

Status report 83 - Advanced Power Reactor 1400 MWe (APR1400)

Overview

Full name	Advanced Power Reactor 1400 MWe
Acronym	APR1400
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	3983.00 MWth
Gross Electrical capacity	1455.00 MWe
Design status	Under Construction
Designers	KEPCO/KHNP
Last update	04-11-2011

Description

Introduction

The Advanced Power Reactor 1400 MWe (APR1400) is a standard evolutionary advanced light water reactor (ALWR) in the Republic of Korea developed in 2002. The design is based on the experience that has been accumulated through the development, construction, and operation of OPR1000, the Optimum Power Reactor 1000MWe, the first standard pressurized water reactor (PWR) plant in Korea. APR1400 also utilizes state-of-the-art proven technology and incorporates a number of advanced design features to meet the utility's needs for enhanced economic goals and to address the new licensing safety issues and requirements for an improved plant safety.

During the 1980s, Korea launched the technology self-reliance program on every aspect of a nuclear power plant construction project and accomplished the nuclear technology self-reliance through the OPR1000 design and construction projects. Currently (as of August 2010), a total of 8 OPR1000 units are in operation with excellent performance and 4 units of the latest version of OPR1000 are under construction and will be in commercial operation in 2010 through 2012. The repeated construction and subsequent operation of OPR1000 units brought forth internationally competitive construction technology and outstanding plant operation and maintenance capabilities. Based on the self-reliant technology and experience accumulated through the design, construction, operation and maintenance of OPR1000, Korea launched the APR1400 development project in 1992 and completed its standard design in 2002.

The reactor type and plant design concept of APR1400 is the result of a two-year research program, Phase I, which was completed in 1994. During this phase, the advance PWR plant designs being developed worldwide were reviewed, and the major design goals for safety and performance as well as the design concept were established to

meet the future demand for ALWRs. The safety and economic goals for APR1400 were set through a comparative study among the ALWR designs available worldwide and the utility requirements on ALWRs. The major design requirements for safety and performance goals set for APR1400 are listed in Table 1.

The basic design satisfying the above design requirements was developed during Phase II. Also, a series of experimental research on the advanced design features newly introduced in the design was performed in parallel with plant design development. The nuclear steam supply system (NSSS) design has been completed with the general arrangement and the major component specifications. With a complete balance of plant (BOP) design, the Standard Safety Analysis Report (SSAR) was developed. At the end of the basic design development in early 1999, a further design optimization was performed to improve the economic competitiveness, operability, and maintainability while maintaining the overall safety goal during Phase III. APR1400 design optimization has been completed in 2001 and acquired the standard design approval from Korean regulatory body in May 2002.

APR1400 was determined to be built as the next nuclear power plant in the Republic of Korea following 12 units of OPR1000 plant being operated or constructed. The site for the first APR1400 plant was decided near the Kori site, and the construction project for the twin units, Shin-Kori units 3&4, is in progress with the goal of commercial operation in 2013 and 2014 for units 3 and 4, respectively. Also, the follow-on APR1400 construction projects are either in the early stage, such as Shin-Ulchin units 1&2 with the commercial operation goal in 2016 and 2017, respectively, or in the planning stages such as Shin-Kori units 5&6 and Shin-Ulchin units 3&4.

General Requirement	Performance requirements and economic goals
<p>Type and capacity : PWR, 1400 MWe</p> <p>Plant lifetime : 60 years</p> <p>Seismic design : SSE 0.3g</p> <p>Safety goals :</p> <p>Core damage frequency < 1.0E-5/Ry</p> <p>Containment failure frequency < 1.0E-6/Ry</p> <p>Occup. radiation exposure < 1 man -Sv / Ry</p>	<p>Plant availability : greater than 90%</p> <p>Unplanned trips : less than 0.8 per year</p> <p>Refueling interval : 18 months or longer</p> <p>Construction period : 48 months (N-th plant)</p> <p>Economic goal : 20% cost advantages over competitive energy sources</p>

Table 1. APR1400 design requirement for safety and performance goals

Since APR1400 has been evolved from OPR1000, the basic configuration of the nuclear steam supply system is same. As shown in Figure 1, APR1400 has two primary coolant loops and each loop has one steam generator and two reactor coolant pumps in one hot leg and two cold legs arrangement. This two loop/four pump configuration of the reactor coolant system is a well proven design concept through highly reliable operation records of OPR1000 plants.

APR1400 has several advanced features such as the direct vessel injection from the safety injection system, passive flow regulation device in the safety injection tank, in-containment refueling water supply system, advanced safety depressurization system, and systems for severe accident mitigation and management. The advanced main control room, designed with the consideration of human factors engineering, with full-digital instrumentation and control (I&C) systems is another example of the design improvement. Especially, the plant general arrangement has been improved with the reflection of operation and construction experiences of OPR1000.

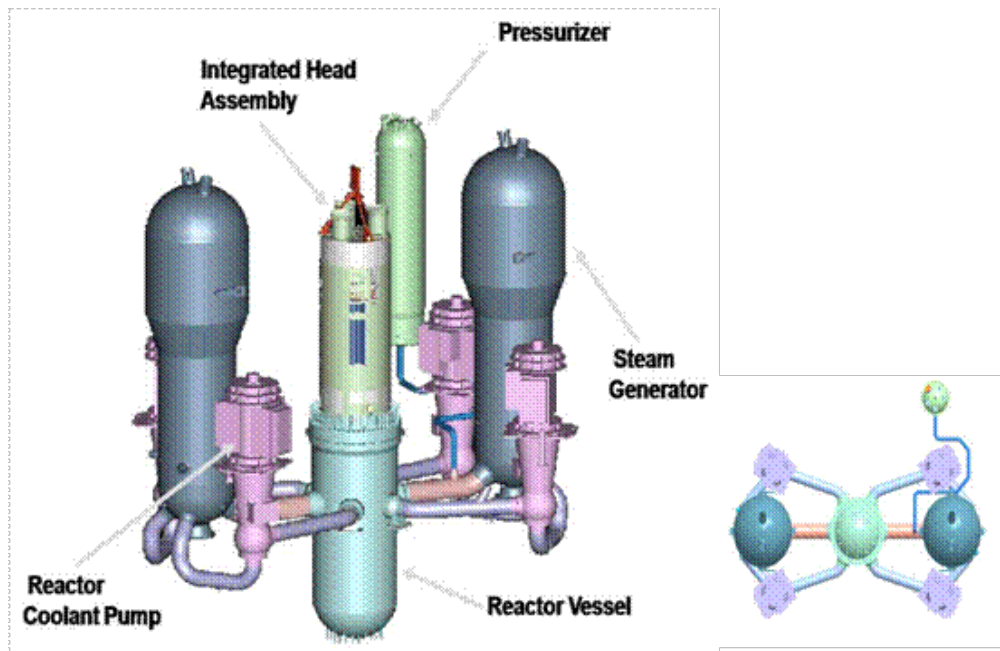


Figure 1. APR1400 Reactor Coolant System Configuration

Description of the nuclear systems

2.1. Primary circuit and its main characteristics

The primary loop configuration of APR1400 is similar to that of the OPR1000, which has two reactor coolant loops. The nuclear steam supply system is designed to operate at a rated thermal output of 4000 MW_t to produce an electric power output of around 1450 MWe in the turbine/generator system. The major components of the primary circuit are the reactor vessel, two reactor coolant loops, each containing one hot leg, two cold legs, one steam generator (SG), and two reactor coolant pumps (RCPs), and a pressurizer (PZR) connected to one of the hot legs. The two steam generators (SGs) and the four RCPs are arranged symmetrically. The steam generators are located at a higher elevation than the reactor vessel for natural circulation purpose. For vent and drain, the elevation of the PZR and the surge line is higher than that of reactor coolant piping. A schematic diagram of arrangements and locations of the primary components and safety-related systems are shown in Figure 2.

In the reactor pressure vessel (RPV) design, four direct vessel injection (DVI) lines are connected to supply core cooling water from the in-containment refuelling water storage tank (IRWST). Level probes are added in the hot leg to monitor the water level during mid-loop operation. The design temperature in the hot leg is reduced from 327°C at the normal operating pressure, 15.5 MPa, of the currently operating nuclear plants to 324°C in order to reduce the possibility of stress corrosion cracking (SCC) in SG tubes. Conventional spring loaded safety valves mounted at the top of the PZR are replaced by the pilot operated safety relief valves (POSRVs). Functions of the RCS overpressure protection during the design bases accidents and the manual depressurization in such a cases of beyond design bases accident of a total loss of feedwater (TLOFW) event are designed to be performed by POSRVs. Also, POSRVs provide a rapid depressurization during severe accidents to prevent direct containment heating (DCH). On the secondary side of the SGs, two main steam lines are arranged on each SG, and each steam line has five non-isolable safety valves, one main steam relief valve, and one isolation valve.

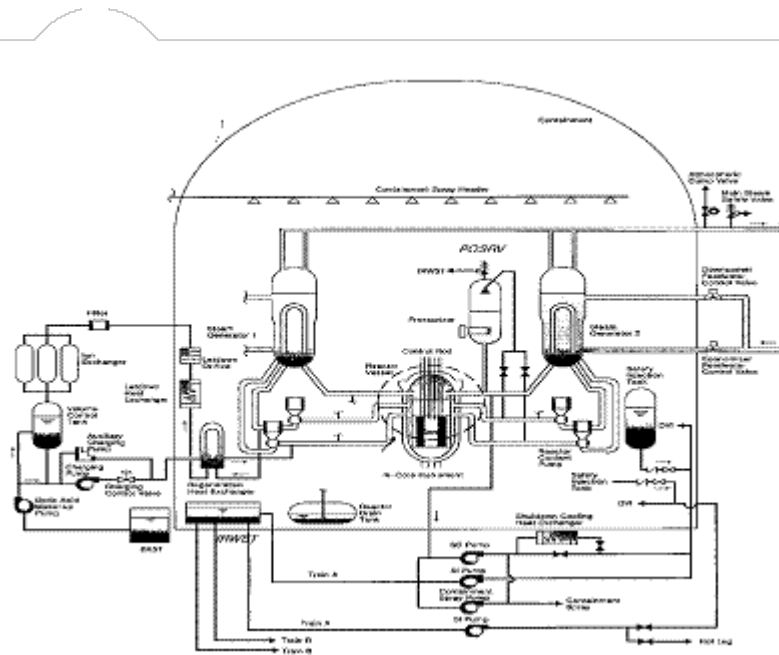


FIG. 2. Schematic diagram of primary components and safety systems

2.2. Reactor core and fuel design

The reactor core of APR1400 is designed to generate 3,987 MW thermal power with an average volumetric power density of 100.9 W/cm^3 . The reactor core consists of 241 fuel assemblies made of fuel rods containing uranium dioxide (UO_2) fuel. The number of control element assemblies (CEAs) used in the core is 93 in which 76 CEAs are full-strength reactivity control assemblies and the rest are part-strength CEAs. The absorber materials used for full-strength control rods are boron carbide (B_4C) pellets, while Inconel alloy 625 is used as the absorber material for the part-strength control rods.

The core is designed for an operating cycle of above 18 months with a discharge burnup as high as approximately 60,000 MWD/MTU, and has an increased thermal margin of more than 10% to enhance safety and operational performance. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber of gadolinium (Gd_2O_3) to suppress excess reactivity after fuelling and to help control the power distribution in the core. The neutron flux shape is monitored by means of 45 fixed in-core instrumentation (ICI) assemblies.

The possibility of utilizing the mixed oxide (MOX) fuel up to 1/3 core is considered in the core design. A total of 8 additional reserve CEAs is installed to increase the reactivity control capability, if necessary, for MOX fuel loadings. Also, the APR1400 core is designed to be capable of daily load following operation.

The fuel assembly consists of fuel rods, spacer grids, guide tubes, and upper and lower end fittings. 236 locations of each fuel assembly are occupied by the fuel rods containing UO_2 pellets or the burnable absorber rods containing $\text{Gd}_2\text{O}_3\text{-UO}_2$ in a 16×16 array. The remaining locations are 4 CEA guide tubes and 1 in-core instrumentation guide tube for monitoring the neutron flux shape in the core.

The advanced fuel assembly, named as PLUS7, provides enhanced thermal hydraulic and nuclear performance and the structural integrity. The mixing vanes with high thermal performance, which induce a relatively small pressure loss, are adopted in all mid-grids to increase thermal margin above 10% that has been confirmed in the Critical Heat Flux (CHF) test. The batch average burn-up is increased up to 55,000 MWD/MTU through optimizing the fuel assembly and fuel rod dimensions and adopting an advanced Zr-Nb alloy as a fuel clad.

The neutron economy is improved by introducing the axial blankets at both end of pellet region and optimizing the fuel rod diameter. The mid-grid buckling strength has been increased using straight grid straps and optimizing grid height. This design improvements increase the seismic resistance for fuel assembly to maintain its integrity even under severe seismic-related accidents. The conformal type contact geometry between the mid-grid spring and fuel rod increase the in-between contact area to improve the resistance capacity for fretting wear. Debris-Filter Bottom Nozzle (DFBN) is adopted to trap most debris before it enters the fuel assembly. This increases the debris filtering efficiency

to reduce fretting wear-induced fuel failure.

2.3. Fuel handling and transfer systems

The fuel handling system is designed for a safe and rapid handling and storage of fuel assemblies from the receipt of fresh fuel to the shipment of spent fuel.

The major equipment of the system comprises the refuelling machine, the CEA change platform, the fuel transfer system, the fresh fuel elevator, the CEA elevator and the spent fuel handling machine. The refuelling machine is located in the containment building and moves fuel assemblies into and out of the reactor core and between the core and the fuel transfer system. The spent fuel handling machine, located in the fuel building, carries fuel to and from the fuel transfer system, the fresh fuel elevator, the spent fuel storage racks, and the spent fuel shipping cask areas.

2.4. Primary components

2.4.1. Reactor pressure vessel

The reactor pressure vessel is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head as shown in Figure 3. The reactor pressure vessel contains internal structures, core support structures, fuel assemblies, control rod assemblies, and control and instrumentation components.

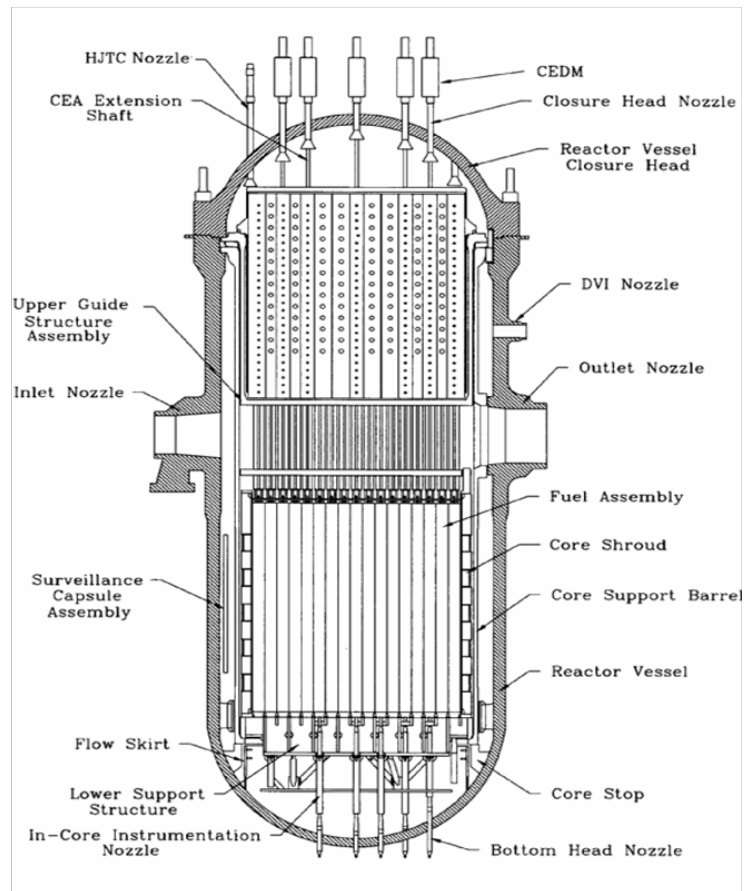
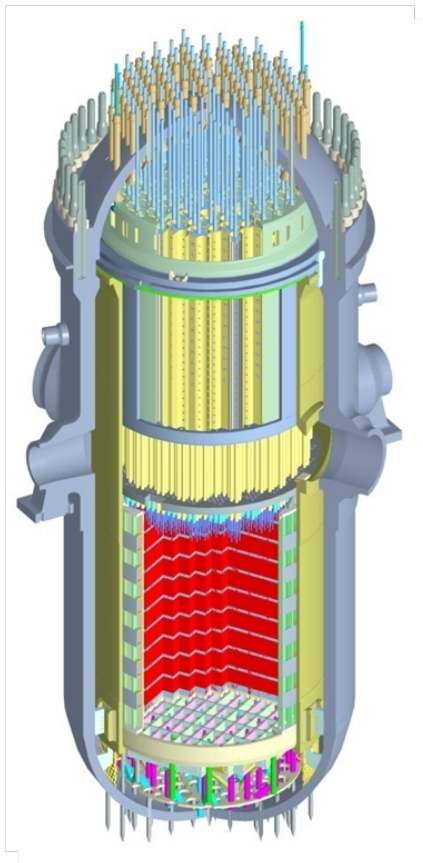


Figure 3. Reactor Vessel and Internals

The structural integrity of the reactor pressure vessel is verified through a structural sizing and fatigue evaluation, which calculates the stresses of the heads, shell and nozzles under thermal and pressure loads.

The direct vessel injection (DVI) nozzles are attached to the reactor vessel for the direct emergency coolant injection as a part of the safety injection system. The location of DVI nozzle is above the cold leg nozzles and determined to avoid the interference with reactor vessel external nozzles and support structure.

The life time of the reactor pressure vessel is extended to 60 years by the use of low carbon steel, which has lower contents of Cu, Ni, P, S as compared to the current design, resulting in an increase of brittle fracture toughness. The inner surface of the reactor vessel is clad with austenitic stainless steel or Ni-Cr-Fe alloy. The reactor vessel is designed to have an end-of-life RT_{NDT} of 21.1°C (70°F).

The reactor vessel is basically manufactured with a vessel flange, a hemispherical bottom head, and three shell sections of upper, intermediate and lower. The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The three shell sections, the bottom head forging and vessel flange forging are joined together by welding. Also, four inlet nozzle forgings, two outlet nozzle forgings, four DVI nozzle forgings, and sixty-one ICI nozzles are also welded. The upper closure head is fabricated separately and is joined to the reactor vessel by bolting. The dome and flange are welded together to form the upper closure head, on which the control element drive mechanism (CEDM) nozzles are welded.

2.4.2 Reactor internals

The reactor internals consist of the core support structures, which include the core support barrel, upper guide structure barrel assembly and lower support structure, and the internal structures. The core support structures are designed to support and orient the reactor core fuel assemblies and control element assemblies, and to direct the reactor coolant to the core. The primary coolant flows in through the reactor vessel inlet nozzles from the reactor coolant pump, passes through the annulus between the reactor vessel and core support barrel, through the reactor vessel bottom plenum and core, and finally flows out through the outlet nozzles of the reactor vessel connected to the hot legs.

The core support barrel and the upper guide structure are supported at its upper flange from a ledge in the reactor vessel flange. The flange thickness is increased to sustain the enhanced seismic requirements. All reactor internals are manufactured of austenitic stainless steel except for the hold-down ring, which is made of high-tension stainless steel. The hold-down ring absorbs vibrations caused by the load to the axial direction of internal structures.

The upper guide structure, which consists of the fuel assembly alignment plate, control element shroud tubes, the upper guide structure base plate, CEA shrouds, and an upper guide structure support barrel, is removed from the core as a single unit during refuelling by means of special lifting rig.

2.4.3. Steam generators

Steam Generator (SG) is a vertical inverse U-tube heat exchanger with an integral economizer, which operates with the RCS coolant in the tube side and secondary coolant in the shell side as shown in Figure 4. The two SGs are designed to transfer the heat of 4000 MWt from the RCS to the secondary system. The secondary system produces steam to drive the turbine-generator, which generates the net electrical power of 1,400 MWe.

Moisture separators and steam dryers in the shell side limit the moisture content of the exit steam less than 0.25 w/o during normal full power operation. An integral flow restrictor has been equipped in each SG steam nozzle to restrict the discharge flow in the event of a steam line break. For the maintenance, inspection of equipment condition, and tube-sheet sludge lancing, each SG has a 21 inches man-ways in the cold leg and hot leg side of primary system.

Also, two man-ways to allow access to the separator and dryer area of the secondary side, an internal hatch over the top of the tube bundle, two 8 inches hand-holes at the tube-sheet region are also provided.

APR1400 SG uses 13,102 tubes per SG with Alloy 690 as tube material to improve the integrity of SG tube. And the upper tube support bar and plate is designed to prevent SG tube from flow induced vibration. In order to improve the operating margin of the steam generator, the SG tube plugging margin is increased up to 10%. The water volume of SG secondary side is sufficient to provide the dry-out time up to 20 minutes in the event of total loss of feed-water

(TLOFW). This design enhances the capability of alleviating the transients during normal operation by reducing the potential for unplanned reactor trip and the plant safety and the operational flexibility.

To improve the operability, the angle of nozzle in the hot leg side of the primary system is modified to enhance the stability of mid-loop operation. The SG water level control system is designed in such a way that the water level is controlled automatically over the full power operating range.

The economizer feedwater nozzle provides a passage of feedwater to the economizer, which is installed to increase the thermal efficiency of the steam generator at the cold side, and experiences a high temperature gradient. The feedwater nozzles are designed to endure the excessive thermal stress which causes an excessively large fatigue usage factor. The downcomer feedwater nozzle attached in the upper shell of SG also provides small portion of feedwater to the downcomer to facilitate internal recirculation flow. 10% of full power feedwater flow is provided to the downcomer feedwater nozzle and the remaining to the economizer nozzle at a reactor power higher than 15%, below which all feedwater is supplied to the downcomer nozzle.

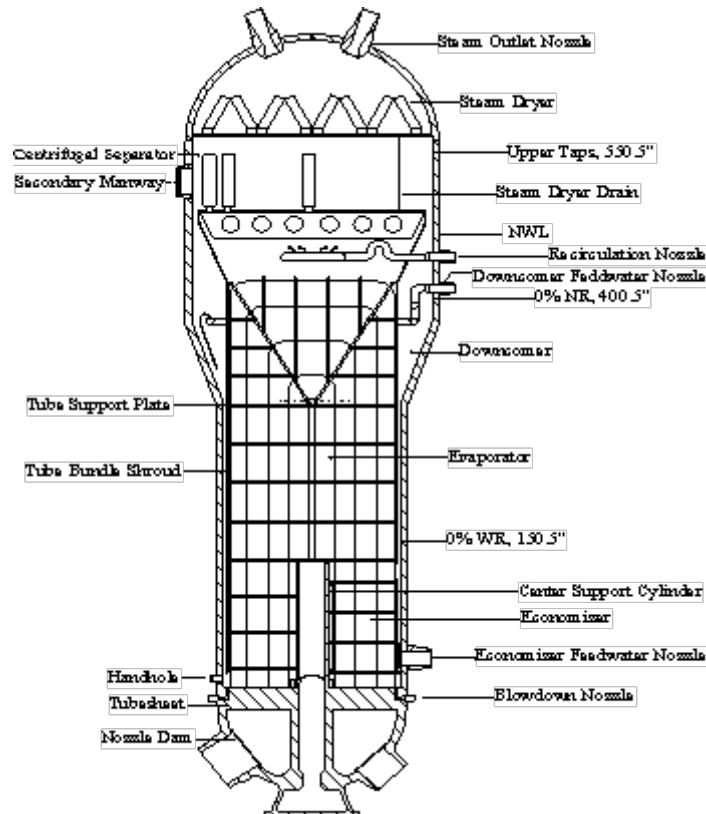


Figure 4. Steam Generator

2.4.4. Pressurizer

The pressurizer is a vertically mounted cylindrical pressure vessel. Replaceable direct immersion electric heaters are vertically mounted in the bottom head. The pressurizer is furnished with nozzles for the spray, surge, pilot operated safety relief valves (POS RVs), and pressure and level instrumentations. A man-way is provided on the top head for access for inspection of the pressurizer internals. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge. The pressurizer maintains RCS pressure and inventory within specified limits in conjunction with the pressurizer pressure and level control systems against all normal and upset operating conditions without reactor trip.

The pressurizer, having a total internal free volume of 68.9m^3 ($2,400\text{ft}^3$), is to maintain an operating pressure and temperature of the reactor coolant system. Four POS RVs are adopted instead of two safety depressurization system (SDS) valves and three pressurizer safety valve (PSV) of the conventional plant. It provides more reliability in

overpressure protection function and more convenience in maintenance activities. The RCS inventory that would discharge through the POSRV under accident conditions is directed to the in-containment refuelling water storage tank (IRWST) and quenched so that the contamination of the containment environment is significantly reduced.

2.4.5. Integrated head assembly (IHA)

The reactor vessel upper closure head area of the conventional plant consists of CEDM cooling system, cooling shroud assembly, heated junction thermocouples, missile shielding structure, and head lif rig. These components are usually disassembled, separately stored, and reassembled during every refuelling outage. The IHA is a structure to combine and integrate all the reactor vessel closure head area structures into one assembly as shown in Figure 5.

The primary purpose of the IHA is to assemble all the head area structures, components, and cable system and their supports into one assembly so that the refuelling time can be reduced from such operational activities as installation and removal of head area components. Also, the IHA contributes to the reduction of radiation exposures to the maintenance crew since the dissembling and assembling time of the reactor vessel head is reduced.

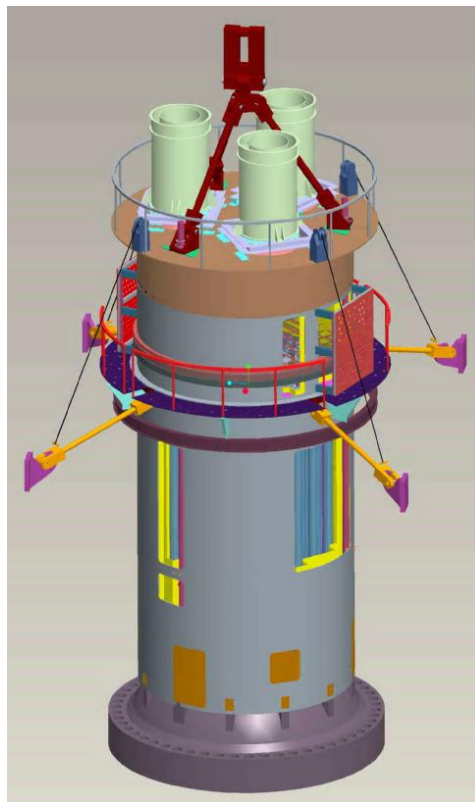


Figure 5. Integral Head Assembly

2.4.6 Reactor coolant pumps

The reactor coolant pumps circulate the coolant between the reactor vessel and the steam generators for heat transfer from the reactor core to the SGs. There are two pumps for each coolant loop, located in each cold leg. The pump is a single-stage centrifugal unit of vertical type, driven by a 13,320 hp electric motor. Leak-tightness of the shaft is ensured by a mechanical seal designed to prevent leaking against the full internal pressure in the pump.

2.4.7. RCS Piping

The Leak-Before-Break (LBB) principle is adopted for the piping system of APR1400, since the pipe whip restraint and the support of the jet impingement shield in the piping system of earlier plants are expensive to build and maintain, and lead to a potential degradation of plant safety. The LBB principle is applied to the main coolant lines, surge lines, and pipes in the shutdown cooling system and the safety injection system. The application of LBB

reduces the redundant supports of the pipe in the NSSS pipe system since the dynamic effects of postulated ruptures in the piping system can be eliminated from the design basis. Therefore, the cost of design, construction and maintenance is reduced.

2.5. Reactor auxiliary systems

2.5.1. Chemical and volume control system (CVCS)

The CVCS of APR1400 is not required to perform safety functions such as safe shutdown and accident mitigation. This system is basically for the normal day-to-day operation of the plant. The components related to charging and letdown function, however, are designed as a safety grade and reinforced to assure the reliability for normal and transient conditions. Two centrifugal charging pumps and a flow control valve provide required charging flow. For normal operation, only one charging pump is used to supply the required minimum flow of 12.6 kg/s.

The letdown flow from the reactor coolant system passes through the regenerative and letdown heat exchanger, where an initial temperature reduction takes place. Pressure reduction occurs at the letdown orifice and the letdown control valve. Following temperature and pressure reduction, the flow passes through a purification process at the filters and ion exchangers. After passing through the purification process, the letdown flow is diverted into the volume control tank (VCT) which is designed to provide a reservoir of reactor coolant for the charging pumps and for the dedicated seal injection pumps for the reactor coolant pumps.

2.5.2. Component cooling water system

The component cooling water system (CCWS) is a closed loop cooling system that, in conjunction with the Essential Service Water System (ESWS) and Ultimate Heat Sink (UHS), removes heat generated from the plant's essential and non-essential components connected to the CCWS. Heat transferred by these components to the CCWS is rejected to the ESWS via the component cooling water heat exchangers. The system is designed to have the cross connection between two divisions to enhance the plant availability and maintenance flexibility.

2.5.3. Reactor coolant gas vent system (RCGVS)

The RCGVS is a part of the safety depressurization and vent system (SDVS). The reactor coolant gas vent valves are mounted at the top of the pressurizer and reactor vessel head. The size of the vent line is increased to have sufficient capacity to vent one-half of the RCS volume in one hour assuming a single failure.

2.5.4. Steam generator blowdown system

The functions of the SG blowdown system are to control SG secondary side water chemistry and to remove sludge from the SG tube support plates. One flash tank can accommodate normal and high capacity blowdown flow rates. To remove dynamic loading due to two-phase flow, the flash tank for blowdown is located in the auxiliary building near the containment. Bypass lines to the condensers are installed to overcome unavailability of the flash tank or the processing system.

2.5.5. Primary sampling system

The primary sampling system is designed to collect and deliver representative samples of liquids and gases in various process systems to the sample station for chemical and radiological analysis. The system permits sampling during normal operation, cooldown and post-accident modes without requiring access to the containment. Remote samples can be taken from the fluids in high radiation areas without requiring access to these areas.

2.6. Operating Modes

APR1400 is designed to be used for various operating modes not only for the base load full power operation but also for a part load operation such as the load following operation. A standard 100-50-100% daily load follow operation has been considered in the reactor core design as well as in the plant control systems.

In addition, various load maneuvering capabilities are considered in the design such as up to 10% step change in load, +/- 5%/min ramp load changes. Also, it has the house load operation capability during a sudden loss of load up to 100% (full load rejection) in which plant control systems automatically control the plant at 3~5% power level without causing any reactor trips or safety system actuations.

In case of turbine generator trip from any power levels including full power, APR1400 prevents reactor trip and maintains reactor power at reduced level using reactor power cutback system (RPCS) and other control systems. This feature shortens outage time to return to power operation after a problem shooting and enhances plant safety by preventing unnecessary reactor trips. Also, APR1400 control system automatically controls plant parameters and prevents reactor trip during a loss of one or two operating main feedwater pumps event occurring at 100% power operation with 3 main feedwater pumps in service.

2.7. Fuel Cycle and Fuel Options

APR1400 is a typical PWR plant using slightly enriched uranium and, hence, is not designed as a breeder or a high-converter reactor. However, the reactor core and other related systems are designed to use MOX fuel up to 1/3 of core. The spent fuel treatment plan is beyond the plant design scope.

Description of safety concept

3.1. Safety requirements and design philosophy

Safety is a requirement of paramount importance for nuclear power. One of the APR1400 development policies is to increase the level of safety significantly. Safety and economics in nuclear power plants are not counteracting each other but can move in the same direction, since the enhancement of safety will also yield an improved protection of the owner's investment. Therefore, safety has been given top priority in developing the new design. To implement this policy, in addition to the plant being designed in accordance with the established licensing design basis to meet the licensing rules, APR1400 was designed with an additional safety margin in order to improve the protection of the investment, as well as the protection of the public health.

In order to implement this safety objective, quantitative safety goals for the design were established in a probabilistic approach:

- The total Core Damage Frequency (CDF) shall not exceed 10^{-5} /RY for both internal and external initiating events and 10^{-6} /RY for a single event and an incident occurred in the high pressure condition.
- The containment failure frequency shall be less than 10^{-6} /RY.
- The whole body dose at the site boundary shall not exceed 0.01 Sv (1 rem) for 24 hours after initiation of core damage with a containment failure.

To achieve the above quantitative goals, the defense-in-depth concept remains as a fundamental principle of safety, requiring a balance between accident prevention and mitigation. With respect to accident prevention, the increased design margin and system simplification represent a major design improvement and the consideration of accident mitigation call for the incorporation of design features to cope with severe accidents as well as design basis accidents.

In addition to the public safety, a concept of investment protection is implemented in APR1400 design. An example of a design requirement that aims at investment protection is the stipulation that a small break loss-of-coolant-accident (LOCA) with a break size smaller than 150 mm in diameter should allow the continued use of the reactor

with its fuel inventory after the repair of the ruptured pipe (and/or other damages in the reactor coolant system).

The enhanced margin could benefit the operability and availability of the nuclear power plants. For example, the margin can alleviate transients, thereby avoiding unexpected trips, and be used for later system modification or adaptation of new regulatory restriction. A few examples of the design requirements following this philosophy are the requested core thermal margin of more than 10%, sufficient system capacity for the prolonged operator response time on the transient events, and station blackout coping time.

3.2. Safety systems and features (active, passive and inherent)

The safety systems and features of APR1400 are designed to be a hybrid system in which active and passive systems perform necessary safety functions. The major safety systems are the safety injection system, safety depressurization and vent system, in-containment refuelling water storage system, shutdown cooling system, auxiliary feedwater supply system, and containment spray system.

The major design characteristics of APR1400 safety systems are as follows:

- Improved reliability of SIS through four trains design of mechanical equipments and eliminating the interconnection between SIS and SCS
- Simplified operation of SIS by merging the high pressure injection, the low pressure injection, and re-circulation modes into one safety injection mode
- Lowered susceptibility of IRWST by the external hazards through locating RWST in the containment
- Enhanced plant safety by adopting the advanced features such as the FD in SIT, the IRWST, and the Direct Vessel Injection (DVI) of SIS
- Improved reliability of CSS through the interconnection design between CSS and SCS
- Increased reliability of AFS by using two 100 % motor-driven pumps, two 100 % turbine-driven pumps, and two independent safety-related emergency feed-water storage tanks located in the auxiliary building

3.2.1. Safety injection system (SIS)

The main design concept of the Safety Injection System (SIS) is simplification and diversity to achieve higher reliability and better performance. The safety injection system is comprised of four independent mechanical trains without any tie line among the injection paths and two electrical divisions. Each train has one active Safety Injection Pump (SIP) and one passive Safety Injection Tank (SIT) equipped with a Fluidic Device (FD) as shown in Figure 6. To satisfy the LOCA performance requirements, each train provides 50% of the minimum injection flow rate for breaks larger than the size of a direct vessel injection line. For breaks equal to or smaller than the size of a direct vessel injection line, each train has 100% of the required capacity. The low pressure injection pumps with common header installed in the conventional design are eliminated, and the functions for safety injection and shutdown cooling are separated.

By adopting the passive flow regulator, fluidic device, in the SIT and the In-containment Refuelling Water Storage Tank (IRWST) system, the current operation modes of high pressure injection, low pressure injection, and re-circulation are merged into only one operation mode in case of LOCAs. Accordingly, the low pressure pumps are eliminated from the SIS and the water for the safety injection is taken from the IRWST only.

The core cooling water is designed to be injected directly into the reactor vessel so that the possibility of the spill of the injected flow through the broken cold leg is eliminated. For this purpose, four safety injection lines are connected directly to the DVI nozzles located above the hot and cold legs on the upper portion of the reactor vessel.

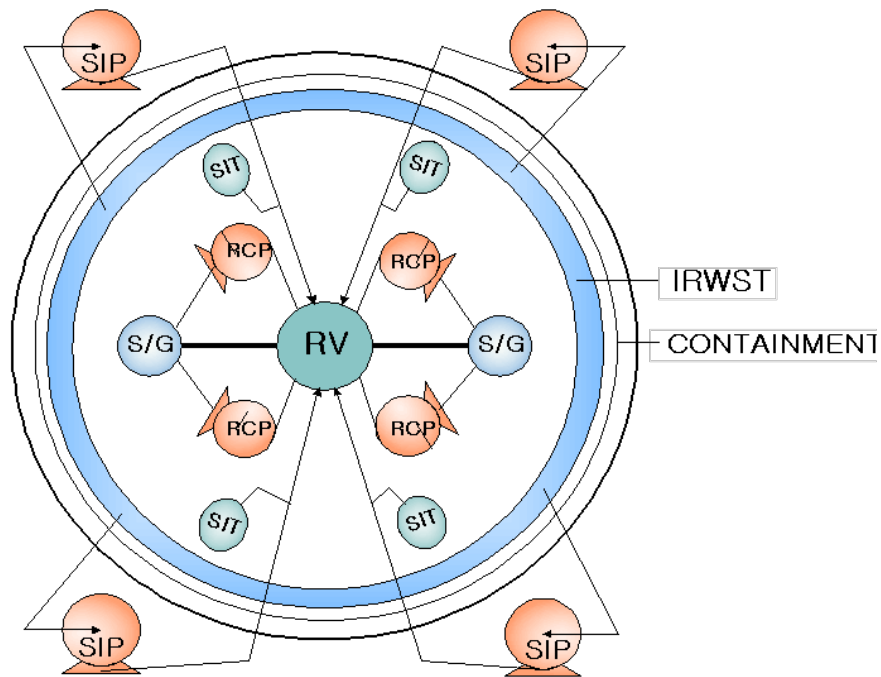


Figure 6. Safety Injection System

3.2.2. Fluidic device (FD)

The APR1400 uses a FD, installed inside the SIT as a passive design feature, to ensure effective use of the SIT water. This design feature enables the APR1400 to achieve the goals of minimizing the ECC bypass during a blowdown, and of preventing a spillage of excess ECC water during the refill and reflood phases of a LBLOCA.

The basic concept of the FD is vortex flow resistance. When water flows through the stand pipe, which is installed in a rectangular direction with the exit nozzle, it makes low vortex resistance and a high flow rate. When the water level is below the top of the stand pipe, inlet flow is switched to the control ports which are installed in a tangential direction with the exit nozzle, and it makes a high vortex resistance and low flow rate as shown in Figure 7.

The APR1400 FD provides a high discharge flow rate of SI water, when the SIT starts to inject, which is required during the refill phase of LBLOCA. When the refill phase is terminated, the discharge flow rate of the SI water drops sharply but is still large enough to remove any heat decay during the reflood phase. Because of the strong vortex motion in the FD, the pressure loss coefficient of the low flow rate period is almost ten times higher than that of the high flow rate period. The difference in the pressure drop helps extend the total duration of the SI water injection and also enables the low pressure safety injection pump to be removed from the SI system.

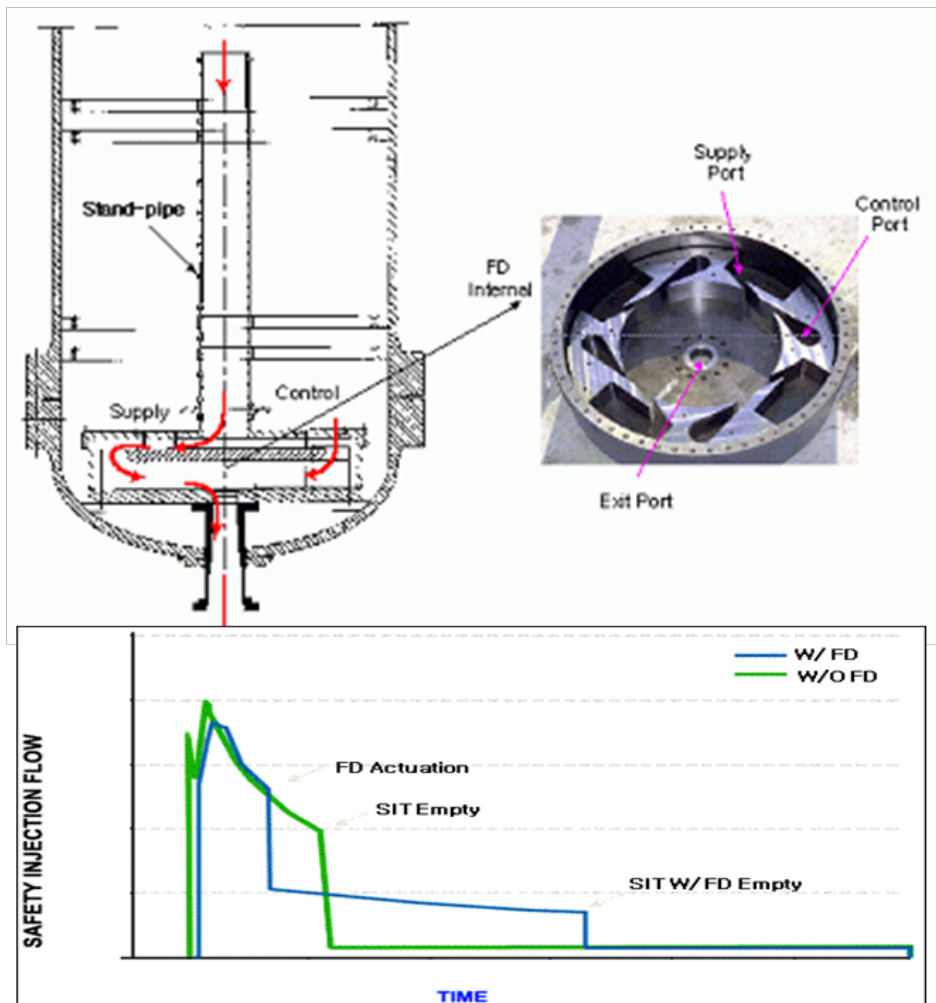


Figure 7. Fluidic Device in SIT

3.2.3. In-containment refuelling water storage tank (IRWST)

The in-containment refuelling water storage tank is located inside the containment, and the arrangement is made in such a way that the injected core cooling water can return to the IRWST. It consists of an annular cylindrical tank along the containment wall at low elevation, a holdup volume tank (HVT), and four inside sumps. The susceptibility of the current refuelling water storage tank to external hazard is eliminated by locating it inside the containment. The IRWST provides the functions of the normal refuelling water storage as well as a single source of water for the safety injection, shutdown cooling, and containment spray pumps.

The IRWST is also used as a heat sink to condense the steam discharged from the relief valves on the pressurizer in case of rapid depressurization of the RCS to prevent high pressure core melt ejection or to enable feed and bleed operation. Moreover, it provides the function of coolant supply to the external reactor vessel cooling system (ERVCS) and the cavity flooding system in case of severe accidents to retain the molten core inside reactor vessel and, if vessel breaks, to cool the molten core in the reactor cavity, respectively. The volume of the IRWST is 2,470 m³ (652,800 gal). This capacity is sufficient for flooding the refuelling cavity during normal refueling operations. It also covers the necessary water volume to flood the HVT and the reactor cavity for the external reactor vessel cooling operation during severe accidents.

3.2.4. Shutdown cooling system (SCS)

The shutdown cooling system is a safety-related system that is used in conjunction with the main steam and main or

auxiliary feedwater system to reduce the RCS temperature in post shutdown periods from the hot shutdown operating temperature to the refueling temperature. After initial heat rejection through the SGs to the condenser or atmosphere, the SCS is put into operation at 176.7 °C and 31.6 kg/cm² A.

To improve shutdown cooling capability and system reliability, and to remove any possibility of intersystem LOCA, the following improvements were implemented for the design of the SCS:

- Increase of the design pressure to 6.2 MPa to protect the inter-system LOCA;
- Reinforcement of decay heat removal function for the reactor emergency conditions;
- Adoption of the partial 4 train concept by introducing exchangeable shutdown cooling and containment spray pumps; and
- Installation of the independent heat exchanger.

3.2.5. Auxiliary feedwater system

The Auxiliary Feedwater System (AFWS) is a dedicated safety system designed to supply feedwater to the SGs for removal of heat from the RCS for events in which the main or startup feedwater systems are unavailable. In addition, the AFWS refills the SGs following the steam generator tube rupture to minimize leakage through the ruptured tubes. The auxiliary feedwater system is an independent 2-division system, one for each SG, and each division has 2 trains. The reliability of the AFWS has been increased by use of one 100% capacity motor-driven pump and one 100% capacity turbine-driven pump for diversity and redundancy and one dedicated safety-related auxiliary feedwater storage tank in each division as a water source in addition to the non-safety grade condensate storage tank as a backup source.

3.2.6. Safety depressurization and vent system (SDVS)

The SDVS is a dedicated safety system designed to provide a safety grade means to depressurize the RCS in the event that pressurizer spray is unavailable during plant cooldown to cold shutdown and to depressurize rapidly the RCS to initiate the feed and bleed method of plant cooldown subsequent to the total loss of feed-water event. The Pilot Operated Safety Relief Valves (POSRVs) are employed for feed and bleed operation. This system establishes a flow path from the pressurizer steam space to the IRWST.

3.2.7. Containment spray system (CSS)

The CSS consists of two trains and takes the suction of its pump on the IRWST to reduce the containment temperature and pressure during the accidents occurred in the containment. The CSS are designed to be interconnected with the Shutdown Cooling System (SCS), which is also comprised of two trains. Also, the pumps of these systems are designed to have the same type and capacity in-between. This design makes the CSS have the higher reliability compared with the conventional plant.

3.3. Experimental test for validation of new design features

APR1400 adopted new safety features such as the safety injection system with direct vessel injection, POSRV with spargers, and fluidic device. To verify these new features, associated test programs were conducted separately. In addition to these separate effect tests, an integral effect test was also performed using ATLAS, Advanced Thermal Hydraulic Test Loop for Accident Simulation, test facility for major design basis accidents including large break LOCA for APR1400.

The DVI nozzles in the APR1400 are located in the upper part of the Reactor Pressure Vessel (RPV) downcomer. In order to evaluate the Emergency Core Coolant (ECC) bypass during the reflood phase of a postulated LBLOCA and to assess the contribution of 4 train SIS to safety enhancement, the performance of SIS was examined by using the MIDAS, Multi-dimensional Investigation in Downcomer Annulus Simulation, test facility, which was designed to

be 1/5 length scale of the APR1400 and to use steam and water as test fluids.

In order to confirm the prototypic performance of the FD design, the full-scale performance tests were carried out in the VAPER, Valve Performance Evaluation test Rig, test facility, which was designed as the same size and operating condition as those of the APR1400 SIT. It was verified from these full-scale tests that the FD performance satisfies the standard design requirement of the APR1400.

Actuation of the power-operated safety relief valves (POSRVs) results in a transient discharge flow of air, steam or a two-phase mixture to the IRWST through the spargers. The discharge of these fluids induces complicated thermal-hydraulic phenomena such as a water jet, air clearing, and steam condensation, and these phenomena impose relevant hydrodynamic forces on the IRWST structure and the components of the SDVS. The relevant test of these loads was conducted at the B&C, Blowdown and Condensation, test facility. And, the test to evaluate steam jet condensation and the resultant thermal mixing in the IRWST was also performed.

The ATLAS is the integral effect test facility, which is designed from the viewpoints of both a global and local scaling based on the three-level scaling methodology to verify the integrated safety of the APR1400 according to adopting the new design features. The APR1400 accidents, which are the reflood phase of the LBLOCA, small-break LOCA scenarios including the DVI line breaks, steam generator tube ruptures, main steam line breaks, and mid-loop operation, have been simulated in the ATLAS.

The ATLAS facility has the following characteristics:

- 1/2-height, 1/144-area, and 1/288-volume scale
- Full-pressure simulation of the APR1400
- Same geometrical configuration as that of the APR1400 including 2×4 reactor coolant loops, a direct vessel injection (DVI), an integrated annular downcomer, etc.
- Maximum 8% of the scaled nominal core power

3.4. Severe accidents (beyond design basis accidents)

In the APR1400 design, severe accidents are addressed as follows:

- For phenomena likely to cause early containment failure, for instance, within 24 hours after accidents, mitigation systems shall be provided or design should address the phenomena although the probability for such accidents is low.
- For phenomena which potentially lead to late containment failure if not properly mitigated, the mitigation system or design measures should be considered in conjunction with the probabilistic safety goal and cost for incorporating such features to address the phenomena.

This approach is to enhance the effectiveness of investment on safety by avoiding undue over-investment on highly improbable accidents. Also, a realistic assessment is recommended for severe accident analyses.

The facilities to mitigate severe accidents are designed to meet the procedural requirements and criteria of the U.S.NRC regulations, including the Three Mile Island (TMI) requirements for new plants as reflected in 10 CFR 50.34 (f) and SECY-93-087. The severe accident management systems consists of a large dry pre-stressed concrete containment, Hydrogen Mitigation System (HMS), Cavity Flooding System (CFS), Ex-Reactor Vessel Cooling System (ERVCS), Safety Depressurization & Vent System (SDVS), Emergency Containment Spray Backup System (ECSBS).

3.4.1. Containment building design for severe accident

In order to maintain the integrity of the RCB and prevent the leakage of radioactive materials against severe accidents, the RCB is designed to have enough free volume for the load to be below ASME Section III Service Level C in 24 hours after severe accidents and to keep hydrogen concentration under 13% in case of 75% oxidation of fuel clad-steam and the 1/4 inch steel plate is installed in the inside of the RCB. In addition, the RCB is constructed with the concrete having compressive strength of 6,000psi after 91 days curing.

3.4.2. Hydrogen mitigation system (HMS)

During degraded core accident, hydrogen will be generated at a greater rate than during the design basis LOCA. The HMS is designed to accommodate the hydrogen production from 100% fuel clad metal-water reaction and limit the average hydrogen concentration in containment to 10% for degraded core accidents. The HMS consists of a system of Passive Auto-catalytic Recombiners (PARs) complemented by glow plug igniters installed within the containment. The PARs serve for accident sequences in which mild or slow release rates of hydrogen are expected, and are installed uniformly in the containment. Whereas, the igniters supplement PARs under the accident of very low probability in which very rapid release rates of hydrogen are expected, and are placed near source locations to facilitate the combustion of hydrogen in a controlled manner such that containment integrity is maintained.

3.4.3. Cavity flooding system (CFS)

The APR1400 reactor cavity adopts a core debris chamber, which is designed to have the heat transfer area of corium more than $0.02\text{m}^2/\text{MWt}$. The flow path of reactor cavity is designed to be convoluted to hinder the transfer of core debris to the upper containment. This design prevents Direct Containment Heating (DCH) by core debris.

The CFS consists of two trains connected with IRWST and two isolation valves are installed on each line as shown in Figure 8. When two isolation valves are open during severe accidents, the cavity cooling water is supplied from the IRWST to the reactor cavity driven by the gravity induced from the difference of water head in-between and then cools down the core debris in the reactor cavity, scrubs fission product releases, and mitigates the molten corium concrete interaction (MCCI).

3.4.4. External Reactor Vessel Cooling System (ERVCS)

The ERVCS is implemented as a severe accident mitigation system used for the purpose of in-vessel retention of corium under hypothetical core-melting severe accident conditions. The ERVCS shall be used only under the severe accident condition and thus is designed on safety margin basis. As shown in Figure 8, one train of shutdown cooling pump, with related valves, pipes, and instrumentation & controls, is provided for initial reactor cavity flooding to the level of hot leg. After the initial flooding by the shutdown cooling pump, the boric acid makeup pump (BAMP) is utilized to refill the reactor cavity, at a flow rate greater than that of boiling caused by decay heat from the molten core.

The ERVCS is designed to be manually operable only when the core exit temperature reaches a certain temperature following a severe accident. The operating procedure for the ERVCS was developed through severe accident analysis and probabilistic safety assessment. The gravity driven cavity flooding system (CFS) provides flooding of the reactor cavity below the reactor vessel. The CFS is a backup system used in case that ERVCS is unavailable and provides corium cooling, should the reactor vessel fail.

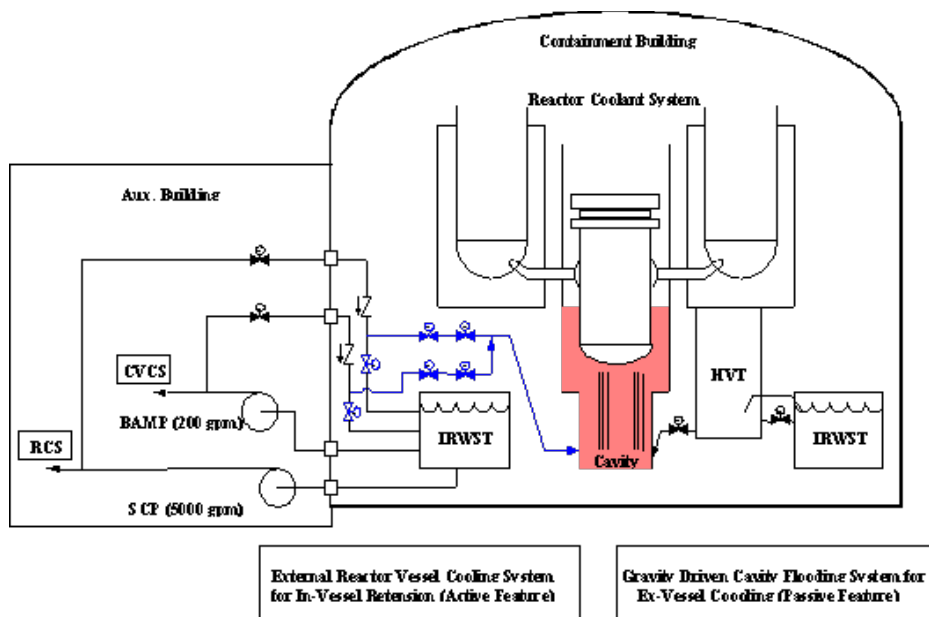


Figure 8. External reactor vessel cooling system and cavity flooding system

3.4.5. Emergency Containment Spray Backup System (ECSBS)

The ECSBS reduces the containment temperature and pressure during the SAs by using the spray water supplied from the temporary water source. The ECSBS contributes to relieving the threat of containment integrity

3.5. Seismic Design

In order to increase the resistibility of the buildings and the structures against earthquake, the buildings and the structures are designed with applying the Safe Shutdown Earthquake (SSE) of 0.3g as a Design Basis Earthquake (DBE). And the seismic input motion enforced in the high frequency range is applied to envelope the design ground response spectrum of Reg. Guide 1.60 standard spectrum. In the meanwhile, according to 10 CFR 50 Appendix S, the seismic analysis considering the Operating Base Earthquake (OBE) for the operation condition is eliminated.

Since the seismic evaluation is performed including the effect of the soil-structure interaction on the soil sites, the APR1400 can be constructed not only on the rock sites but on the soil ones.

Proliferation resistance

Normal fuel manufactured for use in the water cooled reactors is low-enriched uranium (LEU, < 5% U-235) before irradiation. It is not possible to use this material as a weapon. Technically, the plutonium that arises from today's high burnup fuel should be undesirable as weapons material, since the Pu-240 content is in excess of 25%. Also, the discharged fuel assemblies are far too radioactive to be accessible for potential diversion, and when held in shielded casks, far too heavy for normal transport. In addition, plutonium is chemically very toxic, so remote handling is necessary in a reprocessing factory.

Safety and security (physical protection)

The APR1400 design and plant layout considered safety and security in various ways in which the design and

configuration inherently protects the plant against human induced malevolent external impacts and insider action. The physical security system is designed in accordance with the applicable regulations and is expected to provide protection against malevolent acts of sabotage with high assurance.

The main design features for APR1400 safety and security are as follows:

- Thick concrete walls for exterior and a large number of interior walls protect those equipments important to safety and provide a significant deterrent to penetration. The Auxiliary Building is physically separated in 4 quadrants, which provides adequate physical separation and barrier.
- The entry control point to the plant is centralized with security facilities and located in the compound building only for both units for the twin unit plants.
- A robust vehicle barrier system that is located at a safe standoff distance.
- Fencing is employed to establish a perimeter boundary at a sufficient distance such that under normal circumstances, security response force personnel are able to identify and engage a potential land based assault.
- An intrusion detection system is employed adjacent to the protected area boundary fencing to provide indication of unauthorized attempts to enter the protected area.
- A closed circuit television network is used to provide remote monitoring of the protected area boundary.
- An access control system is utilized to permit only properly authorized personnel into designated areas of the facility.

Description of turbine-generator systems

6.1. Turbine generator plant

The turbine generator plant consists of the main steam, steam extraction, feedwater, condensate, turbine generator and auxiliary systems. For these systems, heat balance optimization was made considering system operability, reliability, availability and economy.

The turbine generator system is designed to be capable of operation at 3% house load for a period of at least 4 hours without any detrimental effects in the system, and capable of startup to full load from the cold condition in 8 hours including rotor preheat. The main steam lines and the high-pressure turbine are designed for a steam pressure of 6.9 MPa (1000 psia), and two reheater stages are provided between the high pressure and the low pressure turbines. The generator is a three phase, 4-pole unit operating at 1800 rpm.

The capacity, response and modulation capabilities of the turbine bypass system are designed to make the turbine capable of withstanding a 100% generator load rejection without trip of the reactor or the turbine. The total flow capacity of the turbine bypass system is designed to be 55% of the turbine steam flow at full load steam pressure.

6.2. Condensate and feedwater systems

The condensate and feedwater systems are designed to deliver the condensate water from the main condenser to the steam generator. The condensate pumps consist of three 50% capacity motor-driven pumps (two operating and one standby). The feedwater pump configuration is selected to be three 50% capacity turbine driven pumps because of its ability to allow more reliable operation; all three pumps are normally operating, and the plant can remain at 100% power operation even in the case that one of the feedwater pumps is lost.

During the shutdown and startup, a motor-driven startup feedwater pump provides feedwater from the deaerator storage tank or condensate tank. The startup feedwater pump is capable of providing up to 5% of full power feedwater flow to both steam generators. On-line condensate polishers, which can operate in full and partial flow, as well as in bypass mode, are provided to maintain proper water chemistry during normal power operation. In the feedwater systems, the feedwater heaters are installed in 7 stages and arranged horizontally for easy maintenance and high reliability.

6.3. Auxiliary systems

6.3.1. Turbine bypass system

The turbine bypass system is provided to dissipate heat from the reactor coolant system during the turbine and/or the reactor trip. The OPR1000 and the APR1400 plant have the same capability of relieving 55% of full load main steam flow. In the case of OPR1000, 15% are dumped to the atmosphere and 40% are discharged to the main condenser while the APR1400 plant discharges the total 55% directly into the main condenser.

6.3.2. Turbine building open cooling water system (TBOCW)

The TBOCW system supplies seawater to the service side of the turbine building closed cooling water (TBCCW) heat exchangers. The APR1400 plant does not need the TBOCW pump which is installed in the OPR1000 to supply seawater as a heat sink for the plant. In the APR1400 plant design, the TBOCW system interfaces with the circulating water (CW) system to take the fresh seawater and discharge the heated seawater to the CW discharge conduit. This design concept reduces the plant capital cost.

6.3.3. Condenser vacuum system

The Condenser Vacuum (CV) system supports the plant startup and maintains the condenser vacuum by continuously removing non-condensable gases and air. The system consists of four 33-1/3 % capacity condenser vacuum pumps which are used to draw down the condenser shell pressure. These pumps are also used for "Holding mode" during normal operation without the steam jet air ejectors. In addition, the radiation level in the CV system discharge is continuously displayed on the radiation monitoring system in the main control room. The APR1400 plant is designed to combine the system discharge and the deaerator normal vent flow line to reduce the number of radiation monitors.

Electrical and I&C systems

7.1. I&C Systems

7.1.1. Design concept including control room

APR1400 is, like most of the advanced reactors being developed world-wide, equipped with digitized instrumentation and control (I&C) systems and computer-based control room man-machine interface (MMI), reflecting the status of modern electronics and computer technologies. The I&C and control room concept implemented in the APR1400 design is schematically depicted in Figure 9.

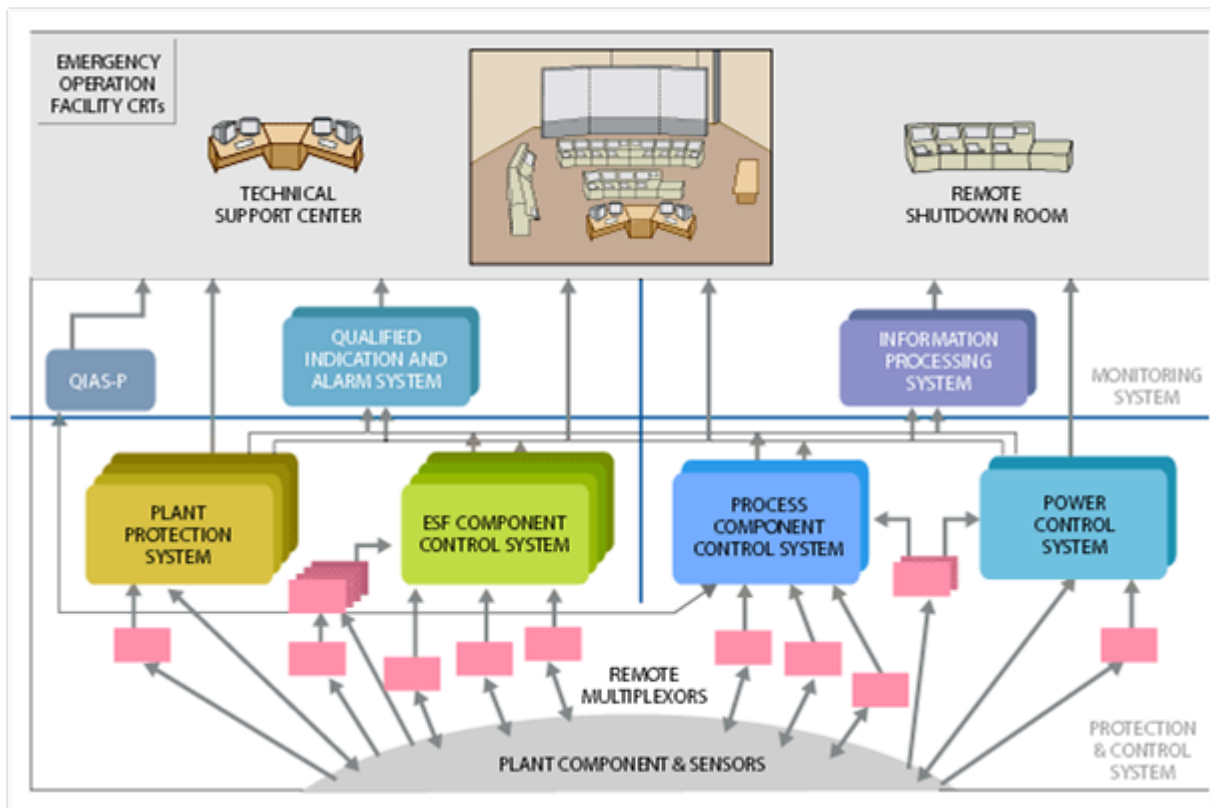


Figure 9. MMIS configuration

The APR1400 I&C system is designed with the network-based distributed control architecture. In this architecture, operator interface functions and control functions for NSSS, BOP and TG are integrated in common design standards and implemented in common digital system for high functionality, easy operation, and cost effective maintenance. Diversity between safety I&C systems and non-safety I&C systems together with hardwired switches are provided for the defense-in-depth against common mode failure of software in the safety I&C systems.

The main features of the I&C system are the use of distributed control system (DCS) and microprocessor-based Programmable Logic Controllers (PLCs) for the control and protection systems, and the use of UNIX workstations and industrial PCs (personal computers) for data processing systems.

To protect against common mode failures in software due to the use of software-based I&C systems, DCS and PLCs will be required in the redundant systems for diversity. For data communication, a high-speed fiber optic network based on standard protocols is used. The remote signal multiplexer is also utilized for the safety and non-safety systems field signal transmission.

Human factor engineering is an essential element of the MCR design and the human factor engineering principles are systematically employed to ensure safe and error-free operation. For the successful completion of the APR1400 MMI design process, a multidisciplinary team of human factor specialists, computer specialists, system engineers, and plant operators worked together as a team from the stage of conceptual design through the validation process.

7.1.2. Reactor protection and other safety systems

The plant protection system (PPS) includes the electrical, electronic, networking, and mechanical devices to perform the protective functions via the reactor protection system (RPS) and engineered safety features actuation system (ESFAS). The RPS is the portion of the PPS that acts to trip the reactor when the monitored conditions approach

specified safety settings and the ESFAS activates the engineered safety systems by safety injection actuation signal and the auxiliary feedwater actuation signal, and etc.

The reactor protection system and other safety-related systems are designed to use the off-the-shelf digital equipment which is commercially available to standardize the components and minimize the maintenance cost with the consideration of diversity. A high degree of conservatism is required in the design of the safety-related systems, and therefore, design principles such as redundancy, diversity, and segmentation have been incorporated in order to achieve both the desired availability and reliability of these systems.

A high reliability of the protection system is ensured by self-diagnostics, and automatic functional tests through surveillance using four independent channels. The redundant and fault tolerant configuration on controllers and the use of fiber-optics to isolate communications will increase system availability and maintainability.

A detailed software development program for software-based Class 1E systems were produced and applied as a guideline to ensure completeness of the software implementation, verification and validation process. Several critical safety systems were evaluated through prototyping and design verification programs.

7.2. Electrical systems

The electrical one line diagram of APR1400 is shown in Figure 10.

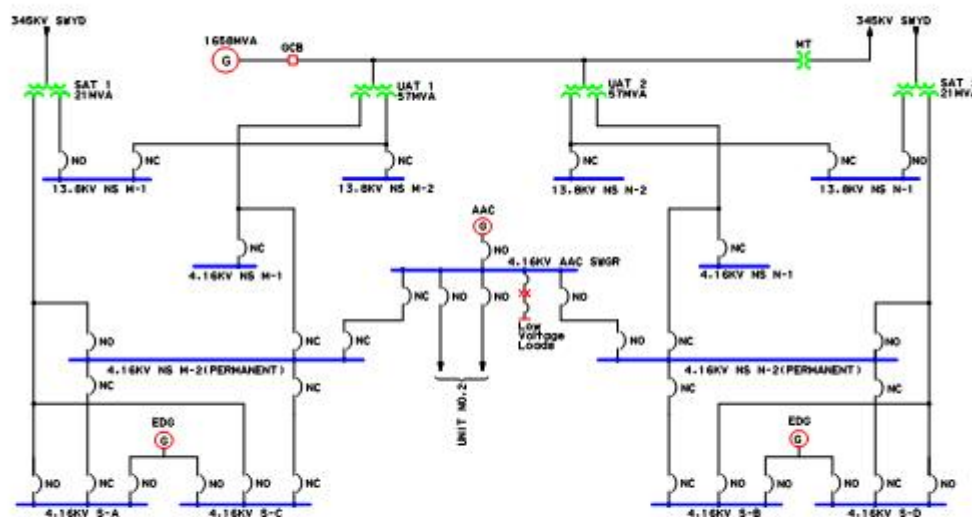


Figure 10. One line diagram

The main features of the electrical system configuration are:

- Two independent off-site power sources of 345 kV
- One main transformer consisting of three single-phase step-down transformers, and two three-winding unit auxiliary transformers for power delivery and supply during normal operation mode
- Two Class 1E emergency diesel generators (DGs) to provide on-site stand-by power for the Class 1E loads
- An alternate AC (AAC) source to provide power for equipment necessary to cope with station blackout at least for 8 hours. For the diversity of emergency electrical power sources, the gas turbine type is selected for AAC
- Automatic transfer of power source from unit auxiliary transformers to standby auxiliary transformers in the event of loss of power supply through the unit auxiliary transformers;
- Four independent Class 1E 125V DC systems for each RPS channel
- Two non-class 1E 125V DC systems and one non-class 1E 250V DC system

- AC voltage levels of 13.8 kV and 4.16kV for medium, 480 V and 120V for low voltages.

7.2.1. Operational power supply systems

The main power system consists of the generator, generator circuit breaker, main transformer, unit auxiliary transformer and stand-by transformer. The generator is connected to a gas-insulated 345 kV switchyard via the main transformer which is made of three single-phase transformer units. Step-down unit auxiliary transformers are connected between the generator and main transformer, and supply power to the unit equipment for plant startup, normal operation and shutdown. The stand-by transformer is always energized and ready to ensure rapid power supply to the plant auxiliary equipment in the event of failure of the main and unit auxiliary transformers.

The normal power source for non-safety and permanent non-safety loads is the off-site power source and the generator. If the normal power source is not available, the permanent non-safety loads are covered by two alternative sources: one from the stand-by off-site power source (via the stand-by transformer) and the other from one non-1E alternate AC power source.

7.2.2. Safety-related systems

The electric power necessary for the safety-related systems is supplied through 4 alternative ways: firstly, the normal power source, i.e., the normal off-site power and the in-house generation; secondly, the stand-by off-site power, i.e., the off-site power connected through the stand-by transformer; thirdly, the on-site standby power supply, i.e., two diesel generators; and finally, the alternative AC source.

Among these power sources, the on-site standby power is the most crucial for safety; it should be available in any situation. The arrangement of the on-site electrical distribution system is based on the functional characteristics of the equipment to ensure reliability and redundancy of power sources.

The on-site power supply is ensured by two independent Class 1E diesel generator sets; each of them is located in a separated building and is connected to one 4.16 kV safety bus.

The alternate AC source adds more redundancy to the electric power supply even though it is not a safety grade system. The non-class 1E alternate AC is provided to cope with Loss-of-Off-site-Power (LOOP) and Station Blackout (SBO) situation which have a high potential of transients leading to severe accidents. The alternate AC source is sized with sufficient capacity to accommodate the loads on the safety and the permanent non-safety buses.

Spent fuel and waste management

Radioactive waste management systems include systems, which deal with liquid, gaseous and solid waste, which may contain radioactive material. The Liquid, Gaseous and Solid Waste Management Systems are located in the compound building.

8.1. Liquid Waste Management System (LWMS)

The design objectives of the LWMS is to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor radioactive liquid waste. Each type of liquid waste is segregated to minimize the potential for mixing and contamination of non-radioactive flow streams. The processed liquid radioactive waste is sampled prior to release from monitor tanks and radiation monitors are provided in the discharge line to provide for a controlled monitored release.

The LWMS provides a means to collect, store, process, sample, and monitor radioactive liquid waste. The LWMS consists of collection tanks, process filter, Advanced Liquid Treatment System (ALTS) packages, process pumps and

vessels, monitor tanks, and appropriate instruments and controls to permit most operation to be conducted remotely. Radioactive wastes are segregated by routing to an initial collection sump or tank. This permits more effective processing of each type of waste and may lead to reduced solid waste volumes. Segregation is accomplished by means of a subsystem that processes different waste categories.

The LWMS is divided into the following major subsystems:

- Floor Drain Subsystem
- Equipment Waste Subsystem
- Chemical Waste Subsystem
- Radioactive Laundry Subsystem (RLS)

8.2. Gaseous Waste Monitoring System (GWMS)

The GWMS is provided with radiation monitors that monitor the discharge from the charcoal delay beds upstream of the discharge to the Compound Building HVAC System. The GWMS consists of two subsystems; the process gas subsystem and the process vent subsystem. The system is designed to collect, process, and monitor releases from all system input streams.

8.2.1. Process Gas Portion of the GWMS

Process gases contain radioactive xenon and krypton fission products from fuel and tramp uranium on fuel surfaces. The process gas portion of the GWMS receives fission gases from the Process Gas Header (PGH) and uses charcoal delay beds to delay the process gases for decay prior to release. The primary input source to the PGH is gases stripped from reactor coolant by the CVCS Gas Stripper and Volume Control Tank. The Reactor Drain Tank process vent surge volumes comprise the balance of the hydrogenated gaseous inputs to the PGH. The flow in the PGH primarily consists of hydrogen and noble gases with some trace quantities of other fission gases and water vapor. The removal of fission gases by the gas stripper maintains the fission gas concentration in the reactor coolant at a low residual level. This minimizes the escape of radioactive gases during maintenance on the reactor coolant system and releases resulting from leakage of reactor coolant.

8.2.2. Process Vent Portion of the GWMS

The process vent portion of the GWMS is designed to collect the low activity aerated gas streams from the potentially contaminated vents headers in the Containment, Auxiliary Building, Turbine Building and Compound Building. The process vents, except from the condenser evacuation system, are monitored, filtered (if required) through the building HVAC systems, and released through the unit vent. The condenser evacuation system is monitored and then discharged through the unit vent.

8.3. Solid Waste Management System (SWMS)

The SWMS is designed to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor solid waste. The SWMS processes both wet solid active waste and dry active waste for shipment to a licensed disposal site.

Primary functions of the SWMS include providing means by which spent resin, filters, etc. from the LWMS and primary letdown systems are processed to ensure economical packaging within regulatory guidelines, as well as handling dry, low activity wastes for shipment to a licensed disposal facility.

The SWMS is subdivided into the following subsystems:

- Spent resin transfer and storage system
- Filter handling system

- Dry active waste sorting and segregation system
- Spent resin drying system (SRDS)

Plant layout

9.1. Buildings and structures, including plot plan

The general arrangement of APR1400 was designed based on the twin-unit concept and slide-along arrangement with common facilities such as the compound building which includes radwaste building and access control building. The general arrangement of the buildings is schematically depicted in Figure 11. The auxiliary building which accommodates the safety systems and components surrounds the containment building. The auxiliary and containment buildings will be built on a common basemat. The common basemat will improve the resistance against seismic events and reduce the number of walls between buildings so that rebar and formwork cost can be reduced.

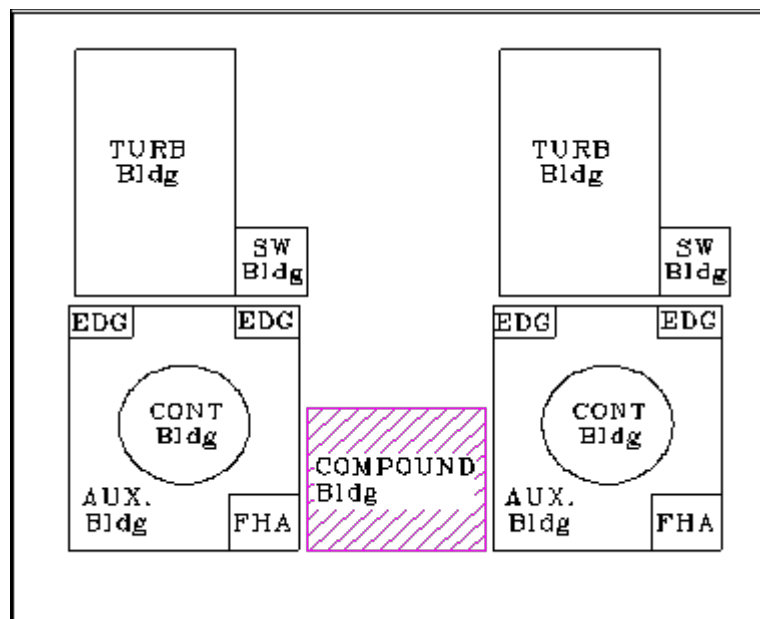


Figure 11. Plant General Arrangement

The layout is highly influenced by safety considerations, in particular, by the physical separation of equipment for the safety systems. The safety injection pumps are located in the auxiliary building in the four quadrants, one pump in each quadrant. This arrangement ensures the physical separation of the pumps, minimizing the propagation of damage due to fire, sabotage, and internal flooding. The emergency diesel generator rooms are also separated and located at the symmetrically opposite sides.

The building arrangement is also designed for the convenience of maintenance, considering accessibility and replacement of equipment. The internal layout of the containment, in particular, is designed to allow the one-piece removal of the steam generator. With proper shielding and arrangement of maintenance space, and careful routing of ventilation air flow, the occupational radiation exposure is expected to be lower than 1 man-Sievert a year.

The design strength of the buildings in the safety category, which are the containment and the auxiliary buildings, is sufficient to withstand the effects of earthquakes up to the safe shutdown earthquake (SSE) of 0.3 g.

9.2. Reactor building

The reactor building is the central building of the plant. APR1400 is a pressurized water reactor, and the reactor building essentially coincides with the containment building. Figure 12 shows a cross-sectional view of the reactor building including a part of the auxiliary building in the vertical direction with the arrangement of major equipment.

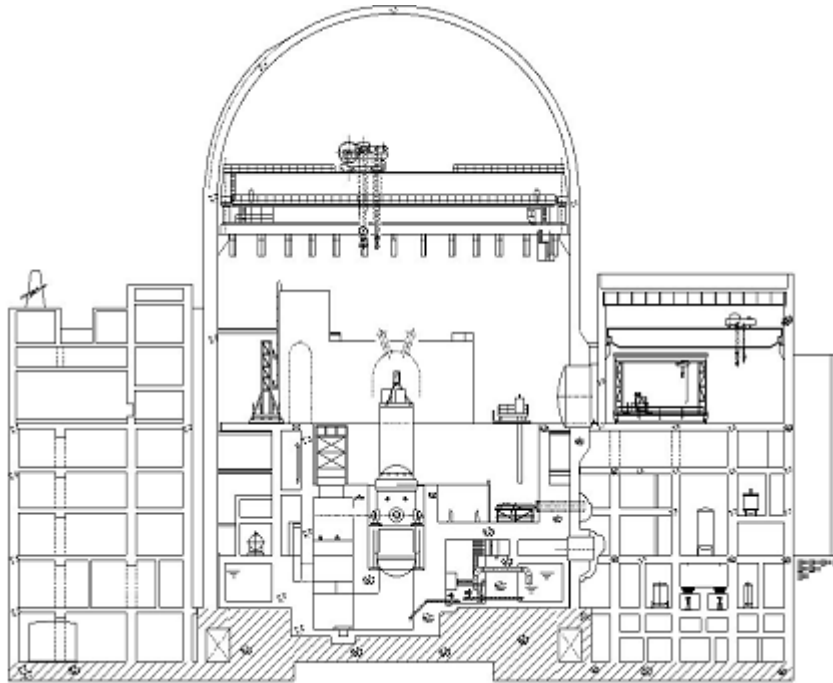


Figure 12. Cross-sectional view of the reactor building

9.3. Containment

The containment building is made of the post-tensioned cylindrical concrete wall with a steel liner, and reinforced concrete internal structures. The containment building houses a reactor, steam generators, pressurizer, reactor coolant loops, In-containment refuelling water storage tank (IRWST), and portions of the auxiliary systems. The containment building is designed to sustain all internal and external loading conditions which are reasonably expected to occur during the life of the plant. The containment building is on a common basement which forms a monolithic structure with the auxiliary building.

The interior arrangement of the containment building is designed to meet the requirements for all anticipated conditions of operations and maintenance, including new and spent fuel handling. There are four main floor levels in the containment: the lowest floor level, called the basement, the highest floor elevation, called the operation floor, and two (2) mezzanine floors in between the basement and operating floors. The two mezzanine floors are designed primarily of steel-supported grating.

The equipment hatch is at the operating floor level, and has an inside diameter of 7.8m (26 ft). This hatch size is selected to accommodate the one-piece replacement of a steam generator. A polar bridge crane is supported from the containment wall. The bridge crane has the capability to install and remove the steam generators. Personnel access to the containment is through two hatches, one located at the operating floor level and the other at the plant ground elevation.

The containment is a post-tensioned concrete cylinder with an internal diameter of 45.7m (150 ft) and a hemispherical top dome. There is no structural connection between the free standing portion of the containment and the adjacent structures other than penetrations and their supports. The lateral loads due to seismic and other forces are transferred to the foundation concrete through the structural concrete reinforcing connections.

9.4. Turbine building

The turbine building houses the turbine generator, the condenser systems, the preheater system, the condensate and feedwater systems, and other systems associated with power generation. The turbine building configuration is simplified for constructability, and the maintainability of the systems is improved by centralizing the condensate polishing system, separating the switchgear building, and rearranging the equipment hatches. There are four main floor levels referred to as the basement, ground level, operating level, and deaerator level.

The turbine building is classified as non-safety related. It has no major structural interface with other buildings except for a seismic interface with the connecting auxiliary building. It is designed such that under SSE conditions, its failure will not cause the failure of safety related structures. The turbine building is located such that the containment building is at the high pressure turbine side on the projection of the turbine shaft. This allows the optimization of the piping and cable routes to the nuclear island. This arrangement also minimizes the risk of damage to safety-related equipment by missiles from the turbine or the generator, in the event of an accident. The vibration problem which occurs during transient loading was minimized by moving the fresh water tank of the steam generator blowdown system to the auxiliary building.

In the APR1400 plant, the 52 inches last stage blade (LSB) of the LP turbine was taken into consideration for the building design. Other items reflected in the general arrangement design are as follows:

- Relocation of the TBCCW heat exchanger into the turbine building ;
- Relocation of the secondary sample room & lab to the compound building;
- Simplification of contour of the turbine building super structure.

9.5. Other buildings

9.5.1. Auxiliary building arrangement

The auxiliary building completely surrounds the containment building and is on the common basement which forms a monolithic structure with the containment building. The diesel generator room is built into the auxiliary building. To assure the safety and reliability, the auxiliary building is designed to enhance physical separation for the mitigation of internal flooding and fire propagation as shown in Figure 13.

The auxiliary building houses pumps and heat exchangers for the safety injection system and shutdown cooling system. Also, the auxiliary feedwater tanks and main control room are located in the auxiliary building. For the convenience of operation and maintenance, there is a staging service area in the auxiliary building for installation work in front of the equipment hatch of the containment.

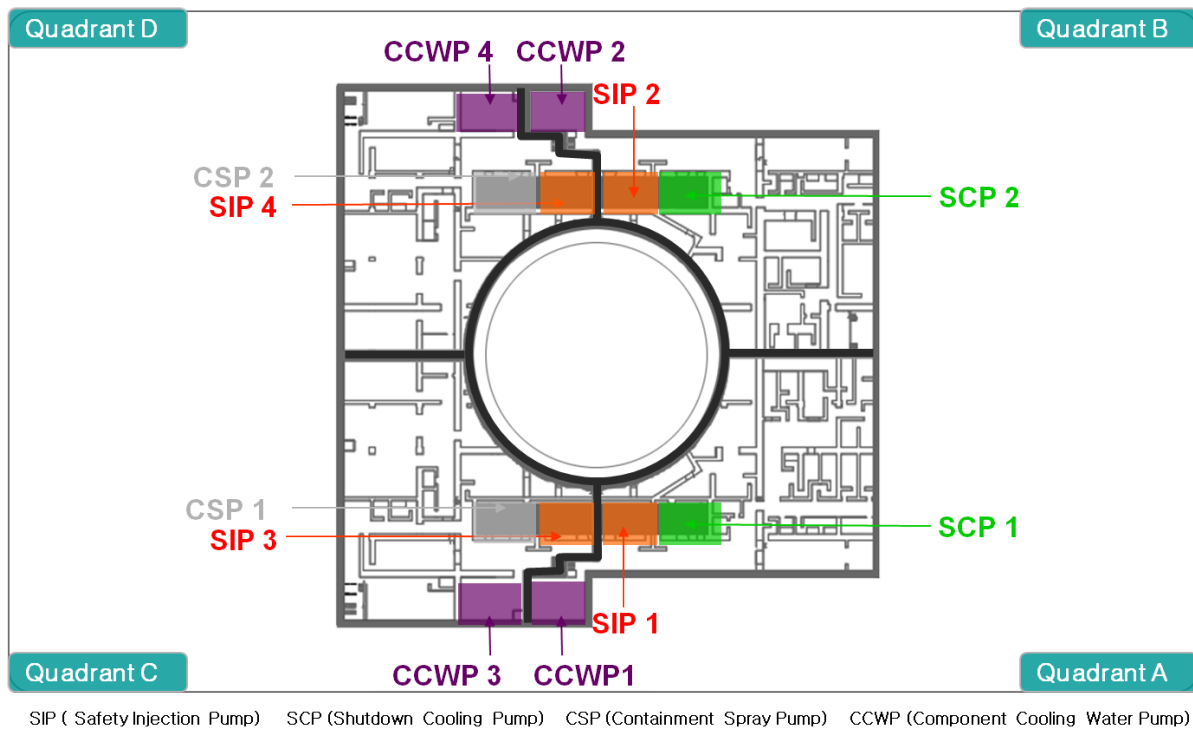


Figure 13. Quadrant arrangement of auxiliary building

The emergency diesel generator (EDG) area is located in the auxiliary building at the ground level. The fuel storage tanks are located on each side of the auxiliary building. The EDGs are arranged as separate entities with dedicated auxiliaries including air supply, exhausts, and cooling systems, so that they are independent of each other in all respects. The EDG areas are arranged to provide routine maintenance facilities and maintenance access space such that work on one EDG will not affect the operability of the other EDG.

9.5.2. Compound building arrangement

The compound building is an integrated building of the radwaste and access control buildings. The compound building consists of an access control facility, a radwaste treatment facility, a hot machine shop, and sampling facilities & lab. The compound building is designed to be shared between two units and is classified as non-safety related.

Radiation shielding is provided wherever required. For the building arrangement design, the protection against natural phenomena and the accommodation of associated environmental conditions were reflected to retain the spillage of potentially contaminated solids or liquids within the building. It has no major structural interface with other building, the access control is made at the ground floor in the compound building.

9.5.3. Switchgear building

The switchgear building is located in the vicinity of the turbine building and all the electrical switchgears are centralized in this area for the convenience of maintenance and efficiency of space allocation.

10.1. Operational performance

The APR1400 design is optimized to achieve the high operation performance and to enhance the convenience of maintenance by incorporating the following improvements. The lifetime design goal of availability factor for APR1400 is above 92%.

10.1.1 Improvements for In-Service Inspection (ISI)

The reactor head is manufactured as one piece by integrating the flange and upper shell based on the advanced forging capacity of manufacturer. In the conventional plant, the flange and upper shell are fabricated separately and welded each other. This improvement reduces the girth seam, for which In-Service Inspection (ISI) have to be performed over the lifetime. Also, the work platforms are installed to enhance the convenience of ISI for steam generators.

10.1.2 Enhanced refueling works

The fuel handling devices are improved to reduce the refueling time. In particular, a fuel transfer tube, connecting the containment building and the fuel handling area in the auxiliary building, is improved to be opened quickly by a remote control so that the radiation exposure is reduced. In addition, a temporary fuel storage rack can be installed inside of refueling pool to be used under an abnormal condition during the refueling. The design of In-Core Instrument (ICI) cable tray is improved to not install and disassemble for every refueling operation and the task related ICI cable is simplified.

10.1.3 Design features for reducing unplanned trip

To reduce the unplanned reactor trip, the core thermal margin is designed to be increased above 10 % through lowering the core outlet temperature and increasing the RCS coolant flow. In addition, the pressurizer volume relative to power is enlarged to enhance a capability for coping with the transients.

The turbine rotor is manufactured as one piece by forging to reduce the susceptibility of Stress Corrosion Crack (SCC). The turbine control system of Mark VI is improved to enhance the reliability and maintainability by the redundant design of controllers and strengthening the diagnostic functions and the visuals for operator. The vibration monitoring functions are improved by strengthening the self-diagnostic functions of the detectors and multi-directional measurements. In addition, earthquake-proof structures are installed to prevent the turbine trip by high vibration.

The exciter is adopted as a static excitation type to reduce the mechanical wearing. The Auto-Voltage Regulator (AVR) is placed in a dedicated room to minimize its malfunction by protecting it from heat and humidity. Also, the filtration abilities of the stator cooling water pipelines is strengthened to not heat up by the reduced coolant flow.

The feed-water flow control system is designed to control the feed-water flow automatically over full operation range and to operate three turbines driven main feed-water pumps during the normal power operation. When one main feed-water pump is tripped during the full power condition, other two main feed-water pumps recover the total feed-water flow to the nominal value of full power condition. This design reduces unnecessary power cutback and unplanned turbine trip.

10.2. Construction

A new construction schedule and constructability enhancement methods were developed based on the experience from the repeated OPR1000 construction. The power block foundation of APR1400 is seismically enhanced with applying 0.3g Safe Shutdown Earthquake (SSE) as a Design Basis Earthquake (DBE) and the reactor containment building (RCB) and the auxiliary building are built on a common base-mat. This design requires a highly increased mat size

and the concrete quantity. Thus, the construction method for this massive concrete structure is reviewed to meet the target duration. The common base-mat foundation is simplified as a flat type so that it may have benefits for the concrete works.

Modularization has been introduced to reduce the construction period and the cost. There are three types of modules as follows: the structural module, mechanical equipment module, and composite module. The structural modules are implemented for re-bar and liner plates. To expand the modular construction, the Steel-plate Concrete (SC) structure module, the mechanical equipment, and the composite module are under development. If the composite module is applied to all buildings in the Nuclear Power Plant, the construction period will be dramatically reduced to less than 40 months through the pre-fabrication at both the factory and the site.

For mechanical and electrical equipment and piping installation, it is recommended to increase the fabrication portion in the manufacturing shop. Approximately 80 items of APR1400 including auxiliary and RCB water chillers and pumps, feed water pumps and turbine drives, charging pumps, turbine building component cooling water heat exchangers, and condensers have been identified to be capable of modularization.

10.3. Design verification by 3D CAD system

To successfully accomplish the APR1400 development from conceptual design to construction, the entire plant design processes have been reviewed using a 3D CAD model. Design output was made with frozen model after the verification of the 3D CAD model. This design verification improved both the quality and the timeliness of the project design.

KHNP developed a new 3D CAD system called Tri-dimensional Design Verification System (TDVS) to improve and streamline the existing engineering process for the main 2D design work and subsidiary 3D review work as shown in Figure 14. In the TDVS, all engineers should use 3D models at every stage of the design process and review 3D models from various points-of-view, and produce deliverables based on the verified 3D models.

Each 3D design is controlled and managed through TDVS to implement design work procedures, and to share and distribute the correct information to the right people without delay.

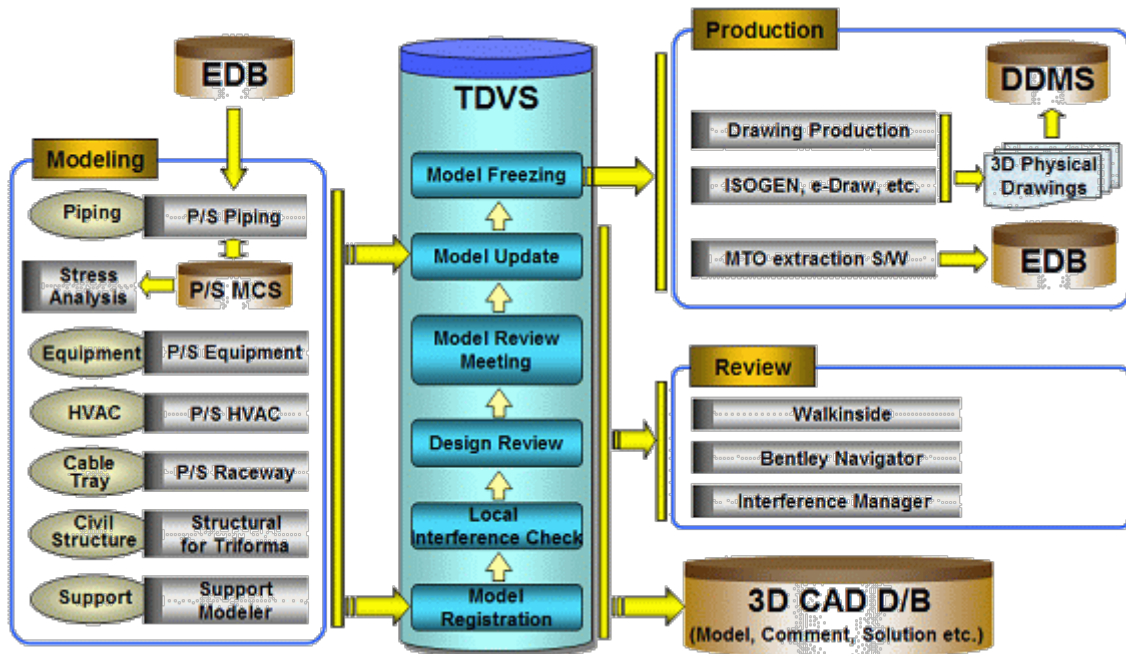


Figure 14. Tri-dimensional Design Verification System (TDVS)

Engineers of each discipline make 3D CAD models themselves. The 3D CAD system is connected with the Engineering Data Base (EDB) system. When the engineer routes the pipe with 3D CAD modeling software, the data for this pipe line, such as line number, line size, specifications, pressure and temperature, come from the EDB so engineers don't have to type the engineering information.

Reviewing a 3D CAD model is done with various exclusive types of software to verify design information and configuration. Engineers from each discipline can check physical interferences using exclusive interference detection software, which can detect interference automatically. Since lots of concrete structures are built in a nuclear power plant, it is very important to check the accessibility of the main equipment prior to construction. Therefore, engineers should use the animation program to make a scenario and check for the carrying in and out of equipment. The cyber character technique of computer games is applied to the plant design. An engineer enters the 3D integrated model as a cyber character, and can look for various components, as if an engineer is actually performing a walk-through inspection. The 3D CAD system is connected to the EDB and Drawing and Document Management System (DDMS). When an engineer reviews a 3D CAD model, an engineer can review the engineering information and related drawings.

Engineers can produce various deliverables such as a drawing and bill of material by using a verified 3D CAD model. After verifying the piping model, an engineer can create a piping plan and section drawing with exclusive software. The software performs hidden-line removal to convert the view from an incomprehensible wire-frame into a standard line-drawing. The software allows automatic annotation and dimensioning of the drawing. This Piping Modeling software extracts and converts piping data and transfers them to drawing generation software. The drawing generation software creates fully annotated piping fabrication isometric drawings with the related bill of material. After verifying the support model, an engineer can create support drawings. Drawing generation software creates plan, section, and isometric views on the drawing with the related bill of material. After verifying the piping model, the engineer can create a piping plan and section drawings with exclusive software. The software allows the semi-automatic annotation of the drawing. The designer points at items in the drawing and the software retrieves their label from the 3D CAD model and allows the designer to place them in the drawing.

10.4. Construction schedule

Along with many new construction methods, the modularization of reactor internals and the mechanical and structural composite modularization technologies have been applied to the construction of Shin-Kori 3 & 4, which is the first plant of APR1400. The modular construction method is applied to the Containment Liner Plate (CLP) and Stainless Steel Liner Plate (SSLP) to reinforce the steel and the structural steel module. This method is also applied to the fabrication of equipments such as the reactor internals and the condenser. The deck plate construction method is applied for the construction and the installation of mechanical and electrical equipments to be conducted simultaneously in the auxiliary building and compound building.

Thus, KHNP expects that the Shin-Kori 3 & 4 will be constructed within less than 51 months for the first unit and 45 months for the second unit. In addition, as shown in the reducible schedule design for new construction method application, KHNP expects that the succeeding plant including the third unit of the APR1400 can be constructed within 40 months.

Deployment status and planned schedule

11.1. Republic of Korea

The first APR1400 plant is under construction in Korea is Shin-Kori nuclear units 3&4 (SKN 3&4). Shin-Kori site, located right next to the existing Kori nuclear power plant unit 1~4 currently under operation, has space for 8 units. Shin-Kori units 1&2, which are 1000MWe OPR1000 design, are other plants under construction in Shin-Kori site with the commercial operation date of December 2010.

SKN 3&4 construction project has been launched by KHNP (Korea Hydro and Nuclear Power Company) with the major supply and construction contracts signed in August 2006. Site grading started in September 2007 and Construction Permit (CP) was issued by Korean government on April 15, 2008 on which power block excavation

started. The first concrete pouring for unit 3 was done on October 15, 2008 as scheduled, while unit 4 follows 1 year behind to unit 3. After a successful structural and building construction as well as main equipment manufacturing, the unit 3 reactor vessel was installed on July 15, 2010, which was 15 days ahead of schedule. Figure 15 is a photo taken on July 9, 2010 and shows site view with the unit 3 reactor vessel arrived for installation.

SKN 3&4 is scheduled with a 51 month construction period from first concrete to initial fuel loading considering the First-of-a-Kind plant of APR1400 design. The initial fuel loading is scheduled on January 1, 2013 for unit 3 after receiving Operation License (OL) from government and commercial operation in September 2013 and 2014 for units 3 and 4, respectively.



Figure 15. Shin-Kori units 3&4

The second APR1400 construction project has also been launched to construct another twin unit, Shin-Ulchin nuclear units 1&2 (SUN 1&2). As of August 2010, site preparation work and CP review are in progress with the expected CP by the end of 2010. SUN 1&2 is scheduled to be in commercial operation in June 2016 and 2017 for unit 1 and 2, respectively. SUN 1&2 design is basically the same as SKN 3&4 except I&C and MMIS equipments and the reactor coolant pump supplier. These equipments will be manufactured domestically based on the result of government funded research and development project for equipment localization.

According to the Korea's national electricity development plan, another 4 units of APR1400 will be constructed after SUN 1&2. Shin-Kori units 5&6 will be constructed right next to SKN 3&4 with the commercial operation target in 2018 and 2019 for unit 5 and 6, respectively. Currently, KHNP and suppliers are preparing for the construction project and contract negotiation. Finally, Shin-Ulchin 3&4 is on the list with 2020 and 2021 completion target.

In 2009, the Korean government established so called "Long-term national energy master plan by 2030". This plan call for further expansion of nuclear energy for electricity in Korea with a target of 41% of installed electricity generation capacity in 2030 by nuclear as compared to the current 26%. In terms of electricity generation, nuclear is planned to provide 59% of total electricity generation in 2030 as compared to the current 35%. According to the master plan, a new reactor design with improved safety and economics needs to be developed and, currently, APR+, a 1500MWe capacity advanced reactor based on APR1400 technology, is under development as a next fleet of power reactors in Korea as well as in global market. The earliest time for APR+ in operation would be 2022.

11.2. United Arab Emirates (UAE)

On December 27, 2009, UAE and Korea signed a historical prime contract for supplying 4 units of APR1400 to UAE, the first nuclear power plant in UAE as shown in Figure 16. Braka has been selected as the construction site on the Persian Gulf coast, west of Abu Dhabi. The Braka Nuclear Power Plant (BNPP) is scheduled to be in commercial operation in May 2017 for unit 1 and one for each following years till 2020 for unit 4. The preliminary safety analysis report (PSAR) for BNPP will be submitted to FANR (The Federal Authority for Nuclear Regulation of the UAE) by the end of 2010, and the CP is expected to be issued by June 2012. Major milestones for unit 1 include first concrete on October 2012, reactor vessel setting in July 2014, operation license and fuel loading by October 2016, and commercial operation on May 1, 2017.



Figure 16. Braka Nuclear Power Plant

The BNPP design will be based on SKN 3&4 as a reference plant, however, a number of design changes will also be implemented mainly related to the site specific environmental conditions that are different from those used in SKN 3&4 such as the seismic floor response spectra, ultimate heat sink temperature, ambient air temperature, and 50 Hz electrical grid. Also, the major safety and security related issues currently on effect in the global nuclear industry such as aircraft crash and cyber security need to be addressed in the BNPP design.

The plant owner in UAE is ENEC (Emirates Nuclear Energy Corporation) and the prime contractor for supplying BNPP in Korea is KEPCO (Korea Electric Power Corporation). KEPCO arranged all necessary subcontractors for suppliers and constructors.

11.3. Others

KEPCO/KHNP is under discussion with several countries from all over the world for a possible supply of APR1400 as well as OPR1000. Also, to be better prepared for the market specific safety and utility requirements, Korean nuclear industries are developing customer specific designs such as APR1000, advanced power reactor 1000MWe, an upgraded OPR1000 design, and EU-APR1400 for European countries. Finally, KEPCO/KHNP started the Design

Certification (DC) process with the U.S.NRC on APR1400 with the goal of docketing in mid 2012 and final certification in 2015. Currently, APR1400 design is under pre-application review process with the U.S. NRC.

References

Standard Safety Analysis Report (SSAR) for APR1400 is the major reference and additional information on APR1400 is available in KEPCO and KHNP websites.

Technical data

General plant data

Reactor thermal output	3983 MWth
Power plant output, gross	1455 MWe
Power plant output, net	1400 MWe
Power plant efficiency, net	35.1 %
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target >	90 %
Seismic design, SSE	0.3
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-5 /Reactor-Year
Large early release frequency <	10E-6 /Reactor-Year
Occupational radiation exposure <	1 Person-Sv/Ry
Operator Action Time	0.5 Hours

Nuclear steam supply system

Steam flow rate at nominal conditions	1130.83 Kg/s
Steam pressure	6.9 MPa(a)
Steam temperature	285 °C
Feedwater flow rate at nominal conditions	1134 Kg/s
Feedwater temperature	232.2 °C

Reactor coolant system

Primary coolant flow rate	20991 Kg/s
Reactor operating pressure	15.5 MPa(a)
Core coolant inlet temperature	290.6 °C
Core coolant outlet temperature	323.9 °C
Mean temperature rise across core	34.66 °C

Reactor core

Active core height	3.81 m
Equivalent core diameter	3.63 m
Average linear heat rate	18.38 KW/m
Average fuel power density	38.44 KW/KgU
Average core power density	100.9 MW/m ³
Fuel material	UO ₂
Fuel element type	Fuel rod
Cladding material	Zircaloy-4
Outer diameter of fuel rods	9.5 mm
Rod array of a fuel assembly	Square lattice , 16x16
Lattice geometry	Square
Number of fuel assemblies	241
Enrichment of reload fuel at equilibrium core	4.09 Weight %
Fuel cycle length	18 Months
Average discharge burnup of fuel	44.6 MWd/Kg
Burnable absorber (strategy/material)	Gd ₂ O ₃ -UO ₂
Control rod absorber material	B ₄ C or Inconel Slug
Soluble neutron absorber	Boron

Reactor pressure vessel

Inner diameter of cylindrical shell	4655 mm
Wall thickness of cylindrical shell	284 mm
Design pressure	17.2 MPa(a)
Design temperature	343.3 °C
Base material	SA508, Grade 3, Class 1
Total height, inside	14800 mm
Transport weight	573 t

Steam generator or Heat Exchanger

Type	Vertical U-tube with integral economizer
Number	2
Total tube outside surface area	15205 m ²
Number of heat exchanger tubes	13102
Tube outside diameter	19.05 mm
Tube material	SB-163 Alloy 690
Transport weight	832.7 t

Reactor coolant pump (Primary circulation System)

Pump Type	Vertical, Single Stage centrifugal
Number of pumps	4
Pump speed	1190 rpm
Head at rated conditions	109.7 m
Flow at rated conditions	7.67 m ³ /s

Pressurizer

Total volume	67.96 m ³
Steam volume (Working medium volume): full power	36.16 m ³
Steam volume (Working medium volume): Zero power	48.25 m ³
Heating power of heater rods	2400 kW

Primary containment

Type	Pre-stressed concrete
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Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	45.72 m
Dimensions - height	76.66 m
Design pressure	0.515 MPa
Design temperature	143.33 °C
Design leakage rate	0.15 Volume % /day

Residual heat removal systems

Active/passive systems	Active
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Safety injection systems

Active/passive systems	Active and Passive
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Turbine

Type of turbines	TC-6F
Number of turbine sections per unit (e.g. HP/MP/LP)	1/0/3
Turbine speed	1800 rpm
HP turbine inlet pressure	6.63 MPa(a)
HP turbine inlet temperature	282.2 °C

Generator

Type	Direct Driven (Water-Cooled)
Rated power	1690 MVA
Active power	1521 MW
Voltage	24 kV
Frequency	60 Hz
Total generator mass including exciter	713 t

Condenser

Type	Steam Surface
Condenser pressure	5.08 kPa

Feedwater pumps

Type	TBN Driven
Number	3
Pump speed	4570 rpm
Head at rated conditions	609.6 m
Flow at rated conditions	0.902 m ³ /s