Status report 78 - The Evolutionary Power Reactor (EPR)

Overview

Full name	The Evolutionary Power Reactor
Acronym	EPR
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	4590.00 MWth
Electrical capacity	1770.00 MWe
Design status	Under Construction
Designers	AREVA
Last update	04-04-2011

Description

Introduction

1.1. Historical technical basis of the EPR™

The EPR™ is an evolutionary development by Framatome and Siemens of the current pressurized water reactors operating in Germany and France. Following the merger of the nuclear activities of Framatome and Siemens in January 2001 the EPR™ project was integrated into AREVA.

After extensive discussions the operators and suppliers were able to establish a common safety basis for the future product despite the differing national technical codes and standards. The definition of fundamental safety requirements for future pressurized water reactors in January 1995 by the French Groupe Permanent Réacteur and the German Commission for Reactor Safety provided a basis for the technical design of the EPR™. The general outline of the Nuclear Island was designed during the basic design phase, started in February of 1995, the objective of which was to generate a preliminary safety analysis report.

The cooperation was also ensured on the part of the nuclear safety authorities by the Franco-German steering committee (Deutscher Franzosich Direktionauschuz, DFD), formed by the German Ministry for the Environment and the French Direction de la Sûreté des Installations Nucléaires (DSIN). Similar to the DFD, the national advisory councils, the German Commission for Reactor Safety and the French Groupe Permanent Réacteur, also formed a joint working group. The technical assessments of the EPR™ concept were performed by common working groups of the German Gesellschaft für Reaktorsicherheit (GRS) and the French Institut de Protection et de Sûreté Nucléaire (IPSN).

The development organization set up, defined jointly by all the project participants, enabled applying the experience feedback of each partner in the most complete and most effective manner.

The European Utility Requirements (EURs), established by the European utilities, represent common utility views on the design and performance of future nuclear LWR power plants. After assessment by a group of fourteen European utilities, compliance of the EPR™ described hereafter with respect to the revision C of the EUR was confirmed in July 2009.

The following description of the EPR™ is based on the Standard EPR™ which already incorporates the feedback from the two on-going constructions in Finland (Olkiluoto 3) and in France (Flamanville 3). Technical data is summarized in Appendix 1, and abbreviations are defined in Appendix 2.

1.2. Design features and rationale

1.2.1. General objectives

From the beginning of the project one of the major targets was to further enhance the safety level of the EPR™ with respect to those, already very high, of the existing nuclear power plants in France and Germany. In coherency with the rules established by the French and German nuclear safety authorities for the next generation of pressurized water reactors, the EPR™ responds to the following principles:

- An "evolutionary" design, so as to draw maximum benefit from the accumulated experience in designing and operating the PWR units now in service in the two countries.
- An enhanced safety level: on one hand, a decreased core melt probability has been achieved by reducing the frequency of initiating events and by ensuring higher availability of the safety systems. On the other hand, actions have been taken to limit the radiological consequences in case of a severe accident. For accidents without core melt, the architecture of the peripheral buildings as well as the associated ventilation systems enable showing the non-necessity of protective measures for the people living near the damaged NPP unit. In the highly improbable but nevertheless envisaged situation of a core melt accident at low pressure, the reinforced reactor building and specific palliative devices will limit radioactive releases. Hence, a few protective measures, very limited in space and time. Lastly, the reactor design enables excluding situations that could lead to large early releases.
- Taking reactor operating problems into account very early in the EPR™ design: In-depth work has been done during the Basic Design phase to ensure that the design will enable attaining personnel radiation exposure levels as low as possible. Equipment maintenance has also been taken into account, by imposing installation rules that ensure good accessibility. The human factor has been integrated into the design effort, to guarantee the best possible prevention of human error in the operation of EPR™ units.

Due consideration was also given to the reduction of the residual risk allocated to beyond design accident sequences.

1.2.2. Main safety principles

The EPR™ design safety approach is that of generation III$^+$ PWRs developed by AREVA and MHI, respectively the EPR™ and the APWR.

Confinement of radioactive products present in the fuel is ensured by three successive physical barriers to prevent their release to the environment: the fuel cladding, the RCPB and the internal containment building with its associated features (extensions, isolation devices). The sturdiness of the barriers is assessed in the defence in depth framework.

The basic safety approach underlying the EPR™ design remains the internationally recognized defence-in-depth principle, which follows in particular the Design Safety Requirements for Nuclear Power Plants (1) of IAEA's Safety Standards Series and the INSAG (2)recommendations. This principle applies to all safety activities, one of which being the design, subject to partially overlapping requirements. Hence, a failure would be detected and corrected or mitigated by appropriate measures. Its application throughout the design ensures a gradual protection against transients, anticipated operating incidents and accidents of the most varying aspects. The latter include those resulting from component failure or human error within the plant, as well as external events.

The application of the defence in depth principle to the design is secured by a series of defence levels (intrinsic characteristics, components, procedures) aimed at preventing accidents and at providing an appropriate protection should the prevention fail.

Five different levels of defence have been identified.

The first level concerns all the preventive measures designed to reduce the risk of occurrence of abnormal situations. The second level integrates all the control and limitation systems that can intervene to limit the amplitude of transients that may result from the failure of the first level of defence. The third level includes all the safety systems designed to control the consequences of accident situations. A systematic analysis of multiple failures in redundant systems was conducted by AREVA on the EPR™ to show that even in such situations core melt is avoided.

The fourth level of defence in depth consists in defining control systems that make it possible to avoid the failure of the containment, even in the highly improbable case of a degenerative sequence going all the way to core melt at low pressure. The EPR™ integrates this requirement by use of simple systems having only a limited impact on the general architecture, and not risking interference with the operation of the NPP unit. The fifth level, finally, which concerns the organization on and off site to respond to any emergency situation that could lead to radioactive releases, could, in all logic, be simplified in view of the above-described improvements. But even though they are now less necessary, these measures were maintained.

An appropriate implementation of barriers and defence-in-depth together allows ensuring that the three basic safety functions - reactivity control, cooling the fuel and confining radioactive substances - are correctly ensured.

1.2.3. Systems architecture

The fluid systems architecture developed is the result of an intensive exchange of information about design and operating experience between the EPR™ designers and the participating electric utilities. The use of probabilistic evaluations at the very outset of the project was useful for defining the following orientations:

- Redundancy.

Four-train redundancy is used for the main safety systems (safety injection, emergency steam generator feed water supply) and the associated support systems (electrical power supplies and cooling systems). Some systems have kept two-train architecture (backup borating system) or three-train (spent fuel pit cooling system). The four-train architecture along with a four-loop primary system design contributes to the simplicity of operation mentioned earlier. It procures flexibility to adapt the design to the maintenance requirements during operation, but also during outages, when the redundancy level is increased due to the fact that the residual power is lower, as is the load on the systems that may need to intervene.

- Physical separation.

The different trains of the safety systems are installed in four divisions of the plant unit for which strict physical separation is applied. A common mode failure that would result from an external aggression (flood, fire, etc.) is therefore excluded by design.

- Functional diversity.

The risk that would result from common mode failures that could affect redundant systems has been reduced by systematically seeking functional diversity. If a redundant system is completely lost, there will always be a diversified system that enables performing this function and bringing the EPR™ unit to safe conditions (complete loss of the residual heat removal system, loss of steam generator feed water supply or complete loss of the medium-pressure safety injection system).

1.2.4. Control of severe accidents

Fully meeting the safety objectives with respect to severe accidents leads to incorporating particular measures; the main ones are the following:

- Containment by-pass situations are excluded, in particular high-pressure core melt situations which can endanger the integrity of the containment. In existing NPP units, the high reliability of the depressurization and residual heat removal systems make it possible to practically exclude this risk. In the EPR™, a supplementary line of defence is provided: two trains of motor-driven valves controlled by the reactor operator enables palliating the other lines of defence.
- Exclusion of violent phenomena that can result from the production of hydrogen is provided by catalytic recombiners. The pressure increase that would result from the combustion of residual hydrogen (assuming recombiners) is taken into account in the containment design.
- Corium spreading and cooling can take place in a dedicated spreading area next to the bottom of the reactor pit, whose walls and floor are covered with a refractory material that prevents the erosion of the structural concrete. An entirely passive device enables covering the layer of hot material with the water from the In-Containment Refuelling Water Storage Tank (IRWST), located next to the corium spreading chamber.
- The pressure inside the reactor building is controlled by a dedicated containment heat removal system. The cooling capacity is also used to maintain an isothermal layer in the structural concrete under the spreading area, which ensures its long-term integrity.
- The design of the NPP unit buildings enables collecting possible leaks through the containment penetrations and filtering them before their release through the stack.

All these measures make it possible to meet the very strict radioactive release objectives that have been imposed for next-generation reactors.

Some technical data of the EPR™ plant are provided in Appendix 1.

2.1. Main characteristics of the primary circuit

The Reactor Coolant System (RCS) configuration is that of a conventional PWR four-loop design and can be considered well proven (figure 2.1-1). The sizing of the EPR™ reactor pressure vessel (RPV), steam generator (SG) (especially secondary side) and pressurizer (PZR) incorporates increases in the respective volumes compared to current 4-loop PWR designs.

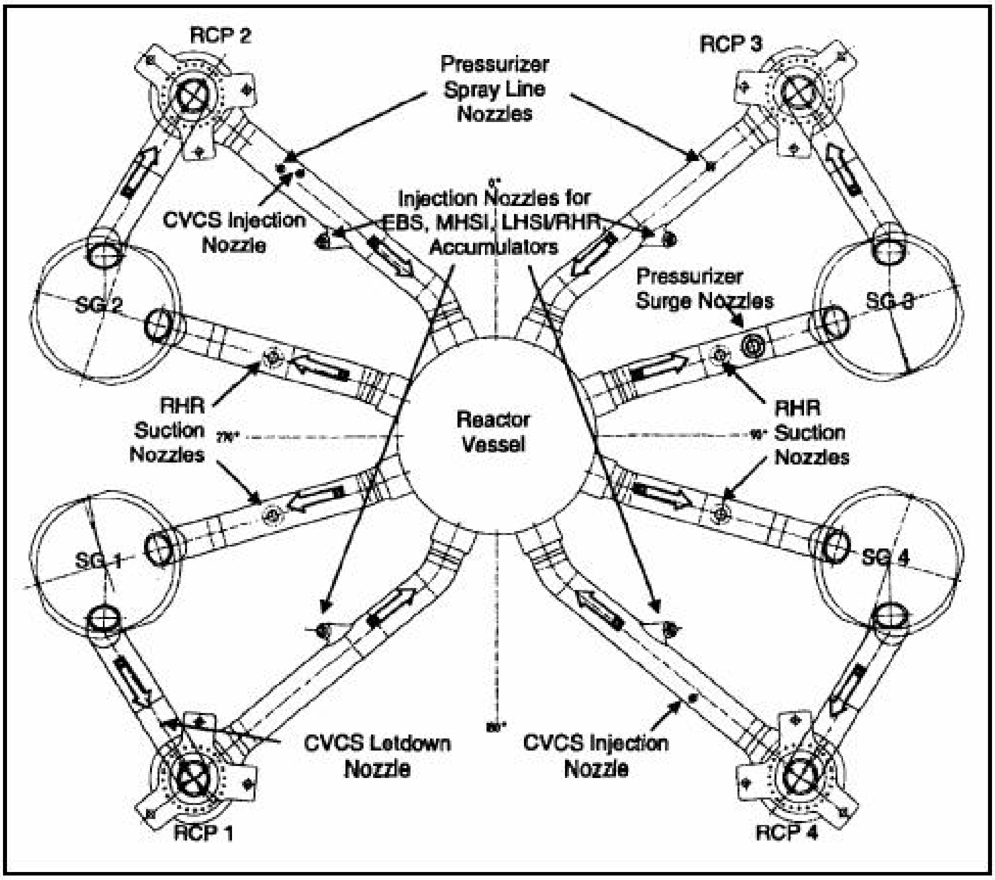


Figure 2.1-1: RCS layout

In the RPV design, the free water volume between the level of the reactor coolant lines and the top of the active core is increased in order to improve the mitigation of LOCA (smaller breaks) by prolonging the period until the core begins uncovering or by minimizing the core uncover depth. The increase of this volume also contributes improving the mitigation of accidents during shutdown conditions, particularly in mid-loop operation (e.g., with loss of RHR), by extending the period for operator action.

The increased pressurizer water and steam volume, with associated pressure and level scaling, favours reducing the number of load cycles on relevant systems and components. Normal operating transients are mild; the potential for reactor trips is thus minimized. Actuation of pressurizer safety valves can be avoided for events such as loss of condenser.

The large water and steam volumes of the SG secondary side provide the following advantages:

- Smooth pressure and water level changes during normal operating transients, thus further reducing the potential for unplanned reactor trips,
- Significant time delays to mitigate a steam generator tube rupture (SGTR) prior to filling of the SG,
- Extended dry out time beyond the required 30 minutes for the most limiting event of a total loss of feed water supply (including emergency feed water).

Overpressure Protection (OPP) of the Reactor Coolant Pressure Boundary (RCPB) in both hot and cold conditions is performed by the PZR safety valves in parallel with the reactor protection system and associated equipment. The OPP prevents opening of non-isolable valves during all anticipated operational occurrences and accidents having the potential for radioactive releases.

2.2. Reactor core and fuel design

2.2.1. Overall features

The main features of the core and its operating conditions result in a high thermal efficiency of the plant, low fuel cycle costs, and flexibility for extended fuel cycle lengths. The reactor core consists of an array of 241 fuel assemblies, 36 more than in the four operating units of the French N4 series.

The EPR™ core design has a high degree of flexibility with respect to cycle length adaptations, allowing fuel cycle cost reductions by high burn ups and low leakage loading patterns. By design, the core has stabilizing reactivity coefficients under all operation conditions, thus meeting basic safety objectives.

2.2.2. Fuel assemblies

The EPR™ core can accommodate different fuel assembly designs such as AFA 3GL or HTP. Basically designed for UO_2 fuel elements, the EPR™ core has the capability to be loaded with up to about 50% of MOX-fuel assemblies [1].

Each fuel assembly, of 4.20 m active length, consists in a 17 x 17 lattice of 265 fuel rods and 24 guide tubes mechanically joined in a square array. The fuel rods are mechanically restrained axially and radially in the fuel assembly structure by eight M5^{TM} intermediate grids and two end spacer grids.

The fuel rods are made of Zircaloy tubing containing UO_2 ceramic pellets, the initial enrichment of which is less than or equal to 5.0 Wt% U^{235}. The cladding and tubing are made of highly corrosion-resistant and low-growth M5^{TM} alloy.

The top nozzle of the fuel assembly is the structural element that interfaces with the top core plate. The top nozzle also supports the hold-down springs of the fuel assembly, which are used to prevent hydraulic lift-off of the fuel assembly during operation.

The robust FUELGUARD^{TM} bottom nozzle of the fuel assembly serves as the structural element that interfaces with the bottom core plate. The bottom nozzle shape directs and equalizes the flow distribution and also filters out small debris.

Some fuel assemblies contain burnable absorber (Gd_2O_3) to suppress high excess reactivity, especially in the first core.

2.2.3. Rod Cluster Control Assemblies and Reactivity Control

The core has a fast shutdown system consisting of eighty-nine RCCAs. All RCCAs are of the same type, each with twenty-four individual and identical absorber rods fastened to a common spider assembly. These rods are made of stainless steel tubing containing neutron absorbing materials. The material of the absorbers is a hybrid Ag-In-Cd (AIC) alloy and B_4C design, with AIC in the lower part and B_4C in the upper part.

The coolant contains soluble boron (B_{10} enriched) as a neutron absorber. The boron concentration in the coolant is varied to control slow reactivity changes necessary for compensating Xenon poisoning or burn-up effects during power operation and for compensating large reactivity changes associated with large temperature variations during cool down or heat-up phases.

2.3. Fuel handling systems

The spent fuel assemblies are transferred to the fuel pool located in the fuel building. The protection of the fuel building is achieved by full hardening. The inner building structures are decoupled from the outer protection wall in order to ensure the integrity of the spent fuel pool under external aggression. The fuel storage and handling equipment include storage of spent and new fuel assemblies outside the containment of sufficient storage capacity for full core unloading during outage.

The new fuel assemblies are stored in the fuel building to enable easy access thereto. Inside the reactor building, a loading machine transfers the spent and new fuel assemblies into or out of the reactor. The fuel assemblies are transferred from the containment to the fuel building and vice versa via a transfer tube. The fuel transfer tube is closed from both sides during normal operation. Fuel assemblies are handled through the bottom of the spent fuel pool using a fuel cask handling device.

2.4. Primary circuit components

2.4.1. Reactor pressure vessel (figure 2.4.1-1)

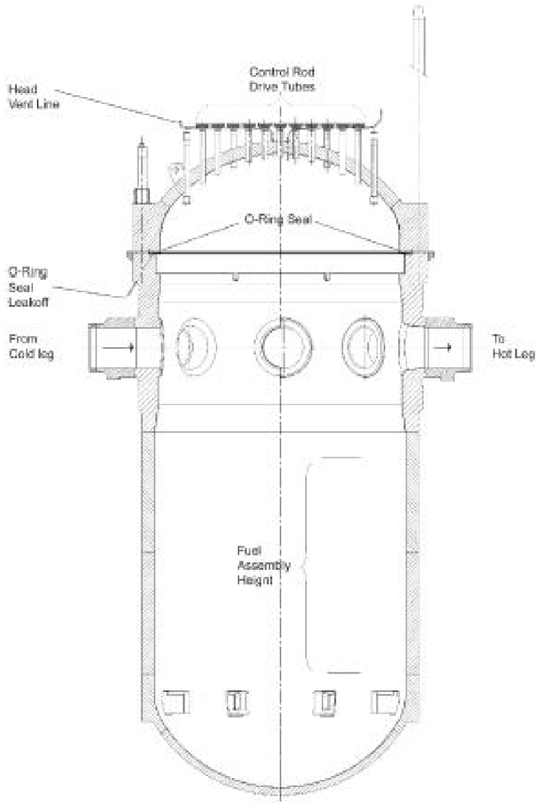


Figure 2.4.1-1. Reactor pressure vessel

The RPV is the main component of the RCS. The vessel is cylindrical, with a welded hemispherical bottom and a removable flanged hemispherical upper head with gasket.

The RPV is designed for a life time of 60 years. This is achieved by provision of a rather large water gap and a neutron heavy metallic reflector around the core. There is a high safety margin against brittle fracture in the design of the RPV based on the material and the low total neutron fluence. The RPV is made of low-alloy steel. The complete internal surface of the RPV is covered by stainless steel cladding for corrosion resistance.

The cylindrical shell of the RPV consists of two sections, an upper and a lower part. To minimize the number of large welds, which reduces the extent of in-service inspections, the upper part of the RPV is machined from a single massive forging piece and fabricated with eight nozzles. The nozzles are "set-on" the nozzle shell, thus requiring substantially less weld bead than would otherwise be required and facilitating non-destructive examination from inside the RPV. The nozzles are located as high as practicable above the core upper edge to increase the hydrostatic pressure for re-flooding and to avoid the loop seal effect.

The RPV closure head consists of two single-piece forgings, the closure head dome and the closure head flange, welded together with a circumferential weld.

The lower part of the RPV is made of two cylindrical shells at the reactor core level, one transition ring, and one bottom head dome. The bottom head is a hemispherical shell connected to the RPV body through the transition ring. There are no penetrations in the bottom head.

2.4.2. Reactor internals (figure 2.4.2-1)

The RPV internals consist of one lower and one upper sections.

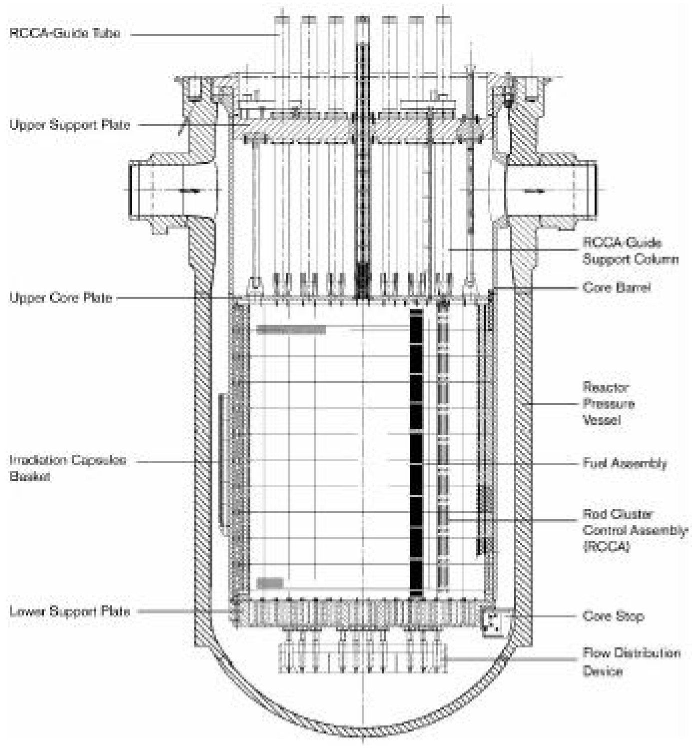


Figure 2.4.2-1. RPV internals

The upper internals are located in the upper plenum of the core barrel. They enclose the upper end of the reactor core and accommodate the RCCA guide and the reactor core instrumentation. The upper internals consist of the Upper Support Plate (USP) with skirt and flange, the perforated Upper Core Plate (UCP), and the various support columns in between.

The lower internals are made of the core barrel, the lower core support structure, the neutron reflector, and the flow distribution device. These are vertically supported by a ledge machined into the flange of the RPV. Their movement is restricted vertically inside the RPV by an annular hold-down spring located between the flanges of the lower and upper internals. This design prevents them from lifting off the RPV ledge.

The fuel assemblies are placed directly on a flat perforated plate, machined from a forging of stainless steel and welded all around to the core barrel. The space between the polygonal outside shape of the core and the cylindrical inner surface of the core barrel is filled by a stainless steel structure which reduces the fast neutron leakage to the RPV wall and flattens the power distribution in the core. This neutron reflector allows savings of fuel enrichment.

2.4.3 Steam generators (figure 2.4.3-1)

The EPR™ steam generators are vertical shell, natural circulation, U-tube heat exchangers with integral moisture separating equipment. They are fitted with an axial economizer which enhances the heat exchange efficiency between the primary side and the secondary side and increases the outlet steam pressure by about 0.35 MPa as compared to a boiler type SG having the same tube surface.

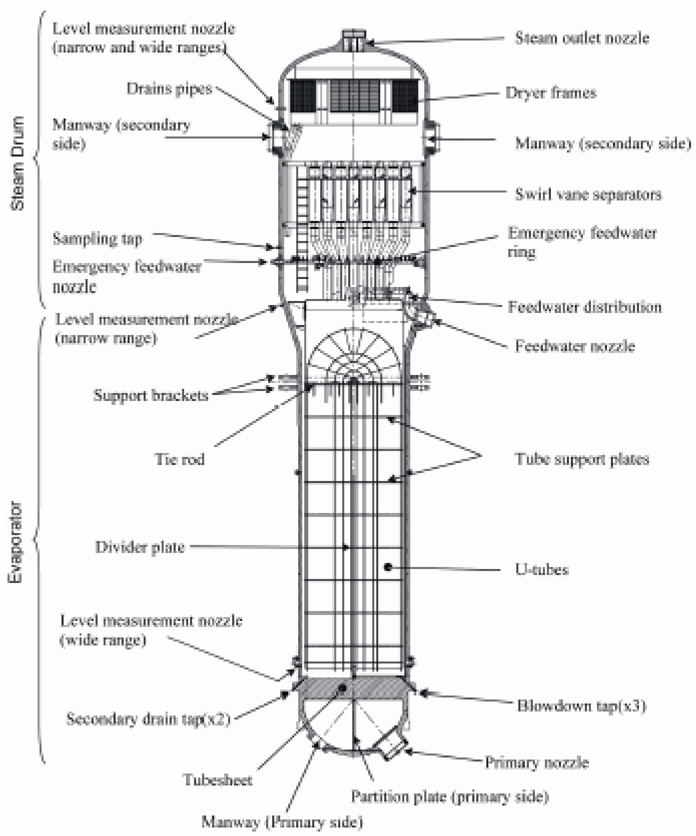


Figure 2.4.3-1. Steam generator

The tubes are made of Inconel 690, a widely used material in SGs throughout the world and highly resistant to corrosion. They are supported by perforated plates.

Parts in contact with primary coolant are made of corrosion resistant alloys or cladded with austenitic stainless steel (tube sheet, channel head).

2.4.4. Pressurizer (figure 2.4.4-1)

The PZR consists of a vertical cylindrical shell, closed at both ends by hemispherical heads. It is of conventional design but with enlarged water and steam volumes. The spray systems consist of three separate lines, two main lines provided for normal operation (connected to cold legs) and one auxiliary line connected to the CVCS, each equipped with a spray nozzle. The spray lines nozzles are welded laterally near the top of the upper cylindrical shell through a blind cover. This design is easy to dismantle, inspect and replace. The spray system delivers a permanent flow to the spray nozzles to minimize thermal transients upon fast valve opening.

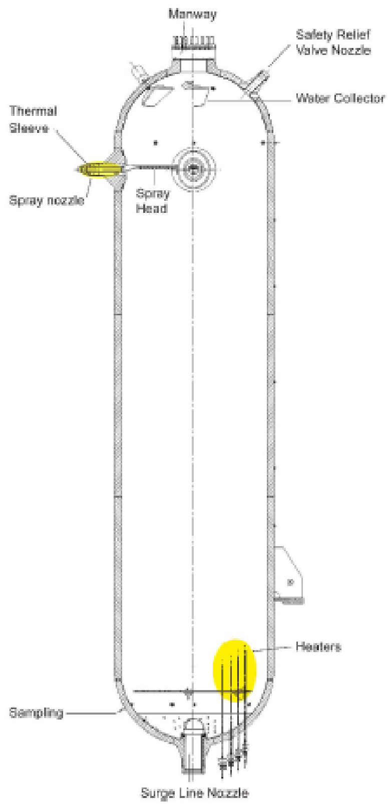


Figure 2.4.4-1. Pressurizer

The PZR electric heaters, installed vertically in the bottom head, have flanged connections with the penetrations in order to be easily replaced and inspected.

All pressure boundary parts, except for the heater penetrations, are made of ferritic steel, with austenitic stainless cladding on all internal surfaces in contact with the reactor coolant.

The PZR water volume is large enough to compensate for coolant expansion between 0% and 100% power under normal conditions and prevents the heaters from being uncovered during out-surges.

The large steam volume prevents frequent actuation of the pressure control equipment during normal operation.

2.4.5. Reactor coolant pumps (figure 2.4.5-1)

The reactor coolant pumps (RCP) are vertical, single-stage, shaft seal units, driven by air-cooled, three-phase induction motors. The casing of the hydraulic unit is made of austenitic stainless steel. They are of well-proven design, as already used in all PWR plants built by Areva in France and abroad.

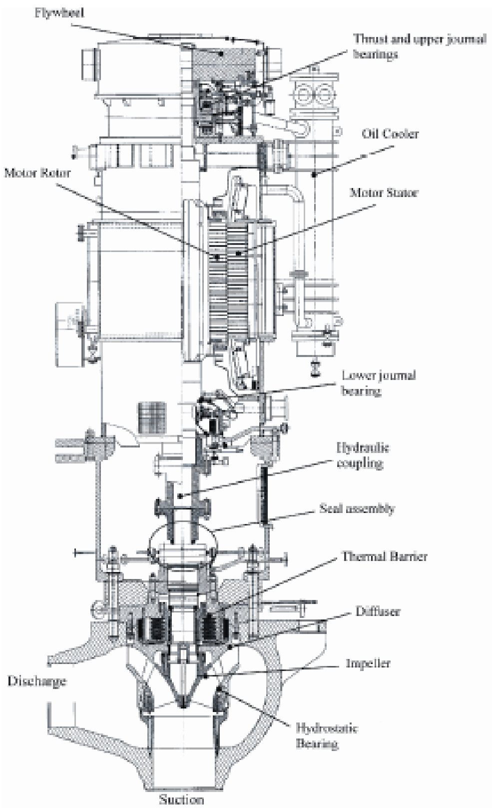


Figure 2.4.5-1. Reactor coolant pump assembly

The RCP is equipped with a standstill seal which ensures shaft leak-tightness without the need for an active seal water supply system when the pump is idle, e.g. in the event of a station blackout. The standstill seal system is actuated when the RCP is at rest after closure of all seal leak-off lines.

2.4.6. Main coolant lines

The RCS piping is designed according to the Break Preclusion (BP) concept which consists in a high quality in design, construction and surveillance. It enables to rule out a catastrophic failure of a main coolant line as regards its possible mechanical effects. This eliminates the need to design RCS components, piping and supports to accommodate the dynamic effects of large or double-ended ruptures. Consequently, large pipe whip restraints and jet impingement shields are not required.

Nevertheless, a mass flow equivalent to a double area break of a main coolant line is still assumed with realistic assumptions for the design of e.g. the emergency core cooling system and of the containment design.

2.5. Auxiliary systems

2.5.1. Chemical and volume control system (figure 2.5.1-1)

The Chemical and Volume Control System (CVCS) is an operational system which controls the water inventory, the water quality and the boron concentration in the primary system. Additionally, the system adjusts the chemical composition of the RCS and removes dissolved gases by degasification of the letdown flow. The CVCS provides a flow path for the continuous letdown and charging of RCS water. The CVCS maintains the RCS water inventory at the desired level via the PZR level control system and provides Reactor Coolant Pumps (RCP) seal water injection and auxiliary spray for PZR cool down when the normal pressurizer spray is unavailable.

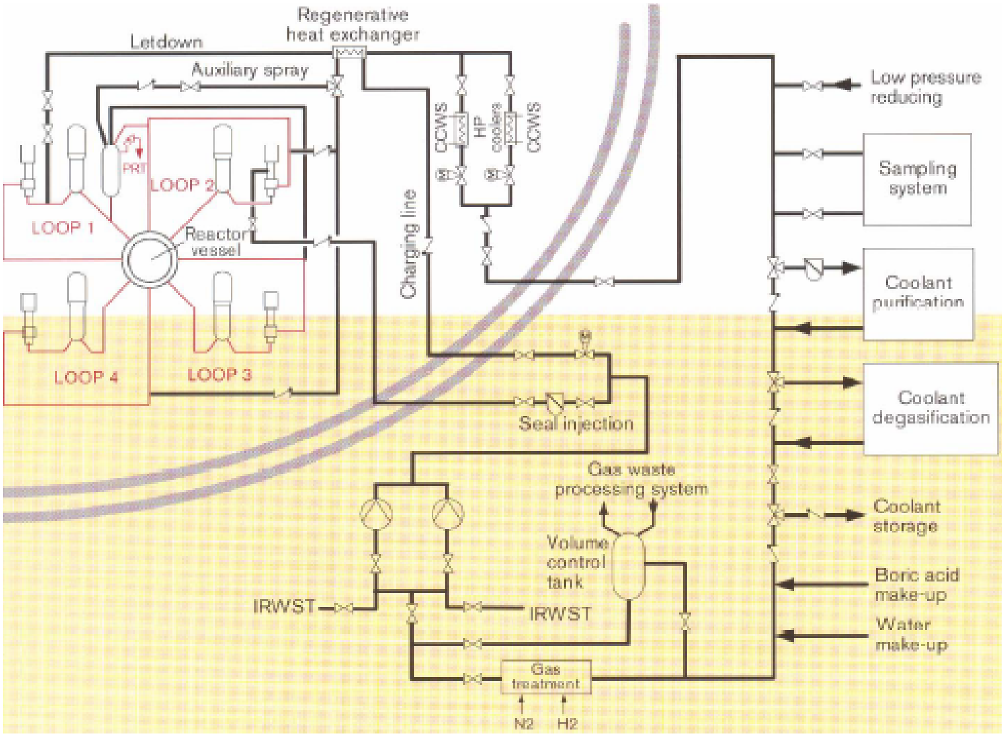


Figure 2.5.1-1. Chemical and Volume Control System

The CVCS is normally in continuous operation during all modes of plant operation from normal power operation to cold shutdown.

Major components of the CVCS are protected from external hazards by the building design and are physically separated or provided with protection from internal hazards.

2.5.2. Component cooling water system (figure 2.5.2-1)

The component cooling water system (CCWS) has the capability to transfer heat from safety-related systems and operational cooling loads to the heat sink via the essential service water system (ESWS) under all normal operating conditions.

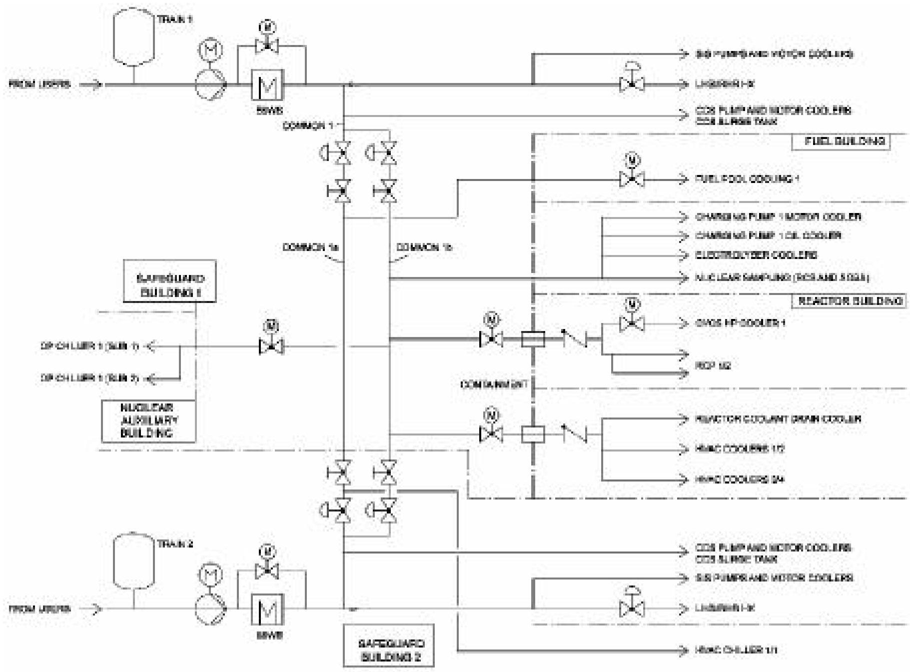


Figure 2.5.2-1. Component Cooling Water System (trains 1 & 2)

The CCWS performs the following safety functions:

- Heat removal from the safety injection/residual heat removal system to the ESWS;
- Heat removal from the spent fuel pool cooling system to the ESWS;
- Cooling of the thermal barrier of the reactor coolant pumps;
- Heat removal from the Heating, Ventilation and Air Conditioning (HVAC) chillers of Division 2 and 3.

The CCWS consists of four separate safety classified trains (1 to 4) corresponding to the four layout divisions (1 to 4) and two separate common loop sets.

2.5.3. Essential service water system (figure 2.5.3-1)

The ESWS consists of four separated safety-classified trains that provide cooling of the CCWS heat exchangers with water from the heat sink during all normal plant operating conditions, transients, and accidents.

The safety functions of the ESWS are to provide cooling water to:

- The four CCWS/ESWS heat exchangers which, in turn, cool components of the safety systems;
- The fuel pool cooling system, as long as fuel assemblies are in the spent fuel storage pool.

2.5.4. Component Cooling Water System dedicated to severe accidents

In the case of severe accidents the containment atmosphere, the IRWST and the corium spreading area would be cooled by the CHRS. The CHRS pumps and heat exchangers would in turn be cooled by two additional CCWS trains completely independent from the main four-train CCWS.

These two separate dedicated trains cool both (i) the CHRS, when it is used during a severe accident involving core melt and, (ii) the backup train of the FPCS.

2.5.5. Essential service water system dedicated to severe accidents

Two additional essential service water trains are dedicated to cool the CHRS heat exchangers for severe accident mitigation. They are completely independent from the main four-train ESWS.

2.6. Operating modes

The EPR™ has the capacity to be permanently operated in automatic mode at any power level between 25 and 100 % of rated power (RP), primary and secondary frequency controls possibly being permanently in operation if need be.

With respect to load variations, the capacities of the EPR™are defined for two power ranges:

- a "standard" range between 60 % and 100 % RP in which the plant is expected to operate most of the time,
- a "less usual" range of operation between 25 and 60 % RP.

The greatest advantages of this part load diagram are limited thermal stresses, especially to the RCS components, and small reactivity changes during load variations due to temperature coefficient variation (moderator effect).

The main steam relief and safety valves respective set points are set above the response value of the main steam bypass station, thus possible excess steam is dumped via the main steam bypass into the condenser and not released to the atmosphere.

Over the complete power range (0% - 100%) the RCS pressure is kept constant by the RCS pressure control.

In addition, the EPR™ is designed to withstand without tripping of the reactor, events like turbine trip, full load rejection, trip of one reactor coolant pump, trip of one feed water pump, and malfunction of a single control system.

2.7. Standard Fuel cycle (open, closed)

After unloading spent fuel from the reactor, two options may be envisaged, according to availability of uranium resources, technical feasibility, industrial policy and costs.

A first option consists in letting the spent fuel radioactivity decrease for several years and then to store it permanently. This so-called "direct storage" or "open cycle" option is followed by many countries using nuclear energy to produce electricity, the USA among others.

A second option consists in considering that the potential residual value of the spent fuel can be of further economic use by sorting the products it contains. This so-called "closed cycle" option with geologic storage of vitrified wastes is the one adopted by France [2], as well as by the UK and Japan.

2.8. Alternative Fuel options

The EPR™ is designed for being mainly operated with UO2 fuel, but having however a capacity for MOX recycling of 50%, or 100% with limited adjustments. The main features of the core and its operating conditions are selected for reaching, on the one hand a high thermal efficiency of the plant and low fuel cycle costs and, on the other hand an extended flexibility with respect to the selection of the fuel cycle length (from one up to two years) and a high manoeuvrability.

The fuel rods are made of M5^{TM} tubing containing originally:

- Either uranium dioxide ceramic pellets, of which the U^{235} initial enrichment is less than or equal to 4.95 wt%;
- Or uranium dioxide ceramic pellets made of depleted uranium where Pu is added as fissile material in the form of PuO_2. The maximum envisaged fissile Pu-enrichment in fuel rod is consistent with the value of 12.5 wt% for total Pu content which is a factory limit.

2.9. Spent nuclear fuel and disposal plans

The EPR™ uses the same types of enriched uranium elements as previous generation French and German LWRs, but it does so more efficiently due to its neutronic design and its ability to reach higher burn-ups. Thus the EPR™ consumes less fuel (17%) than the reactors of the N4 series and produces less irradiated material - and therefore less waste (26%) - for the same amount of energy produced.

After a few years of cool down in the spent fuel pools of the nuclear power plants, the spent fuel elements are shipped to a reprocessing plant. At this location, materials of the spent fuel which can still be valued, such as plutonium and uranium, which represent about 95% of the total mass (excluding oxygen) are separated and recycled as fuel. Materials which cannot be valued, such as high activity-long life wastes, are conditioned in a glass matrix and stored in a specific facility of the La Hague plant.

The plutonium originating from reprocessed spent fuel is recycled in the form of mixed uranium and plutonium oxides fuel (MOX). Presently, MOX spent fuel is not further reprocessed, although representing a potential economic value. The highly radioactive accumulated fission products prevent immediate reprocessing. However, MOX spent fuel constitutes an optional long term energy resource for fast breeder reactors.

The uranium originating from the reprocessed spent fuel can either be re-used in PWRs after being re-enriched in U^{235}, or stored for further utilisation in fast breeder reactors.

In some countries, the spent fuel may be stored in an interim storage facility, pending a decision about either reprocessing, if it is allowed by the law of such countries at that time, or the setting up of a final geological waste disposal facility.

---

[1]. 100% with only limited systems/layout modifications.

[2]. The law 2006-739 of June 28th, 2006 "Loi de programme relative à la gestion durable des matières et déchets et radioactifs" allows for studies and research performed in order to create new storage facilities of highly radioactive wastes or to modify existing ones, at the latest in 2015, fulfilling the needs, particularly in term of capacity and duration.

Description of safety concept

3.1. Safety concept, design philosophy and licensing approach

The EPR™ design follows an evolutionary approach incorporating the operational experience from approximately 100 nuclear power plants in the world (Belgium, Brazil, China, France, Germany, Rep. of Korea, Rep. of South Africa, Spain) constructed in the past by Framatome and Siemens. In addition, experience feedback from other nuclear power plants has been reviewed and design features addressing the generic safety issues identified have been taken into account (e.g. SG tube integrity, overfilling of SG in case of SG tube rupture, ECCS sumps blockage, improvement of SG feed water system availability, improved reliability of the power supply system, containment integrity following a core melt accident).

The EPR™ safety concept is to further enhance the already very high safety level attained at French and German plants. The EPR™ meets the French Nuclear Safety Authority's safety requirements of October 2000 "Technical Guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors". This implies improving the prevention of accidents, including severe accidents, and adding features, mainly related to the containment, to mitigate the consequences of postulated severe accident scenarios - including core melt situations - to avoid the need for stringent off-site countermeasures. The frequency of such postulated accidents has been significantly reduced.

The EPR™ design is based on the following objectives related to current PWRs:

- Increase redundancy and separation;

- Reduce core damage frequency (CDF);
- Reduce large release frequency (LRF);
- Mitigate severe accidents;
- Protect critical systems from external events;
- Improve human-machine interface;
- Extend response times for operator actions.

3.1.1 Licensing approach

EPR™ design assessments have been carried out by international regulatory authorities as well as comparison of EPR™ design principles with current international standards.

Reviews were performed by the French Regulator for the Flamanville 3 EPR™, the Finnish Regulator for the Okiluoto 3 EPR™ and the USNRC assessment for a generic EPR™ design.

The EPR™ Nuclear Island was assessed by a large group of European Utilities. During a year and a half each of the 4800 requirements of the EUR volumes 1 & 2 was analysed by EUR utilities' engineers from information supplied by Areva. In July 15th, 2009 the EUR organisation released a certificate to AREVA which validates the excellent level of compliance of the EPR™ with these requirements (99.5% excluding the non applicable requirements).

Within the MDEP [1] a specific EPR™ Working Group is established including, under the chairmanship of the Finnish Regulator, representatives of the French, US, UK and Chinese regulators. This Working Group which meets regularly is sub structured in several sub-groups which deal with more specialized topics such as I&C, severe accidents, PSA among others.

3.2. Provision for simplicity and robustness of the design

Simplification in all areas of design, construction, operation and maintenance results in a cost-effective design and contributes to improve safety.

The European Utilities who assessed the EPR™ with respect to the EUR concluded that:

"Simplification of the safety system configuration is one of the design approaches that enhance the economy of the EPR™. The simplified system design combines the advantages of a system configuration using diverse backup safety functions with concept of providing a high degree of redundancy with separation of functions. Both objectives basically targeted a very high safety level, but the high degree of redundancy, especially, also provides cost saving possibilities for operation and maintenance.

Important safety systems and their support functions (safety injection, emergency feed water, component cooling, emergency electric power) are arranged in a four train configuration. The high degree of redundancy has significant advantages regarding an optimized preventive maintenance concept that makes possible maintenance and inspection during operation. Thus plant outage time is reduced and plant availability is increased.

Systems and components are designed, constructed and tested according to quality standards commensurate with their importance to safety. The corresponding rules are based on the experience gained from previous generation plants.The design criteria can be summarized as follows:

Simplicity and functional separation:

- The separation of functions is applied, as far as appropriate;
- Contradictory demands on valves in the short term are avoided as a basic principle.

Redundancy and diversity:

- Safety systems are designed to accomplish their safety functions even in case of a component failure or component unavailability (e.g. single failure or preventive maintenance);
- Diversity of systems and components is applied as much as possible to cope with the risk resulting from

common cause failures. Priority is given to functional diversity over equipment diversity.
- Divisional separation: Redundant trains of safety systems are arranged in separated divisions. The divisional separation is also extended to supporting systems such as cooling water, power supply and I&C;

On the basis of the already reached low probability for the occurrence of Severe Accidents in French and German nuclear power plants, engineering efforts were made to further improve the accident prevention level by simplification of safety systems, among others."

3.3. Inherent safety features

The idea of performing safety functions by passive means is not new. All existing PWRs employ passive features like accumulators, gravity-driven control rod insertion or natural circulation in the primary circuit. Besides these, additional passive features have been included in the EPR™ design such as:

- Larger SG and pressurizer volumes providing increased thermal inertia, thus slowing plant response to upset conditions,
- Initial SIS valve line-up (suction from IRWST) meets long term cooling needs without realignment,
- Lower core elevation relative to the cold leg cross-over piping which limits core uncovering during small break LOCAs,
- Passive pressurizer safety valves for both overpressure protection and prevention of spurious opening (passive opening under pressure increase, passive closing under pressure decrease),
- Large dedicated spreading area outside the reactor cavity to prevent the molten core-concrete interaction, by self spreading and subsequent passive flooding of the corium,
- Large water source in the IRWST located inside the reactor building, draining by gravity into the reactor cavity and the corium spreading area,
- Double wall containment with a reinforced concrete outer wall and a pre-stressed concrete inner wall,
- Uninterruptible power supply ensured passively with batteries in case of station blackout and failure of all emergency diesel generators,
- Self-mixing of the containment atmosphere to minimize hydrogen concentrations,
- Removal of the hydrogen from the containment atmosphere by passive autocatalytic recombiners.

In addition to the above features, about twenty passive features were assessed in detail at the beginning of the conceptual phase of the EPR™ [2].

3.4. Defence in-depth description

Defence in depth consists of recognizing that technical, human or organizational failures may occur in a plant lifetime and to prevent them by introducing successive lines of defence comprising five levels, the aims of which are:

First level: to prevent deviations from normal operation, and to prevent system failures. This leads to the requirement that the plant be soundly and conservatively designed, constructed, maintained and operated in accordance with appropriate quality levels and engineering practices, such as the application of redundancy, independence and diversity.

Second level: to detect and intercept deviations from normal operational states in order to prevent Design Basis Conditions 2 from escalating to accident conditions.

Third level: it is assumed that, although very unlikely, some initiating events or the escalation of certain Design Basis Conditions 2 may not be arrested by a preceding level and a more serious event may develop. These unlikely events are anticipated in the EPR™ design basis; inherent safety features, fail-safe design, additional equipment and procedures are thus provided to control their consequences. This leads to the requirement that engineered safety features be provided that are capable of leading the plant first to a controlled state, and subsequently to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material.

Fourth level: to address severe accident conditions in which the design basis may be exceeded to ensure that the radioactive releases would be maintained as low as practicable. The most important objective of this level is the protection of the confinement function. This is achieved on the EPR™ by complementary design features and procedures.

Fifth level: is aimed at mitigation of the radiological consequences of potential releases of radioactive materials that may result from severe plant conditions. This requires the provision of an adequately equipped emergency control centre, and plans for the on-site and off-site emergency response.

3.5. Safety goals (CDF, LERF and operator grace period)

The EPR™ design takes into account general safety objectives which are based on the Technical Guidelines of October 2000 and the EURs. These objectives address the full life cycle including construction, commissioning, operation, maintenance and decommissioning.

The safety approach is based on a deterministic approach, supplemented by probabilistic methods, supported by appropriate research and development work.

The goals of the EPR™ in comparison with those former PWRs are:

- to provide a significantly higher level of safety, focusing on protection of workers and public against the effects of radiations in all plant conditions,
- to limit the impact of the plant to the environment,
- for accident situations without core melt, to avoid the necessity to shelter or evacuate people in the vicinity of the plant,
- for accident situations with core melt, to eliminate by design the likelihood of events which could lead to large early releases; other sequences must be mitigated in order to necessitate only very limited public protective measures in area and in time.

The probability for core melt including all internal events and internal and external hazards is thus well below the 10^{-5}/reactor year (RY) target.

The EUR mean value target for sequences potentially involving either early failure of primary containment or very large release of a cumulative frequency smaller than 10^{-6} /RY is largely met for the EPR™.

3.6. Safety systems to cope with Design Basis Accidents (DBA)

In the deterministic analysis the different internal events are categorized in four categories (DBCs) in accordance with their anticipated frequency of occurrence; DBC1 covers normal operation states, and DBC2 to DBC4 envelope anticipated operational occurrences, infrequent and limiting accidents.

Stringent radiological targets are applied for normal operation and anticipated operational occurrences as well as for accidents.

3.6.1 EPR™ safety systems configuration

Important safety systems (safety injection, emergency feed water, main steam relief, cooling chain, emergency electric power) are arranged in four trains.

The layout comprises four separate divisions, corresponding to the four trains. A simple and straightforward system design approach is favoured, thereby facilitating operator understanding of plant response and minimizing configuration changes. The four train configuration offers the possibility of extended periods of maintenance on parts or even entire systems, useful for preventive maintenance and repair work during normal operation.

3.6.2 Safety injection system (figure 3.6-1)

The safety injection systems (SIS) mitigate loss of coolant accidents of all sizes to ensure limited fuel damages, even in case of large breaks, and specific non-LOCA events, such as main steam line breaks and sequences leading to feed and bleed.

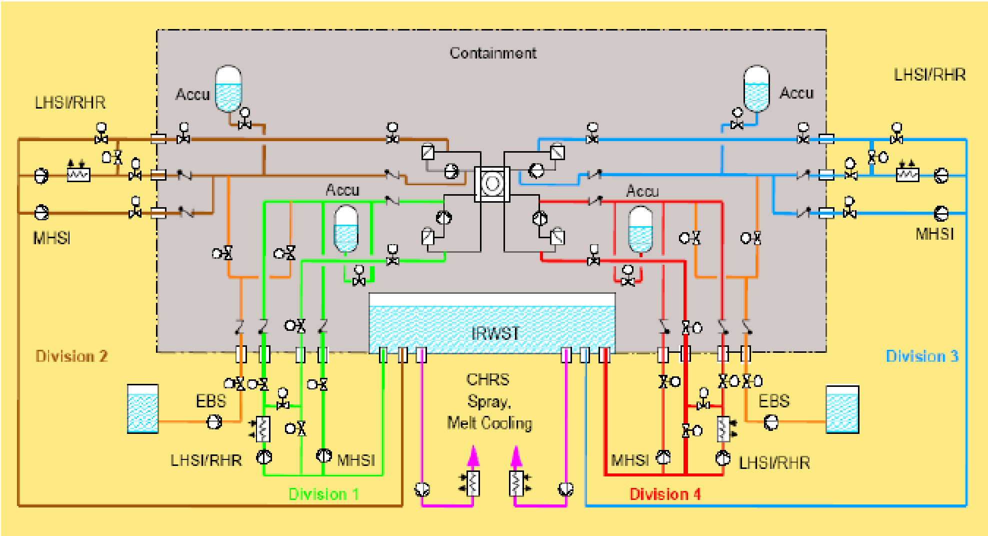


Figure 3.6-1. Safety injection system

The medium head safety injection (MHSI) system feeds into the cold legs of the reactor coolant system. The shut-off head of the MHSI system is sufficient to cope with all LOCA related requirements because the safety grade partial cool down capability is activated simultaneously. The delivery head of the medium head safety injection (MHSI) system is set below the SG safety valve set points so that following a steam generator tube rupture (SGTR), the affected SG is isolated on the secondary side. After the initial transient, the primary and secondary pressures will equalize at a level below the set points of the safety valves of the affected SG, limiting the radiological releases to negligible levels.

During DBC 3 or 4 the low-head safety injection system (LHSIS) transfers the decay heat to the ultimate heat sink via heat exchangers. The LHSIS feeds into the cold leg during the initial phase. In order to stop the core outlet steaming and the steam release into the containment, operation is subsequently switched to hot leg injection. The LHSI injection pressure offers advantages for feed and bleed operation and supports accumulator injection in an optimum way for a large spectrum of break sizes.

Cold leg accumulator injection is provided to cope with large and intermediate break sizes. One accumulator is assigned to each cold leg.

The SIS features an in-containment refuelling water storage tank (IRWST) located at the bottom of the containment.

3.6.3 In-containment refuelling water storage tank

The in-containment refuelling water storage tank (IRWST) provides the source for emergency core cooling water. It is located inside the containment, on the bottom level, between the reactor cavity and the missile protection cylinder. In the case of loss of coolant accidents, or in feed and bleed situations, the safety injection system draws from the IRWST. The water steam mixture escaping through the leak, and through the bleed valve, respectively, is returned to the IRWST. In the case of severe accidents the IRWST would provide water to flood and cool the molten corium once relocated in the spreading area.

In addition, the storage tank provides water for the operational function of flooding the reactor pit and the pools during refuelling.

3.6.4 Emergency feed water system

The emergency feed water system (EFWS) consists of four separate and independent trains, each providing injection to one of the four steam generators. Each EFW pump takes suction from an EFW pool, each located in the corresponding division of the safety buildings. The four EFW pumps are motor-driven, power supplied by emergency buses. In addition, two of them are connected to diversified diesel generators, thus reducing the probability of common cause failure of all emergency power supplies.

Contrary to existing PWRs, the EFWS of the EPR™ does not have operational functions. A dedicated start-up and shutdown system (SSS) is used for start-ups and shutdowns. The SSS is automatically started in case of loss of main feed water, thus minimizing the need for the EFWS.

In the case of a SGTR, the emergency feed water system removes the heat via the intact steam generators. The pressure in the affected steam generator is allowed to increase so as to reduce and eventually eliminate the break flow from primary to secondary side; the maximum pressure will remain at a level below the opening set point of the steam generator safety valve. The EFWS keeps the water inventory of at least one SG above an adequate level to maintain primary to secondary heat transfer, assuming a single failure and maintenance.

The EFWS provides sufficient heat removal capacity and autonomy to ensure continued removal of decay heat for 24 hours with a final RCS temperature not exceeding nominal hot shutdown conditions. This could also be accomplished under the assumptions that neither electric power from external sources nor the ultimate heat sink is available.

3.6.5 Residual heat removal system

In addition to its accident mitigation functions, the LHSIS is part of the operational residual heat removal system (RHRS). The RHRS is designed to transfer residual heat from the RCS, at low RCS temperatures, via the component cooling water system (CCWS) and the essential service water system (ESWS) to the ultimate heat sink, when heat removal via the SGs is not sufficient. Furthermore, it ensures continued heat transfer from the RCS or from the IRWST during cold shutdown or refuelling conditions.

The RHRS consists of four independent trains, each of which uses the LHSI pump and LHSI heat exchanger. The LHSI pump takes suction from a RCS hot leg and discharges via LHSI heat exchangers to a cold leg of the RCS. A bypass line of the heat exchanger is provided to allow control of the cool-down rate. The LHSI heat exchangers are cooled by the associated CCWS train, located in the same division. During normal operation, only two RHR trains are used for cool down and cold shutdown. All four trains are not used unless the RCS temperature is below 100°C.

In case of a break in one of the RHRS trains outside the containment, the affected train is automatically isolated.

3.6.6 Extra borating system

The safety function of the extra borating system (EBS) is to ensure, for any DBC or RRC-A event, a sufficient borating capability of the RCS to allow the transfer to the safe shutdown state.

The EBS consists of two separate and independent trains, each capable to inject the total amount of concentrated boric acid required for reaching cold shutdown from any steady state power operation.

One of the two EBS trains can be used for the periodic hydrostatic test of the RCS.

The EBS consists of two separate and independent trains, each capable to inject the total amount of concentrated boric acid required for reaching cold shutdown from any steady state power operation.

One of the two EBS trains can be used for the periodic hydrostatic test of the RCS.

3.7. Safety systems to cope with severe accidents

The design target of the EPR™ is to restrict off-site emergency response actions (population evacuation or relocation) to the nearby plant vicinity. Maintaining the integrity of the containment is therefore essential. The EPR™ thus includes both preventive measures and mitigating features to prevent base mat melt through and long term containment pressurization, to limit hydrogen deflagration, and radioactive releases to the environment.

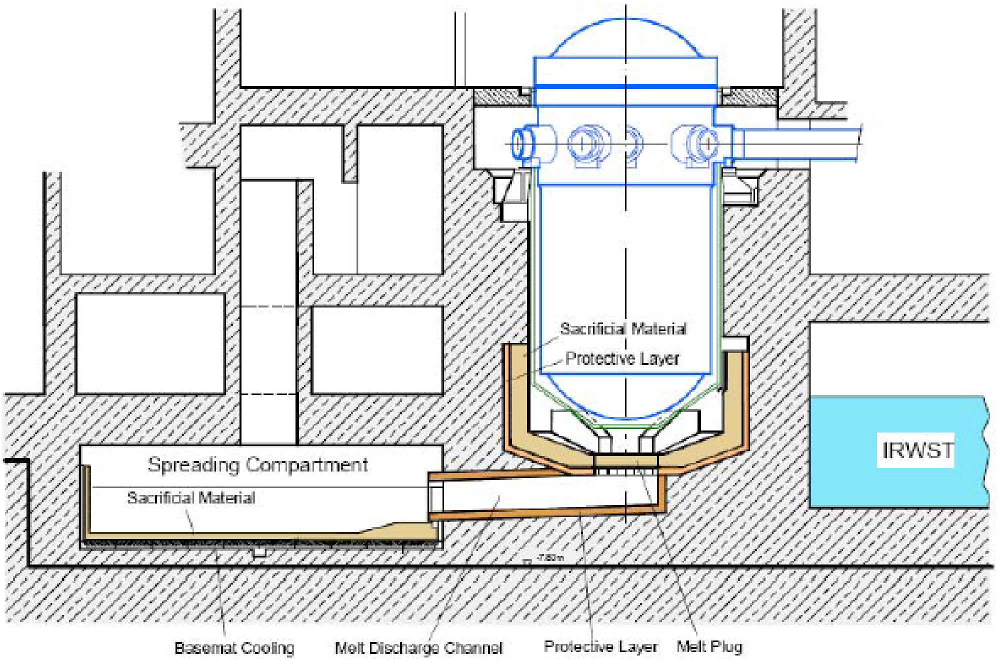


Figure 3.7-1. Core melt retention system

Prevention of:

- Molten core-concrete interaction, by spreading the corium in a spreading compartment provided with a protective layer and a special cooling device (figure 3.7-1).
- High pressure core melt situations, by ensuring a high reliability of the decay heat removal systems, complemented by two dedicated PZR relief valves. The depressurization eliminates the risk of high pressure failure of the RPV, of direct containment heating with potential early containment failure. The consequences of an instantaneous full cross-section break of the RPV at a pressure of about 2.0 MPa are nevertheless taken into account for the layout and support design;
- High hydrogen concentrations in the containment, by passive auto-catalytic H2 recombiners. The prevention of molten core-concrete interaction contributes to reducing the amount of hydrogen;
- Ex-vessel steam explosions, by minimizing the amount of water in the corium spreading area;

Limitation of radioactive releases:

- Corium cooled in the containment and fission products retained by water covering;
- Containment functions preserved by low leak rates, reliable containment isolation function, prevention of base mat melt-through, and ultimate pressure resistance to cope with energetic events;
- Pressure reduction inside the containment by dedicated heat removal;
- Collecting unavoidable containment leakages in the annulus atmosphere and release via the stack after filtration.

3.8. Provisions for safety under seismic conditions

General approach
The first requirement is to ensure the integrity of each of the 3 confinement barriers in case of earthquake.
The second requirement is to maintain the safety functions in case of DBE.
The last requirement is to ensure the possibility to bring the plant to safe shutdown conditions after a DBE.
Beyond the design basis of the standard plant, a seismic margin assessment is carried out for each site.

3.9. Probabilistic risk assessment

In the development of the EPR™ design, probabilistic studies of Level 1 and Level 2 PSA were performed to support and to optimize the design of the plant systems and processes. This practice helped achieve a well balanced design and process and provides added assurance that the overall plant design will comply with the general safety objectives.

3.9.1 Probabilistic targets

The mean value target of the core melt frequency (CMF) for the whole nuclear power plant, including all events and all reactor operating states, is less than 1.0E-05/RY.
The identification of initiating events and the grouping is based on IAEA approach. The associated initiating event frequencies are established using in particular operating experience of existing nuclear power plants (NPP) as well as using national and international data bases.

3.9.2 Internal and external hazards

For flooding and fire, a screening analysis is performed on building level, taking into account potentially affected safety related equipment as well as potential fire/flooding sources. The internal hazard analysis is completed for the PSA for operating license when specific information on component arrangement and cable routing is available.
External hazards are to a certain extent site-dependent. The boundary conditions are chosen in such a way that it should be possible to construct the EPR™ on most potential sites. The external hazard analysis is done for the PSA for operating license when the site specific data evaluations are available.

3.10. Emergency planning measures

The EPR™ has been designed to limit the consequences of a core melt accident as far as possible on the plant itself and its environment. Core melt retention and cooling features ensure that activity release to the environment in case of a core melt accident will remain well below internationally accepted limits for evacuation measures outside the plant.
Nevertheless, it is assumed that the Plant Operator will set up an emergency management team, which will give advices or direction to the operating shift in the control room in case of unforeseen severe event sequences. To support the emergency management team and the Plant Operator in such efforts a specific document “Operating Strategies for Severe Accident” (OSSA), dedicated to severe accident management will be produced.

---

[1]. Multinational Design Evaluation Program is a broad collaborative program between safety authorities to leverage their resources and knowledge for new reactor design review. It provides for the exchange of technical assessments of issues of common interest between regulators. Current members are: Canada, China, Finland, France, Japan, Rep. of Korea, Russian Federation, Rep. of South Africa, the United Kingdom and the United States.

[2]. They were presented during the IAEA advisory group meeting on technical feasibility and reliability of passive safety systems (November 21st to 24th, 1994 – Jülich, Germany).

Proliferation resistance

The EPR™ renders the diversion or undeclared production of nuclear materials or misuse of technology, by host state very difficult due to inherent technical impediments. On the one hand the declared inventory is not appealing and diversion of either fresh or spent fuel elements is made difficult by design. On the other hand undeclared production of weapon grade materials has a very significant cost and is easily detectable.

4.1. Intrinsic barriers that reduce the risk of diversion or misuse pathways

4.1.1. Fresh fuel

The EPR™ is designed for operation with fissile material that has a slender proliferation interest.

Fresh fuel assemblies make use of

- low enriched uranium fuel (LEU), the enrichment is no higher than 5% U^{235} thus far below "weapon-grade", or
- uranium-plutonium mixture known as Mixed Oxide fuel (MOX), the reactor grade plutonium coming from reprocessing of LWR fuel.

Diversion of LEU fuel assemblies for use as feed in enrichment devices and/or diversion of fresh MOX assemblies for processing to separate plutonium are made extremely difficult. A fuel assembly weights more than half a metric ton and requires a specific equipment to be lifted. Furthermore the fuel assemblies are handled and stored in the fuel building which is protected by a heavy shielding and has limited and controlled access.

4.1.2. Fuel under irradiation

When the reactor is operating the reactor pressure vessel is closed and the fuel is inaccessible. Refuelling operation with an open vessel and fuel transfers occur under transparent water shielding permitting direct visual observation for safeguards purposes.

4.1.3. Spent fuel handling and storage in fuel pool

LEU fuel which has gone through a normal operating life (e.g. 3 fuel cycles) will reach a high burnup (> 50 GWd/mt), so it is of limited interest for proliferation purpose: the U^{235} content is below 1% and the poor isotopic quality of the plutonium leads to high neutron emission rate, high heat emission and high level of radiation.

In the spent MOX fuel assembly, the remaining plutonium content has an even worst isotopic quality, and thus a further reduced attractiveness.

In both cases, the spent fuel is highly radioactive and would require a heavily shielded cask to be moved, therefore theft is unrealistic.

Fuel assembly design allows disassembly with a specific device implemented in the fuel pool. This may result in presence of individual fuel rods in the fuel building, however they may be easily safeguarded by surveillance (a fuel rod is more than 4 meters long) and by accounting of rods or group of rods.

4.1.4. Misuse of the reactor for producing weapon grade plutonium

The design of the reactor vessel internal structure does not make room for allowing irradiation of significant amount of specific uranium targets.

Producing plutonium for proliferation purpose would require restricting the fuel assembly irradiation to a very low burnup; otherwise the plutonium isotopic quality would be debased. This would be impractical for an industrial operator.

The normal operating fuel cycle is between 1 and 2 years at the end of which the reactor is shutdown and refuelled, the outage taking at least a week and a half. To discharge fuel assemblies at a very low burnup would require to shutdown the reactor within weeks of start-up: this will result in a large economic penalty due to the loss of power production during the frequent shutdown periods and the need to procure an unusual large number of new fuel assemblies; furthermore this type of operation would be obviously detected by safeguard inspectors.

4.2. Safeguard ability

The design of the plant facilitates the implementation of safeguard inspection controls and accounting measures that constitute extrinsic barriers enforcing the institutional agreements and policies.

Refuelling operation and associated fuel movements are conducted at a low frequency and take place in only two buildings that can easily be monitored. Their integrity is ensured by their structure, designed against external hazards. The few access points allow monitoring and surveillance of all passages.

The EPR™ design includes necessary safeguard measures virtually eliminating any risk of proliferation of fissile materials.

Safety and security (physical protection)

Security provisions are integrated in the design to deal with malevolent actions by protecting sensitive structures, systems and components and by allowing the implementation of security procedures during operation and maintenance activities.

The objectives are the integrity of nuclear materials and NPP safety features, to avoid the threat of radiation for the public or any theft of nuclear fuel.
The bases of the general requirements taken into account come from the European Utility Requirements (EUR).

Stipulations on physical protection are mostly classified information, and only some general principles are given here below.

The plant arrangement and the design of the buildings allow implementing different levels of security areas accessible only after passing access control points.

The plant area is surrounded with the site fence, including the gatehouse and the vehicle barrier. These security features are usually within the plant owner responsibility. Resistance of walls and closure (e.g. doors, grids) is required for the protected area as well as the surveillance of all passage. Within the protected area, the vital area contains all the systems and equipment important to plant safety or security and the storage of the nuclear material.

Boundaries between areas with different security level are structural barriers designed against unauthorized access. Specific features allow monitoring, surveillance and recording of all passages through the boundary between the different areas.

Physical protection of the vital areas against destructive acts from outside is typically based on:

- provision against external hazards, such as the physical separation of redundant systems and the implementation of the air plane crash protection structure (see § 1.2.3),
- design features implemented for coping with station blackout; they provide grace periods in case of destructive acts from outside, this time allowing to restore water inventories and/or to recover damaged plant equipment.
- security provision made to prevent and to detect incorrect inputs in the I&C systems and equipment.

Physical protection against unauthorized manipulations of an authorized person within the vital area takes benefit from the segregation of the redundant trains of the mechanical, electrical and I&C safety systems.

To ensure the plant security of any project, features addressing the same general principles as those considered in the reference design are implemented. For every project, the detailed definition and design of the security provisions have to be achieved independently of the other projects in order to ensure the confidentiality of the information.

Moreover, requirements for safeguarding and measures connected are usually ruled by national regulation and that may lead to specific alteration of the scope.

Description of turbine-generator systems

6.1. Turbine generator description

The saturated steam coming from the SGs via four steam lines is admitted into the turbine High Pressure (HP) casing through four valve chests, each of which is composed of one main steam stop valve and one control valve. The steam flow expands in the HP casing of the turbine, and then is routed through two moisture separator-reheater units where its moisture is removed before steam is reheated. After reheating, the steam flow passes through four reheater stop and intercept valves before being admitted into the (IP) Intermediate Pressure casing (N4) or directly to the three low-pressure (LP) casings (Konvoi), where it further expands.

When entering the LP casings, the steam flow is divided in each casing into two equal flows and expands down to the condenser.

The turbine of the EPR™ is a development based on the impulse type Arabelle turbine product line of the French 1500 MWe series (N4) or on the German Konvoi reaction type turbine. Both turbines are of the multistage tandem compound design, and according to manufacturer's design, consist:

- either of one combined HP/IP cylinder module (N4 type),
- or of a double flow HP turbine (Konvoi type),

and, in both designs, of three double-flow low-pressure cylinder modules.

The rotor of each module is supported by two bearings.

Extractions located in the HP, IP and LP sections supply the various feed water heaters.

The turbine rotor is directly coupled to the generator rotor. The three-phase synchronous generator is directly connected to the exciter.

The generator is a four-pole type, with a hydrogen-cooled rotor and a water- cooled stator. The rotor shaft is directly coupled to the LP turbine shaft.

Turbine trip is initiated if the integrity of any system or component important to turbine operation is endangered. No failure of rotating parts will impair the capability of the reactor to be shut down safely or of the RCS to be cooled down.

The turbine and its auxiliaries are manufactured, erected, tested and commissioned in accordance with the manufacturer's standard practices and in accordance with applicable codes to ensure high reliability of all systems and the mechanical integrity of the turbine generator set.

There is no radioactivity in this system during normal operating conditions. In the event of SG tube leakage, the small amount of radioactivity which may be present in the secondary system is detected by the main steam activity detectors, the SG blow down processing system, and the condenser evacuation system.

Shaft integrity of the turbine-generator is maintained under all normal operating modes, transient conditions, and worst-case failure conditions. The worst-case failure is the loss of one last stage blade. The integrity of the rotor train is maintained by an appropriate bearing, bearing casing, and bearing pedestal anchor bolt design. The mechanical design of these components is set by the dynamic excitation forces due to the loss of a single last stage blade. The forces originate either from the impulse of a single last stage blade loss at over speed or from the unbalance excitation (i.e., loss of last stage blade at over speed and subsequent shutdown of the turbine-generator).

6.2. Main Feed Water (figure 6.2-1)

After expansion in the low-pressure turbines, the steam goes to the condenser. Condensates which collect in the condenser hot well are pumped through four stages of LP feed water heating and delivered to the de-aerator by the condensate pumps. Feed water is pumped from the de-aerator through two stages of HP feed water heating and delivered to the SGs by three high-pressure electric motor-driven pumps, each with a capability of providing 33% of the rated flow. Depending on customer requirement, the main feed water pumps can operate at constant speed or be equipped with a variable speed controller.

The drains accumulating in the feed water heaters, reheaters, moisture separators, and in drains traps downstream of the third stage of feed water heating are cascaded back and subsequently pumped forward by a drain pump. Drains upstream of the third stage of feed water heating cascade back to the condenser.

During normal power operation, the feed water supply to the SGs is provided by the Main Feed Water System (MFWS). A dedicated system, the Start-up and Shutdown System (SSS) supplies the SGs during start-up and shutdown operation of the plant. It is actuated automatically in the event of a low level in the SGs following a reactor trip with a loss of the MFWS. The SSS actuation reduces the frequency of the EFWS actuation and increases feed water reliability.

Feed water is heated in two HP feed water heater stages by the turbine extraction steam system. The condensed steam is cascaded back by the heater drains system to the feed water tank. These systems are not required to operate during or after an accident. The system layout ensures that no malfunction of any component or piping of these systems will affect the safe operation of the plant or any system which is important to safety. Only the function of MFWS containment isolation is important to safety. Thus, the portion of the MFWS from the main feed water containment isolation valves and feed water piping system (from the isolation valve inlets to the SG main feed water inlet nozzles) is safety class.

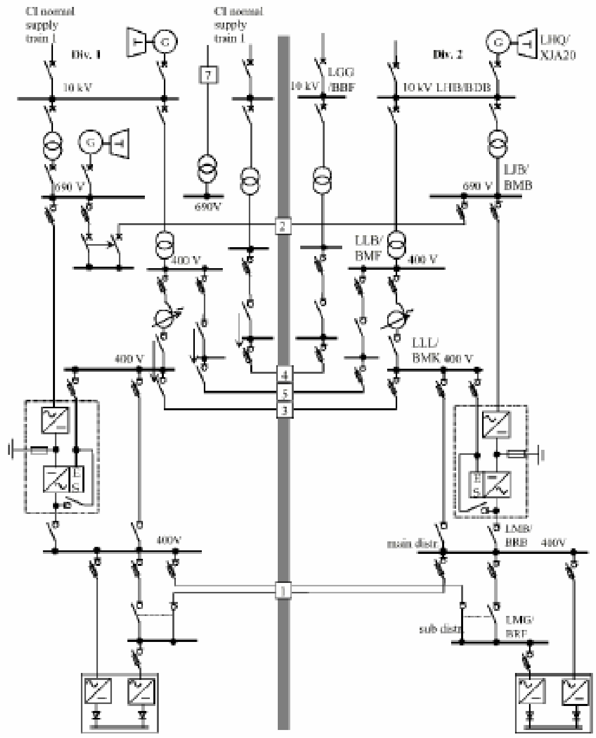
Depending on the site characteristics, the condenser tube material will be stainless steel for river sites or titanium for coastal sites.

The condenser is designed to withstand direct bypass 50% of the turbine rated steam flow to the condenser.

7.1. Electrical systems

7.1.1. Power supply systems (figure 7.1-1)

The Electrical Distribution System (EDS) is designed as a 4-train, 4-division system. Most non-safety related plant loads are powered from the Turbine Island 4-train Normal Power Supply System (NPSS) of the EDS while engineered safety system loads as well as a few non-safety related loads are powered from the Nuclear Island 4 division, Emergency Power Supply System (EPSS) of the EDS. The RCPs and a few non-safety related loads are powered from the Nuclear Island NPSS.



This diagram shows divisions 1 and 2. Divisions 3 and 4 are similar.

Figure 7.1-1. Electrical single line diagram

During plant on-line/power operation, electrical power is supplied from the main generator via the main step-up transformer to the main switchyard and the plant EDS via two auxiliary normal transformers. Each transformer powers two trains and two divisions and about half of the total plant auxiliary load.

During plant off-line/shutdown periods, power to the EDS is supplied from either the main switchyard via the two auxiliary normal transformers, or the standby switchyard transmission grid via a single auxiliary standby transformer. The transfer from ANT to AST is automatic or manual and is initiated if the defined electrical operating conditions are not met for the supply from generator and/or main grid.

The plant can accept a generator load rejection from 100 percent power or less without a reactor or turbine trip while stable operation continues. During such an event, the generator breakers (i.e., those that connect the main step-up transformer and auxiliary normal transformers to the main switchyard) will open, but the connection from the generator to the auxiliary normal transformers via the main step-up transformer remains online. Consequently, the plant can continue to autonomously operate, disconnected from the grid while powering all house loads (house load operation).

7.1.2. Safety related electrical systems

The safety system loads and some non-safety system loads are connected to the EPSS. The safety system loads are those necessary to shut down the reactor safely, maintain it in shutdown condition, remove the residual heat and stored heat and to prevent release of radioactive substances, under accidental conditions. A direct connection between the emergency and normal power supply allows a simple and safe separation from the normal supply in case of emergency power mode.

The EPSS is normally powered directly from the Turbine Island NPSS. However, in the event of a loss of off-site power or voltage and frequency outside the defined range, the EPSS is automatically disconnected from the NPSS while four Emergency Diesel Generators (EDGs) (one for each division) re-power the EPSS. The EDGs are housed in buildings separated from the rest of the plant and are protected against external and internal hazards. The start-up time of EDGs is in accordance with the requirements of the supplied processes. The autonomy requirement is equivalent to three days at full power.

In case of loss of both the off-site and on-site power supply and all EDGs, the loads necessary to safe shutdown of the plant are connected to the Station Blackout (SBO) power supply. Two additional diesel generators, diversified in regards to the EDGs, provide an alternate AC sources for coping with postulated SBO events (i.e., loss of both the off-site power supply and all four of the onsite EDGs). The autonomy requirement is equivalent to 24 hours at full power.

Until the start of the SBO event, the two-hour rated EPSS batteries supply DC power for required loads including inverters and their critical loads (e.g I&C power supply and control voltage). Early in the two-hour period, the SBO diesels are started manually from the control room.

7.1.2.1 Uninterrupted power supply and distribution system

The Uninterrupted Power Supply (UPS) is part of the EPSS and provides continuous and reliable low-voltage DC and AC electrical power for I&C loads such as the plant protection systems and to other loads. EPSS has four UPS trains (one per division) consisting of batteries, battery chargers, inverters, and associated distribution panels.

Each battery charger is sized for the capability to recharge its fully discharged battery while concurrently supplying its largest assigned load to meet demand of various essential steady-state loads. With a full charge and the charger not operating, each battery is capable of supplying power under the worst-case design basis event loads for two hours at full load.

In addition to the four two-hour rated UPS systems, two supplemental 12-hour rated UPS systems are provided, one each for divisions 1 and 4. These supplemental UPS systems are provided for severe accident mitigation and increase the coping time for restoration of AC power.

7.2. Overall I&C architecture

The I&C architecture design is derived from the safety and design philosophy of the EPR™. Particular attention is paid to the Defence in Depth approach to ensure sufficient independence between the I&C systems in direct line with the Defence in Depth criteria. The following main measures are considered:

- The redundant trains are installed in divisions with physical separation and with minimum interconnections.
- Interconnections are energetically isolated against over-voltages from a disturbed division or train (e.g. by means of fibre optics).
- Erroneous signals from a disturbed train are prevented from affecting the other trains by means of majority voting. Necessary safety actions must be performed from the undisturbed trains independent of the state of a disturbed train or division.
- Interlocking and/or unidirectional communication means are preventing failures to be propagated to higher classified systems.
- Appropriate measures are provided to cope with common cause failures (CCF) in order to meet the overall probabilistic design targets

Potential Common Cause Failures (CCF) of I&C technologies are also considered in the design of the I&C architecture, which requires the selection of two diverse I&C platforms to design the I&C architecture.

Figure 7.2-1 provides an overview of the standard EPR™ I&C architecture. The different I&C systems are detailed in the following paragraphs.

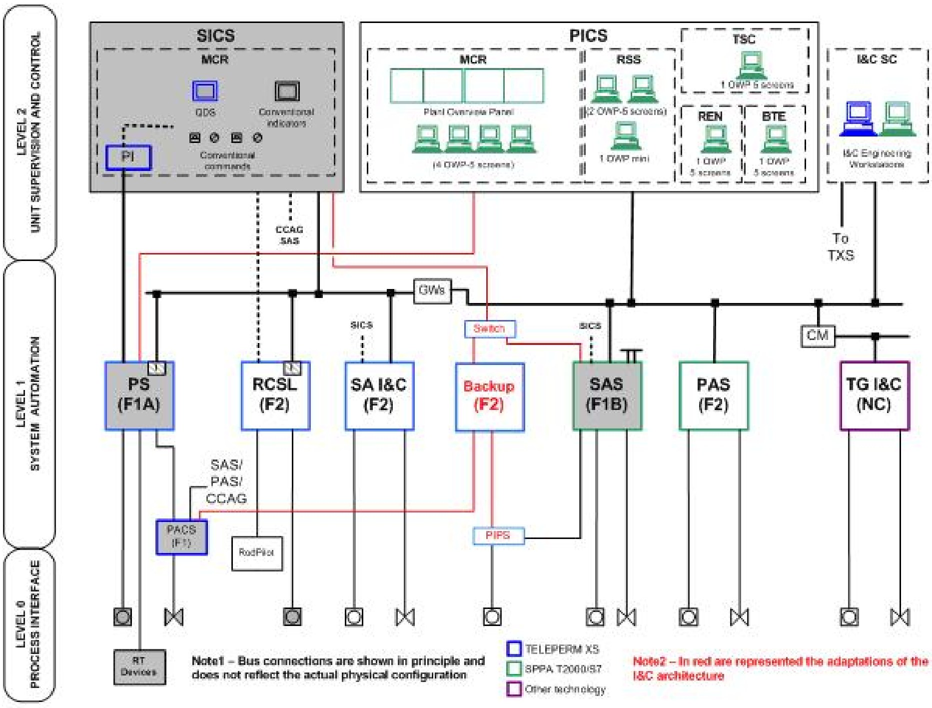


Figure 7.2-1. Overview of the standard EPR ™ I& C architecture.

7.2.1. Level 2: Human-machine interface facilities and control rooms

Process control and supervision is performed centrally from the Main Control Room (MCR). It provides all process information, controls and communication means needed to monitor and control the state of the plant during all plant states, including commissioning, maintenance, refuelling, and operation at power and accident conditions.

The Process Information and Control System (PICS) is a screen-based system with an overview panel. It includes four identical operator work stations (each with 5 screens) used for process control in all plant conditions via operational I&C:

- Two main operator workstations (OWS) dedicated to the operators to control the plant,
- One workstation in observation mode for the shift supervisor and/or the safety engineer.
- A fourth operator workstation for back up purpose if one OWS has failed.

The Main Control Room is also equipped with communication means at the OWP levels and space for administration work.

A Plant Overview panel (POP) consisting of 4 large PICS screens giving an overview on status and main parameters of the plant is visible from all work places and will be used for the co-ordination among the operators.

The complete failure of the PICS is considered in the EPR™ design, so that a backup Safety Information and Control System (SICS) is used in the event of PICS failure. SICS consists in panels comprising conventional controls and indications and Qualified display System (QDS) screens, enabling the operator to maintain the plant in a steady state up to 4 hours, and to safe shutdown (hot or cold) of the plant and to perform post-accident operation in all situations. The area constitutes a safety-relevant human-machine interface, and the related equipment are qualified accordingly.

Further functions that are ensured from the MCR or from adjacent rooms are security surveillance, fire protection monitoring, radiation monitoring, management of maintenance and periodic testing, external and internal communication, access to documentation and to recorded information.

Should the MCR become unavailable or inaccessible, the operators would supervise and control the plant from a remote shutdown station (RSS), designed for transferring and maintaining the plant in safe shutdown conditions The RSS is equipped with:

- A manual actuated switching device disconnecting all HMI equipment in the MCR from the level 1 I&C systems,
- A manual command to trip the reactor,
- Two PICS operator work stations of the same type and functionality as in the MCR, from which the operators can transfer the plant to safe shutdown state and monitor the complete plant,
- Internal and external communication means.

The unit can be controlled in normal operation from the RSS in the case of loss of the external power supply.

7.2.2. Level 1: Automation Systems

System automation functions are implemented in level 1 I&C systems. An overview of the role of each main I&C systems is provided below.

7.2.2.1 Protection System (PS).

The PS main role is to monitor the safety process parameters in all Design Basis Conditions and to actuate F1A protection and safety functions in case of Postulated Initiating Event automatic, including related support systems functions. The PS also provides information on safety parameters for the SICS and PICS.

The PS is a four-fold redundant digital I&C system based on TELEPERM XS and is distributed over the four NI divisions. It is F1A classified.

7.2.2.2 Priority and Actuator Control System (PACS).

The control and monitoring of each individual actuator is ensured by Priority and Actuator Control (PAC) modules, which belong to level 1. All these modules together constitute the Priority and Actuator Control System (PACS). The PACS also manages the priority between commands from I&C functions of different I&C systems actuating on the same actuators such as the PS and the SAS. It is a basic principle that the safety-oriented signal having the highest safety class has priority over the other signals.

The PACS modules belong to the TELEPERM XS platform. Priority management of the PACS is F1A. Each PAC module controls one actuator.

7.2.2.3 Reactor Control Surveillance and Limitation System (RCSL).

The RCSL is mainly devoted to F2 and NC I&C functions which control and monitor the operation of the reactor. This comprises in particular the core control functions for the rods control and the automatic LCO and limitations functions acting on the rods.

The RCSL is a digital I&C system based on TELEPERM XS. It is F2 classified.

7.2.2.4 Severe Accident Instrumentation and Control System (SA I&C).

The SA I&C is devoted to I&C functions which control and monitor the plant subsequently to a severe accident (of category RRC-B).

The SA I&C is a digital I&C system based on TELEPERM XS, distributed in the NI divisions 1 and 4. It is F2 classified.

7.2.2.5 Process Automation System (PAS).

The main role of the PAS is the monitoring and automation of the plant in all normal operation conditions. Moreover, the PAS performs monitoring and control sub-functions (classified F2 and NC) for risk reduction functions as diversified measures to cope with the CCF of the Protection System.
The PAS is a digital system typically based on SPPA-T2000 and is distributed over the four NI divisions and in the Conventional Island

7.2.2.6 Safety Automation System (SAS).

The main role of the SAS is to handle I&C functions for post-accident management needed to transfer the plant from controlled to safe shutdown state (F1B) subsequently to an incident or accident.

The SAS is a digital I&C system based on SPPA-T2000 and distributed in the four NI divisions. It is F1B classified.

7.2.2.7 Backup I&C system.

The EPR™ reactor is designed to withstand the CCF of digital I&C systems. The Backup I&C system main role is to handle a minimum set I&C functions for post accident management to avoid core melt in case of a Design Basis Accident cumulated with a CCF of the SPPA-T2000 I&C platform.

The Backup I&C system is based on TELEPERM XS and distributed technology. It is F2 classified.

7.2.3. Technologies

I&C automation and Human Machine Interface systems are based on utilisation of digital technology.

The I&C architecture is implemented using the two following digital I&C platforms:

- **TELEPERM XS,** developed and maintained by AREVA,
- **SPPA-T2000,** developed and maintained by Siemens.

For the purpose of reducing the risk of Common Cause Failure (CCF) impacting different lines of defence, functionally diverse I&C functions are implemented in different I&C systems, structures and components, taking benefit of the two diverse technologies SPPA-T2000 and TELEPERM XS.

The design features of the system software, libraries of the application software, and main hardware modules for the system platforms TELEPERM XS and SPPA-T2000 are different. Therefore, credit is taken from this equipment and software diversity for the design of the I&C architecture.

Spent fuel and waste management

8.1. Provisions for low consumption of non-renewable resources

The EPR™ incorporates a range of design features that provide positive benefits in terms of use of resources at the front end of the fuel cycle and waste management at the end of the fuel cycle. These design features centre on:

- Increased fuel burn up,
- Longer fuel cycles,
- Reactor core design and fuel loading that provide improved neutron economy,
- Some self burning of plutonium in the fuel,
- Improved primary to secondary circuit heat transfer.

With respect to the front end of the fuel cycle (mining and refining uranium) these design and operational features of the EPR™ are expected to provide savings in the use of natural uranium of 17% over and above those of current French and German PWR reactor designs for the same electrical output.

With respect to the back end of the fuel cycle, discharged fuel (assuming standard UO_2) contains less plutonium and, in the event of reprocessing, gives rise to less fuel cladding and fission products per unit energy produced compared with current French 4-loop PWR plants.

8.2. Provisions for minimum generation of wastes at the source

Like any PWR, the EPR™ will produce a range of liquid, gaseous and solid radioactive wastes. The design and operation of the plant will draw on 20 years experience to ensure these are minimised at source and those that do arise are treated in accordance with best practice, taking account of worker doses, costs and environmental impacts.

The design and operational features are expected to minimise the impacts relative to those of current 4-loop PWR plants.

8.3. Provisions for acceptable or reduced dose limits

One of the main EPR™ design objective was to divide the personnel irradiation doses by a factor of two in comparison with operating PWRs. An annual global dose lesser than 0.4 man.Sv/year was targeted by minimising activation products and by taking benefit from operating experience feedback in designing the structures and systems.

The EPR™ incorporates design improvements aimed at reducing the dose rates from a number of sources. With regard specifically to the radiation levels associated with activated cobalt these improvements include:

- Use of alloy 690 (target cobalt content below 0.018%) instead of alloy 600 (target cobalt content below 0.050%) for the steam generator tubes which minimises the quantity of cobalt in the corrosion products circulating in the primary system,
- Reduction of wear through design modifications, and by replacing materials with a high cobalt content level by alloys without cobalt, used in the past as a hardening agent in Stellite™ alloys. Most of the Stellite™ was eliminated from the EPR™ primary system components and some connecting systems,
- Limiting the amount of cobalt in steel to 6 ppm for steel components transporting liquid subject to irradiation,
- Reduction in the source terms of cobalt 58 and 60, due to optimised primary circuit chemistry control (that also ensures integrity of fuel cladding),
- Controlling levels of cobalt impurities in all primary circuit components,

In addition, the EPR™ piping carrying radioactive fluids is designed for optimum flow conditions, thus avoiding the formation of radioactive deposits. Pipes are constructed so as to preclude, as far as possible, traps and pockets where radioactive materials could accumulate.

The EPR™ is designed for low releases during normal operation. The radiological targets in terms of releases are:

- Liquid w/o tritium : 0.1 T Bq/y
- Gases : 800 T Bq/y

For Design Basis Accidents the radiological targets in terms of doses are:

- Effective dose < 10 mSv
- Organ dose < 100 mSv

8.4. Provisions for low spent nuclear fuel and waste management costs

The EPR™ offers significant advances for sustainable development compared to LWRs in operation today in France and Germany:

- A better utilisation of uranium resources (17% saving on Uranium consumption per MWh),
- 15% reduction on long-live actinides generation per MWh,
- 14% gain on the "electricity generation" versus "thermal release" ratio (compared to 1,000 MWe class reactors),
- A greater flexibility for MOX fuel recycling.
- Ability to limit or reduce the national plutonium inventory, if required, and to better "burn" minor actinides.
- Reduction of ultimate wastes to run after decommissioning, thanks to the 60 years technical life time, without major replacement during, at least, the first 40 years.

Plant layout (figure 9-1).

The plant layout is governed by a number of principles derived from the experience gained through the construction and operation of the French and German nuclear power programmes with an installed capacity of more than 100 000 MW. The plant layout follows, as far as reasonably possible, general rules and recommendations related to personnel circulation, maintenance, In Service Inspection, equipment handling, exchangeability of components, radiation protection, fire protection, industrial safety, routing.

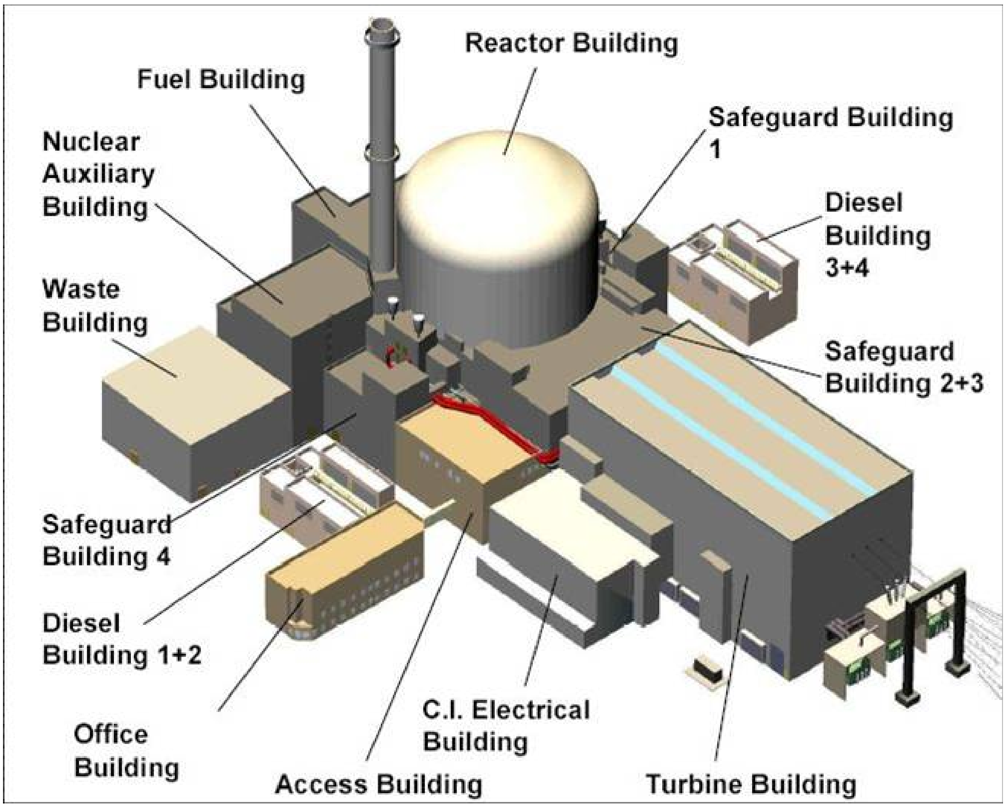


Figure 9-1. EPR™ 3D general view

The plant layout is significantly influenced by the severe accident mitigation requirements and the radiation protection principles.

Design requirements

The plant is designed to withstand the impacts of internal and external events. With respect to earthquake and explosion pressure waves, the buildings and structures are strengthened so that collapsing structures would not jeopardize the function of safety-grade equipment, and that the equipment themselves withstand the dynamic effects inside the buildings.

The loads from internal events (e.g. fire loads, missile loads, jet impingement loads, flooding effects) are included in the design.

Protection against external and internal hazards is ensured by divisional separation of safety-grade systems and physical protection of the containment enclosing the reactor coolant pressure boundary. The risk of inadmissible releases or common-mode failures of safety-grade system is thus consistent with the deterministic design basis and the probabilistic targets of the EPR™.

9.1. Buildings and structures (figure 9.1-1)

The reactor building (RB) and the surrounding safety and fuel buildings (SB, FB) are placed on a common raft. Most of the safety-grade systems, designed with a four-fold redundancy, are located in four independent divisions with complete physical separation. The related electrical systems as well as the instrumentation and control systems are also allocated to these divisions, on a higher building level.

The other buildings, such as the access building and the nuclear auxiliary building, are located in close contact with the SB-4 and FB. The turbine building and the associated conventional electrical building are separated from the NI buildings. The RB is located in the projection of the turbine generator shaft.

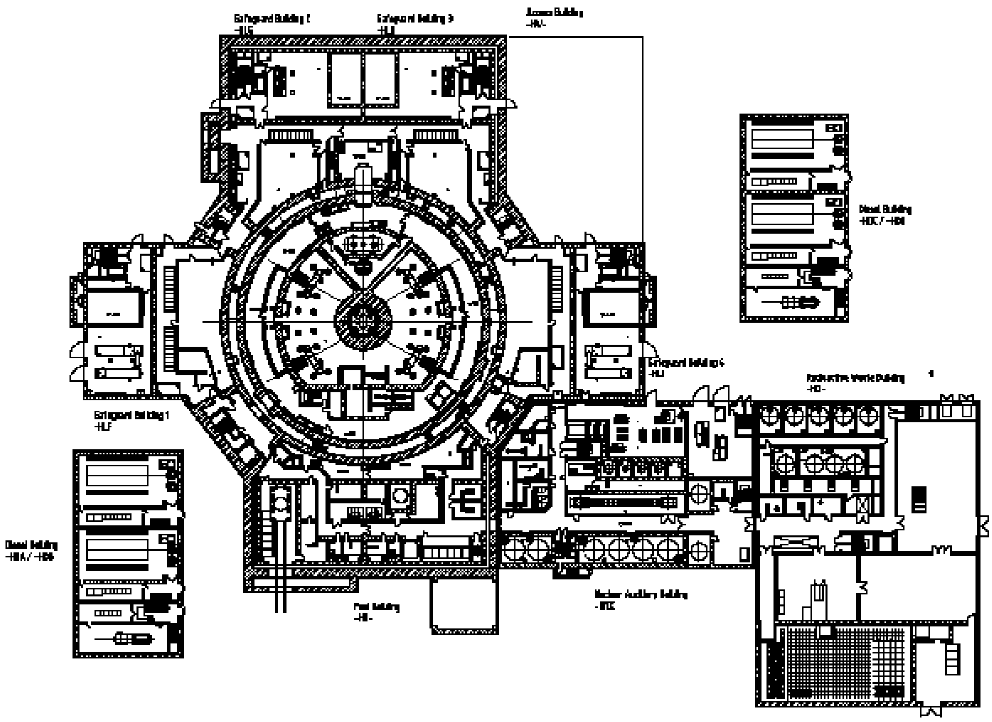


Figure 9.1-1. EPR™ NI buildings – plan view at ± 0.00m

The RB and FB are classified as hot zones. Within the SBs, the safety injection system part is arranged in the inner areas, which are classified as hot zones, whereas the component cooling and emergency feed water systems are installed in the outer areas, which are classified as cold zones.

The primary system is arranged symmetrically in the RB. Concrete walls are provided between the loops and between the hot and cold legs of each loop to provide protection against consequential failures. The pressurizer is located in a separate compartment. A concrete wall around the entire primary system protects the containment from missiles and reduces the radiation from the primary system to the surroundings.

A water pool for storage of the upper core internals during refuelling, and for the entire core internals during inspection, is provided inside the containment for radiation protection.

9.2. Containment (figure 9.1-2)

The limitation of radiological consequences to the environment even under severe accident leads to more constraining design conditions in comparison with existing PWRs, such as the design pressure (0.55 MPa).

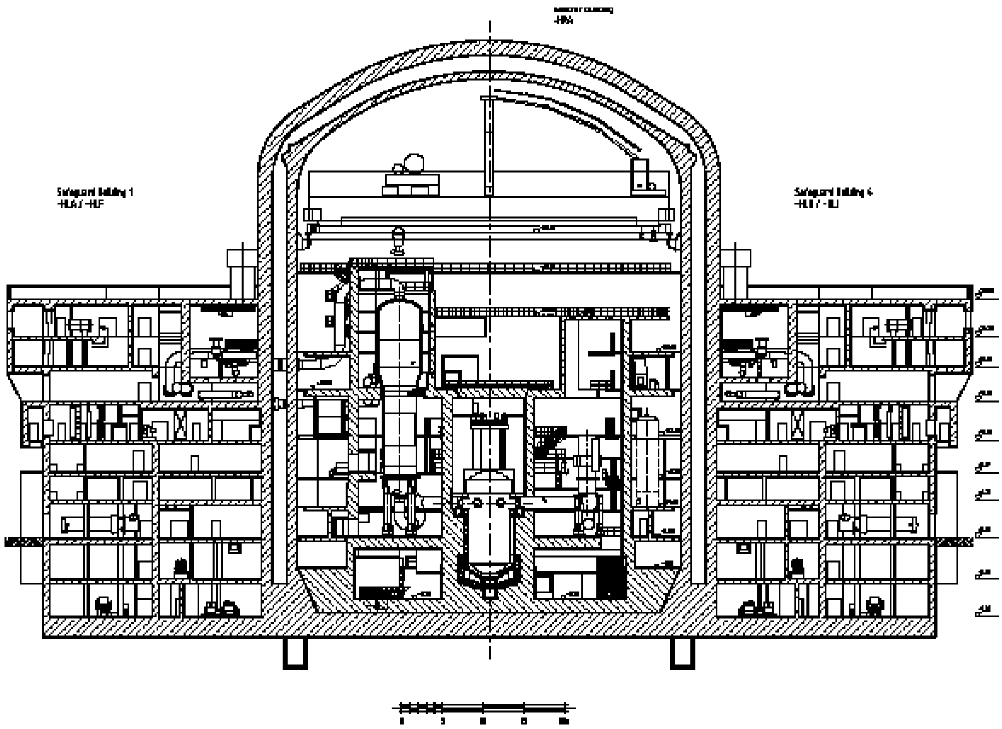


Figure 9.1-2. Vertical cross section of the EPR™ reactor and safety buildings

Therefore the EPR™ has a double concrete containment, consisting in an inner pre-stressed concrete containment with an integral steel liner and an outer reinforced concrete building. The leak-tightness requirement for the inner containment is less than 0.3 % volume per day.

To ensure overall containment leak-tightness, systems for isolation and retention and control of leakages are provided. Leakages through the inner containment are captured by the annulus air extraction system. Personnel access or equipment introduction into the containment are done through permanently closed hatches or air locks with double sealing on both sides.

This principle also applies to the penetrations of the HVAC systems. Fluid systems penetrating the containment are provided with double isolation valves, inside and outside the containment.

The pre-stressed concrete building and the steel liner ensure the capability to perform an integral leak test in air at design pressure.

The structural integrity of the containment is ensured by the thermal inertia of the internal concrete structures, by the safety injection system and the containment heat removal system (CHRS). Its prime function is to limit the pressure increase inside the containment below the design pressure, and to reduce this pressure afterwards. The CHRS heat exchangers and associated active components are located in dedicated compartments of the SB-1 and SB-4. Furthermore, the base mat beneath the spreading area is protected against elevated temperatures resulting from a possible core melt by protective layers and by a dedicated cooling system fed by the CHRS.

The reactor pit bottom is connected to the spreading area, which is designed to collect the core debris and separate it from the IRWST to avoid steam explosion. In a later stage of the accident, a passive means, dedicated melting plugs provide water flow to cool the molten core material. The generated steam is condensed by the CHRS.

A high temperature-resistant protective layer on the reactor pit floor and the spreading area prevents interaction between concrete and the molten core material.

Accumulation of hydrogen, possibly produced during certain accidents, is controlled by passive autocatalytic recombiners.

10.1. Plant Operation

In designing the core of the EPR™, special attention was paid to the flexibility with respect to possible reload enrichments, fuel types, core designs and discharge burn-up.

- Fuel cycle length of 18 months was taken as design basis; additionally following design conditions were assumed :

1. 12 and 24 months fuel cycle length,
2. Full low leakage loading pattern,

- Maximum enrichment of 5% U^{235},
- Average discharge burn-up consistent with 5% w/o U^{235} enrichment : 55 to 65 GWd/mtU,
- Stretch out operation can be performed for up to 70 EFPD after the natural end of the fuel cycle,
- Maneuvering capability must be ensured for the different type of fuel loading,
- Capability for Plutonium recycling (MOX assemblies).

10.1.1 Operating flexibility for load following, ramp up/down

The load following capabilities of the EPR™ are:

- Usual load follow : power level variations between 60% and 100% NP

  Return to 100 % NP possible at 5%/min during 80 % of the fuel cycle

- Unusual load follow : low power level between 25% and 60% NP

  Return to 100 % NP possible at 2.5%/min during 80 % of the fuel cycle

- Extended operation at intermediate power level :

1. For less than 2 days of operation at intermediate power level, no additional restriction on load flexibility,
2. For more than 2 days, additional constraints for returning to full power are accepted,

- EPR™ operating at intermediate power level can contribute to the spinning reserve by its capability of rapid return to full power:

1. Step of 10 % NP followed by a ramp at 5% NP/min,
2. Ramp at 10% NP/min.

10.2. Reliability

The detection of events which require initiating a reactor trip (RT) is four times redundant. Then threshold results are combined in the four divisions in 2/4 voting logics with degradation principles aiming at initiating the reactor trip if three protection channels are unavailable (tests or failure).
When one protection channel is in test, functions are degraded in 2/3 of the remaining channels.

Moreover, the reactor trip initiation is de-energized (trip initiation orientation in case of loss of power supply) and the main trip breaker + contactors are combined in such a way that even in case of test + single failure, RT can still be initiated.

10.3. Availability Targets

The EPR™ plant is designed to follow the availability requirements as provided in the EUR (annual Design Availability Factor - DAF- greater than 90% for a fuel cycle length equal or longer than 12 months over a 20-year representative period including all kinds of Planned Outages, including 10-year ISI outages). An actual availability factor of not less than 92% is expected during the entire service life of the plant, obtained through long fuel cycles up to two years, short refueling outages thanks to design provisions for extended reactor building accessibility, and in-service maintenance thanks to the four train concept.

In order to meet the availability objective of DAF indicated above, the following performances related to Planned Outages and Unplanned Outages are expected:

- Planned Outages (breaker-to-breaker and as planned)

1. Less than 20 days for refueling and regular maintenance Outages (NRO) (breaker to breaker) (EUR requirement),
2. 40 days for In Service Inspections Outages (ISIO) (EUR requirement),
3. Less than 14 days for a refueling-only-outage (ROO) with fuel reshuffling (EUR requirement).

- *Unplanned Outages*

1. < 1 Unplanned Automatic Scram per 7000 hours critical (EUR requirement),
2. < 5 days/year (i.e. <1.4%) of Forced unavailability (or forced outage) (EUR requirement).

10.3.1 Outage extension for special works

The Special Works that cannot be planned at the design stage but must be inevitably performed (such as major repairs or replacements of large components) should not exceed 150 days over 20 year representative period (EUR requirement).

10.4. Provision for design simplification, reduced capital and construction costs

General design objectives of the EPR™ were selected according to the economy approach consisting in optimizing the generation cost, leading to:

- Raising the power level to 1600/1650 MWe;
- Increasing the steam pressure, thus allowing to maximise the steam cycle efficiency;
- Improving the fuel utilization, thus reducing the fuel cycle cost;
- Simplifying the maintenance operations;
- Shortening the refuelling outages;
- Reducing the personnel irradiation doses by a factor of 2;
- Extending Plant life time duration to 60 years;
- Increasing the capacity factor to beyond 91 %, according to fuel cycle length.

An assessment of expected benefits versus cost impact on main RCS components was performed in order to avoid over design. Examples of potential improvements assessed are:

- large free volume between the top of the active core and the level of primary loops for increased margins with respect to core uncover in case of small break LOCAs,
- large pressurizer for smooth plant response in case of DBA,
- large secondary side steam generator volume.

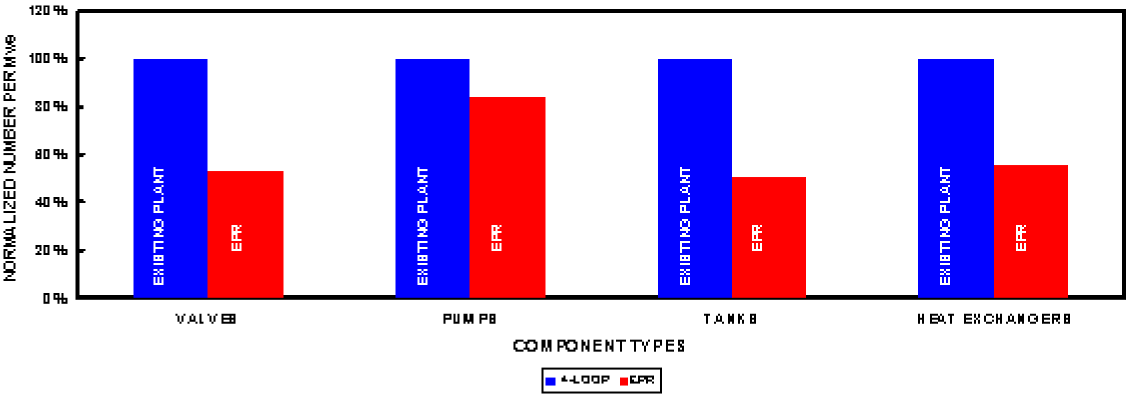
The following table illustrates the cost advantages brought in the primary components design by increasing the plant energy produced (by increasing power output, plant life time, and availability) and by reducing the fuel and O&M costs:

Component	**Reduces**	**Increases :**	**Reduces**	**Improves**

	ISI time, collective dose	- power - life time - availability	costs: - O&M - Fuel cycle	safety
Reactor Pressure Vessel				
Main flange/ nozzle shell in one piece	**X**			
Suppression of bottom mounted in-core penetrations	**X**			**LOCA**
RPV internals				
Neutron reflector	**X**	**Life time**	**Fuel CC**	
Steam Generator				
Increased heat exchange surface		**Power**		
Main feed flow distribution		**Life time**		
Increased steam drum volume		**Availability**		**SGTR**
3 features to reduce thermal stratification in main feed water nozzle		**Life time**	**O&M C**	
I-tubes on EFW distribution ring to reduce thermal fatigue and thermal shocks when actuating EFWS		**Life time**	**O&M C**	
High permeability trefoil tube support plates to prevent clogging		**Life time**	**O&M C**	
Moister separators in stainless steel to prevent erosion/corrosion		**Life time**	**O&M C**	
Channel head partition plate in I-690		**Life time**	**O&M C**	
Reduced number of moisture separators (53 instead of 130)	**X**		**O&M C**	
Addition of a cylindrical part in the channel head under the tube sheet to improve access for control of peripheral tubes	**X**		**O&M C**	
Pressurizer				

Increased volume (75 m^3) to smoothing operating transients		**Availability**		**X**
Safety valves welded directly on nozzles on the upper dome	**X**		**O&M C**	
3 spray lines connected to nozzles located well beneath the safety valves level	**X**		**O&M C**	
Shielding floor between safety valves and spray lines levels	**X**		**O&M C**	
Lower support skirt replaced by 3 vertical supports welded on the cylindrical part providing a complete free access to the lower dome	**X**		**O&M C**	
Connecting flanges allowing quick replacement of the heaters	**X**		**O&M C**	
Reactor Coolant Pump				
Hydrostatic pump bearing and shaft/impeller assembly (Hirth type) allowing a very good pump vibratory behaviour		**Life time** **Availability**		
Addition of a stand still system				**X SBO**
Control Rod Drive Mechanism				
3 features allowing absence of need of any forced air cooling	**X**		**O&M C**	
Double gasket flange to fasten the latch housing on the RPV closure head adaptor	**X**		**O&M C**	
Main Coolant Lines				
Primary piping made from forged austenitic stainless steel; large nozzles (DN>150) machined out of the pipe forgings	**X**			**LBB**
Small nozzles and bosses are set-on welded	**X**			
Surge line piping and small welded nozzles made from forged austenitic stainless steel	**X**			**LBB**

An EPR™ development objective was aimed at simplifying the design of many systems. For example, the number of valves, pumps, tanks and heat exchangers is significantly reduced in comparison with present 4-loop French PWRs, as shown on the figure below, considering on the following systems : Reactor Coolant, pressurizer spray, Reactor Coolant Pump seal and leak off, Safety Injection/Residual Heat Removal, Chemical and Volume Control including Boration and Demineralized Water, Spent Fuel Pit Cooling, Component Cooling Water, Main Feed Water, Auxiliary Feed Water, Emergency Feed Water and Main Steam.



10.5. Construction schedule

The overall construction schedule of a Next-of-a-Kind unit in the series depends largely on site conditions, industrial organisation and policies, and local working conditions. Therefore figures are valid only for the specific project to which they are related. A typical construction time schedule is shown below.

10.6. Provision for low fuel reload costs

The EPR™ large core allows long fuel cycles length at equilibrium cycle of up to 2 years without exceeding a reload batch size of 50% of the core. Increasing the cycle length means a reduction of the batch burn-up.

Longer cycles, although not attractive from the sole fuel cycle cost point of view, could however be beneficial when considering the increased overall availability and its impact on kWh generation costs.

Actual cycle length will depend upon specific fuel assembly design parameters selected by each utility for the fuel reloads, as well as maximum average fuel assembly discharge burn-up authorized by the safety authority of each country.

Development status of technologies relevant to the NPP

The EPR™ evolutionary design is based on experience from the operation of PWR worldwide, primarily those incorporating the most recent technologies: the French N4 and the German KONVOY reactors, both being currently in operation. Many systems and components are similar to those of these reactors, thus forming a proven foundation for the design.

Extensive R&D work programs were undertaken mostly for selecting and justifying design aspects where changes have been made with respect to earlier plants, in order:

- to confirm and/or to provide additional information without necessarily significantly impacting current design choices (e.g. control rod drive),
- to improve and to optimize the design (e.g. core inlet flow distribution device),
- to guide choices of the key new design options and to validate them (e.g. behaviour of corium outside the reactor pressure vessel).

In addition R&D actions provided information supporting further improvements of design tools and/or of their qualification where needed.

Specific R&D not related to severe accidents was, in general, limited to qualification, adaptation or improvement of existing equipment. The main R&D topics covered the following areas:

- Hydraulic behaviour of the internal elements of the reactor pressure vessel (cf. § 2.4.1):

Design of the flow distribution device in the lower part of the reactor pressure vessel was tested on a mock-up for qualification of flow mixing, velocity field and pressure distribution

Another mock up allowed validating the hydraulic behaviour of the upper internals.

These tests were also used for the qualification of computational fluid dynamics models and methods.

- Heavy reflector (cf. § 2.4.2): Hydraulic and cooling characteristics and validation of the neutronic calculation.
- Control rods mechanism and drive line: hydraulic and vibration, material wear and fatigue, mechanical behaviour, including rod drop time.
- Pressurizer relief valves: Prototype testing
- Additional validation of the CATHARE code used for the analyses of the safety injection system performances in specific accidental scenarios (loss of primary coolant, steam generator pipe ruptures)

Consideration of mitigation of severe accident consequences from the outset the EPR™ design led to develop and to implement new features in comparison with existing plants (c.f. § 3.7).

The design approach was primarily based on the general R&D devoted to LWR severe accidents; this R&D work was in progress worldwide and was not specific to the EPR™.

However, specific programs were deemed necessary for optimizing and justifying some design features of the EPR™. The main areas of interest were the development and validation of calculation codes, including benchmark and validation tests at different scales with simulated and actual materials to identify key phenomena. The following topics were addressed:

- Performance of Reactor coolant system depressurization valves,
- Design of the reactor vessel supporting system and the reactor vessel cavity
- Behaviour of the Reactor coolant system in core melt situations
- Stabilization of core meltdown,

- H_2 build-up mitigation (production, distribution, combustion, recombiners qualification),
- Removal of heat from the containment,
- Methods for limitation of radioactive releases,
- Containment wall and liner
- Internal structure of the containment and its liner

Status:

The R&D work was performed either by the designer (e.g. in the AREVA Technical Centres in France and in Germany), or by their main research centre which are partners of the designer in France (CEA) and in Germany. Results of some R&D works initiated by international cooperation (such as those performed under OECD or European Union sponsorship) were also used.

Today, the EPR™ concept has reached a satisfying level of validation as reflected in the construction permit delivered in three countries: the design is completed, plants are under construction and their licensing is well engaged.

No significant additional R&D work is needed for supporting the EPR™ deployment worldwide.

References

[1]. **Safety of Nuclear Power Plants: Design Safety Requirements,** IAEA Safety Standards Series No. NS-R-1; October, 2000.

[2]. **INSAG:** International Nuclear Safety Advisory Group of the IAEA

Technical data

General plant data

Reactor thermal output	4590 MWth
Power plant output, gross	1770 MWe
Power plant output, net	1650 MWe
Power plant efficiency, net	36 %
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target >	92 %
Seismic design, SSE	0.25
Primary coolant material	Light Water
Secondary coolant material	Light Water

Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Indirect

Safety goals

Core damage frequency <	10E-6 /Reactor-Year
Large early release frequency <	10E-7 /Reactor-Year
Occupational radiation exposure <	0.35 Person-Sv/RY

Nuclear steam supply system

Steam flow rate at nominal conditions	2604 Kg/s
Steam pressure	7.72 MPa(a)
Feedwater flow rate at nominal conditions	2630 Kg/s
Feedwater temperature	230 °C

Reactor coolant system

Primary coolant flow rate	33978 Kg/s
Reactor operating pressure	15.5 MPa(a)
Core coolant inlet temperature	295.2 °C
Core coolant outlet temperature	330 °C

Reactor core

Active core height	4.2 m
Average linear heat rate	16.67 KW/m
Fuel material	UO_2 and MOX
Outer diameter of fuel rods	9.5 mm
Rod array of a fuel assembly	17x17
Number of fuel assemblies	241
Enrichment of reload fuel at equilibrium core	4.95 Weight %
Fuel cycle length	24 Months
Burnable absorber (strategy/material)	Gd_2O_3
Control rod absorber material	Hybrid (AIC/B_4C)
Soluble neutron absorber	H_3BO_3

Reactor pressure vessel

Inner diameter of cylindrical shell	4870 mm
Wall thickness of cylindrical shell	250 mm
Design pressure	17.6 MPa(a)
Design temperature	351 °C
Base material	16MND5
Total height, inside	13083 mm
Transport weight	520 t

Steam generator or Heat Exchanger

Type	U tubes with axial economizer
Number	4
Total tube outside surface area	7960 m^2
Number of heat exchanger tubes	5980
Tube outside diameter	19 mm
Tube material	Inconel 690
Transport weight	550 t

Reactor coolant pump (Primary circulation System)

Pump Type	Shaft seals
Number of pumps	4
Pump speed	1500 rpm
Head at rated conditions	102.1 m
Flow at rated conditions	7.87 m^3/s

Pressurizer

Total volume	75 m^3
Steam volume (Working medium volume): full power	35 m^3
Steam volume (Working medium volume): Zero power	50 m^3
Heating power of heater rods	2600 kW

Primary containment

Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	46.8 m
Dimensions - height	57.5 m
Design pressure	0.55 MPa
Design temperature	170 °C
Design leakage rate	0.3 Volume % /day

Residual heat removal systems

Active/passive systems	Active

Safety injection systems

Active/passive systems	Active and Passive

Turbine

Turbine speed	1500 rpm

Generator

Voltage	24 kV
Frequency	50 Hz

Feedwater pumps

Number	3