APR1000 – Advanced Power Reactor 1000

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1. Introduction

1.1 Generalities

The Advanced Power Reactor 1000 (APR1000) is an evolutionary 1000 MWe-class Generation III+ Pressurized Water Reactor (PWR). APR1000 is based on the proven design of Advanced Power Reactor 1400 (APR1400) and Optimized Power Reactor 1000 (OPR1000) and incorporates advanced design features of the Advanced Power Reactor Plus (APR+) and the European Advanced Power Reactor (EU-APR) to enhance plant safety with outstanding performance. The APR1400 and APR+ technologies have been proven through licensing approval, construction and operation. Currently, two units of APR1400 are in operation and eight units are under construction. The APR1400 has been certified by the European Utility Requirements (EUR) organization in November 2017 and also acquired Design Certification by the US NRC in August 2019. The APR+ acquired the standard design approval from Korean nuclear regulatory authority in November 2017. Applying the same regulatory requirements and technical standards as in the APR1400, APR1000 also has been developed to comply with the up-to-date European requirements such as Western European Nuclear Regulators Association (WENRA) and International Atomic Energy Agency (IAEA) as well as Code of Federal Regulations and Regulatory Guides of the U.S.

1.2 Proven Technology and Design features

In the 1980s, the Republic of Korea accomplished nuclear technology self-reliance through the development and construction of the OPR1000. Currently (as of August 2019), a total 12 units of OPR1000 are in operation with outstanding performance.

On the basis of the OPR1000 technology, the APR1400 has been developed by incorporating the operation experience of the OPR1000 series plants and adopting advanced design features. The first APR1400 construction project, Shin-Kori Nuclear Power Plant Units 3 and 4 (SKN 3&4), was stared in 2008, and Operating Licenses (OL) from Korean nuclear regulatory authority have been issued in October 2015 for Unit 3 and February 2019 for Unit 4. Commercial operation of Unit 3 started in December 2016, which places the SKN 3&4 as the first commercially operating Generation III reactor in the world.

APR1400 is internationally recognized for its safety and performance. In 2017, European Utility Requirements (EUR) organization certified the APR1400 design for its compliance with the European Utility Requirements (EUR). In 2019, the USNRC issued final Design Certification (DC) for the APR1400 design. Currently, 4 units in Korea and 4 units in United Arab Emirate (UAE) are under construction.

APR1000 has been developed based on the proven design of the APR1400 and the OPR1000 and by incorporating advanced design features of the APR+ and the EU-APR to enhance plant safety with outstanding performance. Therefore the APR1000 benefits form the proven technology gained through repeated construction and vast operation experience of the OPR1000 and APR1400.

Since APR1000 has been evolved from OPR1000, the basic configuration of the nuclear steam supply systems for both are the same as shown in Figure 1. APR1000 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 well proven design concept through highly reliable operation records. Moreover, the general arrangement of the APR1000 has been improved with the reflection of operation and construction experiences of the OPR1000 and the APR1400.

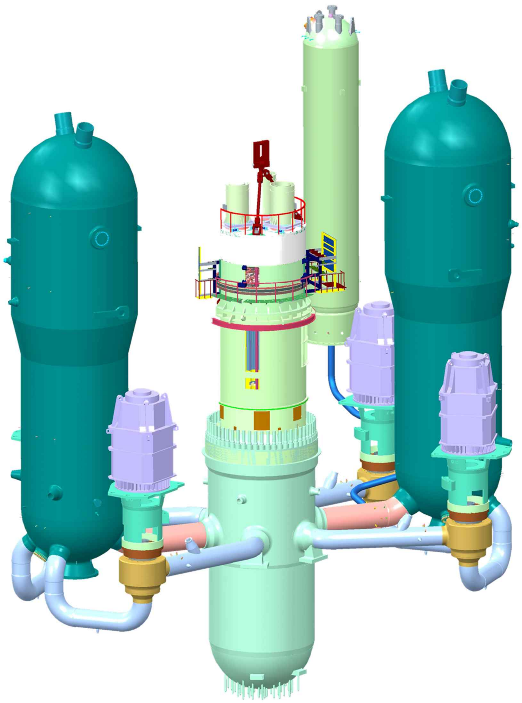


Figure 1. The APR1000 Reactor Coolant System Configuration

APR1000 was developed with adopting advanced safety features of the APR+ and the EU-APR as follows;

- Direct Vessel Injection (DVI) with Emergency Core cooling Barrel Duct (ECBD) of Safety injection system
- Fluidic Device (FD) in Safety Injection Tank
- In-containment Refueling Water Storage Tank (IRWST)
- Passive Auxiliary Feedwater System (PAFS)
- Pilot-Operated Safety Relief Valve (POSRV)
- Fully digitalized I&C and HMIS
- Passive Ex-vessel Corium retaining and cooling System (PECS)

The main design philosophy of the APR1000 incorporates several important features, such as the enhancement of safety, the utilization of proven technologies, the creation of a common design adaptable to each country where it is utilized, and a stable construction cost comparable to that of currently operating PWRs. Achieving a higher level of plant safety is an important goal among the various development policies. The major design requirements for safety and performance goals set for APR1000 are listed in Table 1.

Table 1. The APR1000 design requirement for safety and performance goals

General Requirement	Performance requirements and economic goals
Type and capacity : PWR, 1000 MWe Plant lifetime: 60 years Seismic design: SSE 0.3g Safety goals: Core damage frequency < 1.0^{-5}/RY Large release frequency < 1.0^{-6}/RY Occupational radiation exposure (ORE) < 0.5 man-Sv/RY	Plant availability: more than 90% Unplanned trips: less than 0.2 / RY Refueling interval: 12 ~ 24 months Operability: Fully Digitalized MMIS Construction period: Less than 57 months

2. *Description of the Nuclear Systems*

2.1 *Primary circuit and its main characteristics*

The primary loop configuration of the APR1000 is same as the OPR1000, which has two reactor coolant loops. The nuclear steam supply system is designed to operate at a rated thermal output of 2,825 MWt to produce the gross electric power output of around 1,000 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 SGs and the four RCPs are arranged symmetrically.

The steam generators are located at a higher elevation than the reactor vessel for natural circulation. The steam generators provide an interface between the reactor coolant (primary) system and the main feedwater and steam (secondary) system. The steam generators are vertical U-tube heat exchangers with an integral economizer in which heat is transferred from the primary system to the secondary system. Reactor coolant is prevented from mixing with the secondary steam by the steam generator tubes and the steam generator tube sheet, making the RCS a closed system, thus forming a barrier to the release of radioactive materials from the reactor core to the containment building and the secondary system. For vent and drain, the elevation of the PZR and the surgeline is higher than that of reactor coolant piping. A schematic diagram of arrangements and locations of the primary components are in Figure 2.

Overpressure protection for the reactor coolant pressure boundary is provided by three Pilot Operated Safety Relief Valves (POSRVs) which are connected to the top of the pressurizer. The POSRVs are designed to be performed the functions of the RCS overpressure protection during the design bases accidents and the manual depressurization in the cases of beyond design basis accidents such as a Total Loss Of FeedWater (TLOFW) event. Moreover, additional rapid depressurization valves are installed on the top of the pressurizer to prevent Direct Containment Heating (DCH) during severe accidents. Overpressure protection for the secondary side of the steam generators is provided by spring-loaded safety valves located in the main steam system upstream of the steam line isolation valves.

The main design concepts of the RCS are given below:

The major components of the Reactor Coolant System (RCS) are designed to withstand the forces associated with the design basis pipe breaks in combination with the forces associated with the safe shutdown earthquake and normal operating conditions.

The coolant temperature in the Reactor Vessel (RV) head region is reduced as low as that of the coolant in the cold leg. The reduced temperature of the RV head helps to alleviate the Primary Water Stress Corrosion Cracking (PWSCC) concerns for Control Element Assebly (CEA) guide tubes penetrating the RV upper head.

Design changes are made to the reactor internals to increase the cold coolant flow from the RV downcomer to the upper head region by optimizing the coolant flow path areas into and out of the upper head.

Integrated Head Assembly (IHA) cable routing of the APR1000 is improved from the OPR1000 design. The OPR1000 IHA cables are routed from RDP through the integrated platform which is outside the IHA. The APR1000 IHA cables are routed through simplified cable bridges instead of the separate OPR1000 integrated platform.

The Safety Injection System (SIS) is designed to improve the system operability and reliability. One of the improved design features is the passive fluidic device in Safety Injection Tank (SIT), which regulates the injection flow rate effectively in the event of LOCA. The fluidic device initially delivers safety injection water with high flow rate for a certain period of time, after which the flow rate is reduced.

The Leak Before Break (LBB) scheme is applied to the high energy piping such as RCS piping, the PZR surgeline, and SI/SCS piping.

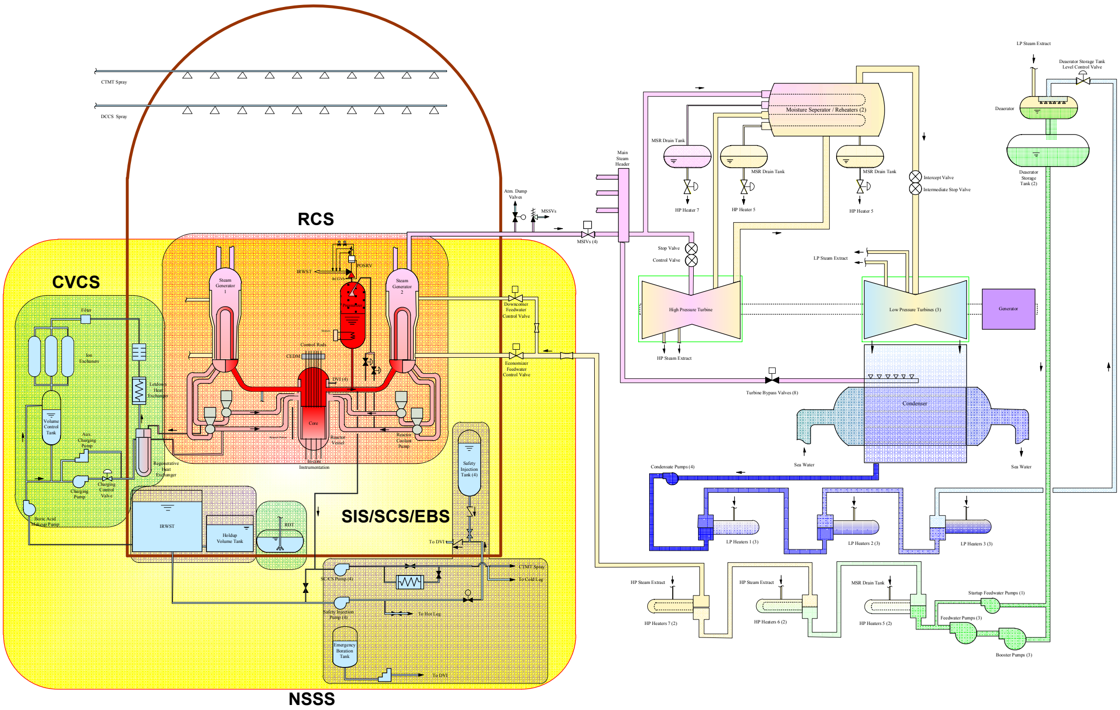


Figure 2. Schematic diagram of primary components

2.2 Reactor core and fuel design

The reactor core of APR1000 is designed to generate 2,825 MW thermal power with an average volumetric power density of 96.26W/cm^3. The reactor core consists of 177 fuel assemblies made of fuel rods containing UO_2 fuel. The number of Control Element Assemblies (CEAs) used in the core is 73 in which 65 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 12 months through 24 months with a maximum rod 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.

Loading mixed oxide (MOX) fuel up to 30% core is considered in the core design. Eight additional reserve CEAs are installed to increase the reactivity control capability, if necessary, for MOX fuel loadings. In addition, the APR1000 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 Gd_2O_3-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 HIPER16^{TM} fuel design has the capabilities of a batch average discharge burn-up as high as 65,000MWD/MTU and the HIPER16^{TM} design has increased overpower margin in comparison with the previous fuel design (PLUS7^{TM}). A schematic diagram of fuel assembly is shown in Figure 3. The HIPER16^{TM} mid-grid design has high through-grid dynamic buckling strength for the enhanced seismic performance. The top nozzle has easy reconstitutability features and holddown spring force was optimized to reduce the fuel assembly bow. The guide tube has high seismic load capability and inner dashpot tube to minimize the fuel assembly bow. The debris filtering and capturing features are implemented in the bottom grid by combining the debris filtering bottom grid and the long bottom end plug to reach the target of zero fuel failure. The bottom nozzle has a low pressure drop features with rectangular flow holes.

The integrity of HIPER16^{TM} fuel has been enhanced by increasing the fretting wear resistance and debris filtering efficiency. The optimized holddown spring force will reduce the HIPER16^{TM} fuel assembly bow. The safety of HIPER16^{TM} fuel has been enhanced by increasing the seismic performance which is related to the spacer grid crush strength and dynamic stiffness.

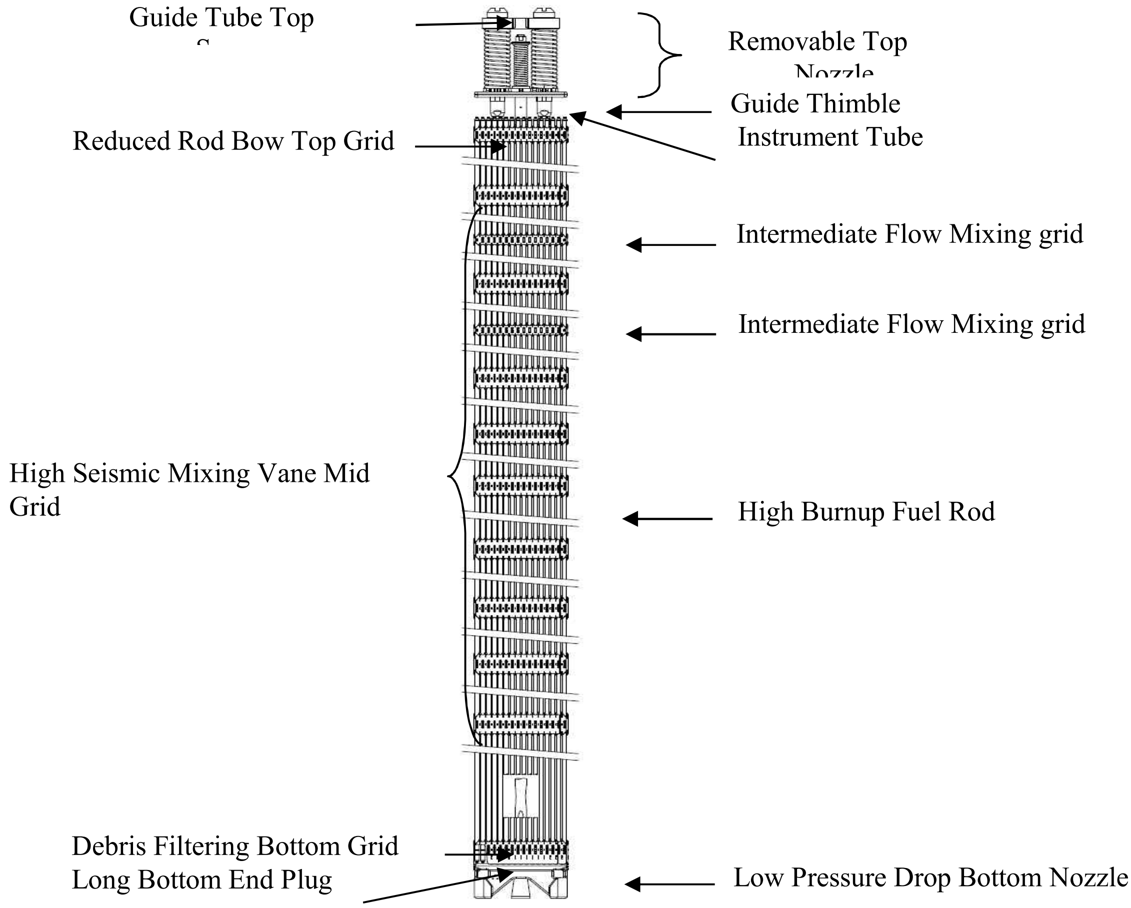


Figure 3. HIPER16™ Fuel Assembly

2.3 Fuel handling and transfer systems

The fuel handling system is designed for the safe and rapid handling and storage of fuel assemblies from the receipt of new 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 new 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 new fuel elevator, the spent fuel storage racks, and the spent fuel shipping cask areas.

The underwater transfer of fuel assemblies provides a transparent radiation shield, as well as a cooling medium for the removal of decay heat. Boric acid is added to the Spent Fuel Pool (SFP) water in the quantity required to assure subcritical conditions.

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 4. The reactor pressure vessel contains internal structures, core support structures, fuel assemblies, control rod assemblies, and control and instrumentation components.

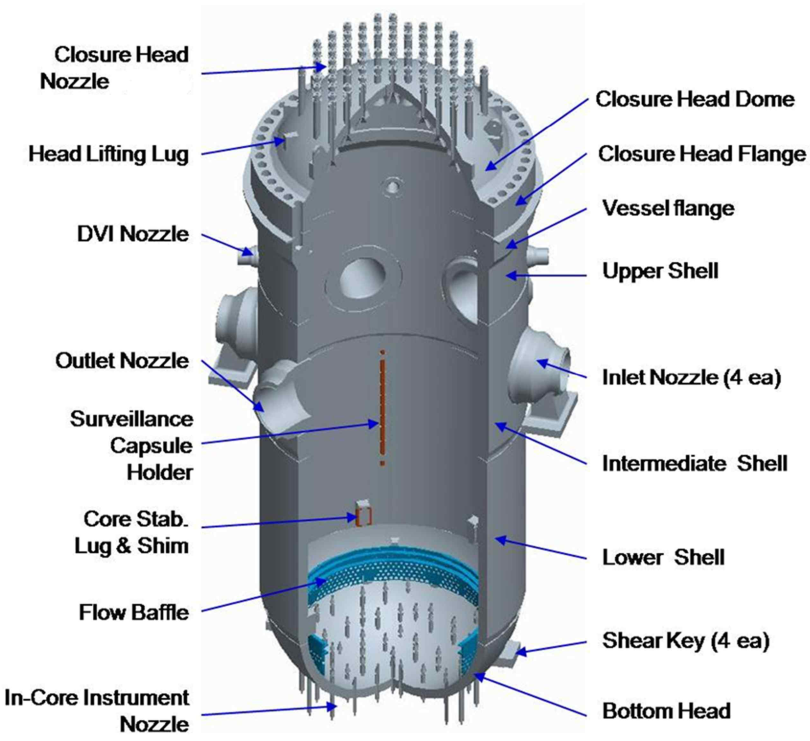


Figure 4. Reactor pressure vessel of the APR1000

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

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 material, resulting in an increase of brittle fracture toughness.

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 (upper, intermediate, and lower), the bottom head forging and vessel flange forging are joined together by welding. Also, four inlet nozzle forgings, two outlet nozzle forgings, and forty five 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. 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.

2.4.2 Reactor internals

The reactor internals consist of the core support structures, which include the core support barrel assembly, upper guide structure assembly and lower support structure, and the internal structures. The core support barrel assembly is designed to support and orient the reactor core fuel assemblies and control element assemblies, and to direct the reactor coolant to the core. The ECBDs are installed outside of core support barrel aligned with DVI nozzles. 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 assembly and the upper guide structure assembly are supported at its upper flange from a ledge in the reactor vessel flange. All reactor internals are manufactured with austenitic stainless steel except for the holddown ring, which is made of high-tension stainless steel. The holddown ring absorbs vibrations caused by the axial load of internal structures.

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

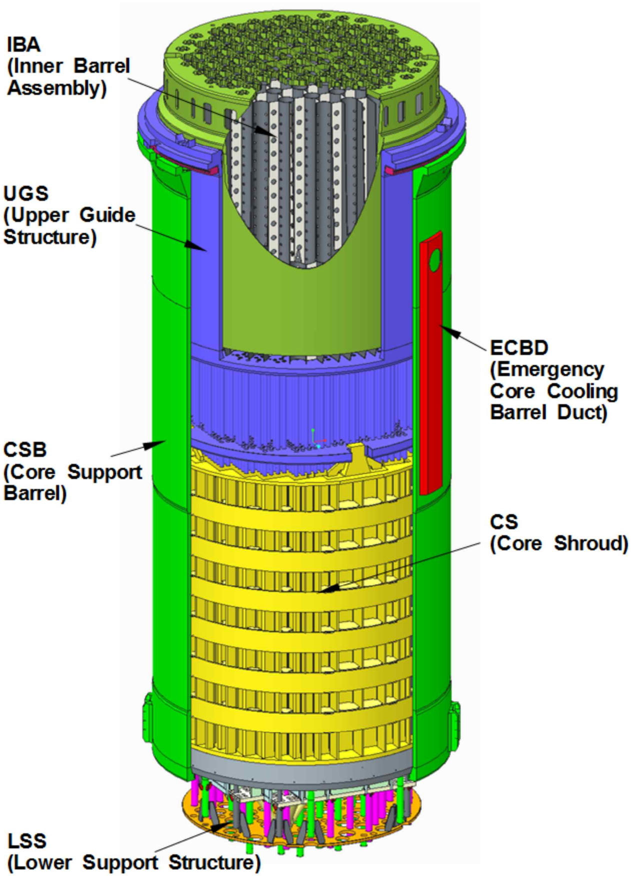


Figure 5. Reactor Vessel Internals of the APR1000

2.4.3 Steam generators

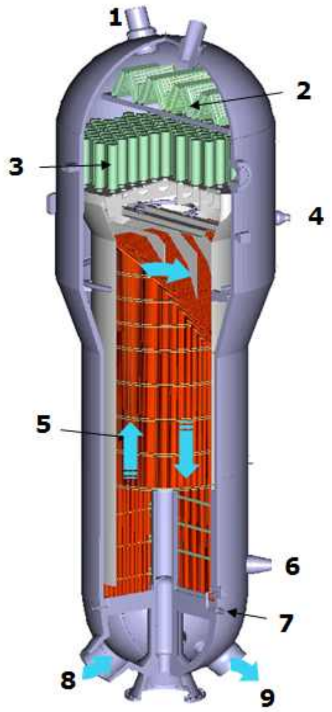
The APR1000 has two Steam Generators (SGs) for the transfer of heat from the primary system to the secondary system. One steam generator is located in each loop. Each steam generator is a recirculating type vertical U-tube heat exchanger with an integral axial flow economizer, which operates with the RCS coolant in the tube side and secondary coolant in the shell side as shown in Figure 6. The two SGs are designed to transfer the heat of 2,825 MWt from the RCS to the secondary system. The secondary system produces steam to drive the turbine-generator, which generates the gross electrical power of 1,000 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 is 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 tubesheet sludge lancing, each SG has manways in the cold leg and hot leg side of primary system. Also, two manways of the secondary side, an internal hatch over the top of the tube bundle, two handholes at the tubesheet region are provided.

The APR1000 SG uses Alloy 690 as tube material. The upper tube support bar and plate are designed to prevent SG tubes 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 20%, below which all feedwater is supplied to the donwncomer nozzle.



1. Steam outlet nozzles
2. Steam dryers
3. Moisture separator
4. Downcomer feedwater nozzle
5. Tube bundle
6. Economizer feedwater nozzle
7. Blowdown nozzles
8. Reactor coolant inlet nozzles
9. Reactor coolant outlet nozzles

Figure 6. 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 surge nozzle, main spray nozzle, auxiliary spray nozzle, Emergency Reactor Depressurization System (ERDS) nozzle, Pressurizer Pilot Operated Safety Relief Valve (POSRV) nozzles, and pressure and level instrumentations. A man-way is provided on the top head for access for inspection of the pressurizer internal. 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 is to maintain the operating pressure and temperature of the reactor coolant system. Three POSRVs 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 IRWST.

2.4.5 Integrated Head Assembly (IHA)

The reactor vessel upper head area consists of the reactor vessel closure head, CEDMs with cables and cable supports, head lift rig, cooling air system with cooling shroud and fans, seismic supports, and missile shield. These components are usually disassembled, separately stored, and reassembled during every refuelling outage. The IHA is a structure to combine and integrate all the structures around the reactor vessel closure head area into one assembly as shown in Figure 7.

The main 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 maintenance activities such 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.

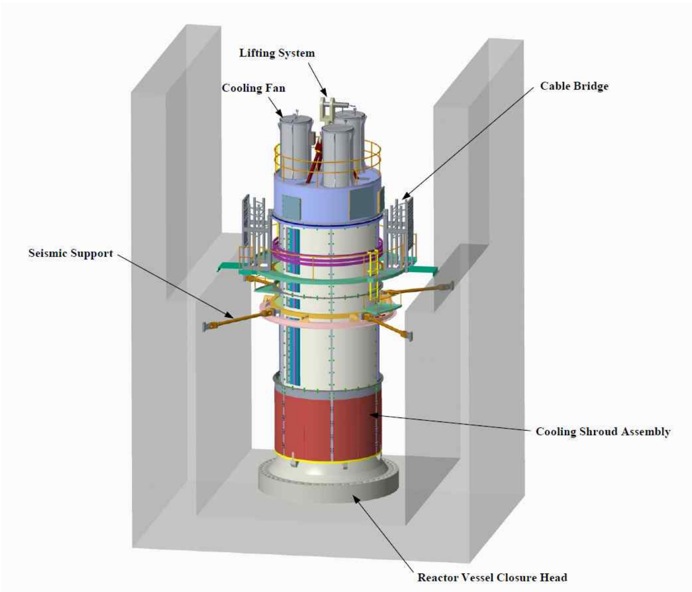


Figure 7. Integrated Head Assembly

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 secondary side of 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 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) concept is adopted for the piping system, since the pipe whip restraint and the support for 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 concept is applied to the main coolant lines, surge lines, pipes in the shutdown cooling system and the safety injection system. The application of LBB reduces the redundant supports of the pipe in the related 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 APR1000 is not required to perform safety functions such as safe shutdown and accident mitigation. This system is basically for the normal 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 the normal and transient conditions. Two centrifugal charging pumps, one reciprocating pump and a flow control valve provide required charging flow. For normal operation, only one charging pump is used to supply the required minimum flow.

The letdown flow from the reactor coolant system passes through the regenerative and letdown heat exchangers, 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.

2.5.2 Component Cooling Water System (CCWS)

The CCWS is a closed loop cooling water system that, in conjunction with the Essential Service Water System (ESWS) and the UHS, removes heat generated from the safety and non-safety components connected to the CCWS. Heat transferred by these components to the CCWS is rejected to the ESWS via the Component Cooling Water (CCW) heat exchangers. The CCWS consists of four (4) separate, closed loops, safety trains. Each train of the CCWS is independent from each other without cross-connection and capable of supporting 100% of the cooling functions required for a safe shutdown and the maximum heat load during accidents.

2.5.3 Reactor Coolant Gas Vent System (RCGVS)

The RCGVS provides a safety-grade means of venting non-condensable gases remotely from the reactor vessel head and the pressurizer steam space during post-accident conditions, when large amount of non-condensable gases may collect in these high spots. The RCGVS is designed to vent gas to reactor drain tank (RDT) or IRWST.

2.5.4 Steam Generator Blowdown System (SGBS)

The SGBS has both non-safety function of the normal and high capacity blowdown and safety function of the Emergency BlowDown (EBD). For normal operation, the SGBS is provided to control the steam generator secondary side water quality and to detect a leak or failure of a steam generator tubes. The SGBS consists of two (2) subsystems, the Blowdown Subsystem (BDS) and Wet Layup Subsystem (WLS). In the BDS, the EBD system consists of two (2) EBD isolation valves and four (4) SGBS spargers located in the IRWST per each SG.

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

The APR1000 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 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% turbine power level without causing any reactor trips or safety system actuations.

In case of turbine generator trip from any power levels including full power, the APR1000 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, the APR1000 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 three main feedwater pumps in service.

2.7 Fuel Cycle and Fuel Options

APR1000 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 30% of core. The spent fuel treatment plan is beyond the plant design scope.

3. Description of Safety Concept and Systems

3.1 Safety requirements and design philosophy

Safety is a requirement of paramount importance for nuclear power. One of the APR1000 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 established licensing design basis to meet the licensing rules, the APR1000 is designed to meet these safety goals with securing an additional safety margin to protect the owner's investment as well as 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.
- The Large Release Frequency (LRF) shall be less than 10^{-6}/RY.

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 calls for the incorporation of design features to cope with severe accidents as well as design basis accidents.

The safety analysis of the APR1000 is performed to demonstrate the performance of the components, its operating systems, and its safety systems under a wide spectrum of anticipated events and postulated accidents. The safety analysis based on deterministic methods is complemented by a Probabilistic Safety Assessment (PSA). The deterministic method starts with the step of specifying the scenarios. This step is composed of initiating events and component failures that are assumed to occur. The scenarios are specified to include even highly unlikely events, and acceptance criteria are specified by applying regulatory requirements. For a probabilistic assessment, realistic quantitative information as regards the occurrence of various events and the conditions and the reliability of components are used to assess the probability of failure of the operational and safety systems of the plant.

The design characteristics for the severe accident mitigation of the APR1000 are intended to prevent or to mitigate containment over-pressurization, hydrogen control issues, direct containment heating and steam explosions, and equipment survivability issues. These severe accident mitigation features of the APR1000 are reviewed along with their effects on the phenomenological response of the plant against the severe accidents to assess how adequately they satisfy the licensing requirements.

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 the 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 Engineered Safety Features (ESF) systems of the APR1000 are designed to be a hybrid system in which the active and passive systems perform necessary safety functions. The major safety systems are the safety injection system, safety depressurization and vent system, refuelling water storage system, shutdown cooling system, auxiliary feedwater supply system, and containment spray system.

The ESF systems provide protection in the highly unlikely event of an accidental release of radioactive fission products from the RCS, particularly as the result of a LOCA. The ESF systems consist of containment system, safety injection system, containment spray system. The containment provides missile shielding for safety class equipment in order to limit the consequences of a failure of the reactor coolant system pressure boundary. Containment isolation features provide for automatic containment isolation upon receipt of a containment isolation actuation signal, containment spray actuation signal, or containment purge isolation actuation signal. In order to maintain the function of the containment in the event of an severe accident resulting in a degradation of reactor core, specific design features incorporated in the APR1000 design are external reactor vessel cooling, reactor pressure vessel thermal insulation, cavity flooding system, and reactor cavity design.

3.2.1 Safety Injection System (SIS)

The SIS or Emergency Core Cooling System (ECCS) provides core cooling in the event of a loss of coolant accident. The Safety Injection System is designed to supply sufficient cooling to prevent significant alteration of core geometry, to preclude fuel melting, to limit the cladding metal-water reaction, and to remove the energy generated in the core for an extended period of time following a loss of coolant accident. The APR1000 SIS utilizes four safety injection pumps to inject borated water into the Reactor Vessel as shown in Figure 8.

The main design concept of the SIS is simplification and diversity to achieve higher reliability and better performance. The safety injection system is comprised of four independent mechanical trains and four electrical divisions. Each train has one active Safety Injection

Pump (SIP) and one passive Safety Injection Tank (SIT) equipped with a Fluidic Device (FD).

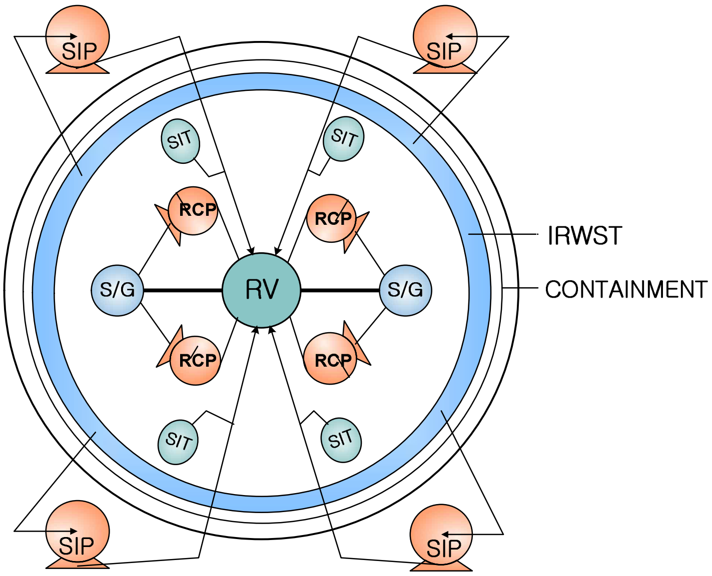


Figure 8. Safety Injection System

3.2.2 Fluidic Device (FD)

The APR1000 uses a passive flow regulator, that is, the FD is installed in the SIT. 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 creates low vortex resistance and a high flow rate. When 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 9. Thereby, this design feature enables the APR1000 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 Large-Break LOCA (LBLOCA).

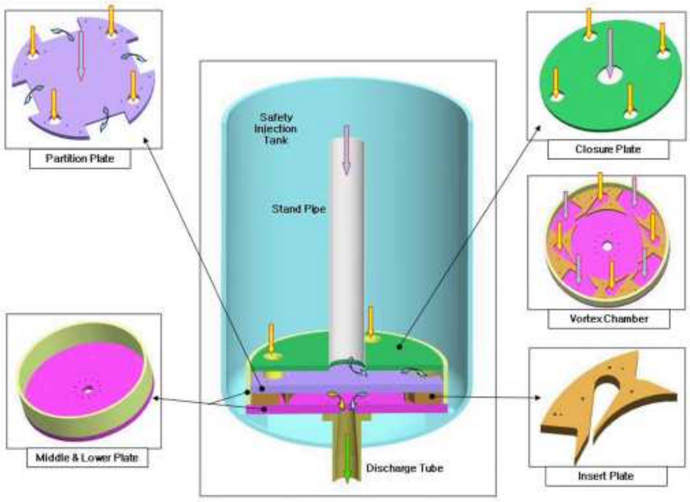


Figure 9. Fluidic Device in SIT

3.2.3 Shutdown Cooling System (SCS)

The SCS 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. The shutdown cooling system consists of four independent subsystems, each utilizing shutdown cooling/ Containment Spray Pump to circulate coolant through a shutdown cooling/Containment heat exchanger.

After initial heat rejection through the SGs to the condenser or atmosphere, the SCS is put into operation at 176.7 ^{o}C and 28.82 kg/cm^2 A. The initial cool-down is accomplished by heat removal to the secondary side of the steam generators and then by releasing steam via the steam bypass system or atmospheric dump valves.

During the shutdown cooling process, the reactor coolant flows out of the Reactor Coolant System through the shutdown cooling nozzles located on each hot leg. Reactor coolant is circulated by the shutdown cooling/containment spray Pump through the shutdown cooling heat exchangers and then returned to the reactor coolant system through the cold leg piping. The cool down rate is controlled by adjusting the flow through the heat exchangers with a throttle valve on the discharge of each heat exchanger.

3.2.4 Passive Auxiliary Feedwater System (PAFS)

The PAFS removes residual heat, via the SG through the Passive Condensation Heat Exchanger (PCHX) which condenses steam on the tube side. Condensate is delivered to the SG economizer feedwater line by using gravity force in the event where the main feedwater system is unavailable.

The PAFS consists of two (2) mechanical trains, which are physically separated each other. Each division is connected from each main steam supply line to downstream of each steam generator economizer check valve. Each train has two (2) Passive Condensation Heat Exchangers (PCHXs) and one (1) Passive Condensation Cooling Tank (PCCT) with 100 % heat removal capacity, two (2) condensate return lines and steam supply lines, associated valves, and instrumentation and controls. Each of the PCCTs is located on the top of the Auxiliary Building and contains two (2) 50% capacity PCHXs.

3.2.5 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 for steam from the pressurizer to IRWST.

3.2.6 Containment Spray System (CSS)

The CSS is a safety system to reduce primary containment pressure and temperature, during DEC-A except the Loss of Ultimate Heat Sink (LUHS) as well as DBAs such as a Main Steam Line Break (MSLB) or Loss Of Coolant Accident (LOCA) and to remove fission products from the primary containment atmosphere following a LOCA. These functions of the CSS are accomplished by spraying borated water to the primary containment atmosphere and the water is supplied from IRWST via SC/CS pumps and SC/CS heat exchangers. The SC/CS pumps start upon the receipt of a Safety Injection Actuation Signal (SIAS) or a Containment Spray Actuation Signal (CSAS). The pumps discharge water from IRWST to their respective spray nozzle headers through the SC/CS heat exchangers and the spray header isolation valves, and finally into the primary containment atmosphere.

3.2.7 Spent Fuel Pool Cooling and Cleanup System (SFPCCS)

The SFPCCS consists of the SFP cooling system and the SFP cleanup system. The SFPCCS is designed to remove the decay heat generated by the stored spent fuel assemblies from the spent fuel pool water, and to purify the contents of the refueling pool during refueling operations. Pool cooling is accomplished by taking heated water from the pool, pumping it through a heat exchanger, and returning the cooled water to the pool. The SFP cleanup system is also used to clarify and purify the spent fuel pool, fuel transfer canal, IRWST, and refueling pool water.

3.3 Diverse Safety Features

3.3.1 Emergency Boration System (EBS)

The EBS is designed to inject highly concentrated borated water into the RCS following an Anticipated Transients Without Scram (ATWS) to shutdown the reactor to a subcritical condition.

The EBS consists of two (2) mechanically separated trains, one Emergency Boration Tank (EBT), and associated valves, piping and instrumentation. Each train contains one Emergency Boration Pump (EBP), and associated suction and discharge paths.

3.3.2 Diverse Containment Spray System (DCSS)

The DCSS is provided to decrease containment pressure and temperature in case of DEC-B by condensing the steam generated in the containment and to reduce the potential for further pressure increase by removing decay heat from the containment atmosphere and from the core debris in the reactor cavity. The DCSS has the capability to remove the iodine and particulate fission product inventories in the containment atmosphere that may result from the DEC-B. In addition, the DCSS provides the means of long-term cooling to maintain the plant in a safe state in the event of DEC-A when the SCS or its supporting systems such as CCWS, ECWS and ESWS are not available.

The DCSS consists of one (1) independent containment spray train with one (1) DCS pump, one(1) DCS heat exchanger, one(1) DCS pump mini-flow heat exchanger, spray nozzles, and associated valves and piping inside the primary containment, and is designed to ensure the heat removal from the containment during the DEC-B.

3.3.3 Diverse Component Cooling Water System (DCCWS)

The DCCWS removes the heat from the components required for mitigation of Design Extension Conditions (DECs). The DCCWS consists of one (1) independent train with providing 100% of the cooling capacity required for mitigation of DECs. The DCCWS rejects the heat to Diverse Essential Service Water System (DESWS) through the Diverse Component Cooling Water (DCCW) heat exchangers.

3.3.4 Diverse Spent Fuel Cooling System (DSFPCS)

The DSFPCS provides removal of decay heat generated by the spent fuel assemblies stored in the Spent Fuel Pool (SFP). The DSFPCS maintains SFP water temperature uniformly and provides adequate cooling to the spent fuel in DECs. Cooling is accomplished by taking heated water from the SFP, pumping it through a Diverse Spent Fuel Pool (DSFP) cooling heat exchanger, and returning the cooled water to the SFP. The DSFPCS is capable of maintaining the SFP water at a low enough temperature to prevent excessive vapor formation or evaporation from the water surface.

3.3.5 Diverse Essential Service Water System (DESWS)

The DESWS provides heat transfer from the DCCWS to the UHS during the DECs. The transfer of heat is accomplished by circulation of DESW through the cold side of the DCCW heat exchangers. The DESWS consists of one (1) independent train with providing 100% of the cooling capacity required for mitigation of DECs.

The DESWS consists of one (1) independent train with one(1) DESW pump, one (1) DESW debris filter, one (1) DCCW heat exchanger, and associated piping, valves, and instruments.

3.3.6 Diverse Essential Chilled Water System (DECWS)

The DECWS is designed to provide and distribute a sufficient quantity of chilled water with through a group of diverse piping system associated piping, to safety-related AHUs and cubicle coolers for total loss of UHS, total loss of CCWS and DEC-B.

The DECWS consists of one independent train. The train consists of one water cooled condenser type chiller, a chilled water pump, an air separator, a compression tank, a chemical additive tank, associated piping, controls and instrumentation. Cooling water for chiller condenser is supplied from Diverse Component Cooling Water System (DCCWS).

The water cooled type essential chiller and pump operate continuously to provide chilled water to the cooling coils of safety-related AHUs and safety-related CCs which are required to operate for total loss of CCWS, total loss of UHS and DEC-B.

3.3.7 Alternate AC diesel generator system

One 100 % capacity safety-related diesel generators, which is independent from the EDGs, is used as an Alternate AC (AAC) source to cope with Station Blackout (SBO) for one unit. The AAC Diesel Generator (AAC DG) is automatically started at the onset of a station blackout.

The AAC DG is also manually started or stopped at the local control panel, or in the Main Control Room (MCR), or Emergency Control Room (ECR).

3.4 Severe Accidents Mitigation Features

3.4.1 Emergency Reactor Depressurization System (ERDS)

The ERDS performs the rapid depressurization of RCS to eliminate the High Pressure Melt Ejection (HPME) scenarios under all Severe Accident (SA) scenarios.

The ERDS consists of the lines, valves, rupture disk, and instruments. A dedicated discharge line from the top of the pressurizer is connected to the containment atmosphere via the SG compartment. The discharge line is designed to remove the saturated steam, water and non-condensable gas in order to ensure RCS depressurization and to prevent a high pressure vessel breach in the event of a SA. The instrument is provided at each discharge line for monitoring the pressure and temperature. The drain line downstream of bleed valves is connected to the RDT for leakage collection.

3.4.2 Passive Ex-vessel Corium retaining and cooling System (PECS)

The PECS provides cooling water source from the IRWST to the core catcher inside the reactor cavity under the reactor vessel and removes heat from molten to guarantee the integrity of the core catcher for prevention of the Molten Corium-Concrete Interaction (MCCI) in the event of severe accidents. Its main functional principle is spreading of the molten corium on the core catcher to make the subsequent flooding, quenching, and cooling more effectively. Spreading of the molten corium significantly increases the surface-to-volume ratio of the molten corium, which results in a fast cool-down and thereby minimizes further release of radioactive substances into the containment.

The PECS is equipped with a Sacrificial Material (SM) layer on the core catcher body, cooling channel, downcomers, cooling water supply subsystem, and monitoring instrumentation.

The cooling water supply subsystem is composed of two trains in compliance with the redundancy and separation principles. Each train has one normal closed isolation valve and two normal open valves. The normal closed valves open only in the event of reactor vessel failure during a DEC-B, but the normal open valve closes by manual action to prevent injection of cooling water due to the malfunction of the normal closed valve during normal operation modes.

5.4.3 Hydrogen Mitigation System (HMS)

The HMS is designed to control combustible gases inside containment within the acceptable limits by the Passive Autocatalytic Recombiner (PAR) in consideration of hydrogen generation during DEC-B. The HMS consists of small, medium, and large sized PARs, which are located throughout the containment.

The HMS is designed to accommodate the hydrogen production equivalent to 100 % of active fuel cladding metal-water reaction, to limit the average hydrogen concentration inside the containment lower than 10 v/o and to finally preclude local detonations.

5.4.4 Double containment

The containment building is composed of double containment structures of a cylinder shell with a hemispherical dome and an annular space between the two structures. The primary containment is made of a post-tensioned concrete with steel-liner for leak-tightness and the secondary containment is a reinforced concrete structure. The containment building houses the safety systems and features required to provide safe shutdown capability.

The primary containment structure is designed to provide biological shielding to plant personnel and the public during normal operation, to confine airborne radioactive materials and to reduce the radioactive release during normal and accident conditions, and to withstand the maximum pressure and temperature inside the containment during accident conditions. During an accident, the temperature and pressure inside the primary containment structures is controlled by built-in safety systems below the design limits. The secondary containment structure is designed in consideration of the intentional aircraft crash loadings and the induced fire effect to maintain the leak-tightness of the primary containment. This is also the physical barrier to protect NSSS main components and relevant systems from external accidents such as missiles and explosion.

3.5 Protection against external hazards

One of the key lessons learned from the Fukushima Daiichi accident is the extension of the design provisions against natural hazards that exceed the conditions considered in the design basis. In this context, the IAEA SSR-2/1 requires that the design of the plant shall provide for an adequate margin to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design derived from the hazard evaluation for the site. The concept of Rare and Severe External Hazard (RSEH) is taken into account in the design of APR1000.

For the standard design, the APR1000 takes into account the typical RSEH parameters. The design for RSEH aims at verifying that the final overall probabilistic evaluation using realistic approach and best estimate rules meet the objectives on core melt and radioactive releases. It is also designed to cover large uncertainties which could exist for hazards, by ensuring that sufficient margin exists with regard to cliff-edge effects.

For the standard plant design, the postulated seismic design ground motions need to cover the majority of the potential nuclear power plant sites. These design ground motions are referred to as the Design Basis Earthquake (DBE). The free-field peak ground accelerations for the DBE are set to 0.3g for use in seismic design of the APR1000. The APR1000 standard plant design considers that the plant is supported by various generic site profiles. Nine (9) generic soil profiles and one (1) fixed-base support condition are considered in seismic analysis. The Soil-Structure Interaction (SSI) effect is taken into account for seismic analysis of APR1000. The seismic analysis results of all analysis cases for generic site profiles and fixed-base condition are then enveloped to produce the final seismic response parameters to be used in the APR1000 standard plant design. The APR1000 standard plant structures housing the items important to safety are designed to withstand the effects of the DBE without loss of capability to maintain their specified design functions.

4. 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. 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

Korea is a member of IAEA, joined NPT on April 23, 1975 to use peaceful nuclear power. Now Korea has no uranium enrichment, reprocessing facilities, and spent fuel waste site, but secure the nuclear material inside the restricted areas and facilities.

5. Safety and Security (physical protection)

The APR1000 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 the APR1000 safety and security are as follows:

- Thick concrete walls for exterior and a large number of interior walls protect those equipment important to safety and provide a significant deterrent to penetration. The Auxiliary Building is physically separated in 2 trains, 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.
- 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.

6. Description of turbine-generator systems

6.1 Turbine generator system

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

Other related system components include a complete turbine generator bearing lube oil system, a high pressure hydraulic fluid system, a digital control & monitoring system, a turbine steam seal system overspeed protective devices, turning gear, a generator seal oil system, a stator cooling water system, and excitation system.

6.2 Condensate and feedwater systems

The Condensate System receives and condenses exhausted steam into the condenser, to collect condensate drained and vented to the hotwell, and supplies condensate to the Feedwater System. Three 50 % capacity motor-driven condensate pumps (two operating and one standby) deliver condensate from the condenser hotwell to in-line full flow condensate polisher with partial flow mode followed by, the steam packing exhauster, and three stages of three Low Pressure (LP) feedwater heater strings to the deaerator. The two lowest pressure feedwater heaters (No. 1 and No. 2) are installed in the condenser necks. The No.3 feedwater heaters are installed horizontally in the heater bay. Isolation valves are provided at the inlet and outlet of each of the three heater strings so that each low pressure feedwater heater string is isolated to prevent water induction into the LP turbine due to the tube rupture or drain control malfunction of No. 1 or No. 2 LP feedwater heater. The system provides condensate to the SG blowdown regenerative heat exchanger for cooling, LP turbine exhaust hood, condensate pumps, feedwater pumps, main condenser block valves for water sealing. The system also provides seal water makeup for the condenser expansion joint and the condenser vacuum breaker valves.

The Main Feedwater System (MFWS) supplies feedwater from the deaerator storage tank to the two SGs at the required pressure, temperature, flow rate, and water chemistry. Three 55 % capacity motor-driven feedwater booster pumps and three 55 % capacity turbine-driven Main Feedwater Pumps (MFWPs) deliver feedwater from the deaerator storage tanks into a common header through three stages of two parallel high pressure feedwater heaters strings. The common header splits into two parallel paths, each of which is connected to a SG. The feedwater flow to each steam generator splits into an economizer and a downcomer feedwater line. Main Feedwater Isolation Valves(MFIVs) meeting single failure criteria are provided in both the economizer and downcomer feedwater lines such that complete termination of forward feedwater flow occurs within 5 seconds after receipt of a Main Steam Isolation Signal (MSIS) and/or Passive Auxiliary Feedwater Actuation Signal (PAFAS).

6.3 Auxiliary systems

6.3.1 Turbine bypass system

The turbine bypass system provides the capability to dump 55% of the rated main steam flow following a loss of the external electrical load and/or a turbine generator trip. During rapid load changes, if there are transient plant conditions where NSSS exceeds the turbine steam requirement, the turbine bypass valves remove excessive energy from the NSSS by dumping steam to condenser in conjunction with the Steam Bypass Control System (SBCS).

6.3.2 Turbine Building Open Cooling Water (TBOCW) system

The TBOCW system supplies cooling water from cooling tower to the service side of the Turbine Building Closed Cooling Water (TBCCW) heat exchangers. In the APR1000 plant design, the TBOCW system interfaces with the Circulating Water (CW) system to take the fresh water and discharge the heated water to the CW discharge manifold.

6.3.3 Condenser vacuum system

The condenser vacuum 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 mechanical vacuum pumps. All four pumps perform the hogging (startup) functions; with three of the four pumps operated for condenser evacuation during plant normal operation. To establish the initial condenser vacuum, the main unit turbine generator is placed on the turning gear, the condensate system is placed in service, and the turbine sealing steam is applied. After the turbine steam seals are established, all pumps initially remove the air from the main condenser to draw down the pressure of the condenser and low-pressure turbine casings.

7. *Electrical and I&C systems*

7.1 *I&C Systems*

7.1.1 *Design concept including control room*

The APR1000 is equipped with digitalized I&C systems and computer-based Man-Machine Interface (MMI) in the control room, reflecting the status of modern electronics and computer technologies. The I&C and control room concept implemented in the APR1000 design is schematically depicted in Figure 9.

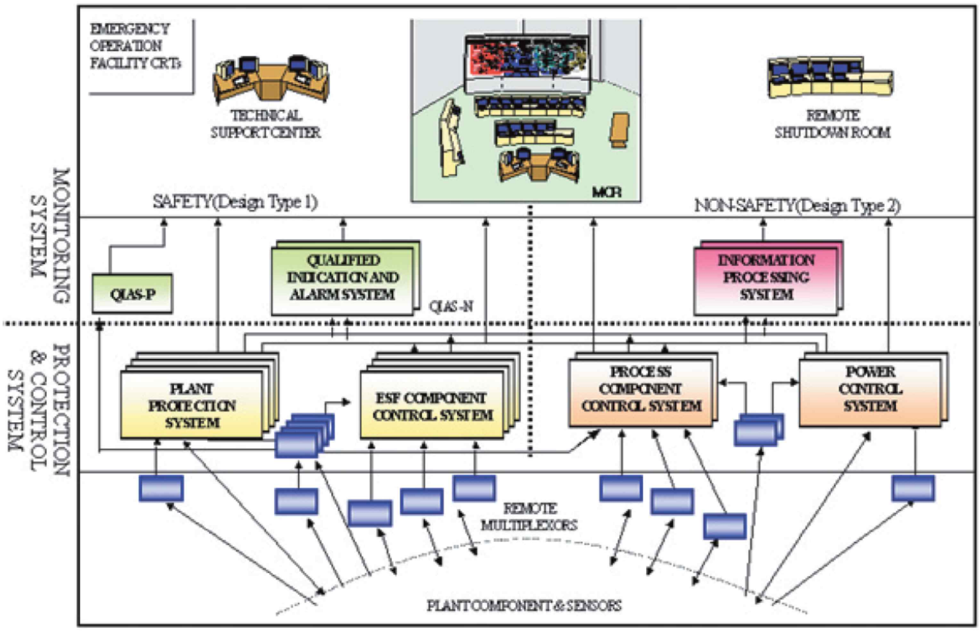


Figure 9. MMIS configuration

The APR1000 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 workstations and industrial personal computers for data processing systems.

The Man Machine Interface System (MMIS) of the APR1000 consists of monitoring, protection and control systems with field devices (i.e. transmitters, RTDs, etc) on a large scale. Operator interface for the MMIS is provided by the Main Control Room (MCR), the Remote Shutdown Room (RSR), the Technical Support Center (TSC) and the local panels. The MCR contains workstations, a Large Display Panel (LDP), and a safety console.

To protect against the 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 field signal transmission of the safety and non-safety systems.

Defense against the Common Mode Failure (CMF) of the digital plant protection systems is one of the key requirements in designing digital I&C systems. The Diverse Protection System (DPS) is designed to be diverse from the Plant Protection System (PPS) against the CMF of the digital plant protection system. Diverse manual ESF actuation is also designed to keep the plant safety against severe situations due to a simultaneous digital system failure of the PPS and the DPS. The open architecture concept is applied to the configuration of the I&C system for high reliability and maintainability. In addition, the stringent software & hardware qualification process is established and followed for the life cycle.

The extensive Human Factors Engineering (HFE) program is incorporated to reduce the possibility of human error in the MCR. During the conceptual and basic design at the R&D stage as well as during the construction phase of the plants, the MMI design has been analyzed and evaluated in an iteratively expanding manner with participation of more than 100 licensed operators and human factors specialists to optimize the design. During the APR1000 development, the evaluation for the MMI design has been performed seven times with full scope dynamic mockups and an APR1000 specific dynamic mockup. The evaluation verified that the MMI are suitable for the human factor principles and guidelines. The new MCR design was also validated to support the normal and emergency operations appropriately.

7.1.2 Reactor protection and other safety systems

The 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 power system is the source of power for station auxiliaries during normal operation and for the Engineered Safety Features (ESF), Diverse Safety Features (DSF), Severe Accident (SA) mitigation systems during AOO, DBA and DEC conditions.

Onsite electrical power systems are connected with two (2) offsite transmission systems which include two (2) physically independent circuits from the transmission network. The offsite power system encompasses the utility grid system, transmission lines, the station Switchyard (SWYD), main generator, Generator Circuit Breaker (GCB), Main Transformer (MT), Unit Auxiliary Transformers (UATs), Standby Auxiliary Transformers (SATs), Isolated Phase Buses (IPBs), and the electrical components associated with them.

Normal power for the station auxiliary loads is supplied either from the main generator when the unit is operating or from the offsite transmission system through the MT when the GCB is open.

Standby power for the station auxiliary loads is supplied from the grid via the SATs when the normal power supply via UATs is not available.

The onsite AC power system, which supplies power to equipment important to safety during all normal operation and accident conditions, includes AC standby power sources, distribution systems, and auxiliary supporting systems. The DC power system, which supplies motive or control power to equipment, includes DC power sources, battery chargers, and their distribution systems.

The electrical power for the systems important to safety is supplied from the following four alternative ways:

- the normal power source of the normal offsite power through the MT or the onsite power through UATs generated by the main generator,
- the standby offsite power connected through the SATs with the grid,
- the onsite emergency power supply from four (4) EDGs,
- the onsite alternate power supply from one (1) AACDG.

As shown in Figure 10, the safety electrical power systems are composed of train A, B, C, and D for safe shutdown in the events of an Anticipated Operational Occurrence (AOO) or a DBA and train E during the DEC-A or DEC-B events. Onsite emergency power for train A, B, C, and D is supplied from the safety EDG, and onsite alternate power for train E is supplied from the safety feature AACDG.

Each train consists of:

- AC medium voltage system,
- AC low voltage system,
- DC power system,
- I&C power system.

Besides their connection to corresponding train of AC power distribution system through a battery charger, each train of the DC power onsite power distribution system important to safety is provided with its own related battery power source.

The EDG is automatically started after receiving an under-voltage signal from the safety medium voltage bus. Each EDG circuit breaker is closed automatically when the EDG output reaches predetermined voltage and frequency, and the circuit breakers of safety loads are closed in sequence automatically. The EDG has enough capacity for the safety loads needed for a concurrent DBA and Loss Of Offsite Power (LOOP).

The AACDG and its supporting auxiliaries are independent and diverse from the EDGs and the offsite power sources. The AACDG is provided with sufficient capacity and capability to cope with and recover from the SBO for a time required to bring and maintain the plant in safe shutdown. The power supply for DEC coping is also ensured by AACDG.

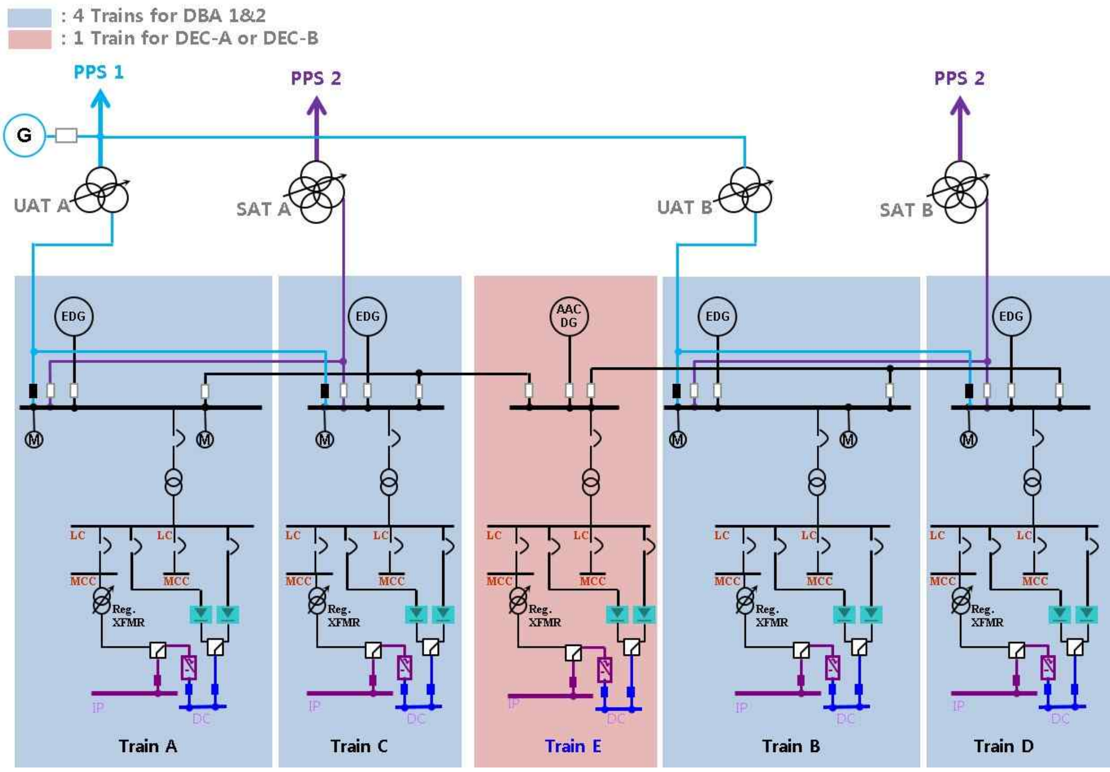


Figure 10 Single Line Diagram of Trains Important to Safety

The electric power system is the source of power for station auxiliaries during normal operation, and for the Plant Protection System (PPS), Engineered Safety Features (ESF), Diverse Safety Features (DSF), and SA mitigation systems during abnormal and accident conditions. The electrical systems are designed in accordance with IEC Codes & Standards as follows:

- The voltages of the onsite electrical power supply system in APR1000 comply with IEC 60038.
- The design, manufacture and test of the MV switchgear in APR1000 comply with IEC 62271-200.
- Short circuit calculations are made in accordance with IEC 60909 series.
- The physical separation and electrical isolation in APR1000 follows IEEE 384 (supplemented by IEC 60709).
- IEEE 323, which envelopes IEC 60780, is applied to the qualification of Safety-related electrical equipment.

In case there are no suitable or applicable European Codes or Standards, U.S Regulatory Guides and/or IEEE Standards are applied.

8. Radwaste Management Systems

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 nonradioactive 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 function of the liquid waste management systems is to collect radioactive or potentially radioactive liquid wastes generated during normal plant operation, including anticipated operational occurrences; to process the liquid waste in order to remove radioactive isotopes; to hold up the liquid wastes for radioactive decay or delayed processing; and discharge the treated liquids to the environment.

The Liquid Radwaste System (LRS) consists of various tanks, pumps, Reverse Osmosis(R/O) packages and associated piping, valves, and instruments.

8.2 Gaseous Waste Management System (GWMS)

The Gaseous Radwaste System (GRS) processes radioactive gaseous waste generated within plant equipment so that the onsite and offsite radiation exposure to personnel are maintained within acceptable limits. The system is located in the Compound Building, which is designed to withstand an operating basis earthquake.

Low activity gaseous wastes are filtered in the HVAC system prior to release to the atmosphere. The low activity gaseous wastes systems consist of the building ventilation exhaust systems, the main condenser evacuation system, and the turbine gland sealing system.

The GRS is comprised of one collection header, one header drain tank, two dehumidifier trains, four charcoal delay beds, one HEPA filter, one nitrogen injection skid, one chiller skid, radiation monitor and two gaseous wastes analyzer. Dehumidifier train consists of one waste gas dryer and one guard bed. The gaseous wastes charcoal delay beds adsorb the radioactive krypton and xenon atoms and delay them for radioactive decay.

After passing through the charcoal delay beds, the gas is discharged through a HEPA filter and a radiation monitor. The gas is purged with nitrogen thereby precluding air migration in the system during low or no flow period. Radiation monitoring is provided on the discharge line from the charcoal delay beds and on the Compound Building HVAC vent. The Compound Building HVAC exhaust radiation monitor is interlocked to shut the discharge line isolation valves on high radiation level.

8.3 Solid Waste Management System (SWMS)

The solid waste management system for the APR1000 is designed to provide holdup, solidification, and packaging of radioactive wastes generated by plant operation, to handle the packaged solid radioactive waste, and to store these wastes until they are transferred to the yard storage area or shipped offsite for disposal. The system is located in the Compound Building, which is designed to withstand a safe shutdown earthquake.

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.

9. *Plant layout*

Plant layout and General Arrangement (GA) of the APR1000 focus on plant safety, economic competitiveness and plant operation convenience in compliance with regulations and utility requirements.

Major buildings of the APR1000 consist of the Containment Building (CB), the Auxiliary Building (AB), the Compound Building (CPB) and the Turbine Generator Building (TGB), as shown in Figure 11. The Nuclear Island (NI) consists of the Containment Building (CB), the Auxiliary Building (AB), the Emergency Diesel Generator building, the Emergency Diesel Generator/Alternate Alternating Power Diesel Generator (EDG/AACDG) building, and the Compound Building (CPB). The Turbine Island (TI) includes the Turbine Generator Building (TGB) and the switchgear area, which are arranged in a radial direction from the Nuclear Island (NI).

The CB is composed of primary and secondary containment. The annulus area is the space between the two containments, as one big volume for treatment of the leakage from the primary containment. The Nuclear Steam Supply System (NSSS) is located inside the primary containment. The secondary containment fully envelops the primary containment to minimize radioactive releases at any plant condition by maintaining sub-atmospheric pressure in the annulus area and to preserve the integrity of the primary containment against external events.

The CB is surrounded by the AB. The AB houses the essential systems performing the important safety functions and the fuel handling area. In addition to the structural components, the components to provide biological shielding and protection against internal and external hazards are incorporated into the AB.

There are the EDG building and the EDG/AACDG buildings, which are separated by the AB. The EDG/AACDG building houses two units of EDGs and one AACDG, while the EDG building houses two units of EDGs.

The CPB is located on the side of the AB and provides an access to the controlled area of the plant. The CPB houses the Structures, Systems, and Components (SSCs) for liquid and solid radioactive waste treatment and retains the spillage of potentially contaminated solids or liquids within the building.

The TGB is independent from the AB and houses the turbine, generator, and the associated supporting systems.

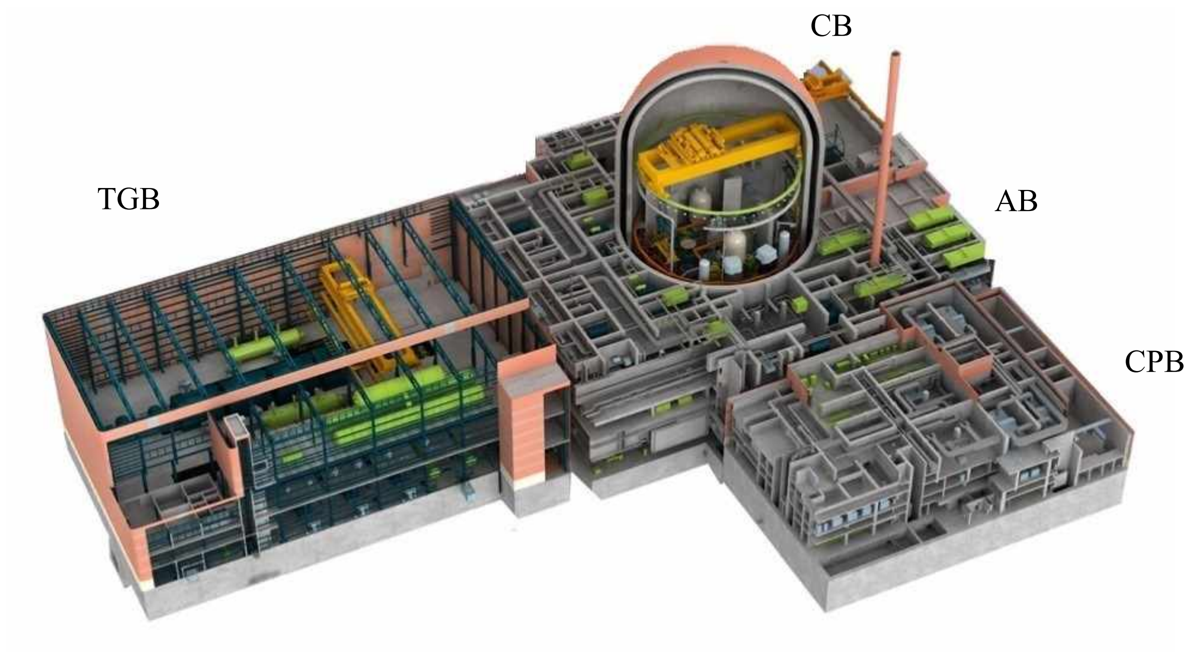


Figure 11. Plant Layout of APR1000

9.1 Site Plot Plan

The APR1000 plant site plot plan is developed reflecting EUR and design practices of the APR1400. The buildings with in the APR1000 standard design can be divided in to Nuclear Island (NI), Turbine Island (TI), ancillary buildings and yard service facilities. The site layout of the APR1000 standard design is shown in Figure 12.

The site plot plan of the APR1000 is developed considering the followings:

- Accessibility: Roadways and personnel flow for operational access and maintenance
- Functional relationship between the main power block, ancillary buildings, and yard service facilities
- Interface among underground facilities (circulating water flow routes, cable and pipe runs, etc.), aboveground facilities, and buildings
- Separation and segregation considering internal and external hazards (fire, flooding, earthquake, aircraft crash, etc)
- Radiation protection and ALARA consideration
- Turbine missile protection
- Emergency exit
- Equipment removal path and space
- Land use optimization
- Security enhancement
- Constructability

In particular, the NSSS and the most of safety systems are located in the Containment Building (CB) and in the Auxiliary Building (AB) to be protected from the external hazards. The buildings which contain safety systems outside the AB are physically separated from each other by the AB. Therefore, in case of the intentional aircraft crash, the required trains of safety systems for safe shutdown and the associated cooling chain remain intact so that core melt does not occur.

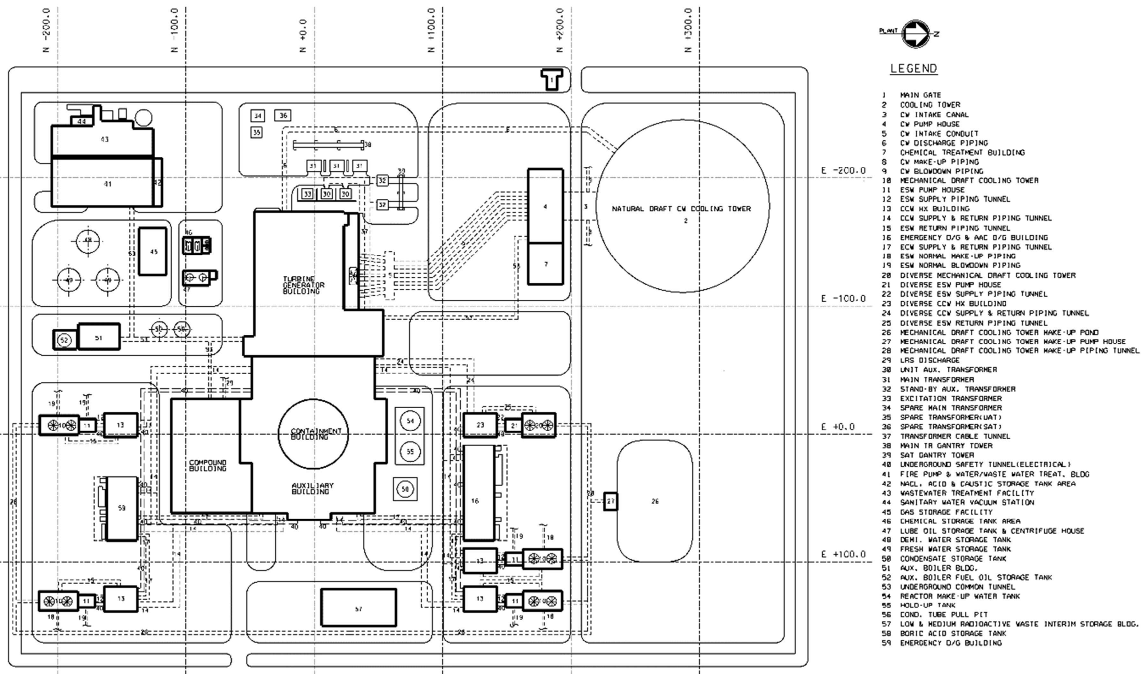


Figure 12. Site Plot Plan for APR1000

9.2 Containment Building (CB)

The CB is composed of double containment structures of a cylinder shell with a hemispherical dome and an annular space between the two buildings, as shown in Figure 13. The primary containment is made of a post-tensioned concrete with steel-liner for leak-tightness and the secondary containment is a reinforced concrete structure. The CB houses safety equipment required to provide safe shutdown capability. The CB is founded on a common basemat with the AB. The inner surface of the primary containment is steel-lined for leak-tightness, and a protective layer of concrete covers the portion of the liner over the foundation slab. The In-containment Refueling Water Storage Tank (IRWST) is designed in an annular-shape and placed inside the primary containment in the lower level of the CB.

The primary containment structure is designed to provide biological shielding to plant personnel and the public during normal operation, to confine airborne radioactive materials and to reduce the radioactive release during normal and accident conditions, and to withstand the maximum pressure and temperature inside the containment during accident conditions. The secondary containment structure is designed to provide an additional radiological barrier for radioactive systems and components inside the primary containment during normal and accident conditions and to protect them from external hazards, such as tornados and aircraft crash. Also, annulus area is provided between the inner radius of the secondary containment and the outer radius of the primary containment above the basemat.

The primary containment is designed to allow personnel access for planned maintenance and inspection activities in any normal operating conditions. Access to the interior of the primary containment is provided through two personnel airlocks. An equipment hatch is provided to permit transfer of equipment into and out of the primary containment structure during plant outages. The space for access, laydown and maintenance aids the installation of equipment during construction, and makes equipment replacement easier over the life of the plant as shown in Figure 14.

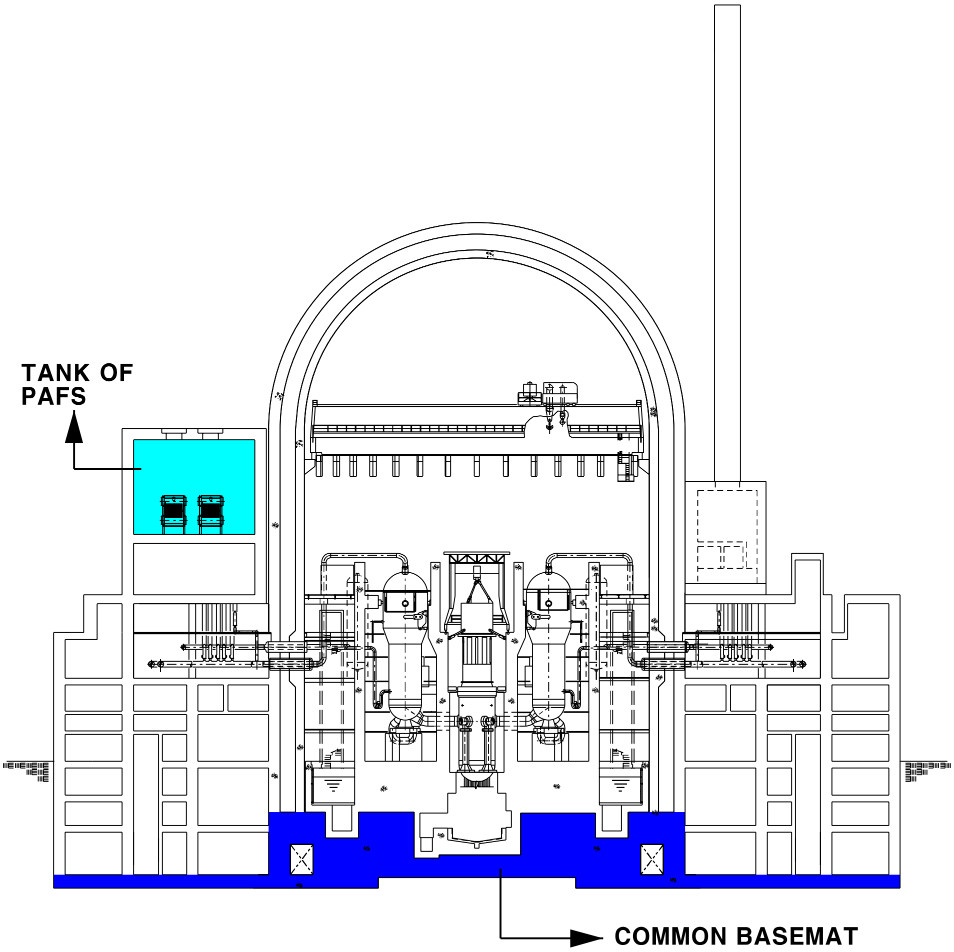


Figure 13. Containment Building (Section)

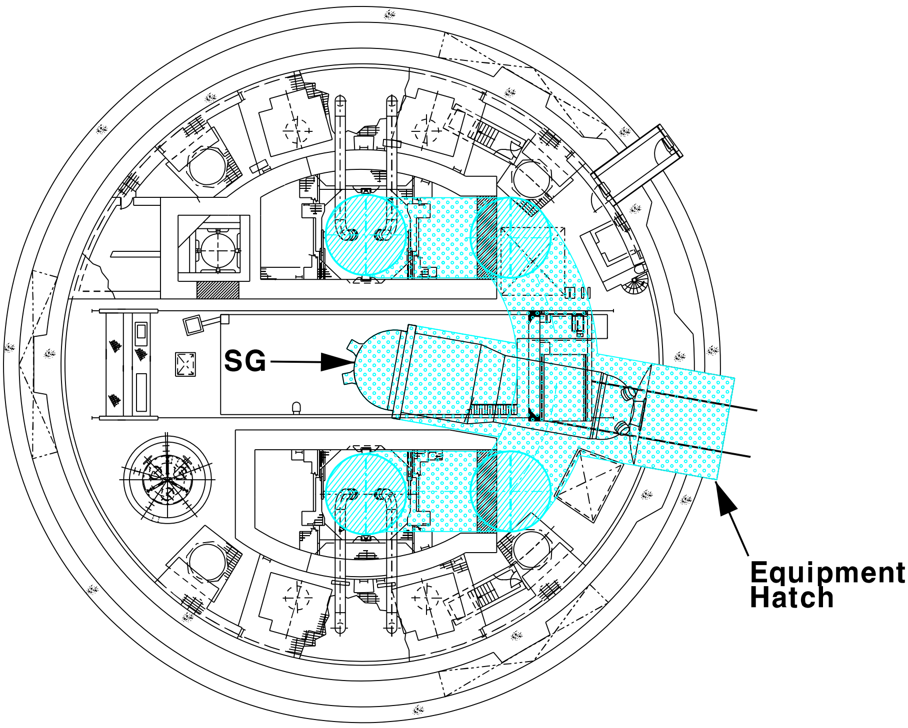


Figure 14. Equipment Hatch for Removal of Large Components

9.3 Auxiliary Building(AB)

The AB is founded on a common basemat with CB and the tanks of PAFS are located on the top floor, as shown in Figure 13. The AB composed of a reinforced concrete structure with rectangular walls and floor slabs and a composite structure with columns and girders, as shown in Figure 13. The AB surrounds the CB with the quadrant arrangement. It houses safety equipment required for safe shutdown.

The AB is designed to withstand internal hazards as well as external hazards such as, earthquake, extreme winds, missiles, fire, flooding, aircraft crash and sabotage. Four independent trains of safety systems essential for safe shutdown are physically separated from one another to prevent a Common Cause Failure (CCF) of all systems and to prevent the propagation of damage from internal events and natural and man-made external hazards. The AB employs concept of quadrant separation for the protection from fire, flooding, explosion, High Energy Line Break (HELB), and missile hazard. Arrangement of components, structures, passages and rooms is optimized to take into account the operational and maintenance convenience. Radioactive areas and clean areas are physically separated to reduce the occupational exposure dose. HELB zones are also physically separated to prevent the hazard propagation to Non-HELB zones.

9.4 Emergency Diesel Generator/Alternate AC Diesel Generator Building

The EDG building and the EDG/AACDG buildings are located at northern and southern side of the AB. These buildings are designed as the reinforce concrete to be protected from the internal and external hazards.

The EDG/AACDG building houses two EDGs and an AACDG, while the EDG building houses two EDGs. Each diesel generator with its ancillary equipment is physically separated in each building.

The AACDG system supplies AC power in the event of SBO when the Loss Of Offsite Power (LOOP) occurs simultaneously with the failure of all EDGs.

9.5 ComPound Building (CPB)

The CPB is comprised of reinforced concrete shear walls, interior walls, slabs, beams, and columns on the basis of a reinforced concrete foundation separated from the adjacent AB. The CPB is located at the southern side of the AB.

The CPB consists of an access control area, a radwaste treatment and drum removal area, radwaste system control room, sampling facilities and laboratory, a main mechanical hot workshop, an Operational Support Center (OSC) to support the MCR, and service areas. It is designed to be protected from hazards such as flooding, snowfall, earthquake, etc. and to accommodate loadings associated with environmental conditions to the extent necessary to retain the spillage of potentially contaminated solids or liquids within the building. All access to and from the controlled area in the plant is applied to the single access control concept as shown in Figure 15.

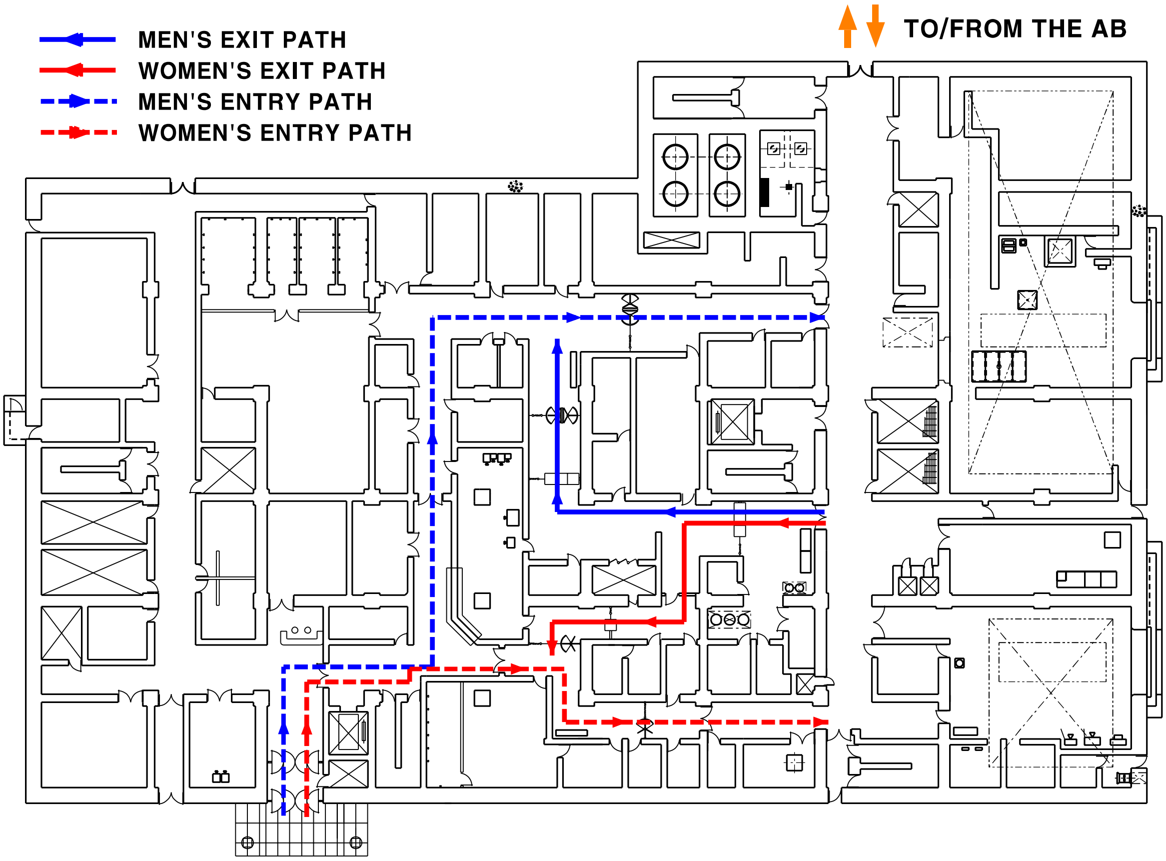


Figure 15. Access Control Area Arrangement

9.6 Component Cooling Water Heat Exchanger Buildings and Essential Service Water Pump Houses

The Component Cooling Water Heat Exchanger (CCW HX) Buildings and the Essential Service Water (ESW) Pump Houses are located at the northern side and the southern side of the AB, considering the effect of turbine missile strike and the economic connectivity between relevant equipment arranged in each building. The CCW HX Buildings and the ESW Pump Houses provide space for the heat exchangers, instrumentations, the pumps, and debris filters. These buildings are concrete structures to accommodate four independent trains of the systems.

9.7 Diverse Component Cooling Water Building and Diverse Essential Service Water Pump House

The Diverse Component Cooling Water Heat Exchanger (DCCW HX) building and the Diverse Essential Service Water (DESW) Pump House are located at the northern side of the AB considering the effect of turbine missile strike and the economic connectivity between relevant equipment. The DCCW HX Building and DCCW Pump House provide space for the heat exchanger, debris filters, and the pump. This building is a concrete structure to accommodate diverse cooling systems for design extension conditions.

9.8 Turbine Generator Building (TGB)

The TGB is primarily composed of basemat, underground shear wall, turbine generator pedestal, and superstructure. The TGB is designed with steel structures based on a common basemat concrete. It is located at the western side of the AB in a radial direction with respect to the CB. In the event of an accident, this arrangement minimizes the risk of damage to safety-related equipment in other buildings due to the turbine generator missile. The TGB is located as close to AB as possible considering the passages for operators and maintenance staffs.

Arrangement of the TG set is designed for optimization of the piping and cable routes to the NSSS. The TGB provides support and housing for the turbine generator and its ancillary systems such as the lube oil system, the hydrogen supply and cooling system, the stator cooling system, the seal oil system, the electro-hydraulic control system, etc.

The TGB also accommodates the systems necessary for constituting the heat cycle such as the condenser system, the preheater system, the condensate and feedwater systems, and other systems associated with power generation.

10. Plant Performance

10.1 Operational performance

The APR1000 design is optimized to achieve the high operation performance and to enhance the convenience of maintenance by incorporating the following improvements:

- 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 to each other. This improvement reduces the girth seam, for which in-service inspections have to be performed over the lifetime. Also, the work platforms are installed to enhance the convenience of ISI for steam generators.

- 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 remote control so that the exposure dose is reduced. In addition, a temporary fuel storage rack can be installed inside the refueling pool to be used under an abnormal condition during the refueling. The design of the In-Core Instrument (ICI) cable tray is improved so that it is not necessary to install and disassemble the ICI for every refueling. This reduces the polar crane load and simplifies the task related to ICI cable.

- Design features for reducing unplanned trip

To reduce unplanned reactor trips, the core thermal margin is increased by more than 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 the capability of coping with the transients.

The turbine rotor is manufactured as one piece by forging to reduce the susceptibility of Stress Corrosion Cracks (SCCs). The turbine control system is improved to enhance its reliability and maintainability by the redundant design of controllers and the strengthening of the diagnostic functions.

The vibration monitoring functions are improved by strengthening the self-diagnostic functions of the detectors and multidirectional measurements. In addition, earthquake-proof structures are installed to prevent a turbine trip caused by high vibration.

The APR1000 adopts a static excitation type to reduce 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 are strengthened not to heat up by the reduced coolant flow.

The feedwater flow control system is designed to control the feedwater flow automatically over the full operation range and to operate three turbine driven main feedwater pumps during normal power operation. When one main feedwater pump is tripped during the full power condition, the other two main feedwater pumps will be able to provide the total feedwater flow to the full power condition. This design reduces unnecessary power cutbacks and unplanned turbine trips.

10.2 Construction

A new construction schedule and constructability enhancement methods were developed based on the experience from the repeated OPR1000 constructions. The APR1000 is seismically enhanced with applying 0.3g Safe Shutdown Earthquake (SSE) as a Design Basis Earthquake (DBE).

Modularization has been introduced to reduce the construction period and cost. There are three types of modules as follows: structural, mechanical equipment, and composite modules. 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 modules, and the composite modules are under development. If the composite modules are applied to all buildings in the Nuclear Power Plant, the construction period is expected to be dramatically reduced to less than 40 months through the pre-fabrication at both the factory and the site.

For the mechanical and electrical equipment and piping installation, it is recommended to increase the fabrication portion in the manufacturing shops. Approximately 80 items of APR1000 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 APR1000 development from conceptual design to construction, the entire plant design processes have been reviewed using a 3D CAD model. Design output was produced with frozen model after the verification of the 3D CAD model. This design verification improves both the quality and the timeliness of the project design.

A new 3D CAD system called tri-dimensional design verification system (TDVS) was developed to improve and streamline the existing engineering process for the main 2D design work and subsidiary 3D review work as shown in Figure 12. 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.

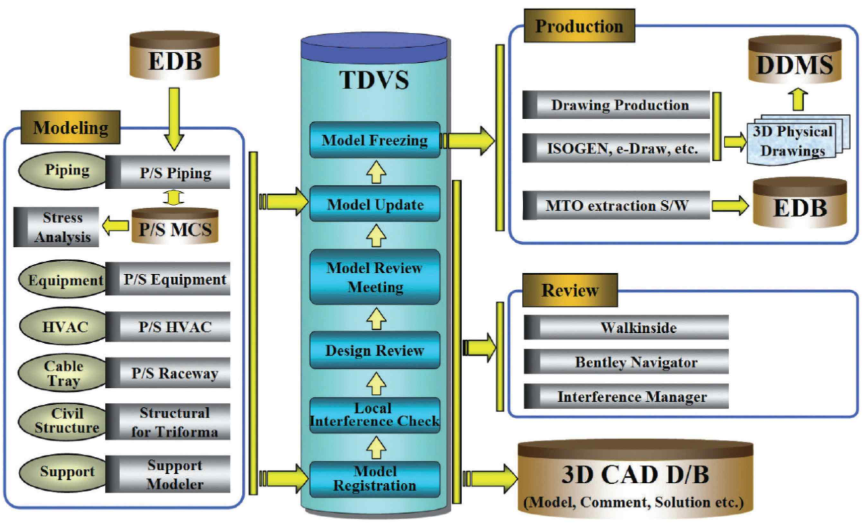


Figure 12. 3D Design Verification System

Engineers of each discipline make 3D CAD models themselves. The 3D CAD system is connected with the engineering data base (EDB) system. For example, when an 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 the equipment. The virtual character technique of computer games is applied to the plant design. An engineer enters the 3D integrated model as a virtual character, and can look for various components, as if the 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, the engineer can review the engineering information and related drawings together at the same time.

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.

Appendix: Summarized Technical Data for the APR1000

General plant data		
Reactor thermal output	2,815	MWt
Power plant output, net	1,000	MWe
Power plant efficiency, net	37.2	%
Mode of operation	Load Follow and Baseload	
Plant design life	60	Year
Plant availability target	More than 90	%
Seismic design, SSE	0.3g	
Primary Coolant material	H_2O	
Secondary Coolant material	H_2O	
Moderator material	H_2O	
Thermodynamic Cycle	Rankine Cycle	
Type of Cycle	Indirect	
Non-electric application	N/A	
Safety goals		
Core damage frequency	Less than 10^{-5}	/RY
Large release frequency	Less than 10^{-6}	/RY
Occupational radiation exposure	0.5	Person-Sv/RY
Operator Action Time	0.5	hour
Nuclear steam supply system		
Steam flow rate at nominal conditions	790 (per S/G)	kg/s
Steam pressure/temperature	7.4 / 294.4	MPa(a)/℃
Feedwater temperature	232.2	℃
Reactor coolant system		
Reactor operating pressure	15.5	MPa(a)
Core coolant inlet temperature	296	℃
Core coolant outlet temperature	327	℃
Reactor core		
Active core height	3.81	m
Equivalent core diameter	3.12	m
Average linear heat rate	17.26	kW/m
Fuel material	UO_2	
Cladding tube material	Optimized ZIRLO	
Outer diameter of fuel rods	9.5	mm
Rod array of a fuel assembly	Square 16x16	
Number of fuel assemblies	177	
Enrichment of reload fuel at equilibrium core	≈ 4 (Batch Average)	Wt%
Fuel cycle length	18 ~ 24	months
Average discharge burnup of fuel	54.1	MWd/kg

Burnable absorber (strategy/material)	Gd_2O_3-UO_2	
Control rod absorber material	B_4C or Inconel slug	
Soluble neutron absorber	Soluble Boron	
Reactor pressure vessel		
Inner diameter of cylindrical shell	4,120	mm
Wall thickness of cylindrical shell	205	mm
Total height, inside	14,642	mm
Base material	SA508, Grade 3, Class 1	
Design pressure/temperature	17.2 / 343.3	MPa(a)/℃
Steam generator		
Type	Vertical U-tube with integral economizer	
Number	2	
Total tube outside surface area	10,009 (per S/G)	m^2
Number of heat exchanger tubes	8,340	
Tube outside diameter	19.05	mm
Tube material	SB-163 Alloy 690	
Reactor coolant pump		
Type	Vertical, Single Stage centrifugal with bottom suction and horizontal discharge	
Number	4	
Pressurizer		
Total volume	51	m^3
Steam volume: full power	25.77	m^3
Heating power of heater rods	1,800	kW
Residual heat removal systems		
Active/passive systems	Passive	
Safety injection systems		
Active/passive systems	Active & Passive	
Active power	1,000	MW
Voltage	22 (3 phase)	kV
Frequency	60	Hz
Condenser		
Type	Steam Surface	
Condenser pressure	8.5	kPa(a)
Feedwater pumps		
Type	Turbine Driven	
Number	3 (each 55 %)	