the combination of coated particle based graphite fuel elements and a molten salt primary circuit coolant based on Li, Na, Be, and Zr fluorides and a gas turbine unit for electricity generation may offer special advantages.

The first practical use of molten salt coolant dates back to the early 1950s when Oak-Ridge National Laboratory (USA) launched a research programme aimed at building a high temperature nuclear reactor with circulating fuel for an aero-engine. A small experimental reactor ARE was built under this programme, followed by the one of a higher power, MSRE. Later on, a number of concepts and conceptual designs of molten salt reactors were elaborated in Japan, France, the Russian Federation (former USSR) and China.

Development of reactor concepts with coolants based on fluoride molten salts progresses along the following two directions:

- Reactors with circulating liquid fuel, based on molten salts; and
- Reactors with fuel-free salt composition used as a coolant.

The MARS concept belongs to the second direction. Concepts of medium and high power high temperature molten salt reactors (abbreviated as VTRS in Russian) belonging to this direction were studied extensively in the Russian Federation (former USSR) [XXVIII-4]. It was anticipated that the VTRS type reactors would use circulating spherical fuel elements similar in design to the fuel of high temperature gas cooled reactors (HTGRs). Various design concepts of the reactor installation were considered including a two-circuit design scheme with natural circulation of salt in the primary circuit and a gas turbine unit with an open air-cycle in the secondary circuit.

Fluoride based molten salt coolants have the following attractive properties:

- Fire resistance, high boiling point, low pressure at high operating temperatures;
- High radiation resistance and chemical inertness with water and air; and

- The capability of heat removal by natural circulation in all NPP circuits and in all modes of plant operation.

The use of coated particle based spherical fuel elements in combination with molten salt coolant is justified by the following factors:

- Molten salt coolants based on fluorides of the abovementioned metals are characterized by good compatibility with graphite over a wide range of temperatures (up to ~1200–1300°C). When impregnated with these salts, graphite becomes inflammable in air;
- In principle, graphite fuel elements with coated particles can be installed in the reactor core as prismatic or cylindrical blocks or fabricated as spheres and placed in the core as a free or fixed-order bed. Spherical fuel elements with coated particles have certain advantages related to a higher possible degree of flexibility in their fabrication (they could incorporate various structures, diameters and fuel enrichment) and subsequent arrangement in the core;
- The MARS core is built as a fixed bed consisting of spherical fuel elements and absorber and graphite elements, with their distribution optimized over the core volume to achieve high fuel burn-up and ensure a low reactivity margin for fuel burn-up;
- Spherical fuel elements considered for the MARS are of a type that has undergone various technological and reactor tests. Specifically, fuel elements of this type are validated for use in HTGRs developed in the Russian Federation, Germany, South Africa and China.

A gas-turbine unit that receives heat from a nuclear reactor with coated particle fuel and high-temperature molten salt coolant can serve as a highly efficient and long-lived autonomous electric power source operating without a water pond as a reject heat sink and without the presence of water in the cooling systems.

Of particular interest is a gas-turbine unit operating in an open cycle with air. Such gas turbine units with various thermodynamic parameters have been built based on aero-engine technology.

A conceptual design of the MARS plant is being developed by a research team of the Russian Research Centre “Kurchatov Institute” (RRC KI, Moscow, Russian Federation). Interactions with the design organization (OKBM, Nizhny Novgorod, Russian Federation) have been started to initiate preliminary design development that would be preceded by an additional analysis of the features adopted at the conceptual design stage.

XXVIII-1.2. Applications

A small nuclear power plant with the MARS reactor can be used for electricity generation and, within heat and power plants, for the production of high temperature heat for process applications or the production of lower-grade heat for heating, agricultural uses, or seawater desalination.

In particular, the assessments performed [XXVIII-5, XXVIII-6] indicate that the following options are feasible with the MARS plant operating in a base load mode, for the conditions of the Russian Federation:

- For regions in the Russian North that are cut-off from any supplies in wintertime – support of agricultural production and the replacement of fossil fuel supplied from afar by locally produced hydrogen;
- For Northern seas in Russia that have a huge potential for bio-production – support of seafood farms to produce protein and mineral rich products, such as crabs, shellfish, water plants, etc.;
- Based on existing waste-free seawater processing technologies – support of the production of a variety of chemical products, mineral fertilizers, metals, and construction materials from brines;
- With electricity produced – on-site support of the refining of oil, gas condensate, and substandard oil-containing products, using new technologies that ensure a ~100% yield of light fractions;
- For the remote and hard-to-access areas – support of the gas processing and liquefaction for further transportation; or support of the coal gasification for areas with deposits of coal;
- Thermal fragmentation (milling) of rocks for the mining of gold and other valuable metals; support of the production of ore concentrates.

Further evolution of the MARS concept provides for an increase of coolant temperature at the core outlet as appropriate structural materials are developed. The resulting concept in which HTGR type fuel elements are cooled by molten salt would be attractive for very high temperature non-electric applications, such as hydrogen production [XXVIII-7].

XXVIII-1.3. Special features

The MARS reactor is designed for those remote and hard-to-access areas that are in need of electric power, heat and water but where power could not be received via electricity grids and where regular supply of fossil fuel is practically impossible. The major features of the MARS plant are, therefore, factory fabrication and fuelling of the complete reactor module, an operation without on-site refuelling, and simplified decommissioning.

The MARS concept is being designed to offer a variety of energy products to autonomous consumers. Co-production of electricity, heat, potable water, chemical products from seawater, seafood, hydrogen for power and process applications, gasification of coal, mining and processing of mineral resources; creation of small habitable areas in the arid lands near the seaside, etc. would be possible with the use of this plant. The MARS plant could be used to provide power to drive a gas pumping station to save gas and to reduce the environmental impact, etc.

The design of a small power plant with the MARS reactor, in principle, allows various deployment options, including floating or pontoon based NPPs, deployment on land as a stationary or transportable unit, as well as underground deployment of the complete NPP.

XXVIII-1.4. Summary of major design and operating characteristics

A simplified schematic diagram of the MARS plant is given in Fig. XXVIII-1.

To assemble the active core configuration, spherical fuel elements similar in design to the HTGR fuel are used, complemented by spherical absorber and graphite elements. At the stage of core loading, performed in a specialized factory, the fixed bed of these spherical elements is arranged, which is kept firmly in place by structural elements and does not change during the fuel lifetime. The reflector material is just circulating molten salt coolant. The configuration of the fixed bed of the active core (spatial distribution of fuel and other elements) enables optimum power and burn-up flattening over the core volume during the fuel lifetime, as well as necessary sub-criticality in a hypothetical loss of coolant event.

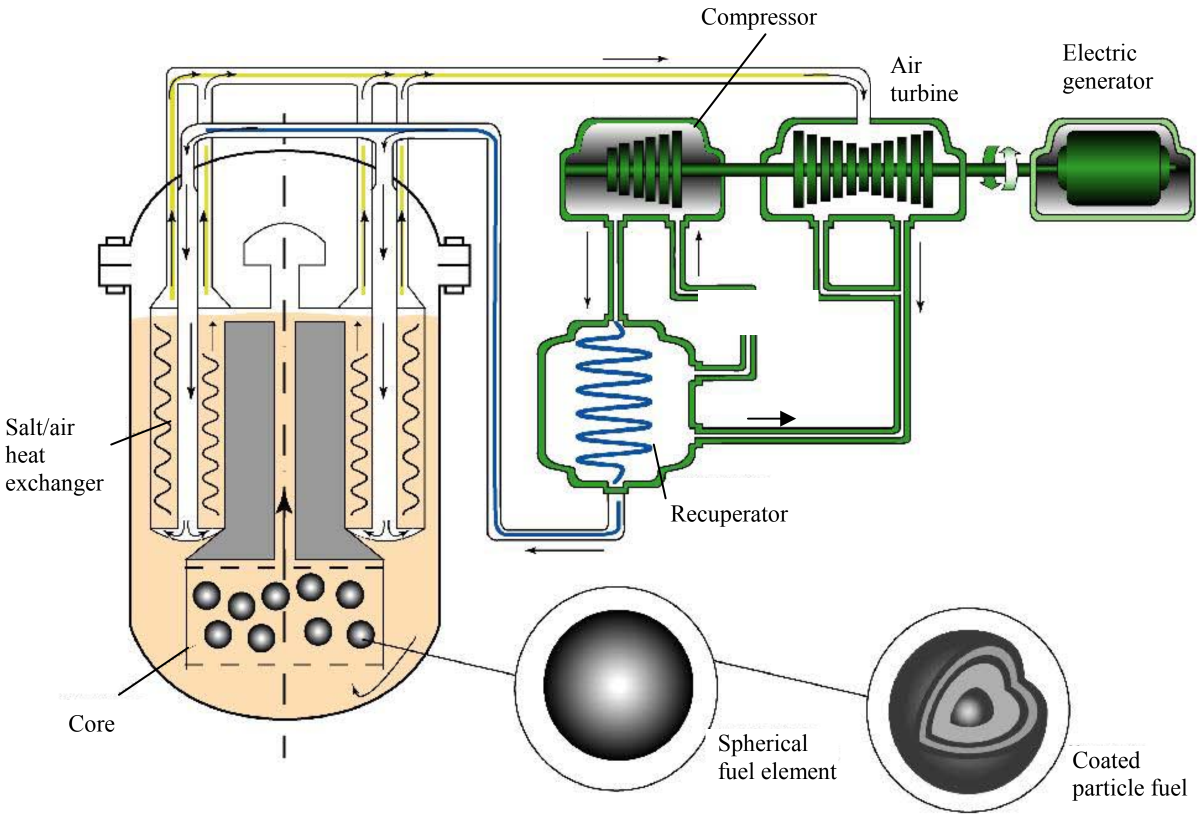


FIG. XXVIII-1. Simplified schematic diagram of MARS.

In the reactor module, molten salt coolant in the natural circulation mode comes upward through the active core and the chimney and enters the "salt-air" heat exchangers. After heat transfer to the air, the 'cold' coolant moves downwards through the annular side reflectors and the module bottom and returns to the active core. The working media of the secondary circuit (heated air) from the heat exchangers is directed to the gas-turbine unit through pipes in the upper head of the reactor module and, after the working cycle, in the recuperator, the residual heat is removed to the atmospheric air that enters the compressor from outside. The application of a once-through (open) air cycle with intermediate cooling in the process of air compression, intermediate heating in the process of air expansion, and heat recovery downstream of the turbine, allows achieving acceptable thermodynamic parameters without using water as a reject heat sink.

An additional path for the removal of a small amount of heat from the reactor is provided by organizing the free convection of the ambient environmental air in the annular gap of the double-walled reactor vessel. Under normal operating conditions, the application of such a technical feature ensures the formation of a thin layer of frozen salt (slag lining) on the vessel inner surface, protecting the vessel against corrosion. Under a loss of heat sink accident, this feature ensures passive removal of the residual heat.

Table XXVIII-1 presents a summary of the major design and operating characteristics.

TABLE XXVIII-1. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS

Major design characteristics	
Installed capacity: - Thermal; - Electric; - Thermal, in heat supply mode	 16 MW(th) 6 MW(e) 8.5 MW(th)
Fuel type	Spherical graphite fuel elements with coated fuel particles (spherical particles with UO_2 kernels and multi-layer ceramic coatings)
Fuel enrichment	10% by ^{235}U
Primary coolant/ circulation type	Fluoride-based molten salts/ natural circulation
Moderator	Graphite matrix of spherical fuel elements, fluoride-based molten salt
Structural materials	- Fuel particle coatings – pyrocarbon, silicon carbide. - Spherical fuel element matrix and cladding – graphite. - Reactor module – Hastelloy-N alloy
Core	Cylindrical fixed bed of spherical fuel elements, spherical absorber elements and graphite spheres; the effective diameter is 3.0 m; the height is 3.0 m. Reflectors – molten salt coolant
Reactor vessel	Double cylindrical vessel of a mono-block type with built-in primary circuit systems. Outer diameter of the reactor module – 4 m. Vessel height – about 10 m.
Number of circuits; thermodynamic cycle type.	For a base load electricity generation mode: two-circuit system with a gas turbine unit operating in an open air-cycle
NPP style	Modular, integral type
NPP operation mode	Basic option: electricity generation in a base load mode; off-peak electric power could be used to power various process applications.
Thermodynamic cycle efficiency	37%
Load factor	0.82
Service lifetime, years	60
Neutron-physical characteristics	
Reactivity coefficients: - Fuel temperature - Coolant temperature - Coolant density	 - 2×10^{-5} Δ(1/k)/k - 2.2×10^{-6} Δ(1/k)/k - 6×10^{-3} Δ(1/k)/ (g/cm^3)
Reactivity effects: - Total void - Burn-up reactivity swing over core lifetime	 -2 % Δ(1/k) - 3% Δ(1/k)

Neutron-physical characteristics (continued)	
Power peaking factors, averaged over fuel lifetime - Radial - Axial - Total	 K_r <1.2 K_z <1.3 K_v <1.6
Reactivity control mechanism; control and protection system	
Burnable poison	Spherical absorbing elements with B_4C absorber; B_4C content in spheres, total number of absorbing spheres in the fixed bed core, and the distribution of the absorber elements are selected to ensure burn-up flattening and minimum burn-up reactivity swing over the fuel lifetime
Control and protection system Worth of each CPS	Twelve control rods arranged in 'rings' coaxial with the core perimeter and grouped in three independent mechanical control and protection systems of 4 rods each; the absorber material is B_4C; each CPS system ensures the compensation of all reactivity effects and can control the operating reactivity margin. 9% $\Delta(1/k)$
Thermal-hydraulic characteristics	
Mode of primary coolant circulation	Natural
Mass flow rate of coolant in the core	29.4 kg/s
Coolant temperatures: - Core inlet / outlet, at normal operation - Lower limit (to avoid freezing) - Upper limit (boiling point)	 550 / 750°C 350°C 1300°C
Fuel temperatures (normal operation): - Maximum/ average - Limit	 1000 / 800°C 1250°C
Fuel temperatures (accidents): - Maximum/ average - Limit	 1300°C 1600°C
Burn-up cycle	
Fuel lifetime	15 or 60 years
Fuel load per a spherical element	7.90 or 31.58 g
Average / maximum burn-up of the discharged fuel	98 / 120 MW·day/kg of heavy atoms
*Non-electric applications**	
Hydrogen production	(8–10) 10^6 nm^3 H_2/year (using processes with a specific energy consumption of 4–6 kW-hour / nm^3 H_2)
Seawater desalination	1 10^7 t/year (using processes with a specific energy consumption of 5 kW-hour / ton of water)

Economics	
Specific capital costs: - Prototype - Commercial unit	 3500 US$/kW(e) 2500 US$/kW(e) Factors contributing to the minimization of capital cost are the following: - Full factory assembly of the reactor module; - Transportability of a ready-for-service reactor module.
Operation and maintenance (O&M) costs	A detailed estimate has not been performed yet. Factors that could contribute to the minimization of O&M costs are the following: • Reduced number of the operation personnel • The refuelling, repair and maintenance performed at a centralized factory.

* All estimates assume purposeful use of 50×10^6 kW-hour of the off-peak electric power generated annually.

Table XXVIII-2 presents basic characteristics of a small nuclear power plant with the MARS reactor for the core lifetime of 60 and 15 years, respectively.

TABLE XXVIII-2. BASIC CHARACTERISTICS OF THE MARS PLANT FOR TWO CORE LIFETIMES

PARAMETER	OPTION 1	OPTION 2
Reactor core and primary circuit		
Thermal power, MW	16	16
Fuel lifetime, years	60	15
Core diameter/height, m	3 / 3	3 / 3
Average core power density, MW/m^3	0.75	0.75
Fuel loading per spherical fuel element, g	31.58	7.90
Fuel enrichment by ^{235}U, weight %	10.0	10.0
Fuel burn-up, GW·day/t	98	98
Maximum fuel temperature, ^{o}C	1000	1000
Fluence of fast neutrons (E≥0.18 MeV) over the fuel lifetime, n/cm^2 - On a fuel element - On the reactor vessel	 2.1 10^{21} 1.0 10^{21}	 0.53 10^{21} 0.33 10^{21}
Primary coolant flow rate, kg/s	29.4	29.4
Coolant temperature at core inlet / outlet, ^{o}C	550 / 750	550 / 750
Secondary circuit		
Total number of heat exchangers (including the back-up ones)	21	6
Heat exchanger diameter/height, m	0.5/4.6	0.5/4.6
Air temperature, ^{o}C - Before the turbine - After the recuperator	 700 232	 700 232
Heat recovery factor	0.85	0.85
General parameters of the plant		
Reactor module diameter/height, m	4 / 10	4 / 10
Reactor unit weight, t	~ 171	~ 132

General parameters of the plant (continued)		
Weight of the gas turbine unit, including the electric generator, t	~ 26.4	~ 26.4
Thermodynamic cycle efficiency at the ambient air temperature of 0°C, %	37	37
Thermal power in heat supply mode, MW	8.5	8.5

XXVIII-1.5. Outline of fuel cycle options

The MARS nuclear power plant (NPP) is being developed as part of an innovative nuclear energy system considered by experts of the Russian Research Centre "Kurchatov Institute" for the future large-scale deployment of small reactors. The system includes the operating small nuclear power plants and the complete infrastructure chain supporting their deployment, operation, transportation, repair, maintenance, refuelling and decommissioning.

The energy system as a whole is schematically shown in Fig. XXVIII-2; it consists of the two parts:

- *External part*, which is a network of safe and simple small power plants requiring minimum qualified operation personnel; and
- *Internal part*, a part of the system closed from the outside world, where these plants are built, repaired, refuelled and reprocessed.

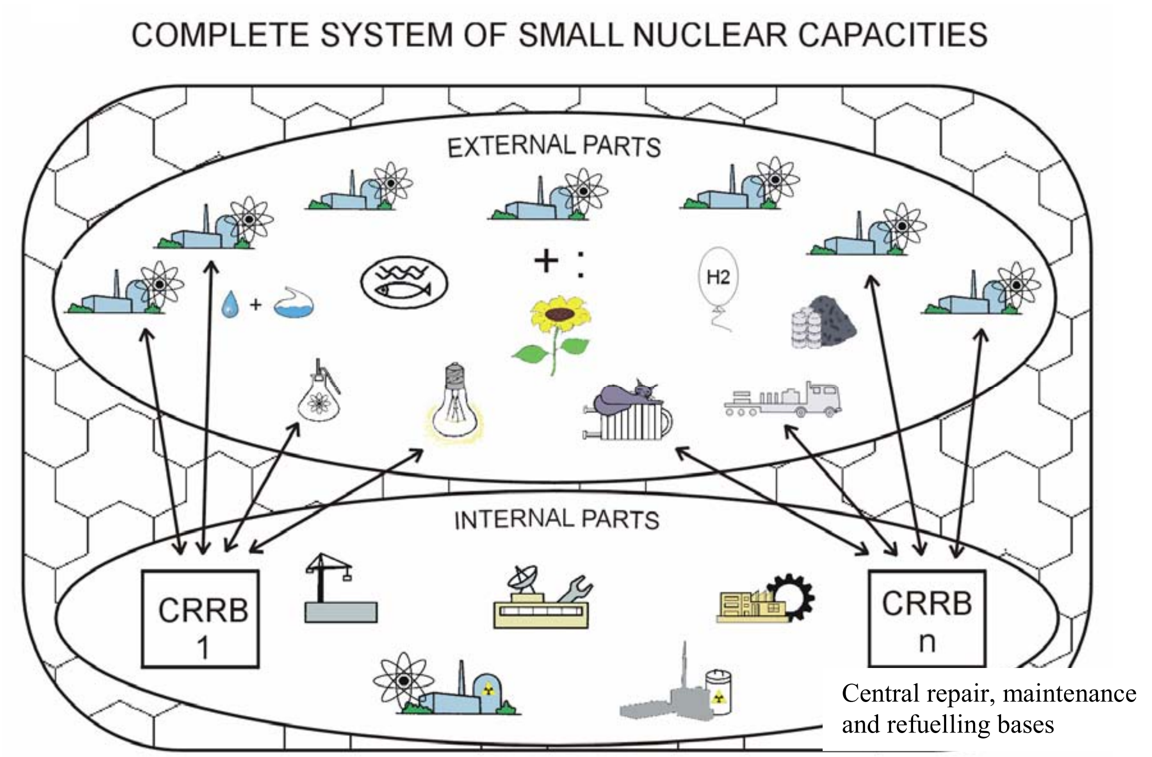


FIG. XXVIII-2. Concept of a nuclear energy system with small nuclear power plants and centralized repair, maintenance and refuelling bases.

Within such a system, all technically complex and radiation-hazardous activities related to fabrication of reactor modules, disposal of decommissioned NPPs and reprocessing of spent nuclear fuel are performed in the internal part, i.e., at specialized and centralized facilities operated under reliable security and proliferation control measures. In turn, the functions of these facilities are supported by a regional enterprise (central repair/maintenance and refuelling base – CRRB) providing a complete range of services to nuclear power sources located in different regions of a country or even in several countries.

Detailed elaboration and optimization of the structure of the abovementioned nuclear energy system with small nuclear power plants appears to be an individual complex task. It includes estimation of the relative capacities of production facilities, repair and maintenance, decommissioning and fuel cycle enterprises, including the facilities for radioactive waste reprocessing and transmutation, and selection of the appropriate technologies in view of the characteristics of the reactors constituting the external part of such nuclear energy system. It is anticipated, in particular, that the internal part of the system could include special reactors with circulating liquid fuel intended for burning of trans-plutonium actinides and transmutation of selected fission products.

At the moment, a standard once-through fuel cycle using low enriched (up to 10% of ^{235}U by weight) uranium fuel without spent nuclear fuel reprocessing is considered as basic for the MARS reactor. It is assumed that MARS spent nuclear fuel will be placed in long-term storage until a decision is made on further use of the energy potential of actinides contained in this spent fuel.

It is expected that in the future the MARS spent fuel will be reprocessed using technologies that are currently under development for such type of fuel (spherical graphite fuel elements with TRISO coated particles). Combinations of the promising methods of mechanical fuel extraction from ceramic compositions and of the existing aqueous reprocessing methods are being elaborated to be applied to spent nuclear fuel of the MARS type. Even more promising could be a fluoride volatilization method, which is being developed currently at RRC KI.

XXVIII-1.6. Technical features and technological approaches that are definitive for MARS performance in particular areas

XXVIII-1.6.1. Economics and maintainability

A possible domain for energy systems with small MARS type NPPs includes vast areas in the Near North and Far North of the Russian Federation. These are regions with numerous (up to 6000) isolated small consumers with a load demand of up to 3–5 MW each, not covered by interconnected or autonomous electricity grids. In view of this, the anticipated demand for autonomous small NPPs is estimated at several thousand units in the Russian market only.

The MARS concept incorporates provisions for reduced capital and construction costs as the following:

- The anticipated production scale for the MARS type reactors is that of several hundred units and, therefore, the learning factor may be taken into account duly when assessing the possibility to reduce capital costs;
- The MARS nuclear installation is devised as a “nuclear cell” and, therefore, its design provides for equipment prefabrication and transportability, altogether contributing to a reduction of the NPP construction costs;
- The MARS turbine-generator unit does not use water as a cooling or heat sink medium and, therefore, process water system and the associated costs are eliminated.

Compared to other power reactors, the MARS NPP might offer certain advantages contributing to a reduction of the operation and maintenance costs, among them:

- Elimination of fuel handling operations during the whole period of reactor operation on a site and transportation to and from the site;
- A substantially reduced number of operation personnel;
- An option of unattended operation, e.g., with monitoring and control via a satellite.

XXVIII-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The MARS concept does no provide for the use of scarce or expensive materials and, therefore, offers a prerequisite for a sustainable upscale of a nuclear energy system on its basis. The operation does not involve consumption of non-renewable resources except for uranium, and the reactor has relatively good characteristics of fuel utilization.

It is anticipated that MARS operation would produce a minimum impact on the environment because harmful substances are not released in inadmissible amounts, natural water resources are not used, heat emissions to the environment are acceptable due to a relatively high efficiency, and chemical pollution of the environment is excluded by design.

As no fuel handling operations are performed on the site, liquid and solid radioactive wastes are not accumulated during the MARS operation.

Because the MARS reactor uses coated particle fuel in a graphite matrix that provides several protective barriers to prevent radioactivity release to the environment, it is anticipated that the exposure of personnel and population during the operation will be negligible.

If the concept of unattended operation is realized, the possibility of radiation impacts on NPP personnel would be essentially reduced.

XXVIII-1.6.3. Safety and reliability

Safety concept and design philosophy

The MARS safety concept provides for retaining of radionuclides in the fuel both in normal operation and accidents, so that radiation exposure of the personnel and population around the NPP falls within the limits prescribed by the regulations. Barriers to the release of fission products from fuel are provided primarily by the coatings applied to each fuel particle. The required retention of radioactivity is ensured by the long-term operability of the fuel elements, which maintain their performance at temperatures up to ~1250°C with the fast neutron (E>0.18 MeV) fluence to the coatings of up to ~2.2 10^{21} cm^{-2}, and by quality assurance during fuel fabrication.

Provisions for simplicity and robustness of the design

The simplicity and robustness of the MARS design under long-term operation is ensured by an appropriate selection of the structural materials and by the incorporation of certain inherent and passive safety features in the original design concept.

Active and passive systems and inherent safety features

The MARS is being designed to incorporate an optimum combination of inherent and passive safety features and engineered active and passive safety systems.

The inherent safety features of the MARS are in many respects defined by the used combination of coolant and fuel properties, as well as by the selection of a working medium in the power conversion circuit.

In particular, the MARS reactor is characterized by the following inherent safety features that reduce potential hazards and prevent the initiating events from progressing into accidents, relying just on the laws of physics:

- Low specific power of the core, low stored energy of the non-pressurized coolant;
- The use of a fuel type that effectively retains radionuclides;
- Reliance on natural circulation in all operation modes;
- Negative reactivity coefficients on temperature and negative reactivity effects.

MARS incorporates the following passive safety systems:

- A double-walled vessel is used to prevent loss of coolant accidents; a free convective flow of air is organized in the annular gap between the walls, which in normal operation facilitates formation of a thin layer of slag lining on the inner surface of the vessel;
- Upon a loss of heat sink to the secondary circuit, the reactor vessel cooling system (passive) ensures that the residual heat and heat removed to the environment are in equilibrium.

Structure of the defence-in-depth

Similar to other nuclear installations, the defence-in-depth concept incorporated in the MARS provides for multiple barriers to radioactivity release from the fuel and for measures to maintain the integrity of these barriers. Such a barrier structure largely leans upon the known properties of the fuel (spherical fuel elements with coated particles), i.e., the retention of a large amount of radionuclides in a ceramic fuel kernel and the prevention of radionuclide release to the coolant by the fuel particle coatings. The graphite matrix of fuel elements that has an ability to absorb certain radionuclides facilitates a reduction of radioactivity release to the coolant. A two-circuit plant scheme provides an additional barrier to radioactivity release to the environment.

Design basis and beyond design basis accidents

At the present stage of development, the list of design basis and beyond design basis accidents for the MARS was defined from operating experience of existing NPPs and is not yet final. Probabilistic analysis of the entire range of possible events has not yet been performed. Typical events initiating a development of transient processes are the loss of heat removal to the secondary circuit and the insertion of positive excess reactivity.

The maximum design basis accident (MDBA) considered for the MARS is an accident with loss of heat removal in all heat exchangers without operation of the emergency shutdown system. The evolution of such an accident results in a decrease of the salt coolant flow rate, in an increase of the salt temperature, and in the actuation of a negative reactivity feedback on coolant and fuel temperature. After some time, the reactor will start to cool down due to reactor vessel cooling by the outside air and through the heat exchanger of the circuit of the stopped gas-turbine unit. The fuel temperature will not exceed the operating limit (1250–1300^{o}C), and the salt temperature would be kept below the boiling point (~1300^{o}C). Estimates show that no other special systems for emergency core cooling will be required - these functions are effectively carried out by the design features ensuring natural circulation of the outside air around the reactor vessel and in the circuit of the stopped gas turbine unit.

A beyond design basis accident (BDBA) considered for the MARS is the complete loss of organized heat removal in all heat exchangers and from the reactor vessel, without the operation of the emergency shutdown system. Accident progression is similar to that of the MDBA but due to the absence of heat removal, the fuel and coolant temperatures increase substantially (the temperature difference between the salt and the fuel will not exceed ~5^{o}C). The time to the start of coolant boiling is estimated at ~500–700 hours. Measures to regain reactor unit cooling shall be taken within this long grace period. In this scenario, radiative and conductive heat removal to the environment becomes comparable to the residual heat generation.

Provisions for safety under seismic conditions

Seismic design of the MARS will be performed in compliance with the procedures prescribed by the regulations. It could be noted, however, that the use of a gas-turbine unit facilitates the assurance of plant seismic resistance, as the specific mass of gas-turbine power conversion systems is ~3–4.5 kg/kW and the specific volume is ~0.04–0.17 m^3/kW, which is 10 times smaller than for the most updated steam turbine plants.

XXVIII-1.6.4. Proliferation resistance

Technical features of the MARS reactor contributing to its enhanced proliferation resistance include the following:

- No refuelling performed on the site during the whole period of operation;
- The reactor vessel is sealed and cannot be opened except for operations at a specialized enterprise;
- The undeclared production of weapon-grade nuclear materials in the reactor is essentially precluded by the thermal neutron spectrum, as well as by the associated necessity to change the design configuration of the distribution of fuel, absorber and graphite elements over the core volume, which, as mentioned above, can be performed only at a specialized factory with the use of the equipment that is not available on the site;
- The fuel type used, fuel particles with multi-layer coatings in a graphite matrix, makes it impossible to extract fuel materials without using a special technology for the removal of the coatings; such a technology can be safely implemented only at a specialized enterprize for the reprocessing of a spent nuclear fuel of such type;
- The quality of fissile materials that can potentially be extracted from the MARS fresh or spent fuel is inadequate to produce weapons and will inevitably require additional fuel operations, such as enrichment.

The abovementioned features of the MARS also facilitate control, protection and accounting of nuclear materials, which, according to the design, can be effectively implemented when handling the fresh and spent nuclear fuel at central repair, maintenance and refuelling bases within the "internal" part of a nuclear energy system shown in Fig. XXVIII-2.

XXVIII-1.6.5. Technical features and technological approaches used to facilitate physical protection of MARS

An improved protection against external impacts and internal human-induced actions of malevolent character is in many respects ensured in the MARS concept through the use of fuel in the form of graphite fuel spheres with coated fuel particles that cannot be easily broken mechanically and have a good radioactivity retention capability even in severe accidents.

The strategy of power operation, which is a continuous base load operation mode, offers advantages in obviating the non-foreseen transients. Transient processes triggered by initial events during the NPP operation progress rather slowly, given the low specific power, a considerable heat capacity of the primary circuit, and the passive heat removal from the reactor vessel; this allows a radical increase in time before active involvement to manage the accidents becomes necessary.

In the requirements for protection against external impacts are more stringent, the nuclear installation could be deployed underground.

XXVIII-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of MARS

Market demands and specific needs of certain markets were taken into account in the conceptual design of the MARS, in particular, by developing alternative energy applications that could complement or substitute for the basic option of electricity generation.

Assessments [XXVIII-5, XXVIII-6] show that the potential markets for small nuclear power plants with MARS type reactors already exist both in the Russian Federation (including district heating and other non-electric applications) and abroad, e.g. in certain developing countries that offer a large market niche for combined power supply and seawater desalination.

Various arrangements for leasing of fuel or reactor installation as a whole, transfer of ownership arrangements, etc. are assumed acceptable for a small NPP with the MARS reactor.

In particular, a possible option for developing countries is leasing of the MARS reactor installation on condition and with a guarantee of its return for disposal to the country of manufacture. Such a leasing scheme is fully consistent with the concept of a nuclear energy system with small NPPs consisting of the “external” and “internal” parts, see Fig. XXVIII-2.

Full-scope fuel cycle servicing could be provided also, using central repair, maintenance and refuelling bases located in the country of origin.

XXVIII-1.8. List of enabling technologies relevant to MARS and status of their development

The enabling technologies of MARS that require further validation and testing are as follows:

(1) Technologies ensuring reliability of the reactor structures and components under conditions of a very long lifetime; provisions for the necessary research and development (R&D) are the following:

- In many respects, experience will be used in the development and operation of the fuel of a selected type, which is essentially a high temperature gas cooled reactor (HTGR) fuel; here, the experience in design, in fuel performance under irradiation, and in radioactivity release to the coolant, available for such fuel, becomes important;
- It is necessary to systematize experimental data on the transport of fission products by molten salt coolant, on dissolution of fission products in the coolant, and on interactions between fission products and structural materials (reactor vessel, heat exchange surfaces, etc.);
- Possibly in the near term, it is necessary to perform a series of tests to validate and demonstrate the performance of a “graphite – molten salt – structural materials” combination;

- It is necessary to examine and optimize the conditions of formation of a protective layer (slag lining) on the inner surfaces of the reactor vessel;

(2) It is necessary to develop the method and technique for compensation of reactivity due to fuel burn-up; and

(3) The technologies for safe freezing / de-freezing of the molten salt coolant need to be developed and mastered.

XXVIII-1.9. Status of R&D and planned schedule

At present, conceptual design for the MARS plant is being developed by an expert team of the Russian Research Centre "Kurchatov Institute" (RRC KI, Moscow, Russian Federation). The main R&D is focussed on the following:

- Optimization of the dimensions and structure of the reactor core and reflectors to achieve a targeted set of design and operating characteristics and passive safety features;
- Development of physical models and codes to validate safety performance of the nuclear installation.

At the current early conceptual design stage, the detailed scope and schedule of future works and the list of organizations and institutions to be involved at further phases of the MARS design development are not yet elaborated, although some preferences already exist.

Interactions with the OKBM (a design organization based in Nizhny Novgorod, the Russian Federation) have been started to make preparations for a preliminary design development, which would be preceded by an additional analysis of the adopted conceptual design features. It is anticipated that this work could be carried out, e.g., under a two-year project financed by the International Science and Technology Centre (ISTC); the proposal for such project has already been submitted.

The time frame within which the design development could be completed is conditioned by the availability of funding. It is estimated that research, design and demonstration (RD&D) for the detailed design development may take 5 to 8 years at a total cost of US$ 50 million. Construction of a prototype will cost at least US$ 20 million.

XXVIII-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The MARS concept is largely based on proven technical features; however, the reactor is rated as innovative because it uses a non-conventional combination of such technical features, incorporates a strong reliance on inherent and passive safety features, and provides for a long-life core operation without on-site refuelling.

Basic design features of the MARS nuclear installation, to be verified with a prototype / pilot plant, include:

- Cooling of the spherical fuel elements by a molten salt coolant; corrosion stability of the fuel and vessel structures, properties of the heat transfer from fuel to coolant and from coolant to the secondary circuit working medium need to be confirmed;
- Reliability of fuel and structures in the conditions of a very long fuel lifetime, under the associated coolant chemistry;
- Reliable performance of the passive systems removing heat from the double reactor vessel, etc.

To make final decisions on the selection of operating parameters, a full-scale pilot plant would be built and used for testing of, *inter alia*, simplified control schemes, including a remote control for unattended operation.

XXVIII-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The AHTR concept [XXVIII-7] proposed by ORNL (USA) uses coated particle graphite-matrix fuel and a molten-fluoride-salt coolant. The principal difference from the MARS concept is that a larger power output (2400 MW(th) and more) is considered as one of the enabling technologies to improve the AHTR economics.

XXVIII-2. Design description and data for MARS

XXVIII-2.1. Description of the nuclear systems

Reactor core and fuel design

The fixed bed core appears as a cylinder 3 m high and 3 m in diameter; it is formed by the spherical fuel elements and absorber and graphite elements. The fixed bed core configuration does not change during core lifetime and enables optimum flattening of the distributions of power and fuel burn-up over the core volume during the whole fuel lifetime. The parameters of spherical elements used to assemble this fixed bed core are given in the Table XXVIII-3.

TABLE XXVIII-3. PARAMETERS OF SPHERICAL ELEMENTS USED IN MARS CORE

Spherical fuel element		
Diameter, mm:	- Outer	60
	- Fuel part	50
Graphite density, g/cm^3:	- In fuel matrix	1.65
	- In outer (cladding) layer of the fuel element	1.65
Coated particles		
Kernel diameter, μm		500
Density of UO_2 in the kernel, g/cm^3		10
Fuel enrichment, % by weight		10
Thickness of the first layer coating, μm		30
Density of PyC in the first layer, g/cm^3		1
Thickness of the second layer coating, μm		50
Density of PyC in the second layer, g/cm^3		2
Thickness of the third layer coating, μm		40
Density of SiC in the third layer, g/cm^3		3.2
Thickness of the fourth layer coating, μm		40
Density of PyC in the fourth layer, g/cm^3		2
Absorber element		
Diameter, mm		45
Absorber material		B_4C
Loading of absorber material, g		0.1
Graphite element		
Diameter, mm		35
Graphite density, g/cm^3		1.65

No operations with fuel are performed during the entire reactor lifetime; therefore, there is no storage capacity for fresh or spent fuel elements on the site. Different from pebble bed high temperature gas cooled reactors (HTGRs) and previous high temperature molten salt cooled reactors with HTGR type fuel (abbreviated as VTRS in Russian), the MARS concept incorporates no pebble transport.

At present, two options of core arrangement for the MARS are being considered for fuel lifetimes of 15 and 60 years, respectively.

Spherical fuel elements used in the MARS concept have undergone various technological and reactor tests. In particular, they were validated for use in the VG-400 and VGM high temperature gas cooled reactors that were under development in the Russian Federation previously, as well as for the HTR MODUL reactor in Germany. In addition to this, spherical fuel elements based on coated particles were successfully operated in the reactors of the Peach Bottom and Fort St. Vrain NPPs in the USA and in the AVR and the THTR-300 reactors in Germany.

The following basic design features were adopted for the MARS fuel (see Fig. XXVIII-3):

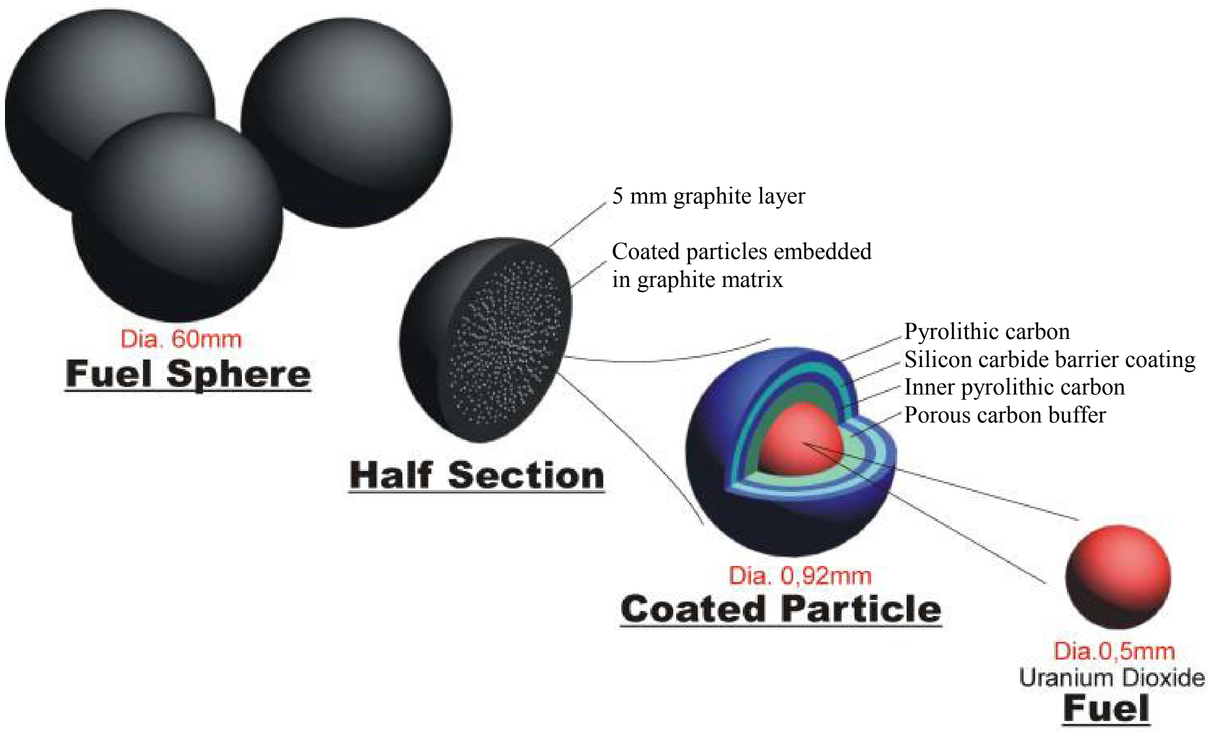


FIG. XXVIII-3. Illustration of the MARS fuel concept.

- A two zone spherical fuel element of 60 mm diameter consists of a graphite matrix with fuel in the form of coated particles and a 5 mm thick cladding of dense graphite. The average fuel element density is ~1700 kg/m^3, which is less than the density of a molten salt coolant at the operating temperature;

- Each coated particle includes an outer coating of isotropic pyrocarbon, a layer of silicon carbide, an inner layer of pyrocarbon, an inner buffer layer of porous pyrocarbon, and a fuel kernel.

Control and protection system

The control and protection system (CPS) consists of 12 absorber rods arranged in 'rings' coaxial with the core perimeter and grouped into three independent mechanical protection systems of 4 rods each. Each mechanical protection system compensates all reactivity effects and controls the operating reactivity margin. The CPS members move in graphite guiding tubes. The absorber material of the CPS rods is boron carbide (B_4C). Cooling is performed by the coolant flowing in the gaps between the graphite tube and the Hastelloy coating of an absorbing sleeve, as well as inside the internal sleeve coating.

A horizontal section of the MARS core with indication of control rod and control rod group positions, and the design of a CPS rod are shown in Fig. XXVIII-4; (1), (2), and (3) are numbers of independent mechanical protection systems.

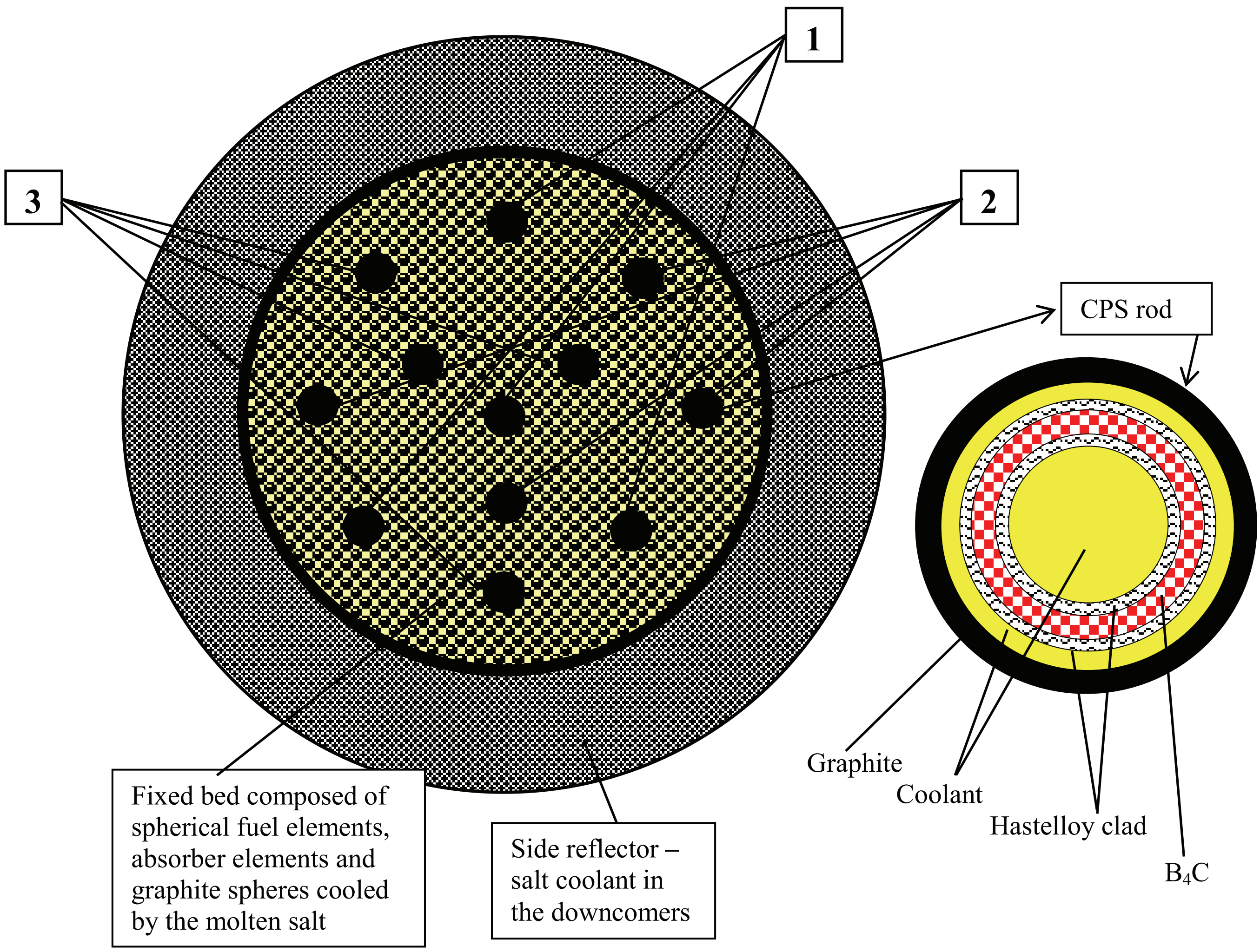


FIG. XXVIII-4. Core map with control rod positions (CPS groups are marked by figures) and the design of an individual CPS rod.

Coolant and structural materials of the core

The fuel-free molten salt coolants considered for the MARS are those for which basic thermo-physical properties are well known (see Table XXVIII-4).

TABLE XXVIII-4. MOLTEN SALT COOLANTS CONSIDERED FOR USE IN MARS

SALT	*NaF-BeF₂*	*LiF-BeF₂*
Molar composition, mol. %	57-43 – eutectic	48-52 - eutectic
Melting temperature, ^{o}C	360	350
Specific heat, J/(kg K)	2172	2720
Thermal conductivity, W/(m K)	1	1.19
Density, kg/m^3	2270-0.37 T	2220-0.4 T
Dynamic viscosity, n s m^{-2}	$3.46 \cdot 10^{-5}exp(5164/T)$	$1.89 \cdot 10^{-5}exp(6174/T)$
Saturated vapour pressure	≤133.3 Pa at T≤800°C <130 kPa at T≤1300°C	Lg (P_{sv})=9.44-10130/T

Fluoride based molten salts are characterized by the following features:

- Good compatibility with graphite over a wide temperature range (up to ~1200–1300°C); graphite becomes virtually non-flammable in air when impregnated with these salts;
- High chemical and radiation resistance, which ensures chemical inertness with respect to in-vessel and ambient materials (water, air), as well as high admissible neutron flux density in the reactor core;
- A small amount of stored chemical and mechanical energy;
- A wide temperature range (up to T≈1300°C) within which the salt coolant retains liquid state with the saturated vapour pressure less or equal to atmospheric pressure; this feature makes it possible to eliminate high pressure vessel for the range of operation temperatures of 700–850°C (the internal pressure on the reactor vessel is effected only by the liquid salt coolant height – hydrostatic pressure);
- Molten salt coolants are transparent and have approximately the same thermal conductivity as water; their thermo-physical properties ensure effective heat removal with natural convection, with a “salt-to-wall” heat transfer coefficient close to that of water;
- The high melting point of salt coolants complicates the start-up and operation of a nuclear installation; however, the same feature offers certain benefits related to the formation of a salt slag layer on the inner surface of the reactor vessel, which minimizes corrosion interactions between the vessel and the circulating salt coolant; retention of the salt in the reactor at hypothetical leaks also improves, as well as operating conditions of the armature based on freeze stop valves;
- Neutron-physical parameters of the salt coolants make it possible to use them effectively both as neutron moderators and reflectors; thermo-physical and neutron-physical properties of $NaF\text{-}BeF_2$ salt are slightly worse than those of $LiF\text{-}BeF_2$, but tritium production using this salt in the reactor is considerably smaller; to improve the neutron balance and reduce the tritium production, the initial enrichment by ^{7}Li shall be at least 99.999%.

The following structural materials are being considered for the MARS reactor:

- Cr18Ni10Ti steel – up to ~ 650°C;
- EI-726 and EP-164 steels – up to ~ 750°C;
- Hastelloy-N – up to ~ 850°C.

Specifically, Hastelloy-N has been specially developed as a basic structural material for reactors with molten salt coolants. Corrosion resistance of this alloy is defined by the presence of the impurities in salt compositions, such as soluble oxides, traces of moisture, fission products, etc.

When LiF-BeF_2 and NaF-BeF_2 fuel-free salt compositions are used, corrosion of the Hastelloy-N is many times less than in a fuel salt. The alloy is not prone to inter-crystalline corrosion, which is elsewhere typically effected by fission products, primarily, tellurium. For Hastelloy-N, the allowable fast neutron (E>0.5 MeV) fluence is greater than 10^{21} cm^{-2}, and the thermal neutron fluence - ~ $5 \cdot 10^{21}$ cm^{-2}.

Primary coolant system

A schematic diagram of the MARS primary coolant system is given in Fig. XXVIII-5.

The molten salt coolant natural circulation loop includes the reactor core filled with spherical fuel elements and absorber and graphite elements; the top, bottom and annular side reflectors; the draught section (chimney); and salt-air heat exchangers. The side reflector material consists of the circulating molten salt coolant.

A compensator tank with molten salt is placed above the draught section. The salt-air heat exchangers (main and back-up) are arranged along the reactor vessel above the core.

The reactor vessel cooled on the outside by natural air circulation, has no side inlets or outlets. All of them are located only in the reactor upper head (CPS rod drives, input and output air pipelines, etc.). A double vessel is used for safety and to ensure that a thin layer of salt (slag lining) protecting the vessel against corrosion is frozen on the vessel inner surface. To reduce the neutron flux incident on the vessel, shielding with a neutron absorber, for example of boron-containing steel, can be installed along the vessel surface.

The projected service lifetime for the reactor core, vessel and equipment is 60 years.

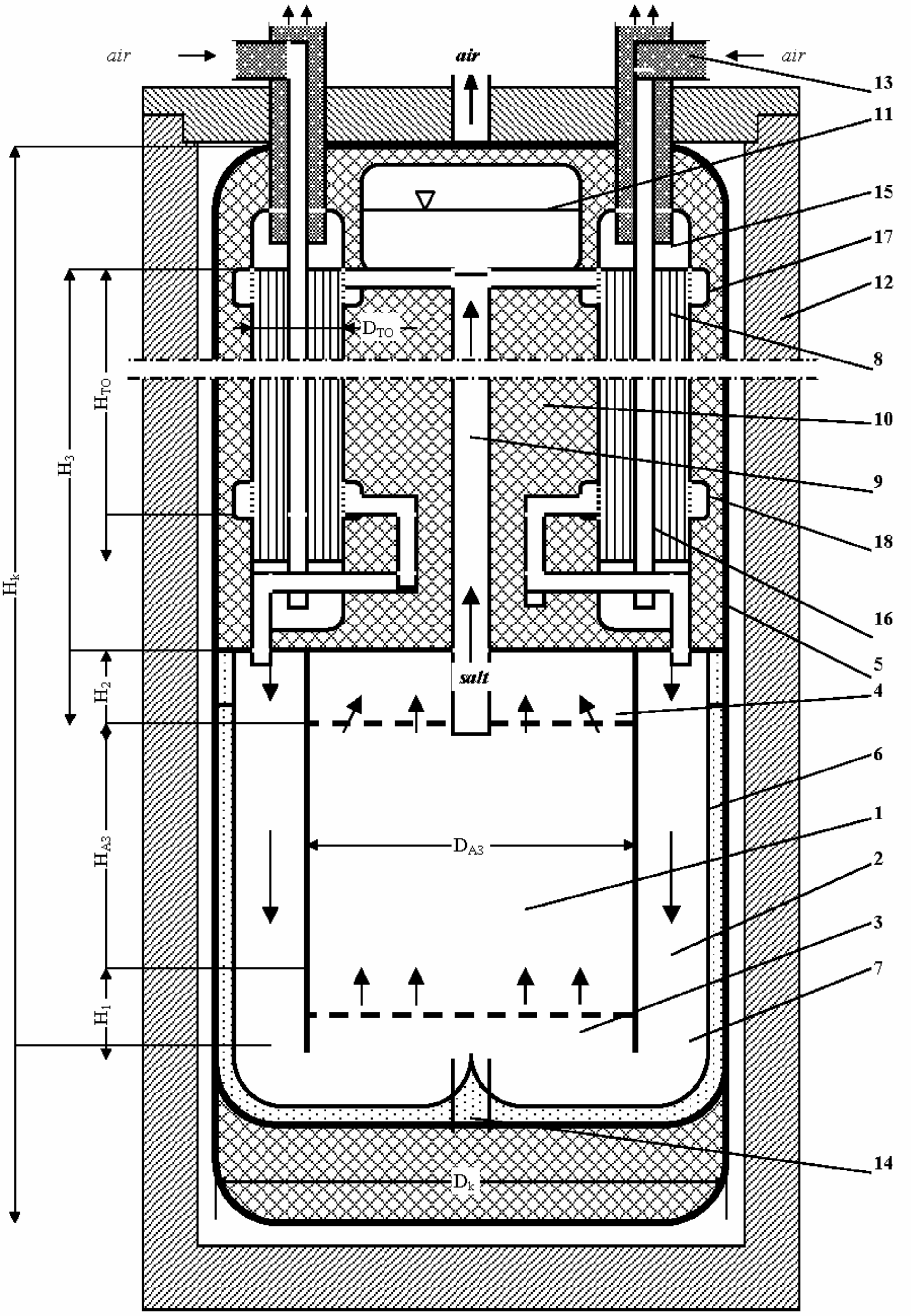
Main heat transport system

The function of the main heat transport system is to remove nuclear heat generated in the spherical fuel elements using natural convection of the molten salt coolant in all operation modes, i.e., under normal operation and in accidents. Fig. XXVIII-6 shows a schematic diagram of heat removal paths for the MARS plant.

The design objectives for the heat removal systems are to achieve a stable natural circulation of the coolant and to ensure that the design limits for coolant temperature (1300°C) and fuel temperature (1600°C) are not exceeded in all regimes. An auxiliary system is provided for by the plant design to heat up the coolant up to its melting point of 350°C and keep the temperature at this level during scheduled hot shutdowns.

Heat from spherical fuel elements is transferred by natural convection of the molten salt coolant in the riser to the “salt-air” heat exchangers, where, in turn, it is transferred to the environmental air serving as a working medium of the secondary circuit. The molten salt coolant never exits the vessel boundary. Heated air is directed to the gas-turbine unit through the pipes located on the cover of the reactor mono-block.

In case of a failure of air circulators in the secondary circuit, a relative share of the heat removed directly from the vessel by natural convection of air through the gap between the walls of the double vessel will increase, while the gas turbine circuit would remove the remaining heat, owing to the natural circulation of air.



1 – Core; 2 – Radial reflector; 3 – Bottom reflector; 4 – Top reflector; 5 – Reactor vessel; 6 – Protective shielding; 7 – Slag layer on the reactor vessel inner surface; 8 – Salt-air heat exchanger; 9 – Draught section (chimney); 10 – Displacer; 11 - Compensator tank; 12 – Biological shielding; 13 – Air inlet to the salt-air heat exchanger; 14 – Air inlet for reactor vessel cooling; 15 – Upper air header of the salt-air heat exchanger; 16 – Lower air header of the salt-air heat exchanger; 17 – Inlet salt header of the salt-air heat exchanger; 18 – Outlet salt header of the salt-air heat exchanger (D_{A3} – Core diameter; H_{A3} – Core height; H_1 – Bottom reflector height; H_2 – Top reflector height; H_3 – Riser height; H_{TO} – Salt-air heat exchanger height; D_{TO} – Salt-air heat exchanger diameter; H_v – Reactor vessel height; D_v – Reactor vessel diameter.

FIG. XXVIII-5. Schematics of the MARS primary coolant system.

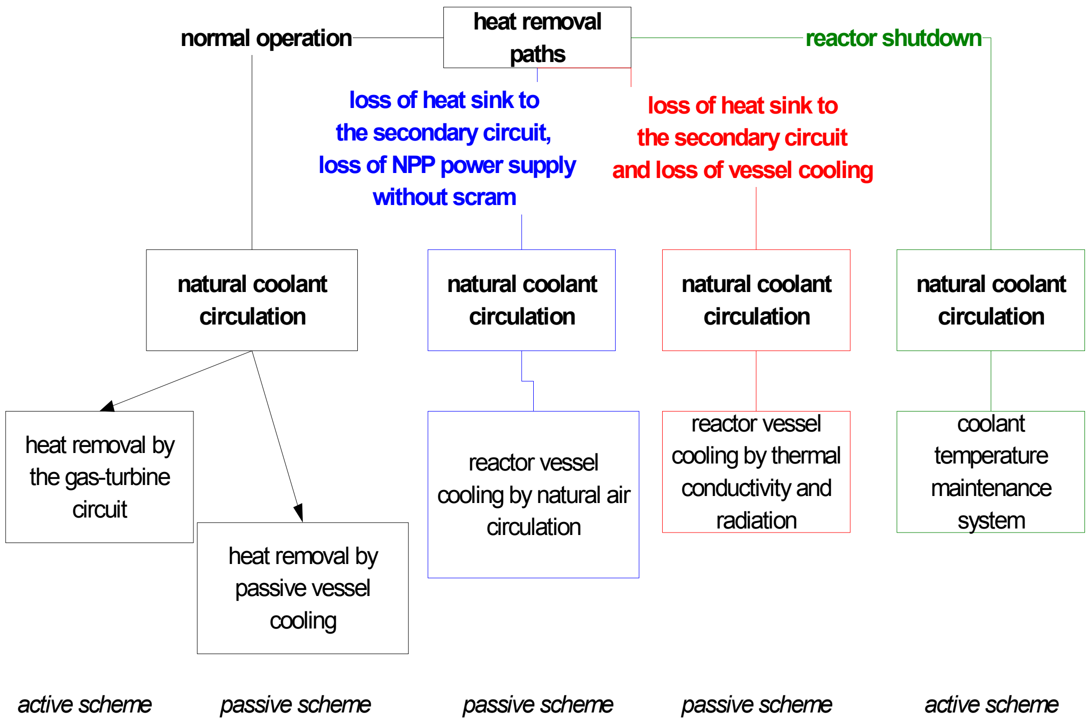


FIG. XXVIII-6. A schematic of the MARS heat removal system.

In the case of a total loss of heat sink to the secondary circuit, heat removal directly from the vessel will play the major role in achieving a balance between the decay heat generated in the core and passive heat removal to the environment. In a hypothetical failure of this passive path, the vessel will be cooled by thermal conduction and radiation from its surface.

XXVIII.2.2. Description of the turbine generator plant and systems

The MARS plant incorporates a gas turbine unit with a once-through (open) air cycle, intermediate cooling in the process of air compression, intermediate heating in the process of air expansion, and heat recovery downstream of the turbine.

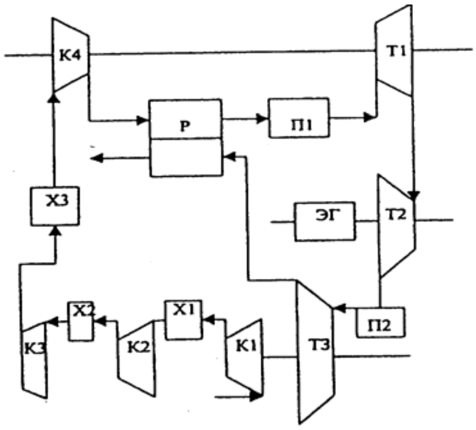
Such gas turbines allow using built-in fossil fuel or hydrogen combustion chambers as an alternative heat source, which could be of particular benefit for the power plant start-up, or scheduled shutdown, or in accidents.

According to the analyses and design studies of the gas turbine units applied within various thermodynamic cycles and schemes, the plant efficiency may vary between 31 and 52% depending on the selected scheme and the inlet air temperature, even if the air is heated up to the same temperature of 700°C.

A scheme of the turbine generator plant selected for the current version of a small NPP with the MARS reactor (intended for electricity generation) is shown in Fig. XXVIII-7.

XXVIII.2.3. Systems for non-electric applications

The currently considered basic design version of the MARS does not provide for incorporation of special systems for non-electric applications in the plant itself. It is anticipated that the off-peak electric power generated by the plant operating in a base load mode could be used to power various process applications, as indicated in Table XXVIII-1.

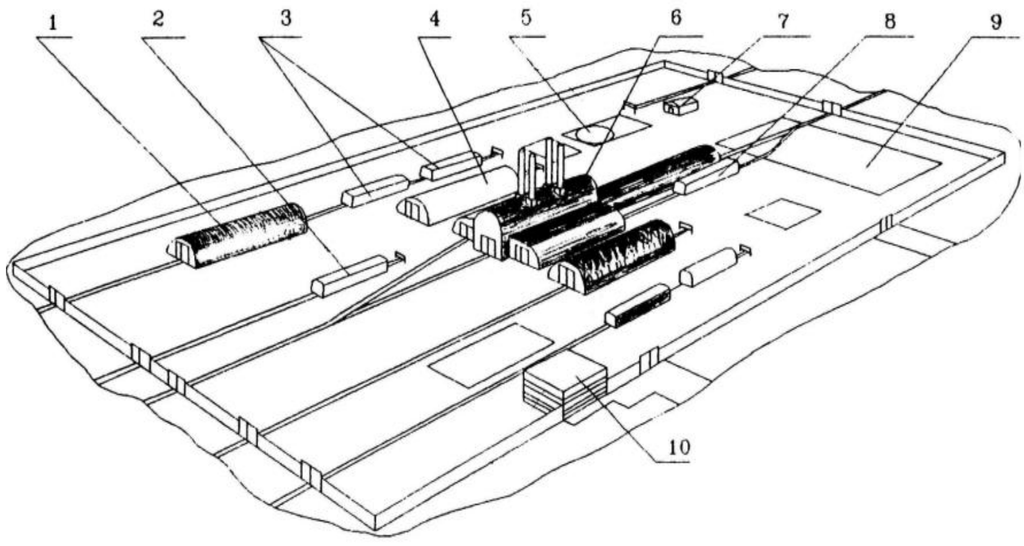


K1, K2, K3, K4 – Compressors; X1, X2, X3 – Intermediate coolers;
T1, T2, T3 – Turbines;
П1, П2 – Intermediate heaters; P – Recuperator; ЭГ – Electric generator

FIG. XXVIII-7. Schematic of the MARS power cycle and secondary circuit systems.

XXVIII.2.4. Plant layout

A general layout of the MARS plant, as proposed for a coastal site, is shown in Fig. XXVIII-8.



1 - Hydrogen production department; 2 - Terminal; 3 - Auxiliary systems;
4 - Main control room; 5 - Fire extinguishing system; 6 - Main building;
7 - Storehouse; 8 - Repair shops; 9 - Area for co-generation facilities;
10 - Office building

FIG. XXVIII-8. Layout of the MARS plant for a coastal site.

In addition to placing the main plant buildings, space is provided for other buildings; this enables the location of special systems for non-electric applications, supposed to operate with the plant electricity in the periods when the major consumer (local electricity grid) is shut off. The plant layout is designed with necessary zoned security arrangements to guard against unauthorized entry.

REFERENCES

[XXVIII-1] ALEKSEEV, P.N., et al., Micro-particle fuel autonomous molten salt reactor (MARS) for small power plants. "Atomnaya Energiya", v.93, iss. 1, p. 3–13 (July 2002).

[XXVIII-2] PONOMAREV-STEPNOY N.N., et al., Micro-particle fuel autonomous molten salt reactor (MARS), PHYSOR 2002, (Proc. Int. Conf., Seoul, Oct. 7–10, 2002) paper D0813.

[XXVIII-3] UDJANSKIY Y.N., et al., MARS reactor – a long-lived "nuclear cell", Russian Nuclear Society Journal, vol. 5-6, p. 22, (2003).

[XXVIII-4] NOVIKOV, V.M., et al., Nuclear reactors of enhanced safety (analysis of conceptual designs), Moscow, Energoatomizdat, ISBN 5-283-03817-3 (1993).

[XXVIII-5] BELOV, I.A. et al., Electricity and hydrogen co-generation using a small autonomous molten salt reactor (of the MARS type) in a system of industrial complexes, Scientific and Practical Conference "Small Power Systems 2002" (Proc. of Int. Conf. Moscow, 2002), p. 477 (Moscow 2002).

[XXVIII-6] SCHEPETINA T.D., et al., Economic effectiveness of the "MARS nuclear cell" as applied to various regions in view of co-generated products, Scientific and Practical Conference "Small Power Systems 2002" (Proc. of Int. Conf. Moscow, 2003), p. 15 (Moscow 2003).

[XXVIII-7] FORSBERG, C.W., et al., Molten-salt-cooled advanced high-temperature reactor for production of hydrogen and electricity, Nuclear Technology, **144**, pp. 289–302 (2003).

ANNEX XXIX

COMPACT HIGH TEMPERATURE REACTOR (CHTR)

Bhabha Atomic Research Centre (BARC),
India

XXIX-1. General information, technical features and operating characteristics

XXIX-1.1. Introduction

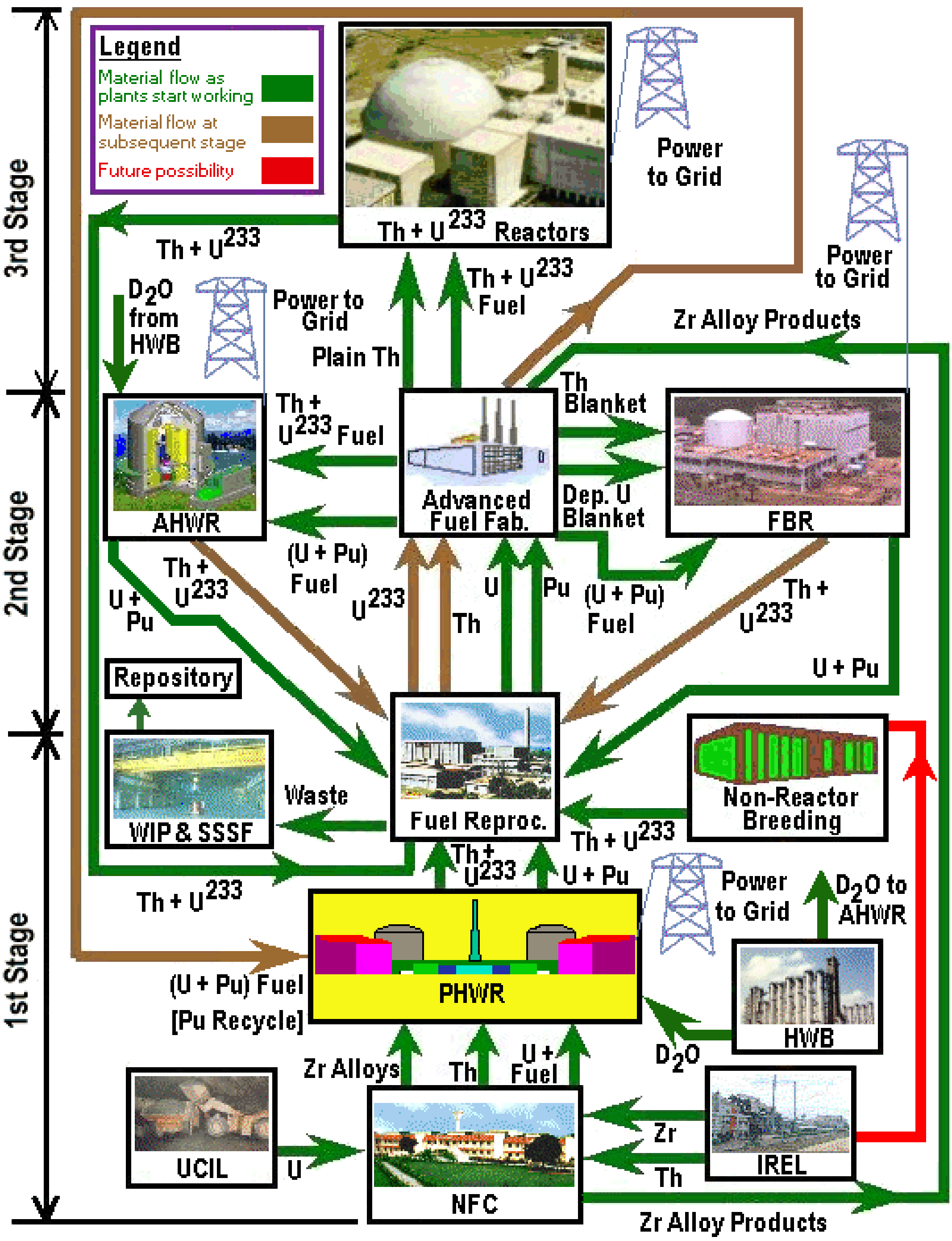
CHTR is an abbreviation for the Compact High Temperature Reactor.

The Indian nuclear programme and its priorities have been primarily defined by an incentive to attain long-term energy security. While formulating this programme more than forty years ago, it was recognized that to meet the long-term large scale concentrated energy needs with known technologies, a thorium based closed nuclear fuel cycle was the only sustainable option. The availability and geographical distribution of energy resources and, in the case of fossil fuel, its environmental impact, eventually limit all other options. The well-known three stage Indian nuclear power programme, shown in Fig. XXIX-1, was drafted accordingly. Considering its position with respect to domestic energy resources, the Indian priority for utilization of thorium has been high.

The first stage of the Indian nuclear power programme has now evolved as a full-fledged commercially successful activity and is an excellent example of the contributions of indigenous R&D to national development. Thirteen pressurized heavy water reactors (PHWR) are currently operating with excellent performance and several others are under construction and planning. Indigenous development and the assimilation of fast breeder reactor (FBR) technology has already culminated in the design of the prototype fast breeder reactor (PFBR) [XXIX-1], which is under construction. This would mark the beginning of a commercial phase in the second stage of the Indian programme. Several modest but important Indian initiatives for thorium utilization, including successful operation of the KAMINI, the only thorium fuelled reactor operating in the world, and several laboratory or prototype R&D programmes, have laid the foundation for future development of thorium fuel cycle technologies for commercial deployment.

In the time frame envisaged for commercial level thorium utilization, the need for large-scale deployment of nuclear power would grow. It will be important to ensure that the next-generation nuclear power plants are cheaper and less demanding of operator skills and maintenance expertise to achieve the required level of performance and safety, than most of the current generation nuclear power plants. Fulfilment of these objectives requires simplification, broad use of passive systems and other innovative approaches in reactor design.

Many necessary technologies have been mastered, however creation of the integral structure of nuclear power capable of long term and wide scale deployment and operation also requires the development of the fuel cycle technologies and related infrastructure. The elaboration of sustainable schemes and technologies for long-term safe disposal of radioactive waste is another important area of consideration. The Department of Atomic Energy of India, with the experience of nearly five decades in almost all technological aspects of nuclear fuel cycle, is pursuing a long-term strategy to address the technologies needed for long-term growth of nuclear power and other applications of nuclear energy.



AHWR = Advanced Heavy Water Reactor; Dep. U = Depleted Uranium; Fuel Reproc. = Fuel Reprocessing; HWB = Heavy Water Board; IRE = Indian Rare Earth; NFC = Nuclear Fuels Complex; SSSF = Solid Storage and Survellance Facility; UCIL = Uranium Corporation of India Limited; WIP = Waste Immobilization Plant.

FIG. XXIX-1. Three stages of Indian nuclear power programme.

A broad view of future research directions to meet India's energy requirements covers the following [XXIX-2]:

- All aspects of closed uranium-plutonium (U-Pu) and thorium-uranium (Th-U) fuel cycles with optimal utilization/ breeding of the nuclear fuel;
- Power producing fast breeder reactors with a short doubling time;

- Thermal reactors of modular design for electricity generation and high temperature process heat applications;
- Scientific and technical developments on fusion power.

Keeping the above long term larger goals in mind, the third stage of the Indian programme must necessarily meet the following objectives:

- Deploying nuclear power on a large scale in the country;
- Making the economic performance more attractive compared to alternative energy options;
- Utilizing thorium as fuel on a commercial scale;
- Attaining higher levels of transparent safety, through utilization of inherent and passive safety features to the optimal extent;
- Providing adaptability for non-electrical applications, in particular, seawater desalination and high temperature process heat applications, including those for generation of non-fossil fluid fuels.

To meet these objectives in the medium as well as long-term time frames, and keeping in view the current international trends in nuclear technology, a roadmap for the third stage of the Indian nuclear power programme has been proposed [XXIX-1], Fig. XXIX-2.

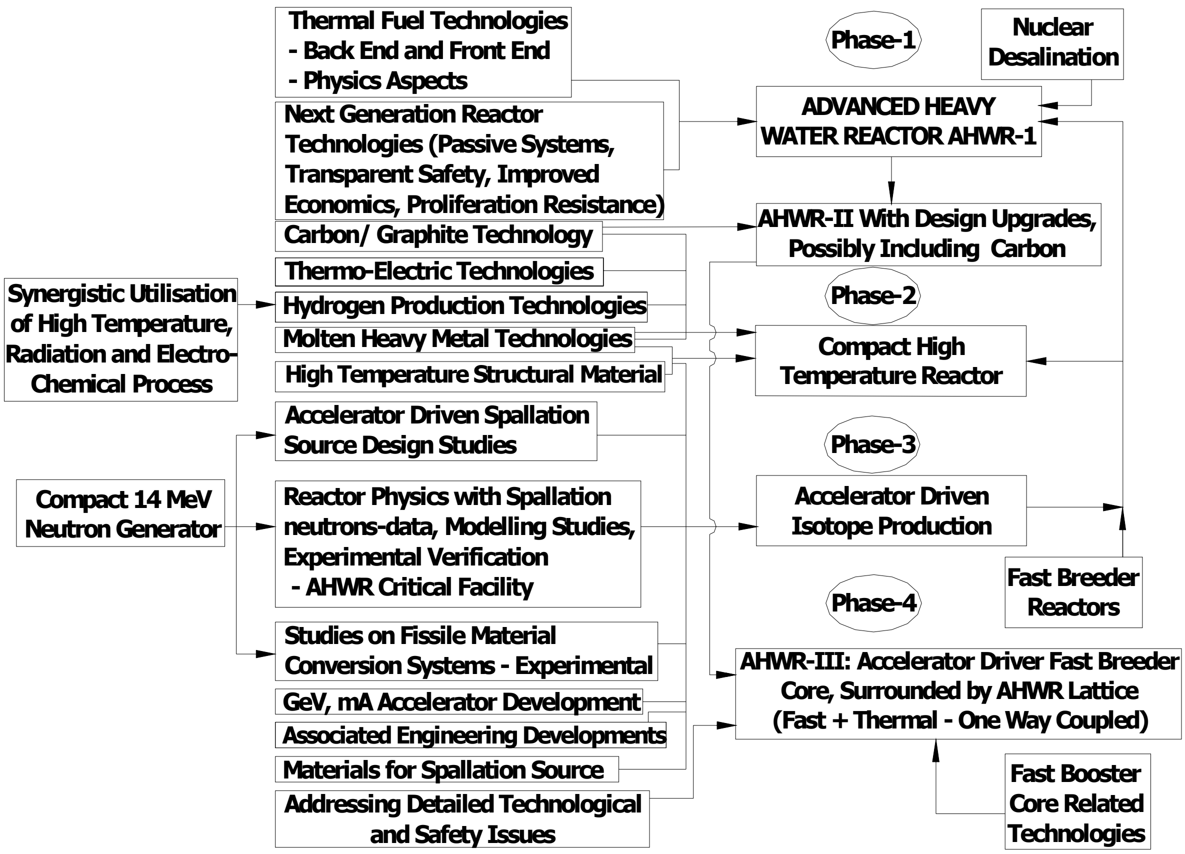


FIG. XXIX-2 Roadmap for the third stage of the Indian nuclear power programme.

Some of the products included in this roadmap, in different time frames, are the following:

(1) A Prototype Fast Breeder Reactor (PFBR) and Advanced Heavy Water Reactor (AHWR);
(2) Compact High Temperature Reactor (CHTR) based power packs;
(3) An accelerator driven system with a fast reactor subcritical core operating together with a mainly thorium fuelled thermal core somewhat similar to that of the AHWR.

The design development of the CHTR has been fully funded by the Department of Atomic Energy (DAE) of the Government of India. The activities on design and technology development for the CHTR are mainly carried out at Bhabha Atomic Research Centre (BARC), a constituent unit of the DAE.

XXIX-1.2. Applications

The CHTR, initially being developed to generate about 100 kW(th), has several advanced passive safety features to enable its operation as a compact power pack to supply non-grid based electricity in remote areas, difficult to access. The reactor is also being designed to operate at 1000°C to facilitate the demonstration of technologies for hydrogen production using the reactor's high temperature process heat.

The CHTR is being developed as a platform to launch a focussed programme [XXIX-3] for the development and demonstration of technologies associated with these two applications. Based on the technologies developed, larger power reactors for the above two applications could be designed.

Cogeneration of electricity and co-production of potable water are also envisaged for the CHTR.

XXIX-1.3. Special features

The CHTR is a land based nuclear installation. It has a long core life of 15 years and is designed to operate without on-site refuelling. The CHTR is being developed as a prototype reactor to develop small power packs to supply electricity in remote areas not connected to the electricity grids. The reactor will have a modular design with shop fabrication of most of the modules. The reactor including the core, the reflectors, the fuel, the reactor shell and the cover plates will weigh no more than 4.0 t, to simplify its delivery to remote locations.

XXIX-1.4. Summary of major design and operating characteristics

Installed capacity (thermal and electric)

The reactor is designed to produce 100 kW(th) of process heat. When used as a nuclear power pack, the CHTR would be coupled to a high efficiency direct thermo-electric conversion system producing about 20 kW(e). In addition, 3 kW(e) of electricity could be produced from the reject heat.

Mode of operation

This CHTR is designed to operate in a load follow mode.

Load factor/ availability

The target lifetime load and availability factors for the CHTR are 80% and 90% respectively.

Some major design and operating characteristics of the CHTR are given in Table XXIX-1.

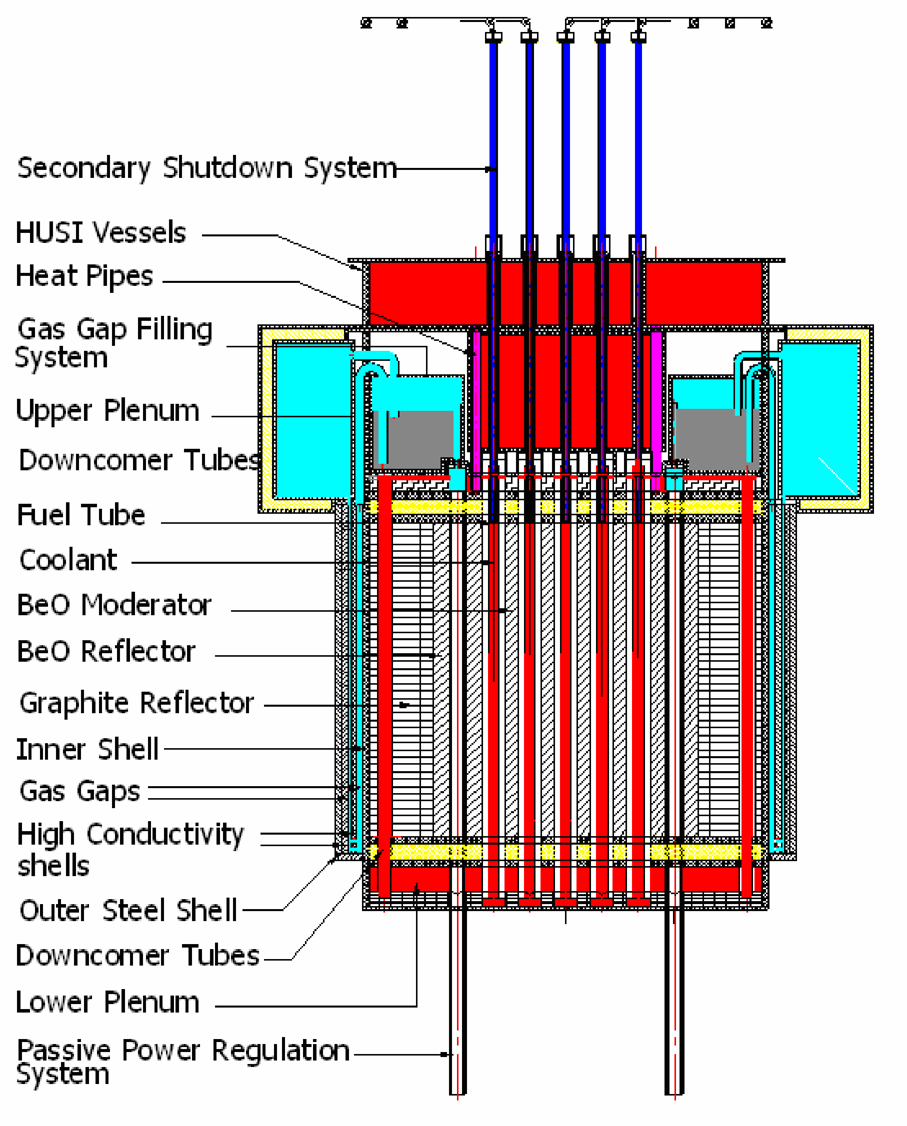
TABLE XXIX-1. MAJOR DESIGN AND OPERATING CHARACTERISTICS OF CHTR

ATTRIBUTES	DESIGN PARAMETERS
Reactor power	100 kW(th)
Core configuration	Vertical, prismatic block type
Fuel	$^{233}UC_2$+ ThC_2 + Gd based TRISO coated fuel particles shaped into fuel compacts with graphite matrix
Fuel enrichment by ^{233}U	33.75%
Refuelling interval	15 effective full power years
Moderator	BeO
Reflector	Partly BeO and partly graphite
Coolant	Molten Pb-Bi eutectic alloy (44.5% Pb and 55.5% Bi)
Mode of core heat removal	Natural circulation of coolant
Coolant flow rate through the core (total)	6.7 kg/s
Coolant inlet temperature	1173 K
Coolant outlet temperature	1273 K
Loop height	1.4 m (actual length of the fuel tube)
Core diameter	1.27 m
Core height	1.0 m (total height of the fuelled part and axial reflectors)
Primary shutdown system	18 floating annular B_4C elements of passive power regulation system
Secondary shutdown system	7 mechanical shut-off rods

Simplified schematic diagram

The component layout of the CHTR is shown in Fig. XXIX-3.

The fuel, moderator and reflector blocks are contained in a reactor shell made of high temperature and liquid metal corrosion resistant material. Top and bottom closure plates of the same material close the reactor shell. Above the top cover plate and below the bottom cover plate, plenums provide for core-outgoing and core-incoming coolant respectively. These plenums have graphite flow guiding blocks to increase the velocity of the coolant between the coolant channel exit and the entry to the downcomer tubes of the reactor. The flow-guiding blocks have passages for the coolant to flow from the inner to outer region of the plenum. The reactor shell is surrounded by two gas gaps that act as insulators during normal reactor operation and reduce heat loss in the radial direction. There is an outer steel shell, surrounded by heat sink. This shell has fins to improve heat dissipation. A passive system has been provided to fill the gas gaps with molten metal in case of abnormal rises in coolant outlet temperature.



HUSI Vessels= Heat Utilization System Interface Vessels

FIG. XXIX-3. Layout of CHTR components.

Nuclear heat from the reactor core is removed passively by a lead-bismuth eutectic alloy coolant [XXIX-4], which flows due to natural circulation between the bottom and top plenums, upward through the fuel tubes and returning through the downcomer tubes. On top of the upper plenum, the reactor has multi-layer heat utilization vessels to provide an interface to systems for high temperature heat applications. A set of sodium heat pipes is in the upper plenum of the reactor to passively transfer heat from the upper plenum to the heat utilization vessels with a minimum drop of temperature. Another set of heat pipes transfers heat from the upper plenum to the atmospheric air in the case of a postulated accident. To shut down the reactor, a set of seven shut-off rods has been provided, which fall by gravity in the central seven coolant channels. Appropriate instrumentation like neutron detectors, fission/ ion chambers, various sensors and auxiliary systems such as a cover gas system, purification systems, active interventions etc. are being incorporated in the design as necessary.

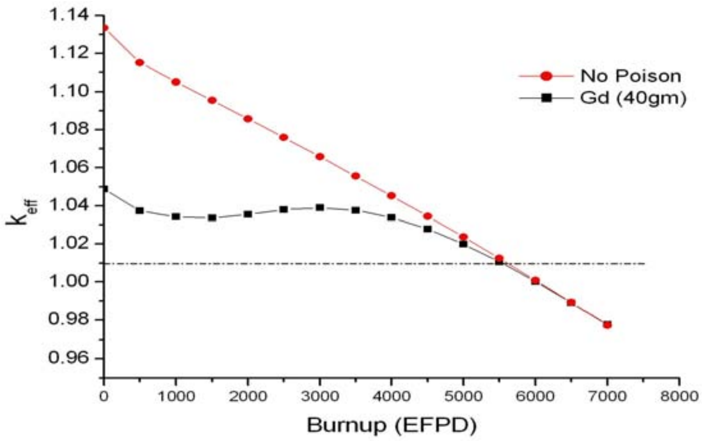
Neutron-physical characteristics

The neutronic design of the CHTR has been carried out with a view of the following objectives:

- All power must be generated from Th/ ^{233}U based fuel;
- The temperature reactivity coefficient for fuel should be negative;
- The fuel should be capable of high temperature performance;
- The fuel burn-up should be high;
- The refuelling interval should be infrequent.

Initially the core was designed for pure ^{233}U. During the analysis of the core physics it was found that 100% ^{233}U results in a positive fuel temperature reactivity coefficient. The coefficient was shown to become negative if fissile fuel is mixed with either fertile isotopes like ^{232}Th and/ or a burnable poison. A combination of 2.7 kg of ^{233}U mixed with 5.3 kg of thorium and 0.040 kg of gadolinium (added only in central fuel tube) has been selected at the present stage of design; this combination of fuel satisfies the reactivity control requirements. The reactivity change due to burning up of fuel is 102 mk[1].

Such fuel being loaded to the CHTR core, the required power of 100 kW(th) can be generated during 15 effective full power years of continuous operation. In this, the Doppler coefficient of reactivity was found satisfactorily negative. The K_{eff} values for this fuel design, as well as for a case with no poison added to the fuel, as a function of burn-up, are shown in Fig. XXIX-4.



EFPD= Effective Full Power Days

FIG. XXIX-4. K_{eff} variation with fuel burn-up.

Reactivity control mechanism

The CHTR incorporates a passive power regulation system (PPRS) [XXIX-5]. This system includes a gas header filled with helium gas at moderate pressure. The header is attached to a niobium driver tube, which contains lead-bismuth eutectic alloy as driven liquid. The driver tube is housed within a control tube that contains an annular control rod made of boron carbide with niobium cladding. The annular space between the driver and control tube contains lead-bismuth eutectic, on which the control rod floats. The space above the liquid

[1] mk = milli k = 1000x ρ (reactivity), where $\rho = (k_{eff}-1)/ k_{eff}$

level is filled with helium. The PPRS gas header, located in the top plenum, is submerged in the coolant and senses the coolant temperature immediately downstream of the heat pipes. Under normal operating conditions, the gas header is surrounded by coolant at 1173 K, the temperature resulting after removal of the reactor power by the heat pipes. Any condition (such as failure of heat pipes), which causes the coolant to return at a temperature higher than the normal, would also cause the gas in the gas header to heat up. This would lead to a rise in gas pressure in the driver tube and would result in a pressure imbalance between the driver and the control tube. This, in turn, would cause the level of liquid in the driver tube to go down and that in the control tube to go up. Since the absorber rod floats on liquid, it would also rise with the liquid level in the reactor core, thus inserting negative reactivity. Depending on the temperature rise sensed, the system would stabilize at a particular value of reactivity insertion. A schematic diagram of the PPRS is shown in Fig. XXIX-5.

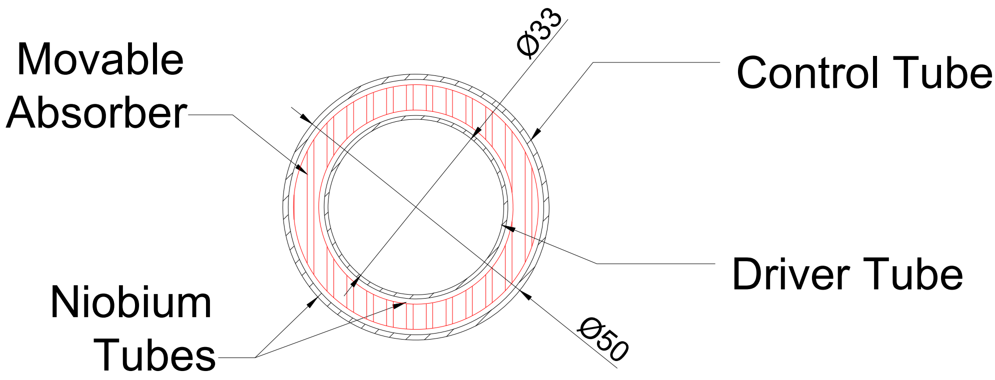


FIG. XXIX-5. Schematic diagram of passive power regulation system.

The PPRS operation was analyzed using a domestic computer code. A typical analytical result is shown in Fig. XXIX-6. With all rods inside the core, K_{eff} under hot operating conditions was found to be 0.850 and with all rods out, K_{eff} would be 1.0489. The total worth of the control rods was found to be 223 mk.

The passive power regulation system, described above, is on itself capable of shutting down the reactor. In addition, the CHTR has been provided with a secondary shutdown system. Under normal operation this system has a set of seven shut-off rods held on top of the reactor core by individual electro-magnets, which are passively released under abnormal conditions when the temperature of the core goes up. The shut-off rods are lifted up by active means. The requirement for these rods was that, when inserted in the fuel tubes, they should be able to bring the reactor to a subcritical state with necessary margin, even when the initial reactivity balance in the reactor is at its maximum. The maximum (with K_{eff} = 1.111) is reached in an uncontrolled cold state therefore, the required worth of the shut-off rods should be evaluated namely for this state.

Calculations were performed using a Monte-Carlo code to determine the K_{eff} values of the reactor in different states. For analysis, three types of shut-off rods were considered, shut-off rods made of Dy_2O_3 encased in molybdenum tubes and tungsten rods of two sizes viz., 20 mm and 15 mm diameters. The K_{eff} values of the reactor, under cold conditions, were determined when the shut-off rods were inserted in (i) all 19 fuel tubes, (ii) the inner 7 fuel tubes, and (iii) the outer 12 fuel tubes. The results are given in Table XXIX-2.

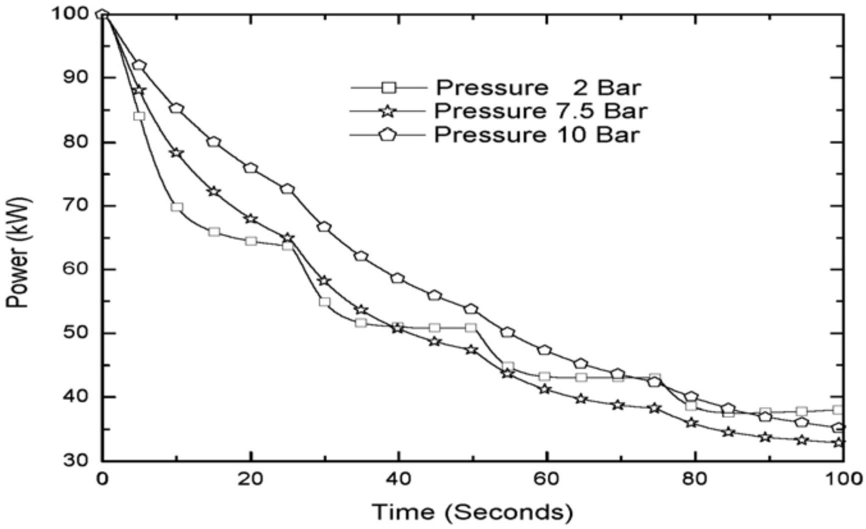


FIG. XXIX-6. Typical result of the analysis of PPRS performance.

TABLE XXIX-2. VALUES OF K_{EFF} FOR DIFFERENT COMBINATIONS OF INSERTED SHUT-OFF RODS

SHUT-OFF RODS IN FUEL TUBES	K_{eff} VALUE		
	DY_2O_3 IN MO TUBE	TUNGSTEN RODS WITHOUT CLADDING (DIAMETER = 20 mm)	(DIAMETER = 15 mm)
All 19 fuel tubes	0.596	0.656	0.776
Inner 7 fuel tubes	0.830	0.863	0.930
Outer 12 fuel tubes	0.850	0.881	0.943

Based on the results shown in Table XXIX-2, the shut-off rods made of tungsten of 20 mm diameter and located in the inner 7 fuel tubes were selected for the secondary shutdown system. The worth of such a system is 258 mk. The maximum worth of a single rod for the selected case was found to be about 40 mk, which provides an estimate of the stuck-rod margin.

Cycle type and thermodynamic efficiency

In the current CHTR design, direct thermo-electric conversion devices are assumed to produce electricity. A cascaded system made of an array of Si-Ge or TAGS (Ti/Ag/Ge/Si), or Pb-Te based thermo-electric generators is being designed for this purpose.

Thermal-hydraulic characteristics

During normal operation of the reactor, core heat is removed by the natural circulation of Pb-Bi coolant. The coolant at 1173 K enters the fuel tube in the lower plenum, absorbs the reactor heat, and at 1273 K reaches the upper plenum. Twelve sodium heat pipes transfer heat from the upper plenum to the system of heat utilizing vessels. Thermal-hydraulic analyses were carried out to study natural circulation and the effect of orificing in the primary loop. A computer model based on the law of conservation of momentum was developed for this analysis; a simplified model of the primary loop is shown in Fig. XXIX-7.

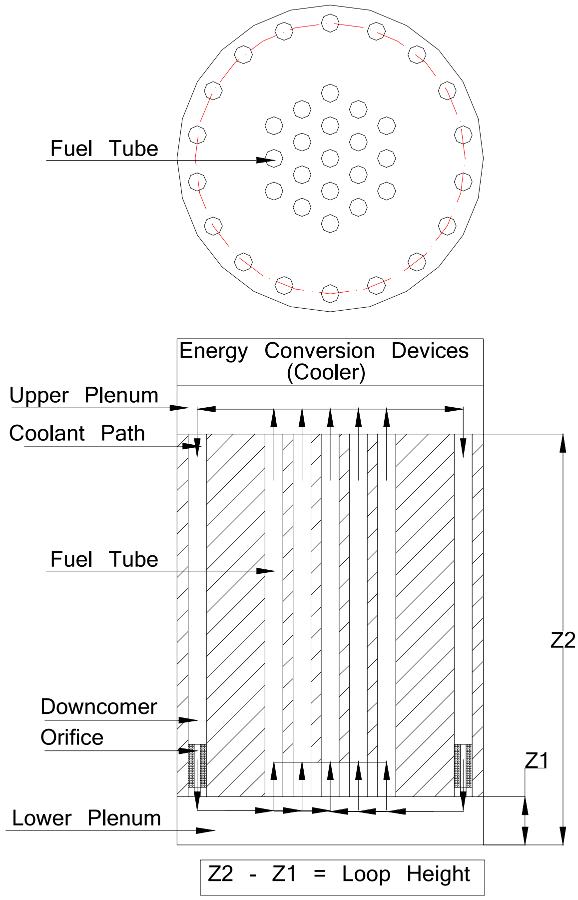


FIG. XXIX-7. Simplified model of CHTR loop used in thermal-hydraulic analysis.

The loop consists of 19 heater tubes, a cooler at the top plenum, and 18 downcomer tubes. Various cases were analyzed. The average mass flow rate and average velocity of the coolant in the coolant tube were found to be 6.7 kg/s and 0.04 m/s respectively. The variation of coolant velocity with internal diameter of fuel channels is shown in Fig. XXIX-8.

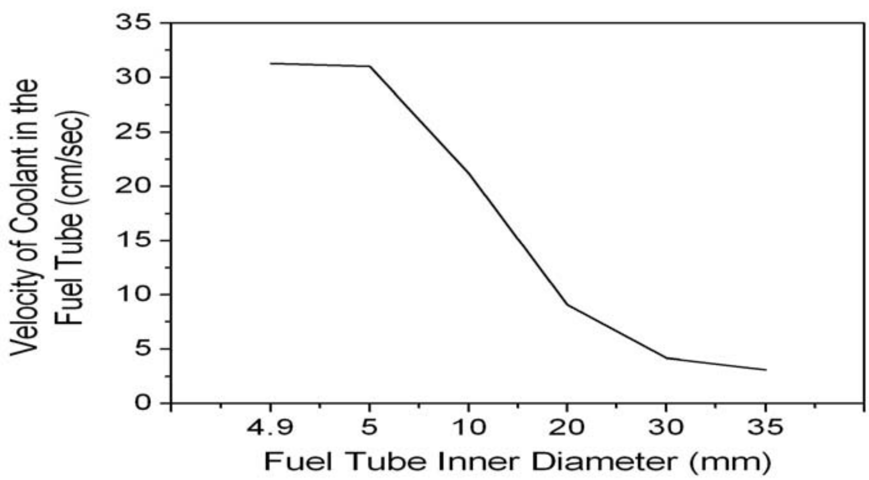
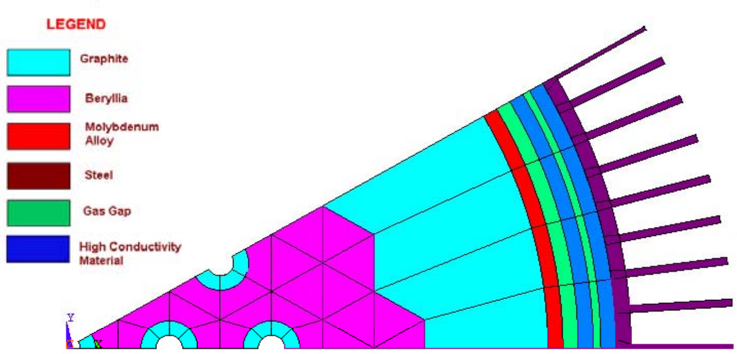


FIG. XXIX-8. Coolant velocity versus internal diameter of coolant channels.

A three-dimensional finite element method (FEM) was used for thermal analysis of the CHTR. Figure XXIX-9 shows a steady state distribution of the reactor middle plane temperature (for the case of inlet and outlet coolant temperatures of 1173 K and 1273 K, respectively). The temperature is seen to be almost constant within the reactor core and the reflector region. The drops in temperature, as expected, occur in two gas gaps provided to prevent loss of heat in the radial direction.



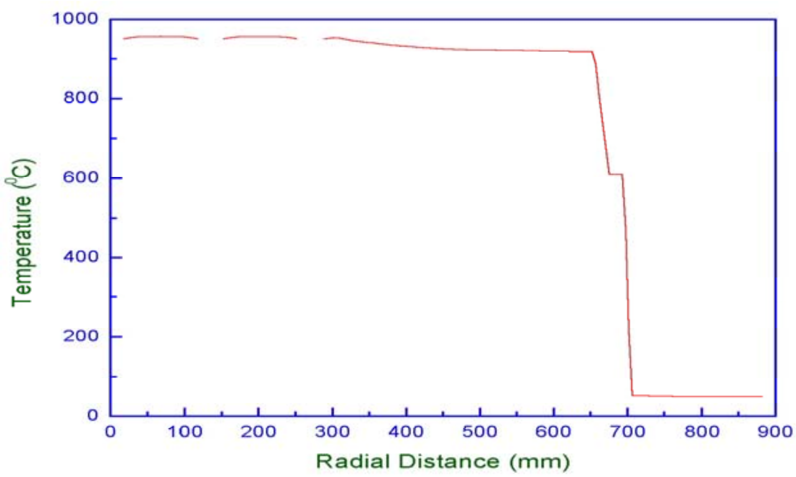


FIG. XXIX-9. Steady state radial temperature distribution within and outside the core.

Maximum/average discharge burn-up of fuel

The average discharge burn-up of fuel is estimated to be 68 000 MW·day/t HM.

Fuel lifetime/period between refuellings

The period between refuelling sis estimated to be 5500 effective full power days (EFPD).

Design basis lifetime for reactor core, vessel and structures

The design basis lifetime for the BeO moderator, BeO reflector, graphite reflector, inner, intermediate and outer vessels and all other non-replaceable components is 50 years. All fuel tubes will be changed after about 5500 EFPD (15 EFP years). It is around 18.5 years (considering about 300 EFPD of operation per year).

Design and operating characteristics of systems for non-electric applications

When used for high temperature process heat applications, the CHTR would include helium circulated through the heat utilizing vessels to transfer high temperature process heat through an interface heat exchanger to the two stages of the Iodine-Sulphur (I-S) process for hydrogen production. In this way, about 320 Nm^3/day of hydrogen gas could be produced.

In addition to this, the waste heat could be used to produce 1.5 m^3/day of potable water.

Economics

No information was provided.

XXVII-1.5. Outline of fuel cycle options

The standard fuel cycle option for the CHTR would depend upon the technology development for reprocessing of TRISO coated particle fuel and fuel compacts. In case of the development of a reprocessing technology, a closed nuclear fuel cycle option would be adopted. Fresh fuel for the reactor would be made from ^{233}U and Th recovered from the spent fuel. Alternately, it would be a once though fuel cycle without reprocessing. The objective then would be to achieve the highest possible fuel burn-up.

Research for reprocessing of TRISO coated particle fuel in the fuel compacts is in early stages. The process would include operations for the extraction of fuel compacts from the fuel tube, dismantling fuel compacts to free fuel particles and mechanical and thermo-chemical treatment of these particles to extract spent fuel.

Since the reactor is modular with no provision for on-site refuelling, its refuelling is assumed to be performed at a centralized refuelling and reprocessing plant.

XXIX-1.6. Technical features and technological approaches that are definitive for CHTR performance in particular areas

XXIX-1.6.1. Economics and maintainability

The compact size, high temperature capability, an option of use as a remote power pack or for hydrogen production, proliferation resistant fuel, etc., could make the CHTR attractive for developing countries.

Being compact in size and modular in construction, the CHTR can be factory fabricated and easily transported to a site by various transport means. This also would reduce the construction cost and time. Besides using high temperature materials and heat pipes, the reactor has no major components like pumps or heat exchangers; hence, capital cost is expected to be low.

The very long life CHTR core, needing refuelling only once in 15 effective full power years, is the feature, which could reduce the O&M costs. The design of the CHTR, with its all passive features, is intended to make the plant capable of unattended operation.

XXIX-1.6.2. Provisions for sustainability, waste management, and minimum adverse environmental impacts

The use of thorium fuel leads to suppression of the generation of minor actinides in the non-plutonium bearing CHTR. Graphite based fuel tubes with low activation and ease in compacting the waste further reduce the amount of wastes generated. An isotope of a certain concern in the thorium cycle is ^{232}U. It is formed via (n, 2n) reactions, from ^{232}Th, ^{233}Pa and ^{233}U and has a half-life of about 69 years. The daughter products of ^{232}U are hard gamma emitters like ^{208}Tl (2.6 MeV) with very short half-lives. As a result, the radioactivity increases with time in the bred uranium isotopes. This presents several technological challenges in the reprocessing and recycling of bred ^{233}U. A laser based separation technique is being developed to clean ^{233}U by removing ^{232}U. Long-lived minor actinides resulting from the burn-up chain are in much lower quantities for thorium fuel cycles, especially if the reactor operates purely in the ^{233}U-Th cycle. Actinides having masses beyond 237 are produced in negligible quantities. This is an important advantage, since the burden of long-lived radioactive waste management is significantly reduced.

To reduce dose limits the CHTR employs liquid metal coolants, off-site refuelling, a ceramic core, and a surrounding heat sink outside the outer steel shell.

Since R & D for the reprocessing of coated particle fuel is in the initial stages, and considering a relatively high burn-up achieved, the CHTR spent fuel is most likely to be sent for direct disposal. Coated particles are particularly adapted to direct disposal, since their coatings can keep leak tightness under long-term storage conditions. Since graphite for disposal is brittle, it can be compacted to reduce the volume of waste.

If R & D for reprocessing of the HTGR type fuel is successful, the unused thorium and ^{233}U could be recovered from spent fuel to make fresh fuel and achieve higher fuel utilization.

XXIX-1.6.3. Safety and Reliability

Safety concept and design philosophy

Since the CHTR is assumed to be used as a power pack for remote areas, it incorporates many design and safety features providing for reactor operation with fewer operator interventions, therefore minimizing skilled man-power requirements for operation. The CHTR strongly relies on inherent safety features and passive systems for reactor control, shutdown and heat removal under normal and abnormal conditions.

Provisions for simplicity and robustness of the design

Some of the provisions for simplicity and robustness of design are the high temperature capability of the fuel, low power density of the core, high thermal capacities and thermal conductivity of the components of the all-ceramic core, high boiling point, chemically inert liquid metal coolant and the use of many passive safety features.

Active and passive systems and inherent safety features

CHTR has the following inherent safety features:

- A strong negative Doppler coefficient of the fuel for any operating condition;
- High thermal inertia of the all-ceramic core;
- Low core power density;
- A large margin between the normal operating temperature of the fuel (around 1373 K) and the leak tightness limit of the TRISO coated particle fuel (1873 K) to retain fission products and gases;
- A negative moderator temperature coefficient;
- Due to the use of the Pb-Bi coolant, which operates at low pressure, there is no over-pressurization and no chance of reactor thermal explosion due to coolant emergency overheating;
- Due to a very high boiling point (1943 K), there is a very large thermal margin to Pb-Bi boiling. This also eliminates the possibility of heat exchange crisis and increases the reliability of heat removal from the core;

- There is a negligible thermal energy stored in the coolant and available for release in the event of a leak or accident;
- The high temperature Pb-Bi coolant is chemically inert. Even in the eventuality of contact with air or water, it does not react violently with explosions or fires;
- No pressure in the coolant allows the use of a graphite coolant channel, improving neutronics of the reactor;
- A low induced long-lived gamma activity of the coolant; in case of a leakage, the coolant retains iodine and other radionuclides;
- For Pb-Bi coolant, the reactivity effects (void, power, temperature, etc.) are negative.

CHTR employs the following passive systems, also described in the subsequent paragraphs:

- Natural circulation of coolant to remove reactor heat during normal operation;
- Passive regulation of reactor power under normal operation;
- Passive shutdown for postulated accidental conditions;
- Passive means of conduction of core heat by filling up the gas gaps with molten metals;
- Passive transfer of reactor heat by heat pipes under normal and postulated accident conditions;
- Passive removal of heat from the reactor core by carbon-carbon composite heat pipes under LOCA.

Structure of the defence-in-depth

Some of the major features of the CHTR design, structured in accordance with various levels of the defence in depth are presented below.

Level-1: Prevention of abnormal operation and failure

The CHTR design features contributing to this level are as follows:

(a) Heat removal from the core under normal operating conditions is accomplished through natural circulation of the coolant, which essentially eliminates the hazard of a loss of coolant flow;

(b) The extent of overpower transients and their consequences is limited by:

- Low core power density;
- A highly negative Doppler (fuel temperature) coefficient, achieved through the selection of an appropriate fuel composition;
- Use of a burnable poison to compensate for reactivity change with burn-up;
- Negative reactivity effects (void, power, temperature etc.) achieved with the use of the Pb-Bi coolant;
- Use of the all-ceramic core with high temperatures margins; and
- The resulting low excess reactivity.

Level-2: Control of accidents within the design basis

The CHTR design features contributing to this level are the following:

- Increased reliability of the control system achieved through the use of a passive power regulation system. This system inserts negative reactivity in the core when temperature increases beyond allowable limits;

- The use of two independent passively operating shutdown systems;
- The use of a high heat capacity ceramic core to prevent fuel temperature from exceeding the design limits for a long time.

The abovementioned design features are expected to result in the reactor operation and safety functions being fully passive and independent of operator intervention.

Level-3: Control of accidents within the design basis

Features of the CHTR that contribute to this level are:

- The use of two independent shutdown systems, one comprising mechanical shut-off rods and the other employing a temperature feedback gas-expansion based passive shutdown system, altogether resulting in an increased shutdown reliability;
- The use of two independent systems to transfer reactor core heat to the outside environment during abnormal conditions, one comprising a gas gap filling system and the other a heat pipe based system;
- The use of an impendent system based on carbon-carbon composite heat pipes transferring heat from the reactor core to the atmosphere in LOCA.

Level-4: Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents,

The features important for this level are:

- Excellent high temperature (up to 1873 K) performance of TRISO coated particle fuel, ensuring that the probability of the release of fission products and gases is very low.
- Large heat capacity ceramic core, resulting in a slow fuel temperature rise with more than 50 minutes being available for a corrective action even when all heat sinks are lost;
- The use of heat sink outside the outer steel shell;
- The erection of the reactor in an underground pit with sealed barrier of reinforced concrete and steel covers is foreseen to provide an additional barrier for the prevention of release of radionuclides.

Level-5: Mitigation of radiological consequences of significant release of radioactive materials

- Passive design features of the previous levels avoid any significant release of radioactive materials and necessity for evacuation or relocation measures outside the plant site.

Design basis accidents and beyond design basis accidents

The major initiating events analyzed are as follows:

- Loss of load accident;
- Loss of coolant accident; and
- Power transients.

A number of inherent and passive safety features in the design of the CHTR prevent the TRISO coated particle fuel from exceeding the limiting temperatures in postulated accidents or abnormal events, among them:

- A variable-conductance heat pipe and gas gap filling system, as two independent systems for heat dissipation to the environment;
- Carbon-carbon composite based heat pipes for heat transfer from the core to the environment in case of LOCA;
- The high heat capacity ceramic core, ensuring slow increases of fuel temperature in power transients.

A highly negative Doppler coefficient of fuel ensures that the average fuel temperature does not exceed 1373 K in case of a power transient; the acceptable maximum temperature of the TRISO coated particle fuel is 1873 K.

Provisions for safety under seismic conditions

Various structures, systems and equipment of the CHTR would be designed for high level and low probability seismic events such as an operating basis earthquake (OBE) and a safe shutdown earthquake (SSE). Seismic instrumentation is also foreseen.

Probability of unacceptable radioactivity release beyond the plant boundaries

The probability of unacceptable radioactivity release beyond the plant boundary is targeted to be less than 1×10^{-7}.

Measures planned in response to severe accidents

Through broad implementation of inherent and passive safety features and passive systems, ensuring reactivity self-control, passive heat removal under all operating conditions, and perfect confinement of radioactivity in TRISO fuel up to very high temperatures, the CHTR aims to eliminate the need for intervention in the public domain beyond the plant boundaries, in all postulated accidents.

XXIX-1.6.4. Proliferation resistance

Some of the important technical features of the CHTR, which reduce the attractiveness of its nuclear materials for use in a nuclear weapon programme, are the following:

- The absolute amount of fissile material in the core is very low, about 2.7 kg ^{233}U at BOL;
- The fuel does not contain any plutonium, the ^{233}Pa produced has a half-life of about 27 days and is converted by decay to ^{233}U;
- The reprocessing of the TRISO coated particle fuel is not available as a commercial technology.
- The radiation field from ^{233}U is very high due to the presence of ^{232}U as contamination.

The same technical features prevent or discourage an undeclared production of weapon grade material in the CHTR. Here, the reactor operation with low excess reactivity is also of certain value.

The discharged fuel will most likely be sent for disposal without reprocessing.

The CHTR has a small core with very low fissile content and can easily be verified. High gamma activity in the discharged fuel can also be used for monitoring.

XXIX-1.6.5. Technical features and technological approaches used to facilitate physical protection of CHTR

Installation of the reactor in an underground pit with sealed barrier of reinforced concrete and steel covers provides a significant barrier to reactor damage arising out of external impacts.

XXIX-1.7. Non-technical factors and arrangements that could facilitate effective development and deployment of CHTR

The two applications for which this reactor is being developed viz., as a source of process heat for hydrogen production by splitting water and as a compact nuclear power pack to supply electricity in remote areas unconnected to the grid system, have a large potential market, especially in developing countries.

Considering the long life core and small overall weight and dimensions, an option of NPP leasing to member states under the IAEA's umbrella could be considered. India has a large infrastructure for manpower training, and could provide specialized training in nuclear related areas to personnel from several IAEA member states, under the IAEA programmes.

XXIX-1.8. List of enabling technologies relevant to CHTR and status of their development

The enabling technologies for the CHTR [XXIX-6] are given in Table XXIX-3.

TABLE XXIX-3. ENABLING TECHNOLOGIES FOR CHTR

OBJECTIVE	ENABLING TECHNOLOGIES	STATUS OF DEVELOPMENT
Development of TRISO coated particle fuel	Production of fuel kernels by the sol-gel technique	The technique exists
	Technology for application of multi-layer coatings	Coating trials initiated on surrogate material
Development of oxidation- and corrosion-resistant coatings	Technologies for application of coatings like PyC, SiC, silicide based coatings	Under development
Development of BeO based moderator and reflector material	Manufacture of high density BeO blocks of different sizes and shapes	Sample pieces were produced by cold pressing and sintering, as well as vacuum hot pressing techniques
Development of liquid metal coolant technology	Natural circulation of Pb-Pi coolant in the primary circuit	Experimental loop planned; the equipment is under fabrication and procurement
	Validated codes for simulation of thermal-hydraulic behaviour of Pb-Bi coolant in primary circuit, under natural circulation	
	Compatibility of materials with Pb-Bi coolant within the design range of parameters	Analytical studies performed. Experimental facility under design
	Freezing / de-freezing technology for Pb-Bi coolant	

Objective	Enabling technologies	Status of development
Development of liquid metal coolant technology	Instrumentation and components like electro-magnetic pumps and flow meters for liquid metal coolant	Under development
Development of passive power regulation system	Passive power regulation system	Experimental set up designed
	Validated computer codes to simulate operation of the passive power regulation system	
Development of passive heat removal systems	Technology for manufacture of heat pipes	Experimental set-up under design
	Technology for testing of heat pipes	Experimental set-up under design
	Technology for gas gap filling system	
	Validated computer codes to simulate gas gap filling system	Experimental set-up under design
Development of graphite and carbon materials	High density isotropic graphite	Under development
Development of high temperature structural materials	Refractory metals	Under development
Development of codes for analysis of compact reactor cores	Validated codes and databases for simulation of compact reactor cores	Codes developed
Development of codes for structural and thermal design of brittle materials	Validated codes and databases for structural and thermal design of brittle materials	Codes under development
Development of hydrogen production technologies	Alkaline electrolizer based electrolysis	Developed
	Solid polymer electrolyte based electrolysis	Under development
	Solid oxide fuel cell for high temperature water electrolysis	Under development
	Thermo-chemical water splitting	Development is being initiated
Direct conversion of energy	Pb-Te and TAGS (Ti/Ag/Ge/Si) based thermo-electric devices for application up to 850 K	Developed
	Si-Ge based devices for application at 1273 K	Under development

XXIX-1.9. Status of R&D and planned schedule

At present, a feasible design of the CHTR has been achieved after completing the conceptual design of the reactor and associated systems. Experimental facilities are being set-up to carry out various studies related to liquid metals, passive safety features and heat removal systems. The manufacturing capabilities for BeO, carbon components, and fuel micro-spheres exist. Trials for TRISO coatings have already started. It is expected that developmental work for various enabling technologies would be completed by 2008-09. Subsequent to the manufacture of fuel and certain systems, it is expected that a critical facility for the CHTR could be set up around 2012.

The design, research and development of the CHTR are being done at BARC with the financial support of the Government of India. At a later stage, relevant agencies will be approached for safety appraisal and licensing for construction.

XXIX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

The CHTR is a high temperature reactor with a coolant outlet temperature of 1273 K. It uses ^{233}U based fuel in the TRISO coated particle fuel form and is cooled by Pb-Bi eutectic coolant. It employs many passive safety features and passive reactor core heat removal systems.

The reactor physics and engineering design of a CHTR based nuclear energy system need extensive qualification through analytic and experimental studies. Many of the enabling technologies need to be demonstrated for the first time. The CHTR would serve the purpose of being a prototype before commercial and larger power reactors are developed based on these technologies.

XXIX-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

No other similar SMRs are under development elsewhere

XXIX-2. Design description and data for CHTR

XXIX-2.1. Description of the nuclear systems

Reactor core and fuel design

The CHTR fuel is based on TRISO (TRI-ISOtropic) coated fuel particles [XXIX-7, XXIX-8]. The kernel of a fuel particle is made of ThC_2, $^{233}UC_2$ and Gd. The kernel is initially coated with a layer of low-density pyrolythic carbon as a buffer layer. On top of this coating, the particle is then further coated with subsequent layers of high-density pyrolythic carbon, silicon carbide and an outer layer of high-density pyrolythic carbon. These particles are mixed with graphite powder as a matrix and made into cylindrical fuel compacts. The fuel compacts are packed in fuel bores in the walls of each of the nineteen fuel tubes. A radial gap between the fuel compact and fuel tubes accommodates fuel swelling. Figure XXIX-10 shows a schematic of the fuel particle and fuel compact. Table XXIX-4 shows the dimensions of the TRISO coated fuel particle. After filling with fuel compacts, the remaining portion of the fuel bores would be filled with graphite and sealed.

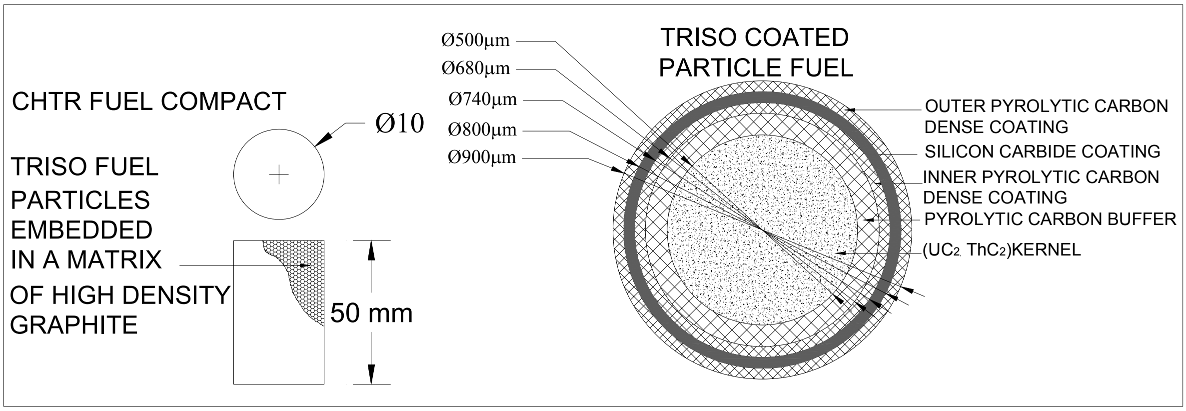


FIG. XXIX-10. Schematic of TRISO coated fuel particle and fuel compact.

TABLE XXIX-4. DIMENSIONS OF COATED FUEL PARTICLE

ITEM	DIMENSION (µm)
(UC_2 + ThC_2 +Gd) kernel, diameter	500
Buffer pyrocarbon layer, thickness	90
Inner pyrocarbon layer, thickness	30
SiC layer, thickness	30
Outer pyrocarbon layer, thickness	50
Coated particle outer diameter	900

The reactor core, as shown in Fig. XXIX-11, consists of nineteen prismatic hexagonal shaped beryllium oxide (BeO) moderator blocks. These 19 blocks contain centrally located graphite fuel tubes. Details of the lattice positions and fuel tubes are given in Table XXIX-5.

TABLE XXIX-5. CORE DESIGN DATA

ATTRIBUTE	VALUE/ DESCRIPTION
Number of fuel tubes	19 with 75 mm outer diameter and 35 mm inner diameter
Fuel tube material	Graphite
Lattice pitch	0.135 m
Active fuel length	0.70 m

Each fuel tube carries fuel inside 12 equi-spaced longitudinal bores in its wall. The fuel tube also serves as a coolant channel. The coolant flows through the central hole of the tube. A typical fuel bed with the BeO moderator, fuel tube and fuel compacts are shown in Fig. XXIX-12.

Eighteen BeO reflector blocks surround the moderator blocks. Graphite reflector blocks surround the BeO reflector blocks. BeO reflector blocks have central holes to accommodate the passive power regulation system. The regulation system works on temperature feedback, and in case of a rise of coolant outlet temperature beyond the design value, inserts negative reactivity into the core.

Main heat transport system

Figure XXIX-13 shows heat removal paths of the CHTR in accidents.

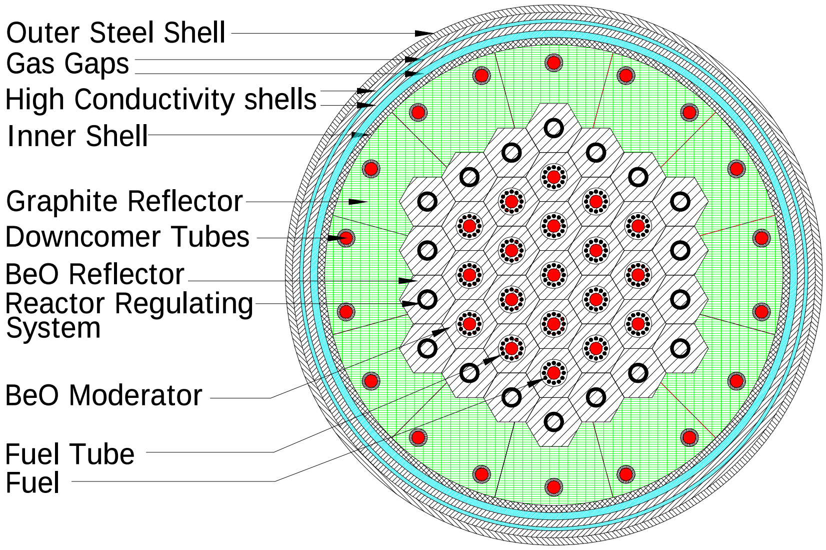


FIG. XXIX-11. Cross sectional layout of CHTR core.

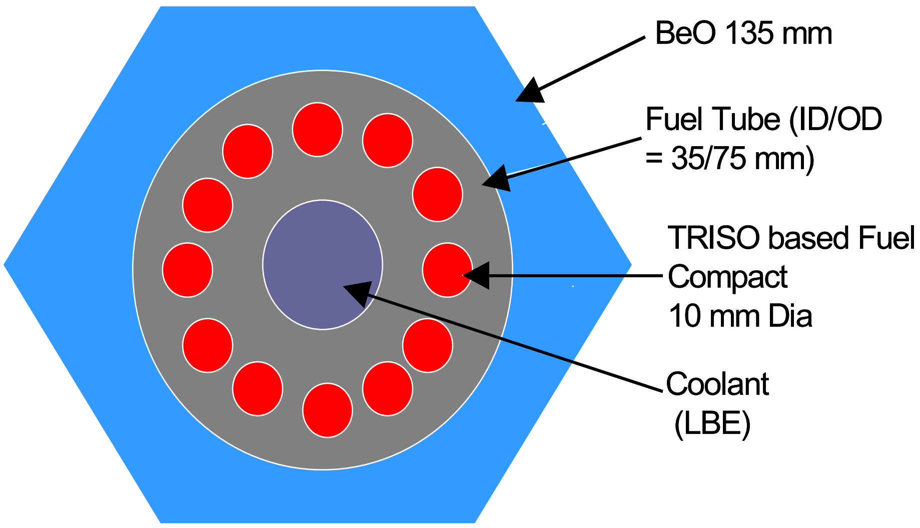
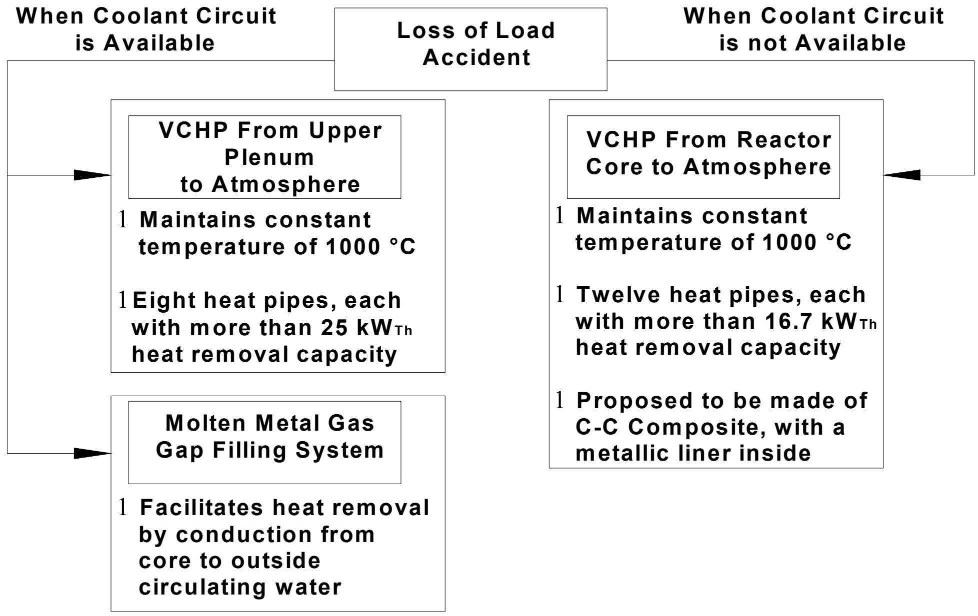


FIG. XXIX-12. A typical fuel bed of CHTR.



VCHP= Variable Conductance Heat Pipe; C-C= Carbon-Carbon

FIG. XXIX-13. Heat removal paths for postulated accident conditions.

For loss of load accident, it has been assumed that the heat utilizing systems will fail together and that the temperature of the interface vessels of heat utilizing systems will continue to increase. The heat removal from the upper plenum of the reactor will then decrease, resulting in an increase of the coolant temperature in the upper plenum. The temperature of the coolant returning to the lower plenum will also increase, leading to an increase in temperature of the coolant at the core inlet. This will eventually affect heating of the core and core temperature will start rising. To obviate such a situation, a system of eight heat pipes has been provided to remove heat from upper plenum of the reactor to the atmosphere and to keep a coolant temperature of 1273 K in the upper plenum. These heat pipes have been designed to remove heat at a rate of 200 kW(th); they are variable conductance heat pipes (VCHP), which can keep the temperature of the heat source (the upper plenum in this case) constant, even if the heat flux at the heat source increases. This system results in maintaining the upper plenum coolant at a temperature of 1273 K or lower, ensuring safety of the core. During normal operation, these heat pipes are designed to deliver heat to interface vessels of the heat utilizing systems. When temperature of the core increases beyond 1273 K, these heat pipes start radiating extra heat to the atmosphere through the finned portion of the heat pipe condenser.

For loss of coolant accident, it has been assumed that coolant is unavailable in the upper plenum, core and lower plenum of the reactor. Due to the absence of a heat removal medium, temperatures of the core will start increasing, leading to heating of all core components. The negative void reactivity coefficient will limit the power and thus, the temperature of the core components. The neutronically limited power would reach 200 kW(th). For this case, a system of 12 variable-conductance heat pipes, made of a carbon-carbon composite with a metallic liner, has been provided. These heat pipes penetrate the core. The condenser end of these heat pipes extends beyond the upper plenum and the interface vessels of heat-utilizing systems to the atmosphere. At the condenser end, these heat pipes have radiator fins to dissipate heat to the atmosphere. In case of a postulated accident due to loss of load or loss of coolant, core temperature will start increasing. As long as the temperature of the core is within

1273 K, these heat pipes will continue to transfer heat to the interface vessels of heat utilizing systems. Since the heat pipes are a variable conductance type, they will not allow temperatures of the core to increase beyond 1273 K.

Passive heat removal system based on molten metal gas gap filling

Under postulated accident conditions and with neutronically limited peak power level, the CHTR is capable of rejecting all the generated heat to the atmosphere by passive means, without fuel damage. To achieve this, a gas gap filling system has been included in the design. The function of this system is to fill the gas gaps at the periphery of the reactor core with liquid metal and facilitate a conduction pathway for the transfer of the reactor heat under postulated accident conditions. At the same time, through low heat conductivity of the filled-in gas, this system effectively prevents heat transfer during normal operation. The neutronically limited peak power was evaluated to be double the normal power of the reactor. The choice of the liquid metal ensures that the fuel temperature in accidents would not exceed 1873 K. The system senses the coolant temperature, and in case of the temperature exceeding the set point, it starts pouring the liquid metal into the gas gap by siphon.

The system consists of a reservoir located above the upper plenum and subdivided into compartments. The liquid metal is stored in the reservoir, which is fitted with siphon tubes and bulbs. One end of the siphon is dipped into the liquid metal and the other opens into the inner gas gap; multiple siphon tubes are employed. The bulb is located immediately downstream of the heat pipes and normally senses a temperature of 1173 K; in case of a failure of the heat pipes, the coolant immediately senses a temperature of 1273 K. This would increase the pressure of the gas inside the bulb, cause the liquid metal to rise inside the siphon tube and ultimately, start the siphon. The liquid metal would then exit into the inner gas gap and fill the outer air gap through holes in the inner gas gap wall. The gas inside the gas gap would be pushed into a gas tank. A connector between the liquid metal and the gas tank would handle the decrease in pressure caused by the fall in level of the liquid metal in the reservoir, such that after some time, the pressure in the reservoir and the gas gaps would be equalized. Table XXIX-6 shows the calculated times taken to fill the gas gap after the start of the siphon. Indium has been assumed as the poured liquid metal. A schematic of this system is shown in Fig. XXIX-14. Return of the poured liquid metal to the reservoir would be accomplished by active means.

TABLE XXIX-6. TIME TAKEN TO EMPTY RESERVOIR AFTER THE START OF SIPHON

NUMBER OF SIPHON TUBES	TIME TAKEN TO EMPTY RESERVOIR AFTER THE START OF SIPHON, S		
	LIQUID INDIUM	LIQUID TIN	LIQUID ALUMINIUM
5	26	25.6	22.8
6	23.2	23.2	20.4
7	21.2	21.3	18.6
9	18.5	18.5	16.1

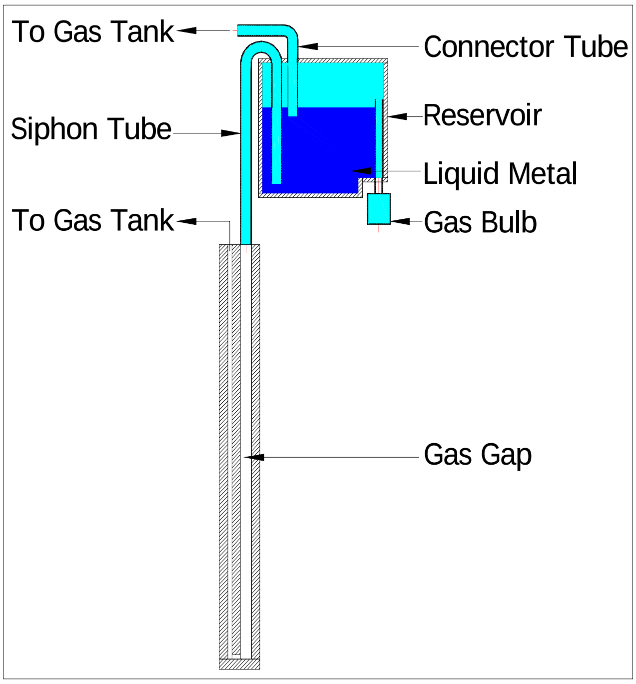


FIG. XXIX-14. Schematic of molten metal gas gap filling system.

Intermediate circuit

A system of vessels filled with liquid Pb-Bi eutectic alloy acts as the intermediate circuit. Each vessel is a molten bath providing heat to the heat utilizing systems. The vessels receive heat from the upper plenum through the heat pipes so that there is very little coolant temperature drop.

XXIX-2.1. Description of the turbine generator plant and systems

High efficiency direct thermal energy to electricity conversion devices has been planned at this stage. If at all, a gas turbine based system was adopted in the design, a commercially available system would be preferred.

XXIX-2.3. Systems for non-electric applications

Hydrogen production by splitting water [XXIX-9]

Water based hydrogen production processes are inherently advantageous due to the abundance of renewable carbon-free resources and the prevention of environmental degradation associated with fossil fuel based hydrogen producing processes.

Thermo-chemical processes form a group of methods, wherein water splitting is carried out by a multiple step chemical reaction to reduce the decomposition temperature requirement from 2773 K (for direct thermolysis) to 823–1123 K depending on the chemical process adopted. Considering the vast Indian thorium resources with the capability of satisfying the country's long term energy needs, nuclear energy is the best long term and sustainable source of energy required for hydrogen production. High temperature nuclear reactors like the CHTR are suitable for supplying heat to endothermic process steps. Large scale hydrogen production using high temperature nuclear reactors offers an attractive concept for future CO_2 free and efficient energy systems. The three most promising thermo-chemical processes viz., iodine-sulphur (I-S), calcium - bromine (Ca-Br) and copper - chlorine (Cu-Cl) have been short listed for development. Apart from the feasibility of the processes, efficiency, stability of the closed loop operation, safety, cost, materials etc., are key issues, which must be addressed in the development. Harnessing the intrinsic potential of these processes for commercial scale production of hydrogen is a scientific and technological challenge. BARC has plans to develop thermo-chemical processes. The Iodine-Sulphur process, which offers the highest quoted efficiency (up to 57%), has been identified for initial R&D. Several laboratories have demonstrated the technical feasibility and close cycle operation of the process. This is a three step process involving formation and decomposition of hydriodic acid (HI) and sulphuric acid (H_2SO_4). The thermo-chemical reactions of the I-S process are given below:

$H_2SO_4 \rightarrow SO_2 + H_2O + 1/2\ O_2$ (1123 K)

$I_2 + SO_2 + 2H_2\ O \rightarrow 2HI + H_2SO_4$ (393 K)

$2HI \rightarrow I_2 + H_2$ (727 K) or (393 K) as per the process selected.

Various reactions involved in the I-S process are shown in Fig. XXIX-15. A multistage R&D plan to develop the I-S process for hydrogen production, as drafted by BARC, is shown in Fig. XXIX-16.

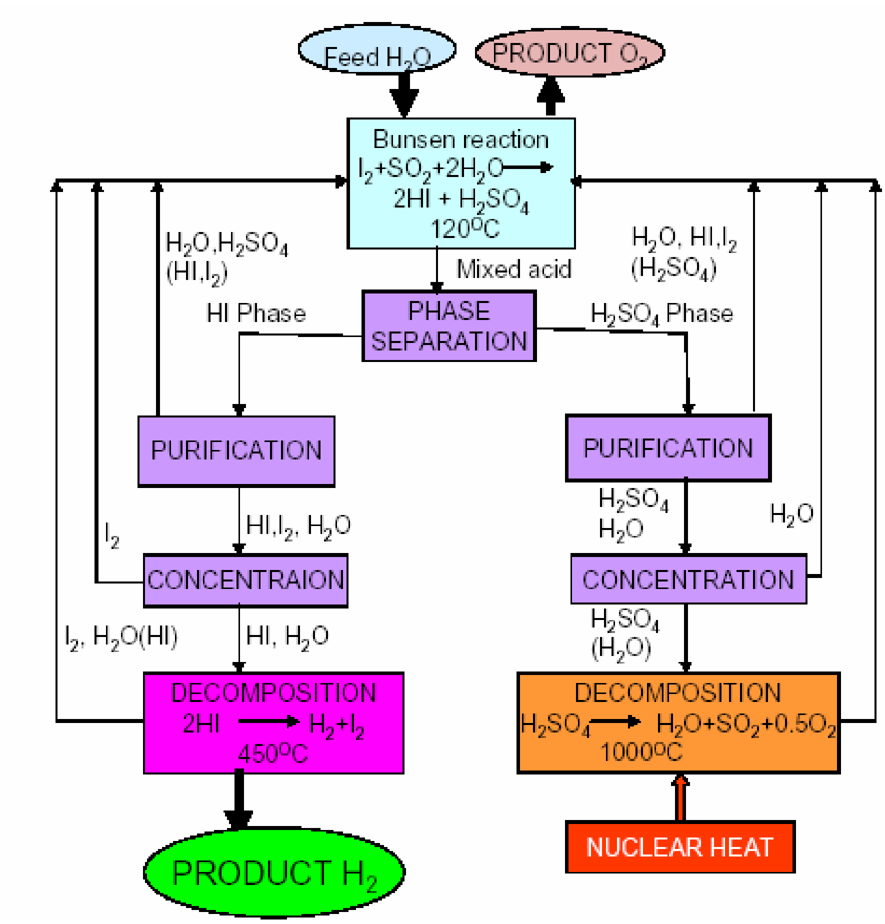


FIG. XXIX-15. Flow sheet for hydrogen production by I-S thermo-chemical process.

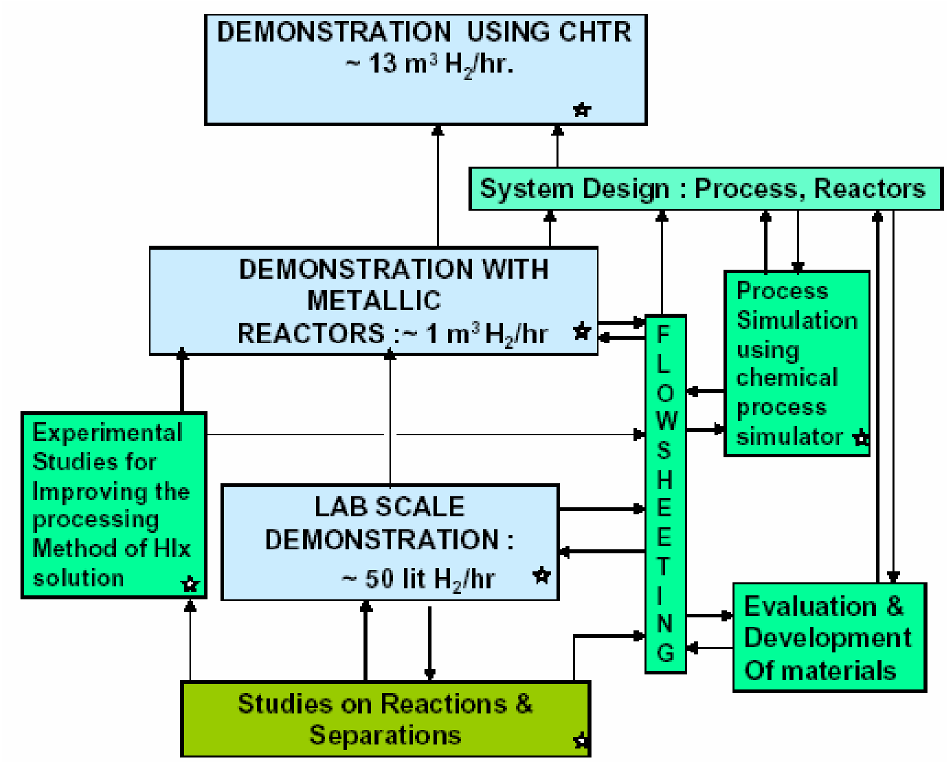


FIG. XXIX-16. Indian R & D plan for hydrogen production by I-S thermo-chemical process.

The system for the I-S process will be integrated into the CHTR through a set of heat exchangers in which the other fluid would be helium. It is necessary to provide process heat at the different temperatures required by the three chemical processes.

XXIX-2.4. Plant layout

The CHTR would be located inside a pit with sealed barrier of reinforced concrete and steel covers, which would protect the reactor against external events.

REFERENCES

[XXIX-1] DEPARTMENT OF ATOMIC ENERGY, Shaping the third stage of Indian nuclear power programme, (DAE monograph, 2001), Mumbai, India (2001).

[XXIX-2] KAKODKAR, A., Shaping the 3rd stage of Indian nuclear programme, Indian Nuclear Society, 12th Annual Conference – INSAC 2001, Indore, India (2001).

[XXIX-3] DEPARTMENT OF ATOMIC ENERGY, Our collective vision, Government of India, Department of Atomic Energy, Document No. 11, Mumbai, India (2004).

[XXIX-4] MAHESHWARI, N.K., VIJAYAN, P.K., SAHA, D., SINHA, R.K., Passive safety features of Indian innovative nuclear reactors, IAEA-TECDOC-1451, Vienna (May 2005).

[XXIX-5] SINHA, R.K., KADOKAR, A., The road map for a future Indian nuclear energy system, Innovative Technologies for Nuclear Fuel Cycle and Nuclear Power (Proc. Int. Conf. June 23–26, 2003) IAEA-CN-108-12, Vienna (2003).

[XXIX-6] SINHA, R.K., KAKODKAR, A., Indian programme related to innovative nuclear reactor technology, Indian Nuclear Society, 13th Annual Conference – INSAC 2002, (Proc. Int. Conf., Mumbai, India, 2002).

[XXIX-7] VAIDYA, V.N., Coated particle fuels for high temperature reactors – Indian, programme, Technical Meeting on Current Status and Future Prospects of Gas Cooled Reactor Fuels (Paper presented at IAEA Meeting, Vienna, Austria, 2004).

[XXIX-8] SATHIYAMOORTHY, D., BARC, Personal communication. 2004–2005.

[XXIX-9] PRASAD, C.S.R., BARC, Personal communication based on a report on hydrogen production, submitted to the Ministry for Non-conventional Energy Sources, India, 2004–2005.

ANNEX XXX

MOLTEN SALT REACTOR FOR SUSTAINABLE NUCLEAR POWER – MSR FUJI

ITHMSI,
Japan

XXX-1. General information, technical features and operating characteristics

XXX-1.1. Introduction

The FUJI is a simplified molten salt reactor (MSR) being designed for operation in a closed thorium-uranium (Th-U) fuel cycle. A direct predecessor of the FUJI is the molten salt breeder reactor (MSBR) [XXX-1, XXX-2] based on the concept of a "single-fluid molten fluoride fuel", developed in the Molten-Salt Reactor Programme (MSRP) at Oak Ridge National Laboratory (ORNL), USA, during 1950–1976. This programme has resulted in the development and demonstration of the basic MSR technology, especially through excellent operation of the experimental molten salt reactor MSRE in 1965–1969; it also produced a conceptual design of the MSBR [XXX-2].

The MSBR was a Th-U cycle thermal breeder applying continuous chemical processing of fuel in situ and periodic core graphite replacement to improve breeding performance [XXX-3].

The FUJI concept was proposed in connection with the philosophy of the thorium molten salt nuclear energy synergetic system (THORIMS-NES) [XXX-4 to XXX-6], explained in more detail in Section XXX-1.5. Different from the MSBR, the FUJI is a concept of a simplified molten salt reactor without continuous chemical processing and periodic core graphite replacement, aimed at attaining near-breeder characteristics in a Th-U closed fuel cycle.

Since 1985, conceptual designs of the FUJI for several fuel cycle options have been developed; including the Pu-burning version (FUJI-Pu) designed to incinerate Pu and minor actinides (MA) from spent solid U fuel or weapons-grade Pu. It was suggested that a miniFUJI pilot plant of about 7 MW(e) is constructed first; this construction has been suggested on the site of the Russian Federal Institute of Technical Physics in Snezhinsk [XXX-7]. A prototype FUJI-Pu and FUJI-233U [XXX-8 to XXX-11] of 100–300 MW(e) could then follow as the next logical steps.

The work of the FUJI had been initiated in the Japan Atomic Research Institute (JAERI, currently within the Japan Atomic Energy Agency (JAEA)); then proceeded at the Tokai University with the cooperation of the Toyohashi Technical and Science University, Fujitsu Corporation, Toshiba Corporation, the Hokkaido University, Electricite de France (EDF), ORNL (USA), Lawrence Livermore National Laboratory (LLNL, USA), Russian Federal Institute of Technical Physics in Snezhinsk, Russian Research Centre "Kurchatov Institute", Joint Institute for Power and Nuclear Research in Sosny (Belarus), Nuclear Research Institute Rez (Czech Republic) and many other organizations; with notable contributions coming also from individual researchers.

The principal stakeholder is the International Thorium Molten-Salt Institute (ITHMSI), President K. Furukawa and Chief Manager Y. Kato. In the USA, Energy Frontiers International (President Dr. J. Pleasant) and Vallecitos Research Associates (President Dr. R. Moir) are cooperating closely with ITHMSI. The Russian Federal Institute of Technical Physics in Snezhinsk (Director, Dr. E. Avrorin; Deputy Science Director, Dr. V. Simonenko) is working with ITHMSI and other research teams in the Russian Federation.

XXX-1.2. Applications

The FUJI nuclear power plant (NPP) is designed to co-generate electricity along with hydrogen production and/or seawater desalination.

XXX-1.3. Special features

The FUJI concept incorporates the following three principles [XXX-4, XXX-5, and XXX-12]:

(a) Thorium utilization [XXX-13];

(b) Application of molten-salt fuel technology; and

(c) Separation of the functions of fissile material production (which is performed in separate breeder reactors) and energy generation at NPPs.

Special features of the FUJI are as follows:

- Modular design, providing for a variety of outputs from multi-module FUJI plants;
- Lifetime core operation without on-site refuelling; the fuel salt composition needs periodical regulation but this operation is performed remotely by using a drain tank of special design and, therefore, the reactor vessel need not be opened during its lifetime;
- Factory fabrication - the reactor vessel with installed graphite moderator would be factory fabricated and assembled;
- Flexible applications - the FUJI concept offers flexibility in the selection of fuel cycle options, such as the following:

(a) The FUJI can operate using any kind of fissile materials (^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu, etc.) or combinations thereof;

(b) Not only fissile materials but also several fission products (FP) and chemical impurities might be flexibly accommodated in the fuel salt without no serious physicochemical issues or penalties on the nuclear performance;

- Flexibility in size and high conversion ratio; the FUJI-233U is size-flexible and near self-sustainable in the fuel without its continuous chemical processing;
- An option of effective Pu and minor actinides (MA) incineration - Pu and MA from the dismantled weapons and spent solid fuel of present-day power reactors could be incinerated directly, by using them as a start-up fuel of the FUJI-Pu. The conversion of spent fuel to fluoride salt can be accomplished economically using the initial fluorination step of the FREGATE process and its modifications, without the refabrication of solid fuel [XXX-14].

XXX-1.4. Summary of major design and operating characteristics

Major design and operating characteristics of the FUJI-233Um – the newest conceptual design of the FUJI family – are summarized in Table XXX-1.

A simplified schematic diagram of the FUJI plant is given in Fig. XXX-1. This simplified figure is common to all MSRs in the FUJI series. The temperatures shown in this figure are typical examples.

The standard fuel salt of the FUJI is ^{7}LiF-BeF_2-ThF_4-UF_4, and the fuel salt flows upward through the core where it is heated, see Fig. XXX-1.

Table XXX-1. SUMMARY OF MAJOR DESIGN AND OPERATING CHARACTERISTICS OF FUJI-233Um

CHARACTERISTIC	VALUE
Major design characteristics	
Installed capacity (thermal)	450 MW
Installed capacity (electric)	200 MW
Mode of operation	Base load and/or load follow
Load factor (target)	90% for base load operation
Availability	90%
Type of fuel	Molten fluoride salt: $LiF-BeF_2-ThF_4-UF_4$
Fuel enrichment	Initial salt composition: 71.75-16-12-0.25 mol.%; with 2.0 weight % of fissile material in heavy metal
Type of coolant	Molten fluoride salt: $LiF-BeF_2-ThF_4-UF_4$
Type of moderator / reflector	Graphite
Type of structural material	Modified Hastelloy-N; composition: Ni(base), Mo(11-13), Cr(6-8), Nb(1-2), Si(0-1); weight %
Core geometry	Cylindrical
Core characteristic dimensions/ power density	Core-I; radius: 2.2 m, graphite fraction: 64 vol. % Core-II; outer radius: 2.8 m, graphite fraction: 71 vol. % Core-III; outer radius: 3.0 m, graphite fraction: 76 vol. % Core height: 2.1 m Power density in the core: 7.3 kW/l
Vessel type	Closed; tank type
Vessel characteristic dimensions	- Inner diameter: 6.84 m - Height: 2.94 m - Wall thickness: 5.0 cm
Number of circuits	Three, including an intermediate molten salt heat transport system
Simplified schematic diagram	See Fig. XXX-1; the reactor vessel contains graphite as a moderator/reflector. Fuel salt circulates through the core, the heat is transported to the secondary (intermediate) circuit through the heat exchanger; from the intermediate circuit heat is supplied to the steam of the power circuit via a steam generator
Neutron-physical characteristics	
Temperature reactivity coefficient	-3×10^{-5}dK/K (initial state)
Void reactivity coefficient	0.07 %dK/%void (initial state) Voiding by boiling will not occur in fuel salt, because the boiling temperature is 1800 K and higher than the maximum fuel temperatures reached even in accident conditions

CHARACTERISTIC	VALUE
Neutron-physical characteristics (continued)	
Burn-up reactivity swing	0.001 dK/30 EFPD (Effective Full Power Days) Since the conversion ratio of FUJI is ~0.97 and since fresh fuel is added to the core periodically (at every 30 EFPD), the burn-up reactivity swing is very small. The above value is applicable throughout the whole operation cycle
Peaking factors	Maximum axial peaking factor in the core (F_Z): 1.3 Maximum lateral peaking factor in the core (F_{xy}): 1.2
Approach to power flattening	Three radial sub-zones in the core
Reactivity control mechanism	
Control rods, type 1	Graphite regulating rods
Control rods, type 2	B_4C based shutdown rods
Other mechanisms	Fuel salt drain system
Number of independent active reactor control and protection (RCP) systems	3
Cumulative worth for each RCP system	(1)Graphite control rod for normal operation: 2 rods; total control rod worth: 0.12%dK (2) Emergency shutdown rod (B_4C particles in clad): 4 rods; total control rod worth: 3.6%dK at one-rod stuck condition (3)Alternate shutdown by draining fuel-salt: well below critical
Thermal-hydraulic characteristics	
Cycle type	Indirect cycle Supercritical steam Rankine cycle; steam conditions at turbine inlet: p=24 MPa, T=810 K
Thermodynamic efficiency	44.4%
Circulation type	Forced
Core inlet coolant temperature	840 K
Core outlet coolant temperature	980 K
Core flow rate	0.711 m^3/s (Fuel salt volume within vessel=21.1 m^3; total =26.4 m^3)
Pressure in the primary circuit	0.5 MPa
Temperature limit for fuel	1800 K (boiling temperature)
Temperature limits for structural materials	Graphite: 3000 K Hastelloy N: 1400 K
Maximum temperature of fuel	985 K
Average temperature of fuel	910 K
Maximum temperatures of structural materials in normal operation	Graphite: 1000 K Hastelloy N: 980 K

CHARACTERISTIC	VALUE
Thermal-hydraulic characteristics (continued)	
Average temperature of structural materials in normal operation	Graphite: 920 K Hastelloy N: 910 K
Operating cycle parameters	
Maximum/average discharge burn-up of fuel	100 000 MW day/ton; maximum burn-up has no meaning for liquid-fuel reactors; the discharged fuel can be used in next reactors
Fuel lifetime	Longer than plant lifetime.
Period between refuelling in effective full power days (EFPD)	Fissile feeding: 2 kg of ^{233}U is supplied to the core in the form of LiF-UF_4 (73-27 mol. %) at every 30 EFPD. Fertile feeding: 67 kg of Th is supplied to the core in the form of LiF-BeF_2-ThF_4 (72-16-12 mol. %) at every 150 EFPD.
Mass balances/ flows of fuel and non-fuel materials	
Initial ^{233}U inventory ^{233}U feed in 30 years (capacity factor 0.90) (^{233}U+^{235}U) inventory after 30 years Natural Th consumption in THORIMS-NES (see Section XXX-1.5)	800 kg 755 kg 1107 kg; fuel salt LiF-BeF_2-ThF_4-UF_4, which contains the mentioned fissile materials, can be used in next MSRs 1000 kg/GW(e)/EFPY. No cladding material is required.
Design basis lifetime for reactor core, vessel and structures	30 years without graphite replacement
Design and operating characteristics of systems for non-electric applications	
Hydrogen production Seawater desalination	120 tons H2/day at 450 MW(th) 28 000 m^3/day from multi-effect distillation (MED) at 450 MW(th) with electricity co-generation Note: the applications could be combined; their shares could be varied
Economics	
Construction cost	1 584 US$/KW(e) for 1 GW(e) plant
Operation and maintenance (O&M) cost	0.58 cent/kWh
Fuel cycle cost	0.30–0.50 cent/kWh
Waste disposal cost	0.10 cent/kWh
Decommissioning cost	0.04 cent/kWh

Centrifugal pumps transfer the outlet fuel salt to heat exchangers where the heat is transferred to a secondary coolant salt of $NaBF_4$-NaF, which transports the heat to a super critical steam generation system, resulting in an overall thermal efficiency of more than 44% [XXX-8].

In the industrial countries, ultra supercritical (USC) turbine systems with a steam condition of 25 MPa and 870 K at the turbine inlet have been proven by the operation of coal and gas-fired power plants, yielding a thermal efficiency of more than 45%.

Typical fast and thermal neutron flux distribution in the FUJI-233Um core is shown in Fig.XXX-2.

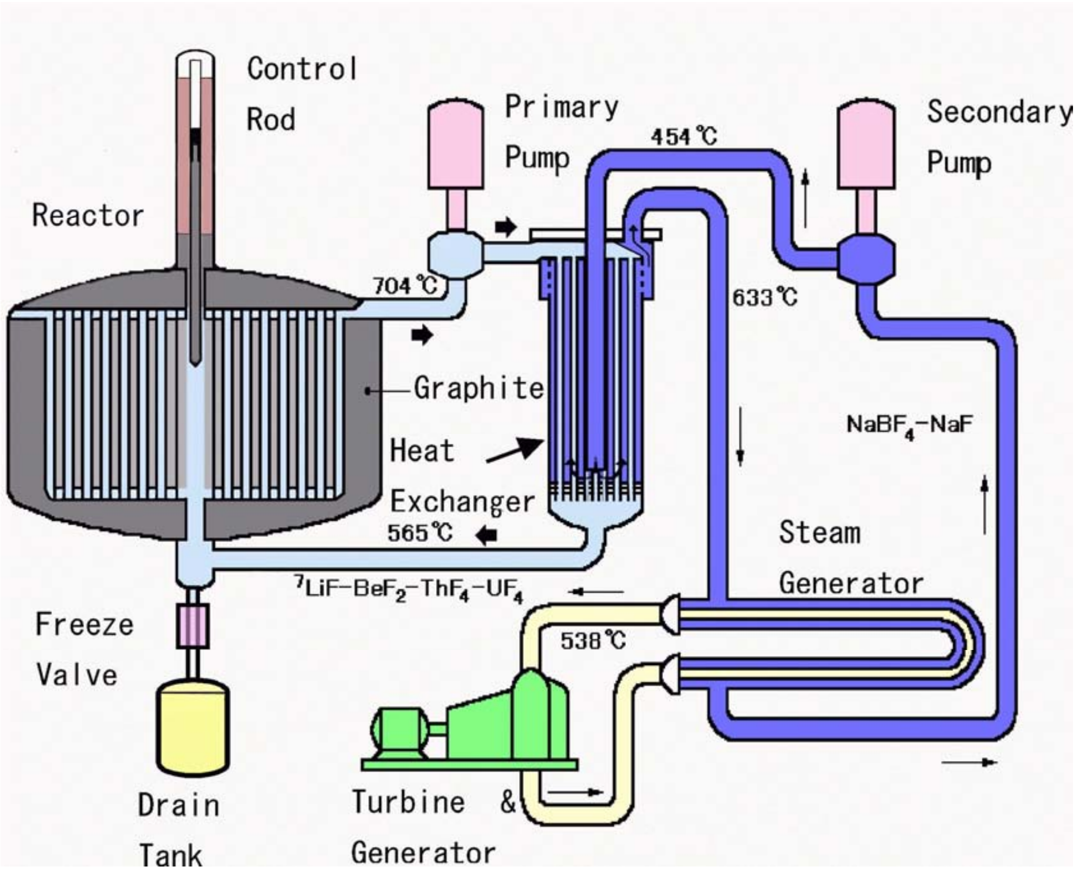


FIG. XXX-1. Simplified schematic diagram of FUJI.

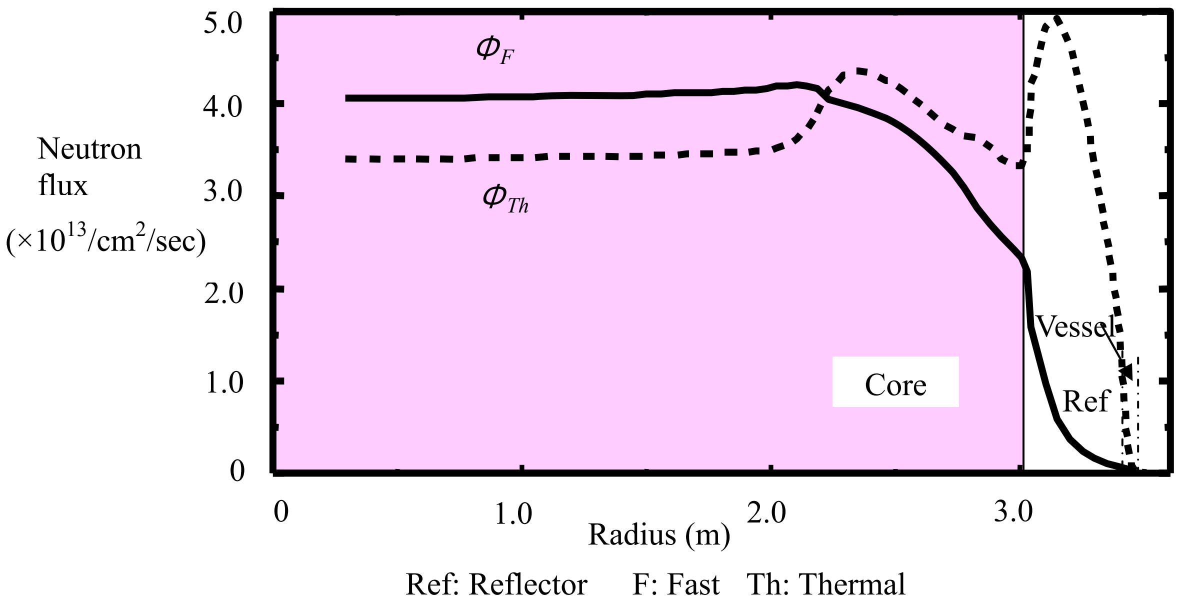


FIG. XXX-2. Neutron flux distribution in FUJI-233Um core.

XXX-1.5. Outline of fuel cycle options [XXX-5, XXX-13, and XXX-15]

Concept of a Thorium Molten Salt Nuclear Energy Synergetic System (THORIMS-NES)

Requirements for nuclear energy system

The concept of a Thorium Molten Salt Nuclear Energy Synergetic System (THORIMS-NES) [XXX-4 to XXX-6, XXX-16] is based on the analysis of the historical trend in substitution of energy sources, as illustrated by the diagram of Fig. XXX-3.

According to this concept, the use of fossil fuel as a major energy source would not continue throughout the 21st century, even though cleaner natural gas would be used in the first half of the century. Nuclear energy has a chance to substitute for fossil fuel as an energy source; to achieve this, certain issues associated with safety, radioactive waste management, proliferation resistance and economics of nuclear energy need to be resolved.

However, according to the same analysis of historical trends in substitution of energy sources, nuclear fission systems are needed only as an interim solution between fossil and solar technologies (fission technologies of energy intensity similar to that of the sun), to ensure global survival into the 22nd century, see Fig. XXX-3. To realize such change, nuclear energy would need to achieve the growth rate with a doubling time of about 10 years and reach the peak output of about 10 TW(e) (~30 times larger than present) by ~2065. A huge nuclear industry would be required to comply with the requirements of such rapid growth and maintenance of the huge nuclear energy system.

Need of a rapid transfer to thorium fuel cycle

In THORIMS-NES concept, the need of a rapid transfer to thorium-based fuel cycle is justified by the following arguments [XXX-4 to XXX-6, XXX-13, and XXX-15]:

- Natural thorium is geochemically 3 times more abundant than uranium;
- The by-product of any uranium based fuel cycle is plutonium, which is generically an attractive material for a weapon programme;
- Shouldering the function of fuel breeding makes power reactor more complex, less economical and potentially less safe; therefore, the functions of fuel breeding and power production could be separated, which would allow power reactors to be more simple, size- and site-flexible;
- Fuel self-sufficiency of power reactors may be a desired quality along with simplicity and flexibility in siting and applications; it is noted that the FUJI-233Um concept without continuous chemical processing and core-graphite replacement may be a good candidate for such a system;
- The breeding of fissile material could be performed separately, not using the fission process but proton spallation or deuterium-tritium (DT) fusion processes. During the 1980s, the technical feasibility of an accelerator molten-salt breeder (AMSB) [XXX-6 to XXX-8 and XXX-12,] was established based on a “single-fluid target/blanket concept” using the same molten-salts as the FUJI, coupling with a proton beam of about 1 GeV. After starting operation of the AMSB, a thorium cycle nuclear energy system could gradually be achieved.

Breeding system with a short effective doubling time

The breeding ratio that might be offered by fission reactors yields a doubling time (DT) that is too long for the nuclear energy systems shown in Fig. XXX-3(D).

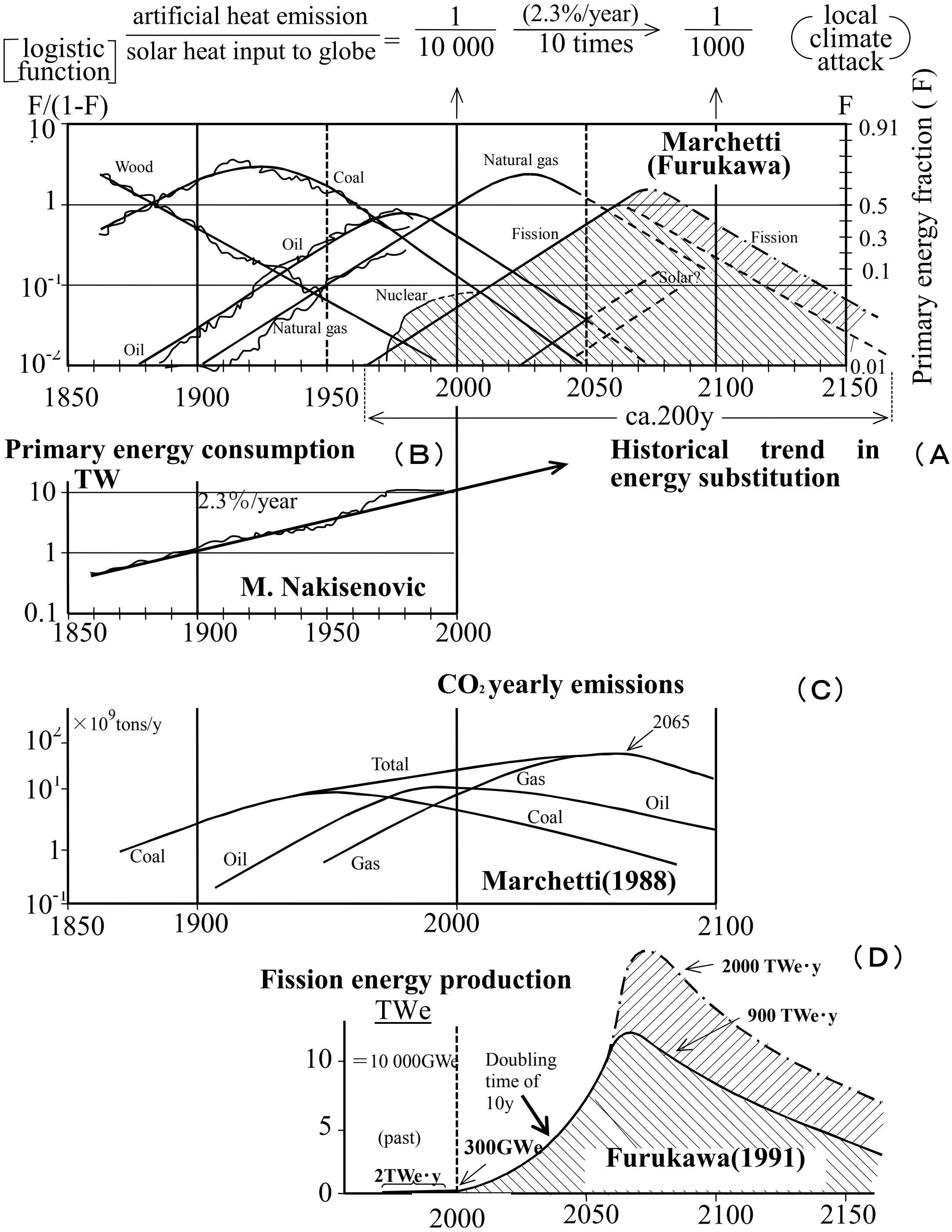


FIG. XXX-3. Global future energy prediction [XXX-4 to XXX-6].

(A) is a further extension of the Marchetti's estimate of historical trends in energy substitution
(B) is yearly growth-rate of the world primary energy consumption;
(C) the predictions of CO_2 yearly emission from fossil fuels; and
(D) nuclear fission energy production estimated based on the assumptions of (A) and (B).

Even in the MSBR developed at ORNL, the expected DT was about 20 years, which is less that than required for the THORIMS-NES system. Therefore, the concept of “a doubling time” for THORIMS-NES is somewhat different from that used by convention for systems with fast breeder reactors; in THORIMS-NES it is assumed that short effective doubling time can be achieved flexibly by increasing the capacity and numbers of the AMSB rather than FUJI-MSR power reactors. In the AMSB, a 1 GeV/300 mA proton accelerator produces about 400 kg/year of ^{233}U even if there is no fissile material but only fertile ^{232}Th in the initial target fuel salt. The initial ^{233}U inventory of the FUJI-233Um of 200 MW(e) is about 800 kg, so the AMSB can support the commissioning of one FUJI-233Um every two years. Also, a high-gain type AMSB with Pu added to the target/blanket salt can produce more ^{233}U and generate a sufficiently large heat output to make the AMSB a self-sustained system. Such AMSB will be able to start up a sufficient number of small and medium sized MSRs to meet the steep growth of energy demand and replace fossil energy. It could be noted that most of the basic technologies in the THORIMS-NES have been proven except the development of a high current proton accelerator for the AMSB.

Positive heritage to the next century

In THORIMS-NES, there is no need for large-scale fuel cycle facilities such as those for solid fuel recycling, spent fuel interim storage, etc. Alternatively, the AMSBs with batch chemical processing facilities would perform effective and efficient nuclear transmutation in the fuel cycle, using a large amount of low-cost excess fission neutrons including high energy neutrons from the AMSB in the recession age of the thorium era, i.e., after about 2070 [XXX-12, XXX-17], as shown in Fig. XXX-3.

Following the decline of nuclear energy, the AMSBs could remain and act as neutron sources for transmutation (incineration) of the remaining nuclear materials, for materials research and development, and for medical use (proton irradiation). The same goals are being pursued by the Spallation Neutron Source (SNS) project in the USA or the Japan Proton Accelerator Research Complex (J-PARC) project in Japan, which are the projects of experimental facilities of intense spallation neutron sources driven by proton accelerators.

Fuel cycle in the THORIMS-NES

Fuel cycle concept of the THORIMS-NES is illustrated by Fig. XXX-4. As it is shown in the figure, the spent fuel-salt after finishing its lifetime in the FUJI reactors will be sent back to regional centres distributed in the world and safeguarded. The salt would be processed in a batch mode to remove ^{233}U (by fluorination) and some fission products, for which the contents will be decided from the integral optimization including such criteria as material compatibility, neutron economy, cost minimization, etc.

The decontaminated diluent salt would then be used to produce make-up fuel for the FUJI-233Um. Specifically, the fertile salt would be charged to the storage tank of the AMSB to keep the ^{233}U content in the target/blanket salt constant, such as ~0.5 mol. %. The salt taken from the storage tank will be enriched by adding the removed ^{233}U and sent as the fuel-salt to the FUJI power station sites, see Fig. XXX-4.

The important aspects of this fuel cycle are the following:

- A single-liquid breeding fuel cycle without solid species is being realized; the total system is simply integrated by one phase of molten fluorides based on the Flibe ($^{7}LiF\text{-}BeF_2$ binary salt) as a solvent;
- The working medium used is suitable for chemical processing - this is a stable ionic liquid with the following characteristics suitable for nuclear systems:

(1) No radiation damage;

(2) Low vapour pressure;

(3) Chemical inertness;

(4) High heat capacity;

(5) High solubility of several widely used ions;

(6) Essentially no possibility to regain criticality due to the lack of hydrogen species;

- The working medium used is a physicochemically well established medium - the basic theoretical chemical data were well established by the ORNL-MSR program [XXX-18];
- The behaviour of the spallation products is very similar to that of the fission products; the existing fission product technology can be applied to treatment of the spallation products without inducing any serious problems, because the concentration of spallation products in the AMSB salt is one order of magnitude lower than that of the fission products in the FUJI, due to the larger number of neutrons generated by spallation reactions as compared to fission.

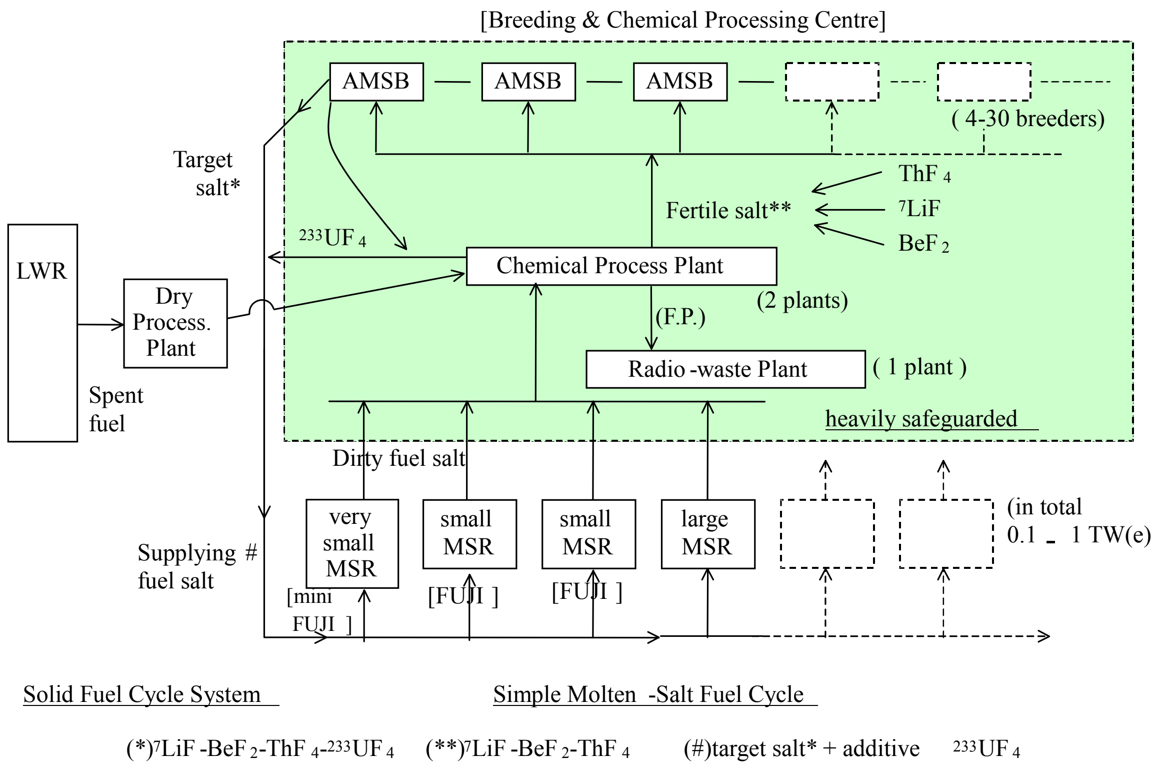


FIG. XXX-4. Schematic of the thorium molten-salt breeding fuel cycle system; green colour envelopes the fuel cycle facilities located within regional fuel cycle centres.

The estimated production of fission products and their final concentrations in salt after 15 years of full power operation of the FUJI reactor are given in Table XXX-2.

Fission product elements of Group I are removed during the operation. Kr and Xe are released to the He cover gas due to lack of solubility. T (tritium) is transferred to the coolant salt through the heat-exchanger metal wall and caught as a water species THO (tritium-hydrogen-oxygen) in the cover gas. This process was investigated in ORNL; the expected radioactivity transfer rate due to tritium is less than 1 Curie /day.

Group II elements are stably dissolved in low quantities in salt; their expected impact is negligible. However, this should be confirmed by reactor operation over a full lifetime.

Group III elements will be floated or segregated in the stagnant salt zone, although their total amount is not large. The behaviour should be reconfirmed by pilot-plant operation.

Synergy with the solid fuel reactor systems could be achieved as shown in the left side of Fig.XXX-4. The spent solid fuel could be treated by a FREGATE type process [XXX-14, XXX-15], for example, as developed in 1980s through the cooperation of France, the Soviet Union and the Czech Republic. This process provides for applying F_2 gas flame reactor fluorination technology, with which Pu-containing fluoride salts are obtained without returning to a solid phase and in a form suitable for direct use in the FUJI reactor. Such process is, therefore, convenient to produce initial fissile inventory of the FUJI, which in this case may act as a plutonium burner. Then, there would be no need in rapid investment for the AMSB development, and the integrated system shown in Fig. XXX-4 could be realized gradually, over a period of more than 30 years.

TABLE XXX-2. PREDICTED ACCUMULATION OF FISSION PRODUCTS IN FUJI, AT THE END OF A 30-YEAR LIFETIME
(in a/o (atomic %), m/o (mole %), and kg)

FP GROUP	PRODUCTION from ^{233}U		AMOUNT dissolved in fuel salt	AMOUNT separated to gas phase
Group I	Xe	27.6 a/o		312.0 kg
	Kr	6.5 a/o		45.9 kg
	T			Circa 0.1 kg
Group II	I	2.6 a/o	27.6 kg [0.032m/o]	
	Br	0.42 a/o	2.8 kg [0.005m/o]	
	Te	4.1 a/o	43.5 kg [0.050m/o]	
	Cs	17.8 a/o	56.0 kg [0.060m/o]	144.0 kg
	Rb	7.2 a/o	0.5 kg [0.001m/o]	51.0 kg
	Sr	11.8 a/o	28.1 kg [0.047m/o]	60.5 kg
	Ba	6.3 a/o	0.3 kg [0.005m/o]	72.0 kg
	Ce	14.1 a/o	166.0 kg [0.170m/o]	
	Nd	16.4 a/o	199.0 kg [0.200m/o]	
	Y	5.9 a/o	1.5-7.5 kg[0.003-0.013m/o]	42-37 kg
	Zr	30.0 a/o	232.0 kg [0.370m/o]	2-10 kg
Group III	Mo	21.6 a/o	[Deposit 175.9 kg]	2-10 kg
	Se	0.9 a/o	6.1 kg [0.010m/o]	
	Sn	0.3 a/o	3.0 kg [0.004m/o]	

XXX-1.6. Technical features and technological approaches definitive for MSR FUJI performance in particular areas

XXX-1.6.1. Economics and maintainability

One estimate of the MSR economics has been reported, although these data are a little old [XXX-2, XXX-19]. Assuming the same power output from a 1000 MW(e) conventional light water reactor (LWR) and a molten salt reactor (MSR), such cost components as capital cost, fuel cycle cost, operating and maintenance cost were compared. The conclusions were as follows:

(1) Capital cost of the MSR could be almost the same as that of the LWR. There are many pros and contras for choosing between these two reactors. The MSR has 3 circuits with an intermediate heat transport system similar to fast breeder reactors. On the other hand, the thermal efficiency is ~30% higher than that in a pressurized water reactor, the core pressure is very low, and the safety system is simplified.

(2) Fuel cycle cost of the MSR could be lower than that of the LWR. This is because the MSR is a high-conversion reactor and requires quite small amounts of thorium and ^{233}U (fissile isotope) to be loaded over the plant lifetime while the LWR requires much larger amounts of natural uranium and large amounts of ^{235}U (fissile isotope) to be loaded during reactor operation. In addition, the MSR uses a liquid fuel and does not require fuel fabrication as does the LWR.

(3) Operation and maintenance (O&M) costs of the MSR could be almost the same or less than those of the LWR according to the publications, although the MSR would need remote maintenance because molten fuel salt of high radioactivity circulates outside the reactor vessel. However, the MSR can operate longer than the LWR and save the downtime.

(4) Plant capacity factor of the MSR could be higher than that of the LWR because it does not need fuel shuffling as does the LWR.

Considering the total component costs, it has been concluded that the economy of the MSR could be almost the same or better than that of the LWR.

More recently, power generation cost for the MSR and a pressurized water reactor (PWR) was re-evaluated at the LLNL [XXX-20], using the original evaluation by the ORNL [XXX-2, XXX-19]. To make a fair comparison, a 1 GW(e) plant size was assumed for both plants. Five cost components were considered, including capital cost, O&M cost, fuel cost, waste disposal cost, and decommissioning cost. Assuming the capacity factor of the MSR as 90% and 80% for the PWR, the results are shown in Table XXX-3; the conclusion is that the MSR could be 20% to 25% cheaper than the PWR in total power generation cost.

In Table XXX-3, only the fuel cycle cost value was re-examined in line with the recent FUJI-233Um design because the original LLNL results were based on a concept of MSR feeding with the denatured ^{235}U. Also, the PWR fuel cycle cost data were re-evaluated using recent data.

TABLE XXX-3. POWER GENERATION COST COMPONENTS OF A MSR (US$ cent/kWh)

	MSR
Capital cost (based on construction cost of 1 584$/KW(e))	2.01
O&M cost	0.58
Fuel cycle cost	0.30 to 0.50
Waste disposal cost	0.10
Decommissioning cost	0.04
Total cost	3.03 to 3.23

The FUJI-series reactors need a simpler infrastructure including a small amount of one-time fuel transportation, could be located closer to consumers due to excellent safety characteristics and small site area, etc. Therefore, the total cost of the FUJI for consumers could be lower.

The MSR has very simple reactor internals and safety systems, which could make its maintenance simpler. On the other hand, fuel salt with high radioactivity circulates in the high-temperature containment and equipment such as primary pumps or a primary heat exchanger and must be inspected by the remote-maintenance equipment. Drive motors, mechanisms of primary pumps and control rods are located outside the high temperature containment making maintenance easier. In this regard, recent developments in remote-handling technology can be applied.

XXX-1.6.2. Provisions for sustainability, waste management and minimum adverse environmental impacts [XXX-5, XXX-12]

As it was already mentioned, the FUJI-233Um MSR concept is being developed for operation within the THORIMS-NES energy system shown in Fig. XXX-3.

Thorium resources on earth are non-localized but geochemically three times more prevalent than the uranium ones. Thorium resources have already been confirmed at about 2 million tons and estimated at about 4 million tons as shown in Table XXX-4.

TABLE XXX-4. ESTIMATED WORLD THORIUM RESOURCES [XXX-21] (thousands of tons)

CONTINENT	COUNTRY	RRA*	RSE**	TOTAL	%
	Greenland	54	32	86	
	Norway	132	132	264	6.4
	Turkey	380	500	880	21.4
Europe	**Total**	**566**	**724**	**1290**	**31.4**
	Brazil	606	700	1306	31.8
	Canada	45	128	173	
	United States	137	295	432	10.5
America	**Total**	**790**	**1125**	**1915**	**46.6**
	Egypt	15	280	295	7.2
	Niger			29	
	South Africa	18		115	
Africa	**Total**	**36**	**309**	**479**	**11.7**
	India	319		391	7.8
Asia	**Total**	**343**	**30**	**403**	**9.8**
World total		**1754**	**2188**	**4106**	**100.0**

* RRA is for resources reasonably achievable

** RSE is for resources supplementary estimated

Thorium resources necessary for a 1000 TW(e)-year production globally foreseen by the THORIMS-NES for the 21st century, Fig. XXX-3, will be only about 2 million tons (assuming one-third of them will fission), which is comparable to about 1.5 million tons of uranium already extracted from the earth. Thorium could be obtained from the “heavy sand” of beaches with relatively little pollution.

Additionally, 0.6 million tons of Li, 0.2 million tons of Be and 2 million tons of F for fuel-salt would be required in the 21st century but one could expect a reduction of one order of magnitude due to the recycling technology. In addition to this, the used Ni alloy and part of the used graphite could be recycled or reused in the THORIMS-NES [XXX-17].

Necessary water resources could be decreased by the high thermal efficiency. Required land resources could be lower owing to the safe, simple and compact system, low radioactive waste and simple infrastructure related to the THORIMS-NES.

The THORMIS-NES features that contribute to minimization of radioactive waste and reduction of adverse environmental impacts are as follows [XXX-17]:

(i) The production of Pu and MA (Np + Am + Cm) in the FUJI-233Um is 0.5 kg and 0.3 g per every 1 GW(e)-year on average, respectively, i.e. very small compared with LWRs (where the corresponding amounts are 230 kg and 25 kg, respectively);

(ii) The required specific volumes of chemical processing, fuel preparation and maintenance are lower for systems with molten salt fuel and high conversion ratio, resulting in a reduced low-level radioactive waste production;

(iii) Fuel-salt can accommodate fairly large amounts of fission products, which would decay and be destroyed while circulating in the molten salt fuel cycle;

(iv) Nuclear transmutation might be performed in a fuel cycle based on molten salt, specifically, in the decline period of nuclear energy (Fig. XXX-3); in this, the incineration of all remaining nuclear materials could be effectively performed within hundreds of years.

Maintenance of the FUJI-233Um primary system would be fully performed by remote handling systems. Ordinarily, operation of the remote handling system would take place after the fuel salt is drained; in this, it could be mentioned that fuel salt does not wet the surfaces of Hastelloy N and graphite.

XXX-1.6.3. Safety and reliability

Safety concept and design philosophy

The FUJI-233Um concept aims to prevent severe accidents and limit their consequences by strongly relying on inherent and passive safety features incorporated in the original design concept.

Provisions for simplicity and robustness of the design

The FUJI-233Um concept incorporates the following features contributing to simplicity and robustness of the reactor installation design [XXX-12]:

(1) The primary and secondary loops operate at a very low pressure (~5 atm), which essentially eliminates accidents such as system rupture due to over-pressurization;

(2) The molten salt coolant is chemically inert and has zero flammability [XXX-12];

(3) There is no possibility for pressure increase in the primary circuit because the boiling point of the fuel salt is very high (about 1800 K) compared with the operating temperature (about 1000 K). In addition to this, the containment has no water inside because FUJI adopts a molten-salt based intermediate heat transport system, which altogether eliminates the accidents with pressure increase in the primary system due to water evaporation or steam ingress;

(4) The fuel salt is critical only where graphite exists in an appropriate fraction. In an accident, the fuel salt exhausted from the core cannot induce a re-criticality accident;

(5) The MSR has a large negative reactivity coefficient on fuel salt temperature that can suppress an abnormal change of the reactor power. The heat capacity of graphite is large and the temperature rise is slow; therefore, it is possible to control it sufficiently, even though the temperature coefficient of the graphite is positive;

(6) Xe, Kr and tritium (T) that are released from molten salt fuel could be effectively trapped in an activated charcoal bed and/or other trapping materials. The container vessels in which the trapping equipment is installed have thick and heat-resistant steel walls and they can be isolated from the off-gas lines by passively operating valves. The production rate of T is estimated at about 6.2×10^{12} Bq/(100 MW day) in normal operation. More than 90% of the T is transferred into the secondary coolant salt and finally, about 98% of the T is transferred to the trapping equipment through an off-gas line [XXX-22]. In this way, the hazard of radioactive gas release from the core under internal and external events and combinations thereof can be decreased;

(7) As the fuel composition can be easily adjusted when necessary, the excess reactivity and the reactivity margin that needs to be compensated by control rods are small. Therefore, the reactivity requirements for control rods are small also;

(8) The delayed neutron fraction of ^{233}U is lower than that of ^{235}U and half of the delayed neutrons are generated outside the core; therefore, the effective delayed neutron fraction in the FUJI-233Um is relatively small. However, safe control of the reactor is possible because of a large negative reactivity coefficient on fuel salt temperature and small overall reactivity margin;

(9) As for the possibility of fire in the reactor-grade graphite, two conditions are essential. One is the sufficient air (oxygen) flow via a chimney effect; another is the external heat source. Even if the primary circuit ruptures, air/oxygen would not enter because nitrogen gas or depleted air (3-5% oxygen) is enclosed in the high temperature containment. Furthermore, there is no possibility of containment breaks by overpressure because pressurization by vapors of the molten salt does not occur. When air enters from outside the containment, fuel salt is transferred to the drain tank and there is no heat source in the core. Conclusively, the possibility of graphite fires in the MSR is essentially suppressed.

Active and passive systems and inherent safety features

The safety functions of a MSR are essentially the same as those for LWRs; they include reactivity control, heat removal from the core, and radioactivity confinement.

For the reactor shutdown, the MSR has more safety systems than an ordinary LWR, as shown in Table XXX-5. For heat removal from the core in accident conditions, the MSR can rely on simplified safety systems and components, as shown in Table XXX-6. For radioactivity confinement, the MSR safety could be superb although the first two barriers are not present, as shown in Table XXX-7.

TABLE XXX-5. REACTOR SHUTDOWN SYSTEMS

Name of a system	Design solution in a MSR	Type of a system / remarks for a MSR
High speed shutdown system (scram system)	Control rods	Active system / small number of rods is sufficient for MSR
Second shutdown system	Fuel salt drain system	Passive system/ no return to criticality in a drain tank
Third shutdown system	Fuel salt density adjustment system	Active system/ also used as a Th make-up system (not used in LWRs)

TABLE XXX-6. SYSTEMS OF HEAT REMOVAL FROM THE CORE IN ACCIDENTS

Name of a system	Design solution in a MSR	Type of a system / remarks for a MSR
Emergency core cooling system; cooling water make-up system	Unnecessary	Unnecessary; drain system could be used as a back-up
Decay heat removal system	Decay heat removal system	Passive system / if drain system is used, the decay heat removal system may be unnecessary

TABLE XXX-7. BARRIERS FOR RADIOACTIVITY CONFINEMENT

Barrier number / name	Design solution in a MSR	Component type/ remarks for a MSR
1. Fuel pellet	None (liquid fuel)	- / Gaseous fission products are continuously removed and trapped
2. Fuel element cladding	None (liquid fuel)	- / Gaseous fission products are continuously removed and trapped
3. Pressure vessel	Core vessel and pipes	Passive/ very low pressure
4. Containment	High temperature containment	Passive / no steam generation; no flammable gas generation
5. Reactor building	Reactor building	Passive/ the same as in a LWR

Design basis accidents and beyond design basis accidents

Design basis accidents (DBAs) for the FUJI MSR are categorized into two types, depending on the initiating events. DBA of the first type is initiated by a single failure of an active component, such as a pump or a control rod, or by a single operator error. DBA of the second type are initiated by failure of a static component such as a pipe, etc. A total of 7 types of DBAs are evaluated for the FUJI MSR, as shown in Table XXX-8.

Most of the DBAs for the FUJI have been analyzed [XXX-23 to XXX-26]; because of the design features of this MSR described earlier in this section, none of these accidents was found to result in any significant consequences.

Severe accidents, i.e. accidents exceeding DBAs, were considered for the first two cases identified in Table XXX-8. They include (i) fuel salt flow decrease accident (an accident with heat removal decrease) and (ii) reactivity insertion accident (an accident with power increase).

In fuel salt flow decrease accident [XXX-23], the primary pumps are locked and control rods do not drop regardless of scram signal. As shown in Fig. XXX-5, the maximum fuel temperature is 900°C, which is below ~1500°C temperature limit for fuel (see Table XXX-1).

TABLE XXX-8. LIST OF DESIGN BASIS ACCIDENTS FOR FUJI MSR

#	DBA NAME	SCENARIO CHARACTERIZATION
1	Fuel salt flow decrease	Stop of all primary circuit pumps, causing decreased heat removal; temperature increase is limited by negative temperature coefficient
2	Reactivity insertion	Reactivity insertion by control rods is small, but start-up of pumps in the cold circuit condition will increase power output; temperature increase is limited by negative temperature coefficient
3	Fuel salt loss by pipe rupture	Corresponds to LOCA in a LWR; lost fuel salt is collected in the emergency drain tank
4	Heat exchanger pipe rupture	As the secondary circuit pressure is slightly higher than that in the primary circuit, boron present in the secondary circuit salt invades the primary circuit; there is no chemical reaction between the two salts
5	Steam generator (SG) pipe rupture	The same as in a PWR plant; high-pressure steam in the SG will be injected into the secondary circuit but would not cause chemical or steam explosion
6	Destructive accident in off-gas system	The same as in a LWR plant; although a MSR has more radioactivity due to gaseous fission products and tritium from the fuel salt, they are trapped in a charcoal, etc.
7	Malfunction of fuel salt adjustment equipment	The equipment is designed to add small or gradual reactivity to the fuel salt

In the design basis reactivity insertion accident (RIA) [XXX-25], the maximum reactivity insertion in the MSR corresponds to the drop of one graphite control rod into the core. Since the worth of a single graphite rod is only 0.06 %δK/K and less than one effective delayed neutron fraction, such initiating event does not result in any prompt criticality of the FUJI.

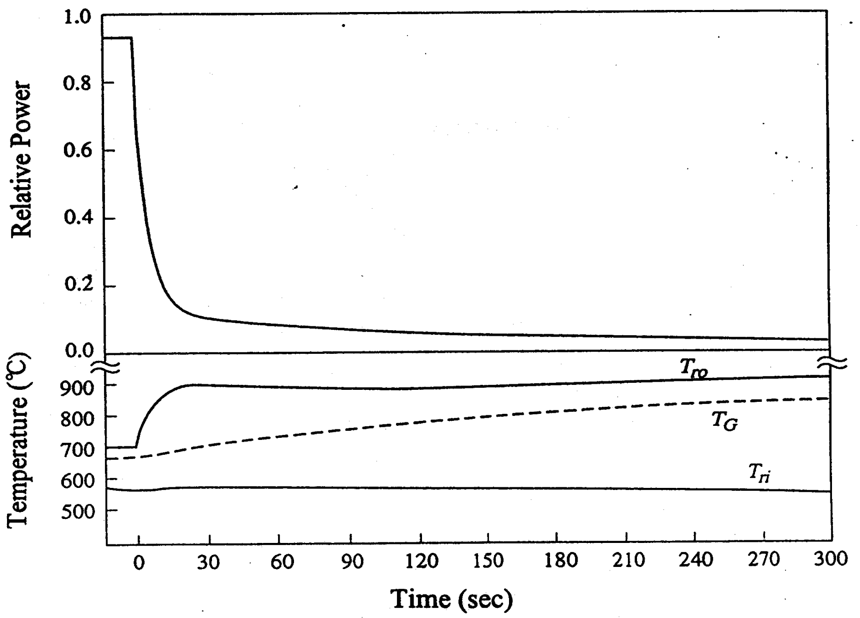
For severe accident analysis, a reactivity insertion of 0.3 %δK/K was assumed, coupled with the failure of the B_4C based control rod scram system. As shown in Fig. XXX-6, the maximum fuel temperature is 1100°C, which is also below ~1500°C temperature limit for fuel (see Table XXX-1).

The results of severe accident analysis performed for the FUJI MSR indicate that the reactor has a sufficient integrity margin in the situations considered.

XXX-1.6.4. Proliferation resistance [XXX-12]

The FUJI MSR design features contributing to an enhanced proliferation resistance are as follows:

- Nuclear fuel of a MSR is in the form of a high temperature liquid contained in the reactor vessel and circulating loop. In the FUJI–series designs, the fuel is a single fluid salt with no distinction between core fuel and blanket fuel. The concentration of fissile material in this single fluid fuel salt is very low (about 2 weight %) in both the FUJI-233U and FUJI-Pu, which reduces the attractiveness of the FUJI fuel for weapon programmes;



T_{ro} – is fuel-salt temperature at reactor outlet; T_{ri} – is fuel-salt temperature at reactor inlet; T_G – is graphite temperature

FIG. XXX-5. Power and temperature change in FUJI under lock of all primary pumps with scram failure.

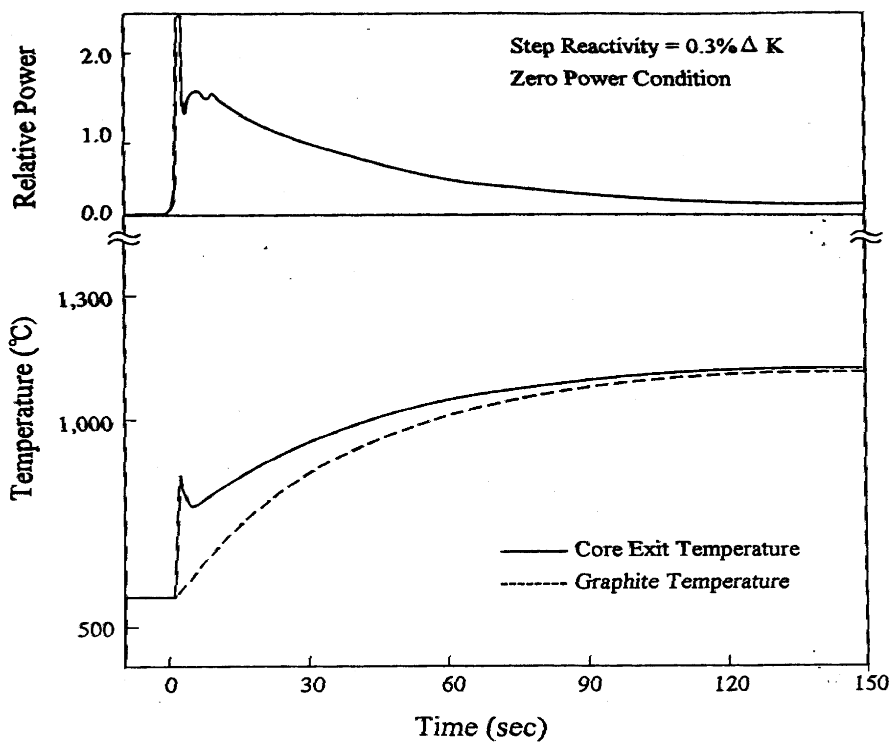


FIG. XXX-6. Power and temperature change in FUJI for reactivity insertion accident with scram failure.

- Excess reactivity in the FUJI MSR is always kept very low, so that any unauthorized extraction of the fuel salt even in small amounts could be easily detected. By virtue of a high conversion ratio, the necessity of addition of fissile material remains small so the plant has no actual fresh or spent fuel storage facilities, which could otherwise be the possible target of diversion or theft;
- When the reactor is out of operation, the fuel salt is drained into a closed tank, which is installed inside the containment-vessel; the design provides for easy cooling of this tank for the salt to get solidified;
- In case of the FUJI-Pu, plutonium isotopes in the fuel salt are soon degraded beyond the weapon quality because there is no production of secondary plutonium due to the use of thorium as the fertile material. In case of the FUJI-233U, ^{233}U in the fuel salt inevitably contains small amount of ^{232}U (about 500 ppm) and its daughter nuclides; this results in a very high radiation dose rate due to high-energy (2.6 MeV) gamma rays from the daughter ^{208}Tl;
- The significant quantity (SQ) of ^{233}U is 8 kg [XXX-27]; this value corresponds to a large volume of the fuel salt (about 250 litres), which conveys a lethal dose (about 1 Sv/hour at 50 cm distance) to any human being who would try to handle it without necessary protection (lead of about 20 cm thickness is necessary to shield this substance for handling, so that only remote technologies might be used for its safe handling);
- If fuel salt is withdrawn from the core and protactinium (Pa) separation is done in a very short time, ^{233}Pa decay to ^{233}U with a half-life of 27 days could be used to produce pure ^{233}U. However, about 50 tons of fuel salt would be necessary to obtain 1 SQ of pure ^{233}U from Pa decay. In addition, it would be very difficult to separate Pa in a very short time because the spent fuel salt has very high radiation. In 2 months after the reactor shutdown, 75% of the ^{233}Pa decays to ^{233}U, and gets mixed with the ^{232}U causing strong gamma radiation;
- If required, the ^{233}U in the fuel salt of the FUJI-233U can be denatured by adding ^{238}U, in very small amounts in virtue of the low concentration of ^{233}U compared with the fertile Th content. This basically maintains the nuclear characteristics, because the production of Pu from ^{238}U would remain small;
- The FUJI-233U does not produce TRU including alternative nuclear materials such as Np, Am and Cm; moreover, the FUJI MSR can incinerate such materials if required. Specifically, the FUJI-233U does not produce any significant amounts of Pu by virtue of the lower atomic number of the fertile fuel isotope (^{232}Th versus ^{238}U). Annual amounts of the fissile material loaded to the primary circuit in the ^{233}U-Th MSR cycle are small compared with the Pu-^{238}U cycle;
- The MSR is a closed single-fluid fuel system with very little excess reactivity and no space in the core for material irradiation such as Pu production from natural uranium;
- For accounting and verification, inspections will be facilitated by low inventory of the annual amount of the fissile material loaded to the core and zero inventory of the spent fuel and by the fact that a single-fluid fuel salt is used in the confinement system. Strong gamma radiation associated with ^{232}U could facilitate monitoring the fuel flow; this radiation would also complicate irregular movement of the fuel from the normal route;

- The FUJI MSR creates the prerequisites for simplified safeguards verification. With a single fuel salt being used in the primary circuit, sample analysis of the fluid fuel salt and an estimation of its volume could become the focus of the physical inventory verification. The total inventory of the fissile material is small and would only change slowly year by year;
- The THORIMS-NES concept provides for a worldwide deployment of the MSR, such as the FUJI, with all fuel cycle operations being centralized within regional fuel cycle centres (see Fig. XXX-4.). In this, the mass of the transport of fissile material between each MSR and regional centre at one time would be small (much less than one SQ) by virtue of the high conversion ratio of the FUJI MSR.

XXX-1.6.5. Technical features and technological approaches used to facilitate physical protection of FUJI MSR

Working medium in the FUJI MSR is a single-fluid fuel salt contained in the closed liquid confinement system and further contained in the high-temperature containment and the reactor building. Fresh fuel for periodical addition to the core is in very small amounts and stored in a closed premise inside the containment, and there is no spent fuel at the plant.

Even under malicious mechanical actions to break the fuel confinement, leaked liquid fuel is accumulated on the catch-pan floor of the containment and drained to the emergency tank. The fuel cannot reach criticality because the fuel salt itself is a sub-critical substance in any shape and quantity; in addition to this, it is easily cooled to a safe solid state.

The FUJI MSR has very low excess reactivity, and even in the case of a malicious action of control rod withdrawal, the reactor would have no prompt criticality accidents with a release of radioactivity to the environment.

These features and small strong building structures also provide an intrinsic protection against external events, such as aircraft crash and missiles.

In a non-operational mode, fuel salt is safely solidified in the drain tank. In the fixed vessel this substance is not transferable and cannot be easily stolen.

Once they enter the FUJI MSR, fissile materials never leave the site throughout the reactor life. This feature of requiring no fuel transportation for recycling is of great benefit for physical protection because, in general, transport between sites could be a vulnerable point in the nuclear fuel cycle.

XXX-1.7. Non–technical factors and arrangements that could facilitate effective development and deployment of FUJI MSR

According to the concept of THORIMS-NES [XXX-12, XXX-28], the innovative nuclear energy system should not only excel in safety performance and economy and non-proliferation but must be adaptable to a steep energy demand growth driven by world population increase and economic development in today's developing countries and also should be effective in saving the planet from global warming and pollution. To meet in full these requirements of the THORIMS-NES, an innovative nuclear energy system must be widely deployed worldwide within a reasonably short period. As it has been shown in section XXX-1.5, a nuclear energy system with the FUJI-233U reactors and regional fuel cycle centres might be a good candidate to meet the requirements of the THORIMS-NES.

Power plant leasing might be useful for those developing countries that would not develop indigenous fuel cycle facilities. Reduced obligations for spent fuel management, as offered to the customer via the FUJI MSR operation with regional fuel cycle centres, as well as long-life reactor operation without on-site refuelling (with small portions of fuel being automatically added to the primary circuit on a periodic basis) could be the attractive features for those countries that would select a NPP leasing option.

XXX-1.8. List of enabling technologies relevant to FUJI MSR and status of their development

The FUJI MSR design takes full advantage of the design and operating experience of the MSRE reactor of ORNL (USA) [XXX-1].

The main enabling technologies of the FUJI MSR are as follows (reference is made to the FUJI project called 'F-plan' XXX-8]):

(1) *Neutronic design [XXX-12, XXX-19]:* the MSR is a thermal reactor and the dominant neutron spectrum follows the Maxwell distribution similar to that of a LWR. The core configuration is very simple; therefore, there are no significant concerns about the reactor physics model. Although it is true that nuclear cross-section measurements for isotopes of the ^{233}U/Th cycle and the corresponding integral experiments are not as abundant as those for the ^{235}U-^{238}U cycle of LWRs, the criticality examination for a MSR is much simpler than that for solid-fuel reactors and actually does not require exact reactivity values to be obtained by critical assembly examination. In a MSR, the final approach to criticality is achieved by a slow addition of fissile salt to the storage tank from which the salt is supplied to the fuel pump bowl to recover salt overflowing from the pump bowl. This approach was verified 40 years ago and currently there are more sophisticated models and nuclear data than were then available.

(2) *Fuel chemistry [XXX-3, XXX-10]:* there are no serious problems except that the examination of detailed PuF_3 solubility data in relation to other fission product ions is necessary. Here, it could be noted that molten salt is an ionic liquid and is not prone to radiation damage, different from a solid fuel.

(3) *Structural materials:* a modified Hastelloy N alloy has been tentatively selected for the FUJI MSR, although the endurance tests should still be performed in a miniFUJI pilot plant; the database on high-temperature performance of the Hastelloy N should be prepared as soon as possible after deciding on its final specifications. The selected Hastelloy is a low-brittle alloy similar to Inconel; its fabrication and welding are expected to pose no major problems.

(4) *Core graphite:* The homogeneous graphite suitable for a MSR can be produced based on past developments. However, irradiation tests should be performed using a powerful reactor for irradiation testing, such as the MS-4 in Dimitrovgrad, Russian Federation. Further development to improve radiation resistance of graphite is necessary because such improvement is important for achieving smaller reactor dimensions and lower electricity costs.

(5) *High temperature containment technology:* this is a new technology not used in the MSRE. However, since the miniFUJI is a compact small reactor, mock-up tests could be performed to validate this technology.

(6) *Turbine-generator plant:* this is a new component, not tested in the MSRE. Hastelloy N is a Nickel-based alloy suitable for steam atmosphere, and the miniFUJI plant could be used to demonstrate the operation of a super-critical turbine system, including load-following capability, the reliability of structural materials for the steam generator, etc.

(7) *Several components and instruments:* the MSR is a high-temperature molten material reactor and in that it is similar to liquid metal cooled fast breeder reactors (LMFBRs).

Therefore, certain efforts in Na technology development for LMFBRs could be useful for safer, more reliable MSR designs. This is especially true for mechanical pumps and steam generators.

(8) *Chemical monitoring:* changes in the chemical behaviour of salts are very slow but further development of the continuous in-situ technologies of chemical control could be recommended.

(9) *Remote maintenance:* The primary fuel salt system has high radioactivity and requires fully remote operation and maintenance. The technologies need to be developed, making use of the recent significant progress in the robotics applicable under high temperatures and radiation conditions.

XXX-1.9. Status of R&D and planned schedule [XXX-5, XXX-12, XXX-29]

Status of R&D

The development of the FUJI-series MSR is thoroughly based on the research and development (R&D) results previously obtained in ORNL (USA) [XXX-1]. Many major reactor engineering problems have been clarified or solved until now. The MSRE successfully operated at ORNL in 1965–1969. The FUJI MSR has basically the same reactor core structure as the MSRE.

At the time of this report, the preliminary design stage for optimization of the FUJI core characteristics such as fissile inventory, fuel conversion ratio (CR), average power density, core lifetime and control was underway. An example of the preliminary results of the design development (for the FUJI-233Um concept) is shown in Table XXX-1.

The fuel salts need no preliminary irradiation tests due to the proven absence of radiation damage.

Two principal solid materials in the reactor are the structural alloy Hastelloy N and graphite.

Hastelloy N alloy

Hastelloy N composed of Ni, Cr, Fe, Mo and other elements (Table XXX-9) serves as the main container material. To reduce high temperature embrittlement of the Hastelloy due to He (a fission product), two modified Hastelloy N alloys were developed in which the contents of Mo, Si and B were reduced and Ti (1.5~2.0%) and Nb (2%) were added. From the thermodynamic analysis, less noble Cr is most reactive among the alloying constituents. A Cr-depleted zone was observed on the surface exposed to the MSRE fuel salt for 22 000 hours at 650°C [XXX-30]; the depth of the degraded zone did not propagate any further than 0.2 mil = 5μm.

Advanced corrosion tests simulating non-isothermal dynamic conditions had been performed in natural and forced convection loops. Figure XXX-7 shows the weight change of a standard and a modified Hastelloy N specimen for over 22 000-hour exposure to the MSRE fuel salt at a maximum temperature of 704°C with a temperature difference of 170°C [XXX-31]. The corrosion of the specimens resulted in a weight loss in the hot leg and a weight gain in the cold leg. The estimated corrosion rate of Hastelloy N was 0.02 mil/year but modified Hastelloy N exhibited better corrosion resistance. These corrosion levels are acceptable for the FUJI MSR design although careful dehydration of salt and graphite is essential. Standard Hastelloy N exposed to fuel salt under irradiation revealed material embrittlement due to inter-granular attack, where grain boundaries were degraded due to the presence of tellurium (Te), a fission product.

TABLE XXX-9. CHEMICAL COMPOSITION OF HASTELLOY N [XXX-3]

ELEMENT	CONTENT (% by weight)	
	Standard alloy	Preferable modified Hastelloy N (Ti-modified) - (Nb-modified)
Nickel	base	base
Molybdenum	15 - 18	11 - 13
Chromium	6 - 8	6 - 8
Iron	5	0.1
Manganese	1	0.15 - 0.25
Silicon	1	0.1
Phosphorous	0.015	0.01
Sulphur	0.020	0.01
Boron	0.01	0.001
Titanium and Hafnium		2 --> 0 (1976)
Niobium		(0 to 2) --> (1 to 2) (1976)
Cobalt		Low enough

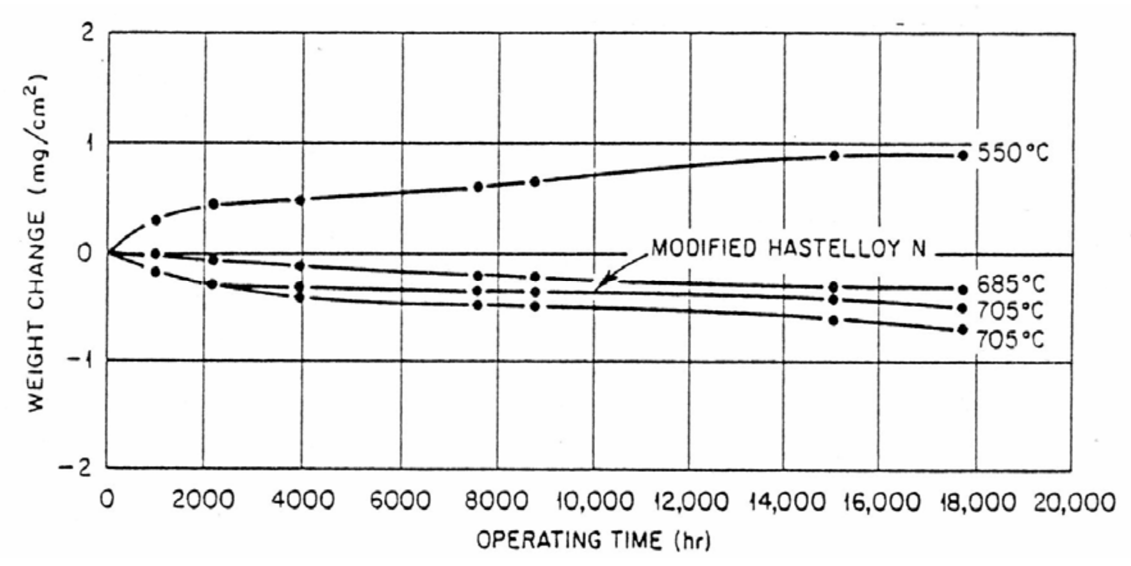


FIG. XXX-7 Weight change versus time of Hastelloy N specimens exposed to fuel salt in thermal-convection loop NLC-19A (ORNL, USA) [XXX-31].

Regarding the latter, the MSRE experience suggests that Te possibly converts to an innocuous telluride (e.g., CrTe) by the reaction:

$$CrF_2 + Te + 2UF_3 \rightarrow 2UF_4 + CrTe,$$

where the reaction equilibrium is controlled by varying the U^{4+}/U^{3+} ratio, that is, the redox potential, by adding either Be (reducing) or NiF_2 (oxidizing).

The MSRE experience also indicates that Te inter-granular attack could be prevented with the control of the redox potential. Figure XXX-8 shows the U^{4+}/U^{3+} ratio, that is, the redox potential, versus the extent of cracking. Little cracking appeared at U^{4+}/U^{3+} ratio ≤ 60; cracking was extensive at the ratio >80.

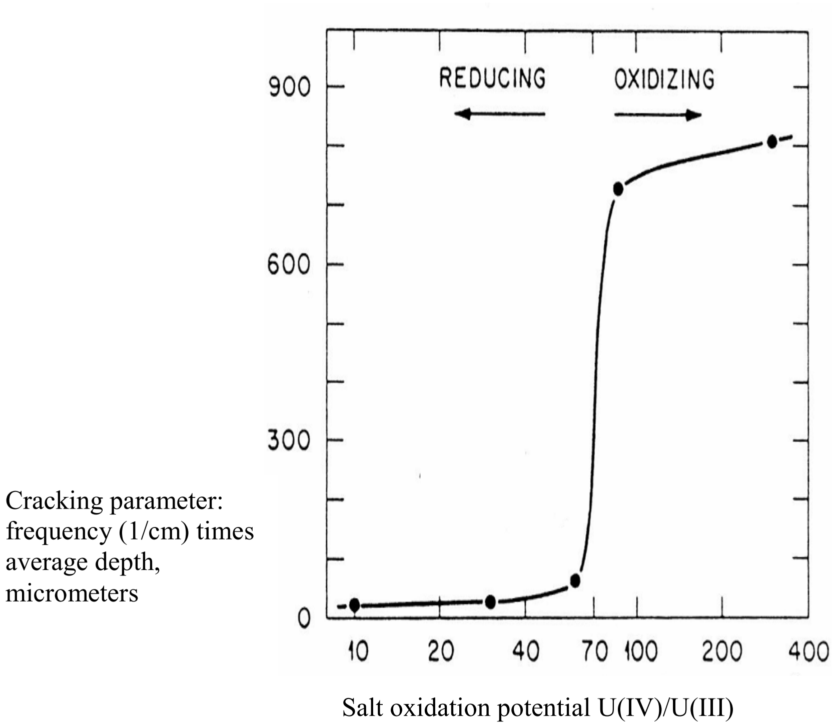


FIG. XXX-8. Cracking behaviour of Hastelloy N exposed to MSBR fuel salt containing $CrTe_{1.266}$ during 260 hours at 700°C [XXX-32].

To improve Hastelloy N performance with respect to tellurium (Te):

(a) Hastelloy N was modified by the addition of 1 to 2% of Nb, significantly reducing the susceptibility to Te inter-granular attack (Fig. XXX-9) [XXX-32];

(b) The redox potential control was found essential [XXX-32]. The potential should be kept within the region of a stable Te compound ($U^{4+}/U^{3+} < 60$) and beyond the region of U-carbide deposition on graphite ($U^{4+}/U^{3+} > 6$).

The problem of the Te attack could, therefore, be solved by applying both the measures (a) and (b). Alternatively, the Russian Federation has developed a candidate material for a MSR. Under similar test conditions, the Russian alloy similar to standard Hastelloy N showed the maximum corrosion rate ≈ 6μm/year [XXX-33] and no traces of the Te attack.

Graphite

In the FUJI MSR, graphite is used as a moderator and reflector material and is directly immersed in the fuel-salt. The basic requirements for graphite were defined through the research on the MSBR, ORNL [XXX-1]. Graphite should be stable against neutron irradiation, be impenetrable by the fuel salt and should not absorb Xe and Kr. Extensive irradiation studies have been performed and numerous data accumulated.

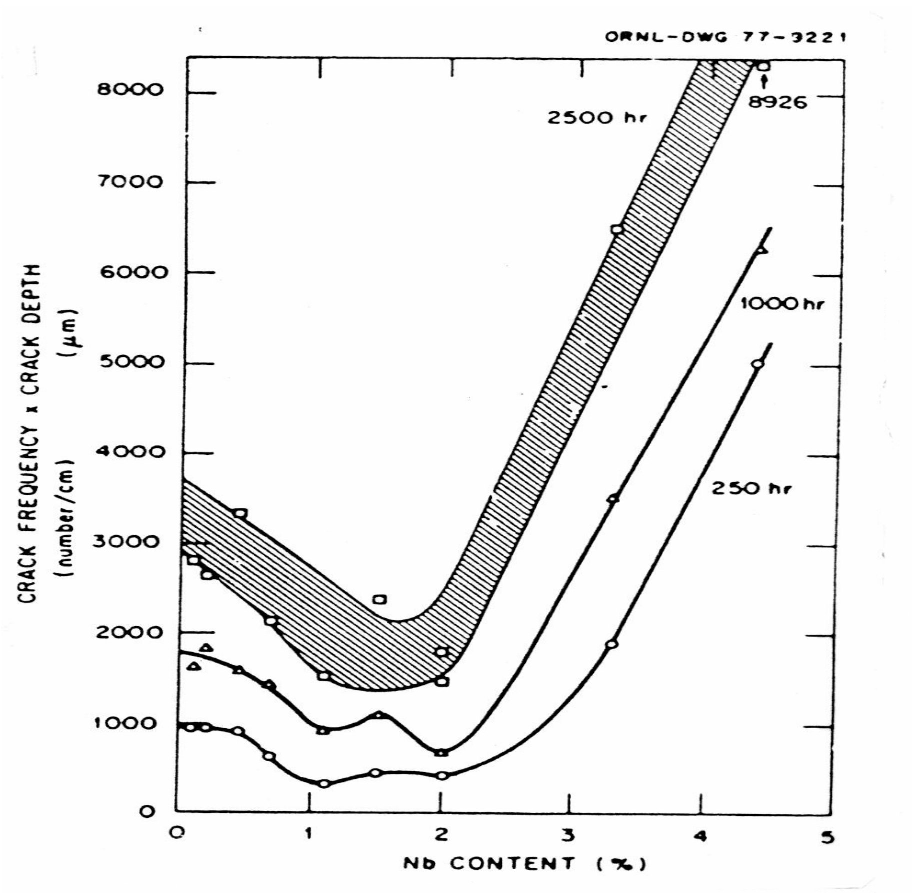


FIG. XXX-9. Variations of a severity of Hastelloy N cracking with Nb content; samples were exposed for indicated times to a salt containing Cr_3Te_4 and Cr_5Te_6 at 700°C [XXX-34].

With irradiation, point defects are formed and agglomerated one with another in each crystallite, causing its remarkable growth in the c-axis direction and a little shrink in the other two directions [XXX-35]; such changes cause the material distortion. The lifetime of the material is determined by failure criterion and decreased by the radiation-induced degradation of thermal conductivity.

Figure XXX-10 shows volume changes of the monolithic graphites (selected as best within the MSBR programme) under irradiation by fast neutrons (>50 keV) at 715°C [XXX-1]. Their lifetimes correspond to the fast neutron fluence between 2 and 3 × 10^{22} n/cm^2, the value at which they revert to their original size after shrinkage [XXX-1]. Similar results were obtained in other investigations, e.g. those performed by the EDF of France and in the former USSR.

Although the core graphite of MSBR was designed assuming replacement every 4 years, graphite of the FUJI will not be replaced over the full reactor lifetime. The effective sealing of graphite against fuel salt penetration is resolved by choosing a pole-diameter of less than 1 μm, in consideration of the surface tension. It can be stated that graphite presents no serious problems for the MSR, although large size homogeneous graphite is not easy to fabricate.

If the irradiation limit for graphite is further increased, performance of the FUJI MSR would be significantly improved, resulting in lower electricity generation costs. The Toyo Tanso Corporation in Japan holds the top share of isotropic graphite in the world and manufactures nuclear-reactor grade isotropic graphite IG-110, supplied as the reactor-core graphite for the high temperature gas cooled reactor HTTR at the Japan Atomic Energy Agency (JAEA) and the HTR-10 reactor at the Tsinghua University of China.

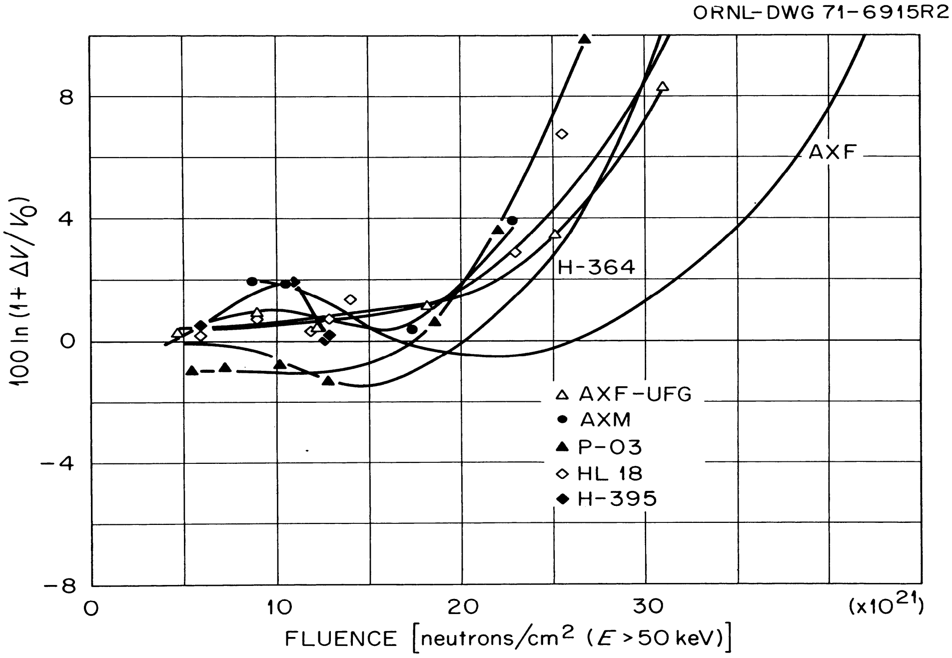


FIG. XXX-10. Volume changes for monolithic graphites irradiated at 715°C (ORNL, USA).

Tests with high quality graphite samples, including graphite irradiation with energetic particles such as carbon ions or high-energy electrons, are planned to be performed to understand the mechanisms of damage more precisely and to develop better materials for long-life cores of the FUJI-series reactors.

Reuse or recycle of materials

After the end of the FUJI MSR lifetime, all reactor components would be sent back to the regional centres for reuse or disposal. Low-cobalt Hastelloy N could mostly be recycled. The irradiated structures would be cooled for one year; then, the contaminated surfaces would be grinded off and, finally, the structures would be re-melted in a vacuum. The processed alloy would then be used to manufacture new structures and components.

The graphite irradiated at low levels would be reused as a reflector material, after grinding the contaminated surfaces to a 0.1 mm depth.

Monitoring

Development of the monitoring techniques is necessary to ensure sound and efficient operation of the FUJI-MSR. Fortunately, the reactor system does not require continuous monitoring of the major fuel constituents such as Li, Be, Th, F and U [XXX-36]. Therefore, electrochemical on-line monitoring of the redox potential has been developed; it is only the U^{4+}/U^{3+} ratio, which responds to the corrosive atmosphere and to the distribution of fission products and tritium in the reactor system. On-line monitoring of the U^{4+}/U^{3+} ratio in the MSRE has proven the results previously obtained in thermodynamic and spectroscopic analyses shown that observations agreed with those from thermodynamic and spectroscopic analyses, in the presence of a Ni/NiF_2 reference electrode [XXX-37]. The U^{4+}/U^{3+} ratio can easily be kept within the suitable region by varying the time during which the beryllium (Be) is dissolved in the melt.

Further R&D

The major further R&D necessary for the FUJI MSR are mainly related to the structural materials and components; they include:

- Study of the additional modifications of the Hastelloy N alloy, obtained by adding rare earth elements, increasing Cr and reducing Co contents;
- Additional analysis and testing of components with low tensile strength, such as the tubing elbow;
- Preparation of the modified Hastelloy N data for ASTM standard and ASME coding (tensile test data; ductility data; creep test data; toughness data; for both base and welded metal).

Planned schedule

The basic programme for developing the THORIMS-NES is structured in three plans:

- *F-plan:* Fission reactor development including the miniFUJI and FUJI in several versions;
- *D-plan:* Dry reprocessing of spent fuel and target/blanket salts including not only molten salt but solid fuel of ordinary reactors such as the LWR, fast breeder reactors (FBR), heavy water reactors (HWR), etc. for producing molten fluoride fuel salt of the FUJI or target/blanket salt of the AMSB;
- *A-plan:* Accelerator molten-salt breeder (AMSB) development in several versions.

A skeleton of the THORIMS-NES development plan is shown in Fig. XXX-11. As shown in part I of this figure, once the project funding is available, several test loops, components and instruments would be prepared and operated for the education and training of the project staff. Decisions on material specifications are important and high temperature tests including irradiation tests would begin.

As shown in part II of the figure, the design of the miniFUJI could be finalized in about 4 – 5 years conservatively. The construction of the miniFUJI could be completed in another ~3 years. After charging the reactor installation with salts and several preliminary tests, the miniFUJI could become critical on the 8th – 9th year of the project.

After obtaining experience to reconfirm and modify previous MSR data from miniFUJI operation, detailed design and certain R&D for the FUJI MSR would be performed, to be completed on the 9th year of the project.. Several kinds of innovative designs would be considered in parallel at this stage, although somewhat conservative designs such as the FUJI-233Um (described in Table XXX-1) could be recommended for the first prototype power station. Additional design optimization would be necessary to meet the requirements of flexibility in reactor operation, especially as comes to the use of different types of fuel within the same core configuration. The reactor could then go critical on the 12th – 15th year of the project. This is the medium-term programme shown in part III of Fig. XXX-11.

After that, efforts toward actual commercialization might be gradually started. Based on the experience obtained from the preceding projects, not only medium or large-sized FUJI but in parallel, the AMSB could be developed. Several preliminary experiments could be completed using the 1 GeV accelerator with several mA proton beams. In addition to the abovementioned, the following studies could progress in parallel:

- D-plan development - at first, the treatment of spent solid fuel could be accomplished by simplifying the FREGATE process [XXX-14, XXX-15];
- Study on the social acceptance of the THORIMS-NES concept in the world - depending on the results of this assessment, several modifications or several versions of the system fitting the specific demands of each region could be completed, including the FUJI and the AMSB themselves. The DT-fusion application might also be realized;
- Design and technology development and infrastructure-building activities for several regional fuel cycle centres could proceed step-wise in parallel with the abovementioned activities.

XXX-1.10. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed

In the 1960s, ORNL (USA) succeeded significantly in construction and operation of the experimental molten salt reactor, the MSRE. An approval for the next prototype molten salt reactor, the Molten-Salt Test Reactor (MSTR, 250 MW(e)), was pursued in the 1970s but never achieved [XXX-3]. The reason was mostly political, related to a 'moratorium' on fuel breeding imposed at that time in the USA [XXX-19].

The basic MSR technology appears very promising for the utilization of Th resources, although the development of a continuous chemical processing, such as proposed for the MSBR, is rated as requiring long-term R&D efforts. The FUJI-series molten salt reactors do not need a system of continuous chemical processing of fuel-salt; therefore, the technology and operating experience of the MSRE reactor of 1960s can be applied to the FUJI.

The miniFUJI – a prototype of the FUJI MSR – will be needed to perform validation of the integral MSR technology including an electricity generation system (that was not present in the MSRE). It could also contribute to education and training of the MSR engineers and plant personnel. After the operation of the miniFUJI, it is projected that a rapid commercialization of the FUJI-series molten salt reactors might be started.

XXX-1.11. List of other similar or relevant SMRs for which the design activities are ongoing

The FUJI-233Um is a simplified basic MSR variant that could be effectively modified in several directions in the future, after the basis for the MSR technology is established. Many possible versions of the FUJI have already been examined, including the FUJI for excess plutonium incineration (FUJI-Pu). An underground version of the FUJI, the FUJI-UG, is being considered [XXX-38]. A combined system of the FUJI and the Free Piston Stirling Engine, the FUJI-STR, is being examined [XXX-34]. Higher temperature versions of the FUJI could also be considered for future advanced process heat applications, such as hydrogen production.

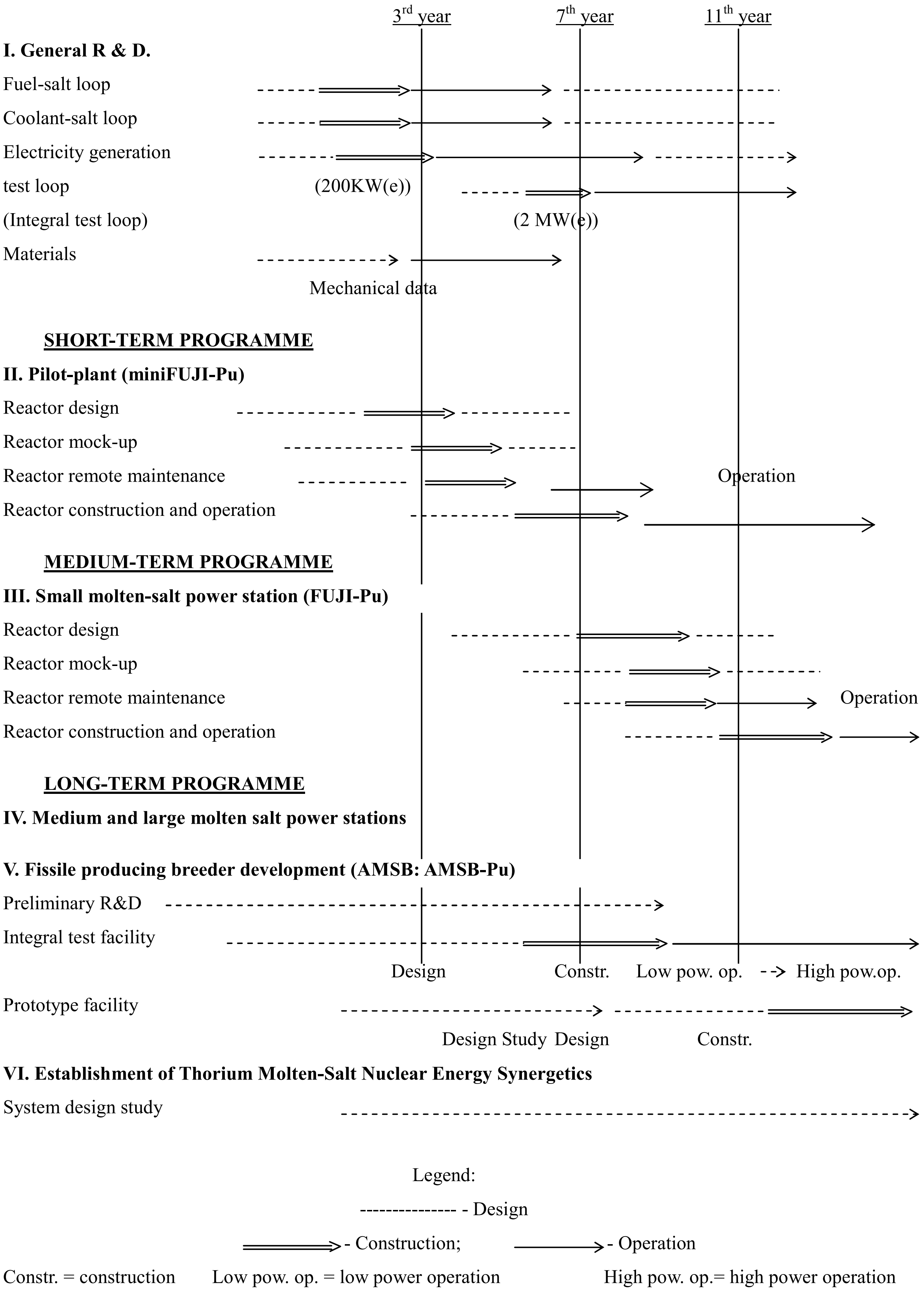


FIG. XXX-11. Developmental schedule for THORIMS-NES.

XXX-2. Design description and data for FUJI MSR

XXX-2.1. Description of the nuclear systems

Reactor core and fuel design

The FUJI-233Um is substantially designed for the Th-^{233}U fuel cycle; however, in the first 20–30 years of operation it could rely upon fuel loads based on ^{235}U or ^{239}Pu mixed with Th [XXX-39].

A vertical cut of the FUJI reactor vessel is given in Fig. XXX-12. Shown are the reactor vessel, the internal graphite moderator, and the control rod. A pilot plant, named the miniFUJI, is also shown in Fig. XXX-12, for comparison.

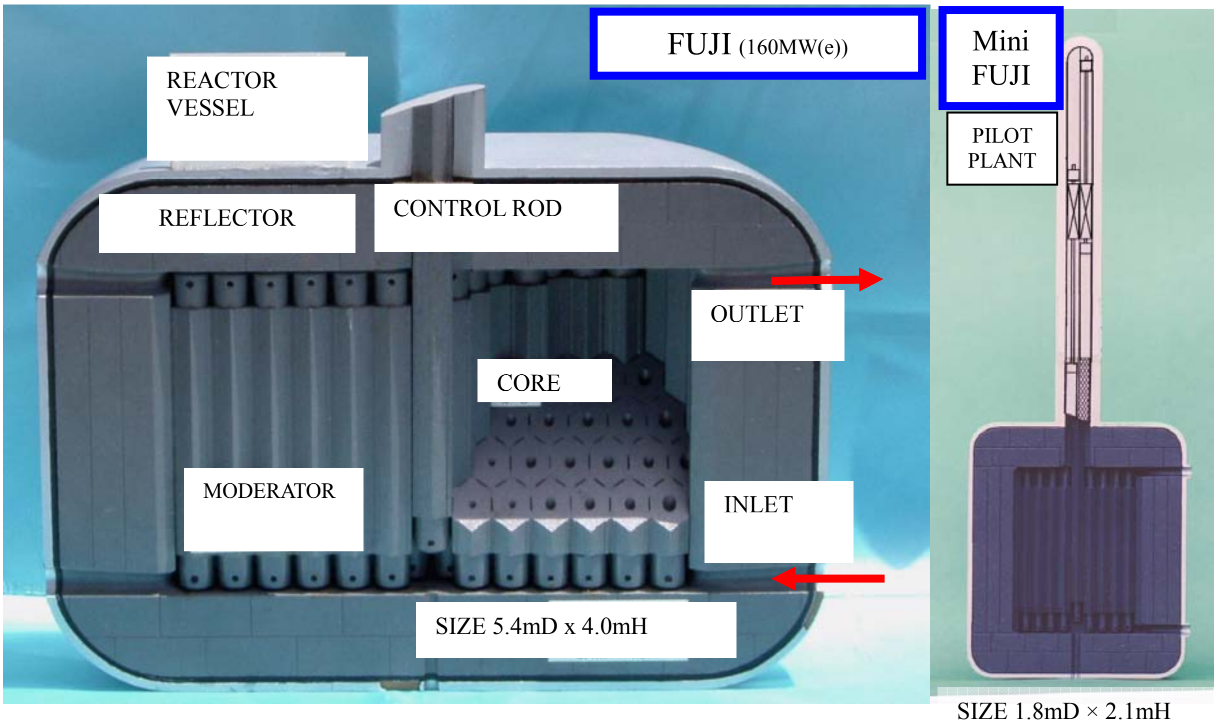


FIG. XXX-12. Vertical cut of the FUJI-series reactor vessel:FUJI MSR core (left) and miniFUJI core (right).

The core is constituted of directly immersed hexagonal graphite rods with a central hole and a thin ditch on each flat side for the fuel salt path, Fig. XXX-13. The volume fraction of fuel salt is different in each radial zone of the core; for example, there are three such zones in the core of the FUJI-233Um. The main solid material inside the reactor vessel is graphite; it occupies about 90 % of the total in-vessel volume.

Figure XXX-14 gives a vertical cross-section of the primary fuel salt system. The primary fuel-salt system is located in the airtight high temperature containment, operating at about 770 K. As it is shown in Figure XXX-14, the FUJI-233U incorporates diverse and redundant safety systems.

A main drain tank located below the reactor room has a natural convection cooling system. Also for decay heat removal, a reliable system such as the heat pipe could be located in the drain tank.

FIG. XXX-13. Core graphite rod (fuel salt flows through the hole in its centre).

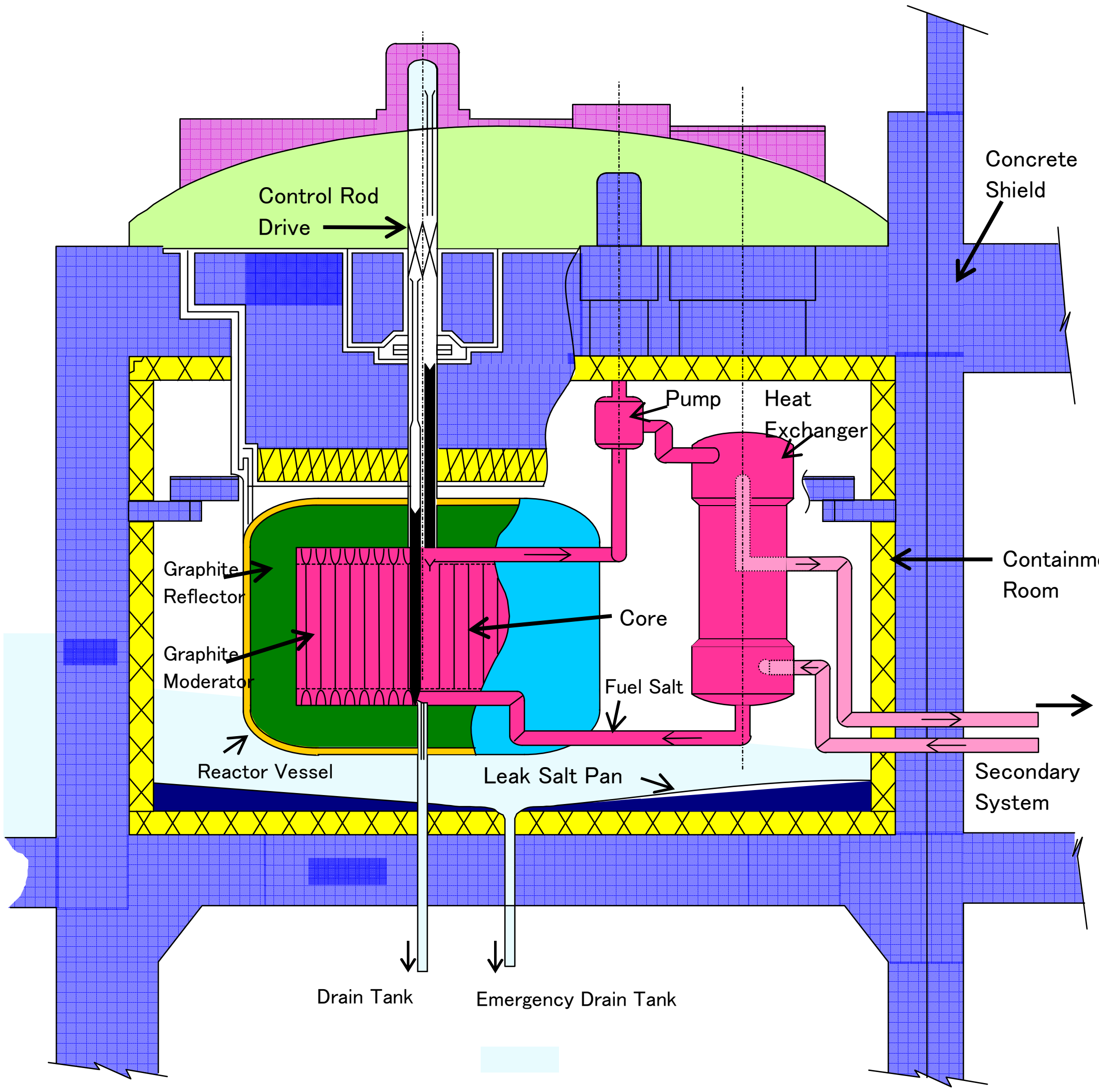


FIG. XXX-14. Vertical view of the FUJI primary system.

The emergency drain tank acts as a back-up to the main drain tank; it is located in the borated water pool, see Fig. XXX-17.

A large catch-pan is set up on the bottom of the high temperature containment and in the event of a fuel salt spill, the salt is guided to the emergency drain tank and cooled down to a freezing point by the outside water.

Main heat transport system

A scheme of the FUJI MSR main heat transport system with specification of heat removal path in normal operation and in accidents is shown in Fig. XXX-15.

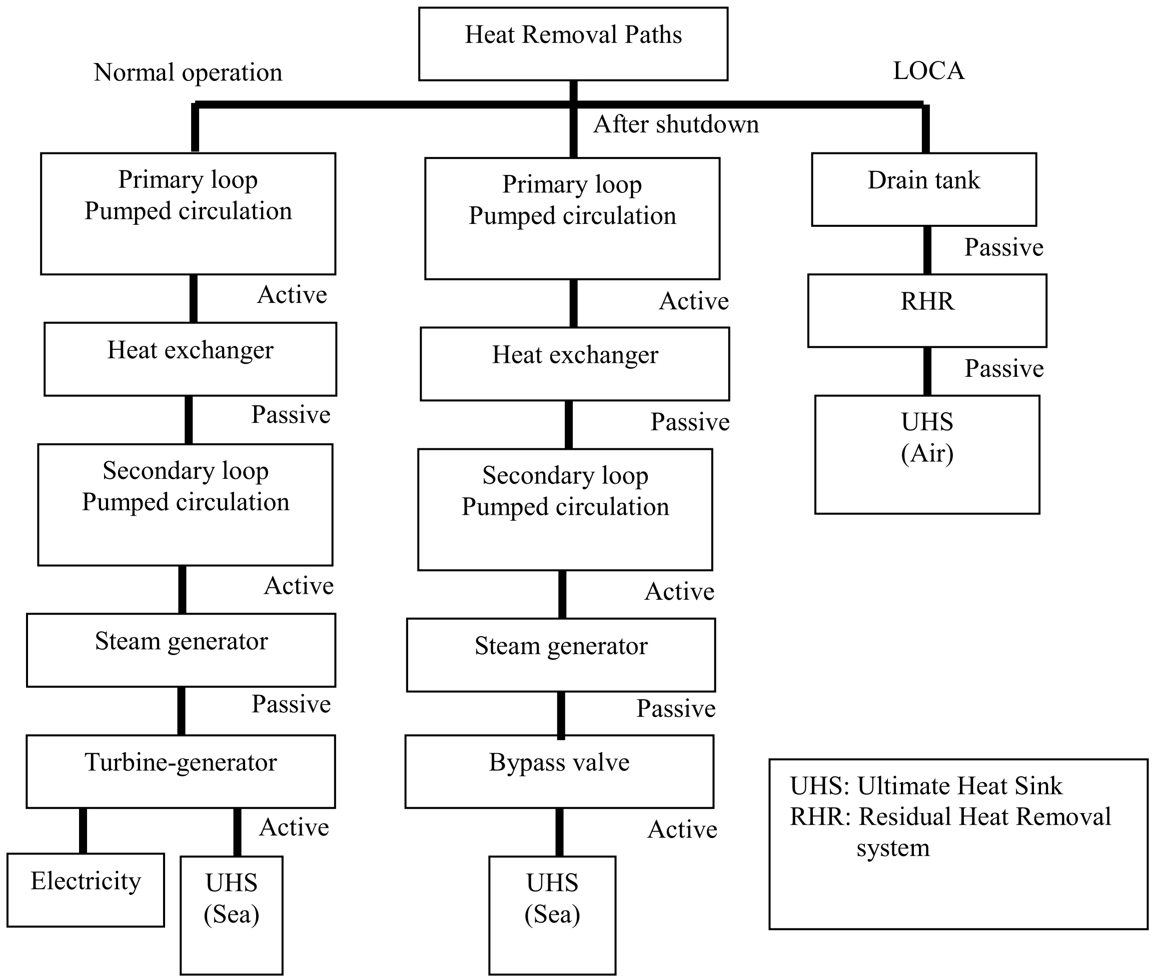


Fig. XXX-15. Main heat transport system of FUJI.

XXX-2.2. Description of the turbine generator plant and systems

Commercially available equipment could be used for a supercritical turbine generator plant of the FUJI. As it was already mentioned, a combined system of the FUJI and the Free Piston Stirling Engine, the FUJI-STR, is being examined [XXX-34].

XXX-2.3. Systems for non-electric applications

As it was already mentioned, the FUJI-series reactors could be used within cogeneration plants producing electricity and hydrogen and/or potable water. No further details were provided.

XXX-2.4. Plant layout

Land based FUJI-233U of 150 MW(e)

The overall configuration of a 150 MW(e) power plant with the FUJI-233U nuclear installation is shown in Fig. XXX-16.

The main components of the primary fuel salt system (marked red in the figure) are installed in a high temperature gas-tight containment.

Since the FUJI design eliminates core graphite replacement during the reactor lifetime, the reactor vessel has no large flanges for the exchange of graphite moderator blocks. The drain tank is located below the reactor vessel. The trap systems for Xe and Kr gases escaping from the fuel salt and for tritium (T) generated mainly in the secondary coolant salt are not shown in this figure.

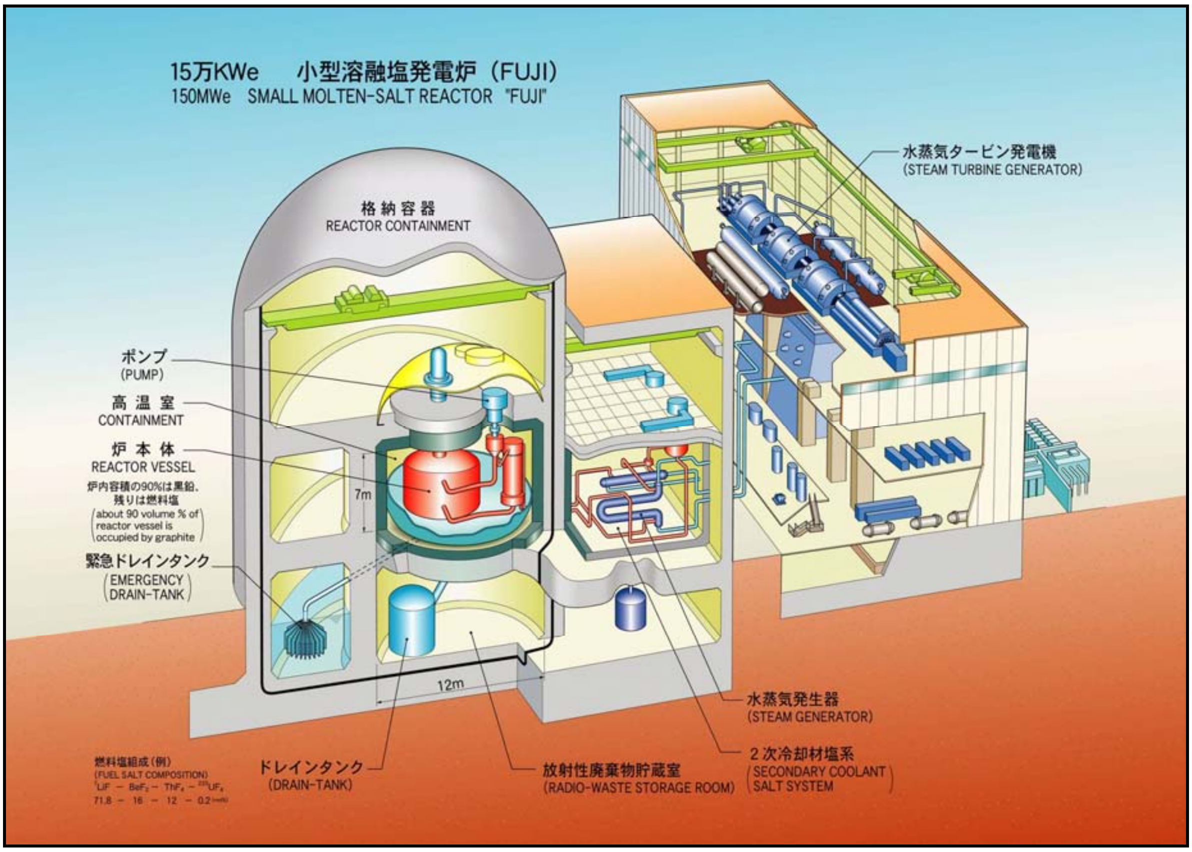


FIG. XXX-16. Plant layout of a 150 MW(e) land based FUJI.

Underground power plants with FUJI

Nuclear islands of the FUJI power plants could also be located underground, as shown in Fig. XXX-17. Such location may provide an additional degree of protection against certain external events, including those of human-induced malevolent origin [XXX-38]. The location depth should not exceed ~10 m in order to keep the plant economic characteristics competitive.

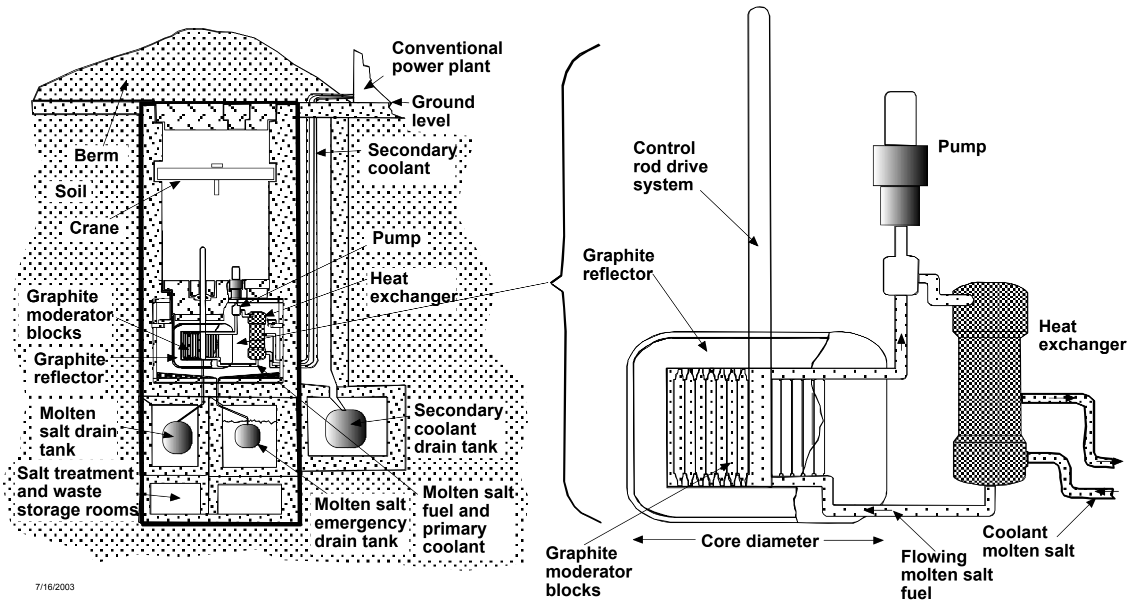


FIG. XXX-17. Underground location of a FUJI nuclear island [XXX-38]; the non-nuclear parts of the balance of plant, located upon the ground, are not shown in this figure.

REFERENCES

[XXX-1] ROTHENTAL, M.W., HAUBENREICH, P.N., BRIGGS, R.B., The Development Status of Molten-salt Breeder Reactors, ORNL-4812, Oak Ridge, USA (1972).

[XXX-2] ROBERTSON, R. C., Editor, Conceptual Design Study of a Single Fluid Molten Salt Breeder Reactor, ORNL-4541, Oak Ridge, USA (1971).

[XXX-3] McNEESE, L.E., et al., Program Plan for Development of Molten-salt Breeder Reactors, ORNL-5018 , Oak Ridge, USA (1974).

[XXX-4] FURUKAWA, K., KATO, Y., OHMICHI, T., OHNO, H., The combined system of accelerator molten-salt breeder (AMSB) and molten-salt converter reactor (MSCR), Japan-US Seminar on Thorium Fuel Reactors (October 1982, Nara, Japan), Thorium Fuel Reactors, Atomic Energy Society of Japan (1985) p. 271–281**.** [Russian Translation: “Atomnaja Technika za Rubezhom”, 1983 [6], p. 23–29] (1983).

[XXX-5] FURUKAWA, K., LECOCQ, A., KATO, Y., MITACHI, K., Summary report: Thorium molten-salt nuclear energy synergetics, Journal of Nuclear Science and Technology, 27, p. 1157–1178 (1990) [ref to: MARCHETTI, C., NAKICENOVIC, N., RR-79-13, IIASA(1987); MARCHETTI, C., 7th World Hydrogen Conf., Moscow, Sept. 1988; Perspectives in Energy, **2**, 19 (1992)].

[XXX-6] FURUKAWA, K., MINAMI, K., MITACHI, K., KATO, Y., High-safety and economic small molten-salt fission power stations and their development program - Th molten-salt nuclear energy synergetics (THORIMS-NES), Alternative Energy Sources VIII (Paper presented at Int. Conf., Miami, USA, 1987) Hemisphere Publications, Vol.2, pp.2–22 (1989).

[XXX-7] SIMONENKO, V.A., FURUKAWA, K., GREBYONKIN, K.F., MITACHI, K., Concept of the molten-salt reactor technology implementation. ICENES’96

(Proc. of the Int. Conf., June 1996, IPPE, Obninsk, Russia, 1997) p.175–179.
[XXX-8] FURUKAWA, K., MITACHI, K., KATO, Y., Small molten-salt reactor with rational thorium fuel cycle, Nuclear Engineering & Design, **136**, p. 157–165 (1992).
[XXX-9] FURUKAWA, K., et al., Design study of small molten-salt fission power station suitable for coupling with accelerator driven molten-salt breeders, 4th ICENES (Proc. of Int. Conf.), "Emerging Nuclear Energy Systems" World Science, p. 235–239 (1987).
[XXX-10] FURUKAWA, K., NAKAMURA, N., MITACHI, K., KATO, Y., KOMATSU, K., Simplified safe small molten-salt reactor "FUJI"for a global measure of greenhouse effects, Energy and Environmental Progress-I, Editor Veziroglu, T.N. (Proc. of 9th ICEE, Miami, USA, 1989), Vol. C, p. 341–355, Nova Science Publications (1991).
[XXX-11] INTERNATIONAL ENERGY AGENCY, OECD NUCLEAR ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Three Agency Study: Innovative Nuclear Reactor Development: Opportunities for International Co-operation, IEA Publications, 9 rue de la Federation, 75739 PARIS Cedex 15, Printed in France by Louis Jean (2002).
[XXX-12] FURUKAWA, K., et al., Thorium cycle implementation through Pu－incineration by thorium molten-salt nuclear energy synergetics, IAEA-TECDOC-1319, Thorium Fuel Utilization: Options and Trends, Vienna, p.123–137 (November 2002).
[XXX-13] FURUKAWA, K., URI GAT, ENGEL, J. R., Supporting the thorium reactor, Nuclear News / March 1994, p.21 (1994).
[XXX-14] BACHACEK, V., PEKA, I., VESELY, V., Uranium branch for FBR for reprocessing by fluoride volatility methods, presented at Trans. ANS (1982), later issued as the following paper; NOVY, P., PEKA, I., CHOCHLOVSKY, I., Technological regulations for the uranium branch of the fluorine reprocessing method for fast reactor spent fuel, report OJV-9081 CH, T. Nuclear Research Institute, Rez, Czech (1989).
[XXX-15] LECOCQ, A., FURUKAWA, K., Chemistry of dry process in the THORIMS-NES concept, International Seminar on Molten-Salts in Nuclear Technology (Paper presented at int. seminar, Dimitrovgrad, Russia, June 19–23, 1995).
[XXX-16] FURUKAWA, K., et al., New primary energy source by thorium molten-salt reactor technology, Electrochemstry, **73,** p. 552–563 (2005).
[XXX-17] FURUKAWA, K., LECOCQ, A., KATO, Y., MITACHI, K., Radioactive waste management in global application of Th molten-salt nuclear energy synergetics with accelerator driven breeders, Specialist meeting "Accelerator-Driven Transmutation Technologies for Radwaste and Other Applications" (24–28 June, 1991, Saltsjobaden, Stockholm, Sweden), LA-12205-C Conf. SKN Rep.No.54; UC-940, p. 686–697 (1991).
[XXX-18] BAES, C.F., The chemistry and thermodynamics of molten-salt reactor fuels, Journal of Nuclear Materials, 51, p. 149–162 (1974).
[XXX-19] ENGEL, J.R. et al., Conceptual design characteristics of denatured molten-salt reactor with once-through fueling, ORNL/ TM-7207 (1980), ORNL, USA.
[XXX-20] MOIR, R.W., Cost of electricity from molten-salt reactors, Nuclear Technology, 138, p. 93–95 (April 2002).
[XXX-21] INTERNATIONAL ATOMIC ENERGY AGENCY, Thorium Based Fuel Options for the Generation of Electricity: Developments in the 1990s, IAEA-TECDOC-1155, Vienna (May 2000).

[XXX-22] MAYS, G.T., et al., ORNL/TM-5759, ORNL, USA, p. 186 (1977).
[XXX-23] SHIMAZU, Y., Locked rotor accident analysis in MSBR, Journal of Nuclear Science and Technology, 15, p. 935 (1978).
[XXX-24] Molten Salt Breeder Reactor, Special Research Committee Report, the Atomic Energy Society of Japan, (1981/4) (in Japanese).
[XXX-25] SHIMAZU, Y., Nuclear safety analysis of MSBR, Journal of Nuclear Science and Technology, 15, p. 514 (1978).
[XXX-26] SHIMAZXU, Y., Safety aspects on depressurization accident of a molten-salt reactor, ICAPP'03 (Proc. Int. Conf., Cordoba, Spain, 2003).
[XXX-27] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safeguards Glossary 2001 Edition, International Nuclear Verification Series No. 3, IAEA (2001).
[XXX-28] FURUKAWA, K., KATO, Y., URBAN, C., Thorium molten salt nuclear energy synergetics for the global energy demand, INAC-2002 (Proc. of Int. Conf. ,Brazil, 2002) XIII ENFIR, R-14 (2002).
[XXX-29] FURUKAWA, K., MITACHI, K., LECOCQ, A., FURUHASHI, A., CHIGRINOV, S.E., Th Fuel cycle deployment through Pu incineration by THORIMS-NES, IAEA-TECDOC-916, Advanced Fuels Reducing Actinide Generation, (Proc. of Technical Committee Meeting, Vienna, 21–23 Nov.1995), p. 115–128 (1996).
[XXX-30] McCOY, Jr., H.E., An Evaluation of the Molten Salt Reactor Experiment Hastelloy-N Surveillance Specimens- First Group, ORNL-TM-1997, ORNL, USA (November 1967).
[XXX-31] KOGER, J.W., ORNL-4832, ORNL, USA. p. 124 (1972).
[XXX-32] KEISER, J.R., Status of Tellurium-Hastelloy N Studies in Molten Fluoride Salts, ORNL/TM-6002, ORNL, USA (1977).
[XXX-33] IGNATYEV, V., NOVIKOV, V.M., SURENKOV, A.I., FEDULOV, V.I., Preprint IAE-5678/11, Russian Research Centre "Kurchatov Institute", Moscow, Russia (1993).
[XXX-34] ERBAY, L.B., Examining the power generation by the Stirling heat engine combined with the molten-salt reactor, ECOS'99 (Proc. of Int. Conf., June 8–10, 1999, Tokyo) p. 429–434.
[XXX-35] REYNOLDS, W.N. Radiation Damage in Graphite, Chemistry and Physics of Carbon, 2, p. 121 (1966).
[XXX-36] MSR Program Semiannual Progress Report, August 1972, ORNL-4812, ORNL, USA, p. 146 (1972).
[XXX-37] MSR Program Semiannual Progress Report, February 28, 1969, ORNL-4396, ORNL, USA, p.200 (1969).
[XXX-38] MOIR, R., TELLER, E., Thorium fueled underground power plant based on molten salt technology, Nuclear Technologies **151**, p. 334–340 (Sept.2005).
[XXX-39] MITACHI, K., FURUKAWA, K., MURAYAMA, M., SUZUKI, T., Nuclear characteristics of a small molten-salt power reactor fueled with Pu, Emerging Nuclear Energy Systems ICENES'93, Editor H. Yasuda, p. 326–331, World Science Publications (1994).

CONTRIBUTORS TO DRAFTING AND REVIEW

Adamovich, L.	N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET), Russian Federation
Banerjee, S.	Reactor Engineering Division, Bhabha Atomic Research Centre (BARC), Trombay, Mumbai, India
Bolshukhin, M.A.	Experimental Design Bureau of Machine Building (OKBM), Russian Federation
Budylov, E.	State Scientific Centre Institute of Physics & Power Engineering (IPPE), Russian Federation
Chaki, M.	Hitachi, Ltd., Power & Industrial Systems R&D Laboratory, Japan
Dulera, I.V.	Reactor Engineering Division, Bhabha Atomic Research Centre (BARC), Trombay, Mumbai, India
Fomichenko, P.A.	Russian Research Centre “Kurchatov Institute”, Russian Federation
Furukawa, K.	International Thorium Molten-Salt Institute, Japan
Gabaraev B.	N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET), Russian Federation
Greenspan, E.	Department of Nuclear Engineering, University of California, Berkeley, USA
Grishanin, E.I.	All-Russian Institute of Atomic Machinery (VNIIAM), Russian Federation
Handa, N.	Toshiba Corporation, Japan
Hibi, K.	Mitsubishi Heavy Industries, Ltd., Japan
Ishida, T.	Kobe University, Japan
Ivanov, E.	Russian Research Centre “Kurchatov Institute”, Russian Federation
Kambe, M.	Nuclear Energy Systems Dept., Komae Research Laboratory, Central Research Institute of Electric Power Industry (CRIEPI), Japan

Kaplar, E.	Russian Research Centre “Kurchatov Institute”, Russian Federation
Kato, Y.	International Thorium Molten-Salt Institute, Japan
Konomura, M.	FBR System Engineering Group, System Engineering Technology Division, Oarai Engineering Center, Japan Atomic Energy Agency (JAEA), Japan
Kurachenkov, A.V.	Experimental Design Bureau of Machine Building (OKBM), Russian Federation
Kuznetsov, V.	International Atomic Energy Agency, Vienna, Austria
Lepekhin, A.N.	Experimental Design Bureau of Machine Building (OKBM), Russian Federation
Minato, A.	Nuclear Energy Strategy Office, Central Research Institute of Electric Power Industry (CRIEPI), Japan
Modro, S.M.	Idaho National laboratory (INL) USA
Moorthy, R.I.K.	Reactor Engineering Division, Bhabha Atomic Research Centre (BARC), Trombay, Mumbai, India
Moriwaki, M.	International Atomic Energy Agency, Vienna, Austria
Reyes Jr., J. N.	Oregon State University (OSU), USA
Rogov, V.A.	Experimental Design Bureau of Machine Building (OKBM), Russian Federation
Sakashita, Y.	Advanced Systems Design & Engineering Dept., Toshiba Corporation, Japan
Samoilov O.B.	Experimental Design Bureau of Machine Building (OKBM), Russian Federation
Sefidvash, F.	Federal University of Rio Grande Do Sul (UFRGS), Brazil
Sekimoto, H.	Research Laboratory for Nuclear Reactors, Tokyo Institute of Technology, Japan
Shimazu, Y.	Division of Quantum Energy Engineering, Graduate School of Engineering, Hokkaido University, Japan

Sienicki, J.	Lead Fast Reactor Design and Safety, Nuclear Engineering Division, Argonne National Laboratory (ANL), USA
Sinha, R.K.	Reactor Engineering Division, Bhabha Atomic Research Centre (BARC), Trombay, Mumbai, India
Su'ud, Z.	Department of Physics, Bandung Institute of Technology (ITB), Indonesia
Takahashi, M.	Research Laboratory for Nuclear Reactors, Tokyo Institute of Technology, Japan
Toshinskiy, G.I.	State Scientific Centre Institute of Physics & Power Engineering (IPPE), Russian Federation
Trifanov, G.A.	State Scientific Centre Institute of Physics & Power Engineering (IPPE), Russian Federation
Tsiklauri, G.	Pacific North-West National Laboratory (PNNL), USA
Vasilyev, B.A.	Experimental Design Bureau of Machine Building (OKBM), Russian Federation
Wade, D.	Reactor Analysis Division, Argonne National Laboratory (ANL), USA
Yoshioka, R.	Information & Control Solution Division, Toshiba Corporation, Japan
Zelenskiy, V.N.	State Scientific Centre Institute of Physics & Power Engineering (IPPE), Russian Federation