

MDEP

Technical Report

TR-APR1400-01

APR1400 Working Group activities

Design Description and Comparison of Design Differences between APR1400 Plants

Participation

Regulators involved in the MDEP working group discussions:	KINS (Korea), FANR (United Arab Emirates) and US NRC (United States)
Regulators which support the present report:	KINS (Korea), FANR (United Arab Emirates) and US NRC (United States)
Regulators with no objection:	-
Regulators which disagree:	-
Compatible with existing IAEA related documents:	Yes

FOREWARD

The APR1400s are in different licensing stages among member countries and there are differences in the design of APR1400s submitted for licensing applications. The APR1400 MDEP Working Group members agreed to identify the design differences between these reactors and to discuss the reason for the differences, such as design improvements or regulatory requirements which are different for each member country.

The designer's explanation on the purpose of the design changes and the working group member's review are compiled in this report.

ACKNOWLEDGMENTS

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Chapter 1. Overview of the APR1400 Plant

1.1 Introduction

The APR1400 is a pressurized water reactor with the electrical power of 1400 MWe designed by Korea. The APR1400 is an evolutionary reactor developed from the proven Optimized Power Reactor 1000 (OPR1000) currently in operation at the Ulchin Nuclear Generating Station in Korea and System 80+ that has received design certification from the U.S. NRC in June 1997.

The first APR1400 plants, Shin-Kori 3 has received operating license from Korean Regulatory Body in Oct. 2015 and is in commercial operation. Besides current construction project of Shin-Kori 3&4, Shin-Hanul 1&2, Shin-Kori 5&6, and Shin-Hanul 3&4, there are two overseas projects related to APR1400; UAE project, US NRC project.

In UAE, Barakah NPP unit 1~4 are under construction. Commercial operation of unit 1 plant is expected in 2018. Construction permit of units 3&4 has been received at Sep. 2014.

For US NRC design certification, documents have been developed based on the proven technology to fully comply with the latest NRC requirements and guidance. The official design certification process has been initiated from March 2015.

Thermal power including RCP power is 4,000MW and the design life time is 60 years for main components such as a reactor vessel, a pressurizer, a steam generator, an RCP and RCS main pipings.

1.2 Building and Structure

The reference model of APR1400, Shin-Kori 3&4, has twin unit based building arrangement for economics. As shown in Fig. 1.1, Shin-Kori 3&4 and BNPP 1&2 have slide along type twin unit arrangement. There is compound building between units for hot machine shop, access control and so on. APR1400 models for NRC-DC has been developed based on single unit arrangement. Except the NRC-DC project,

APR1400 plants take sea water as ultimate heat sink. APR1400 for NRC-DC uses a cooling tower using fresh water as ultimate heat sink.

For reference model of APR1400, the safety related buildings and structures are designed with the application of the Safe Shutdown Earthquake (SSE) of 0.3g as a Design Basis Earthquake (DBE) to increase their strength against earthquakes. The exceptions are as follows;

- In UAE project, soil structure interaction is considered with PGA of site specific SSE of 0.165g.

The containment building is a prestressed concrete structure in the shape of a cylinder with a hemispherical dome and is founded on a common basemat with auxiliary building. The cylindrical portion of the containment structure is prestressed by a post-tensioning system consisting of horizontal (hoop) and vertical (inverted U) tendons. The interior surface of the containment shell and dome is steel-lined for leak-tightness. A protective layer of concrete (fill slab) covers the portion of the liner over the foundation slab. The containment structure concrete provides biological shielding for normal and accident conditions.

The auxiliary building houses Emergency Diesel Generators (EDGs) and Fuel Handling Area (FHA). The layout of auxiliary building, particularly for the physically separated arrangement of safety equipment, is designed to enhance plant safety. As examples, four train Safety Injection Systems (SISs) and two set of EDGs are arranged that each one is placed in the physically separated division of auxiliary building. This configuration prevents the propagation of system damage by internal and external events such as flood, fire, security and sabotage. Other internal structures are also arranged to improve maintainability, accessibility, and convenience of equipment replacement.

US NRC has established 10CFR50.150 (Aircraft Impact Assessment (AIA) requirement). This requirement is applied to new reactor design after July 13, 2009. The first APR1400 plants, Shin-Kori 3&4 project has been started before that time and therefore, AIA requirement has not been applied to SKN 3&4 project. This requirement has been applied from Shin-Kori 5&6 project in Korea. AIA requirement is applied to APR1400 plants for UAE, USA. The approaches are slightly different from each project. In Shin-Kori 5&6 and BNPP projects, the wall thickness is changed from 4.0ft to 4.5ft for both the containment and the auxiliary building to

avoid perforation by aircraft crash. In NRC DC project, the containment wall thickness is changed from 4.0ft to 4.5ft to enhance anti-perforation strength. The auxiliary building outer wall thickness, 4.0ft, has not been changed. It means that the perforation by aircraft crash is allowed. Instead, critical safety function is maintained by independent remote control console although main control room is unavailable by aircraft crash.

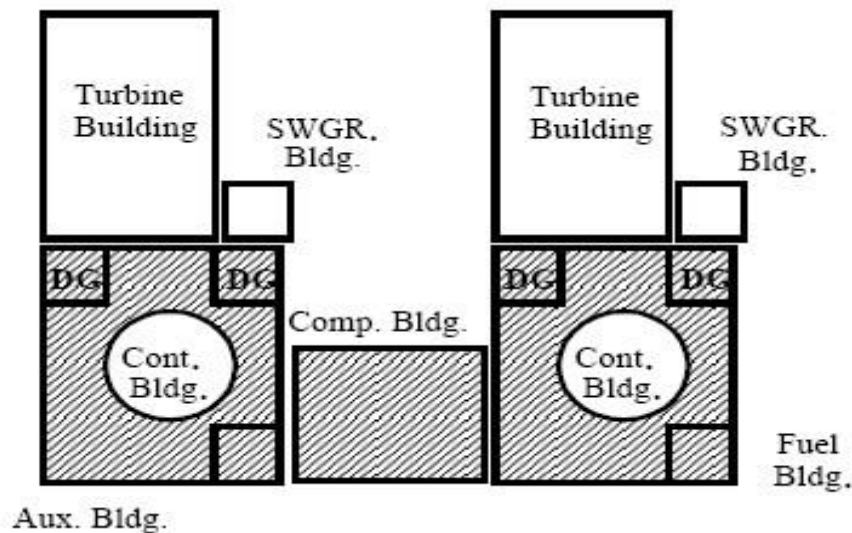


Figure 1.1 Building Arrangement of APR1400 (2 units basis)

1.3 Reactor and Reactor Coolant System

The APR1400 NSSS generates 4000 MWt, producing saturated steam. The NSSS contains two primary coolant loops, each of which has two reactor coolant pumps, a steam generator, one 42-inch (1.07 m) ID hot leg pipe and two 30-inch (0.76 m) ID cold leg pipes (Figure 1.2). In addition, the safety injection lines are connected directly to the reactor vessel. An electrically heated pressurizer is connected to one of the loops of the NSSS. The pressurizer has an increased volume (relative to previous design) to enhance transient response. Pressurized water is circulated by means of electric-motor-driven, single-stage, centrifugal reactor coolant pumps.

The RCS operates at a nominal pressure of 2250 psia (158.2 kg/cm²A). The reactor coolant enters the reactor vessel, then flows downward between the reactor vessel shell and the core support barrel, up through the core, leaves the reactor vessel, and

flows through the tube side of the two vertical U-tube type steam generators (with an integral economizer) where heat is transferred to the secondary system. Reactor coolant pumps return the reactor coolant to the reactor vessel. Major design parameters of RCS is summarized in Table 1.1

Two steam generators produce steam for driving the plant turbine-generator. Each steam generator is a vertical U-tube heat exchanger with an integral economizer. Each unit is designed to produce saturated steam when provided with the proper feedwater. Moisture separators and steam dryers on the shell side of the steam generator limit the moisture content of the steam during normal operation at full power. An integral flow restrictor is incorporated into each steam generator steam nozzle to restrict flow in the event of a steam line break.

The steam generator features several design enhancements including high performance moisture separators and steam dryers, increased overall heat transfer area and integral feedwater economizer. The steam generator tubes are made of the advanced material, Ni-Cr-Fe alloy 690(TT). It also has a larger secondary feedwater inventory which extends the "dry-out" time, thus enhancing the NSSS's capability to tolerate upset conditions and improving operational flexibility. Finally, the APR1400 steam generator design has a greater tube plugging allowance, thus; permitting the NSSS to maintain rated output with a significant number of tubes plugged.

The reactor vessel consists of a vertically mounted cylindrical vessel welded with a hemispherical lower head and a removable hemispherical upper closure head. The internal surfaces that are in contact with the RCS coolant are clad with austenitic stainless steel to prevent corrosion. The reactor vessel is basically manufactured with three shell sections of upper, intermediate and lower, a vessel flange, and a hemispherical bottom head. The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The three shell sections, the bottom head forging and vessel flange forging are joined together by welding, along with four inlet nozzle forgings, two outlet nozzle forgings, four Direct Vessel Injection (DVI) nozzle forgings, and sixty one (61) in-core instrument (ICI) nozzles. 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 reactor vessel and internals are schematically shown in Figure 1.3.

The reactor core consists of 241 fuel assemblies, 93 Control Element Assemblies (CEAs), and 45 In-Core Instrument (ICI) assemblies. The core is designed for the refueling cycle to be above 18 months with a maximum discharge rod burn-up of 60,000 MWD/MTU and for the thermal margin to be increased above 10 %. Important design parameters are summarized in Table 1.2

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 UO₂ pellets or the burnable absorber rods containing Gd₂O₃-UO₂ 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. Each guide tube is attached to fuel assembly spacer grids and to upper and lower end fittings to provide a structural frame to position the fuel rods.

An advanced fuel, named as PLUS7, is used in the APR1400 reactor core. Compared with conventional fuel, PLUS7 provides enhanced thermal hydraulic and nuclear performance, and structural integrity. The mixing vanes with high thermal performance, which induce a relatively small pressure loss, are adopted in all mid-grids to increase thermal margin above 10 % that has been confirmed in the Critical Heat Flux (CHF) test. The batch average burn-up is increased up to 55,000 MWD/MTU through optimizing the fuel assembly and fuel rod dimensions and adopting an advanced Zr-Nb alloy as a fuel clad. Design parameters of nuclear fuel are summarized in Table 1.3.

The reactor coolant is circulated by four vertical, single stage, single bottom suction, single horizontal discharge, centrifugal pumps. The pump shaft is sealed by controlled leakage mechanical shaft seals and driven by a vertical AC induction motor. RCPs for Shin-Kori 3&4 and NRC DC use 60 Hz electricity and RCPs for UAE use 50Hz electricity.

Table 1.1 Design Parameters of APR1400 RCS

Parameter	Value
Thermal Power, MWt (Including Heat Addition from	4000

Pumps)	
Design Pressure, psia (kg/cm ² A)	2500 (175.8)
Design Temperature (except pressurizer), °F (°C)	650 (343.3)
Pressurizer Design Temperature, °F (°C)	700 (371.1)
Coolant Flow Rate, lb/hr (kg/hr)	166.6 x 10 ⁶ (75.6 x 10 ⁶)
Cold Leg Temperature, Operating, °F (°C)	555.0 (290.6)
Hot Leg Temperature, Operating, °F (°C)	615 (323.9)
Normal Operating Pressure, psia (kg/cm ² A)	2250 (158.2)

Table 1.2 Design Parameters of APR1400 Steam Generators

Parameter	Value
Number of units	2
Heat transfer rate per SG, Btu/hr (kcal/hr)	6.830 x 10 ⁹ (1.721 x 10 ⁹)
Number of Tubes per SG	13,102
Average Active Tube Length per SG, ft (m)	63.62 (19.391)
Primary Side	
Design pressure/temperature (psia/°F) (kg/cm ² /°C)	2500/650 (175.76/343.33)
Coolant inlet temperature, °F (°C)	615 (323.88)
Coolant outlet temperature, °F (°C)	555 (290.55)
Coolant flow rate, each, lb/hr (kg/hr)	83.3 x 10 ⁶ (37.78 x 10 ⁶)
Secondary Side	
Design pressure/temperature, psia/°F (kg/cm ² A /°C)	1200/570 (84.36/298.88)
Steam pressure, psia(kg/cm ² A)	1000(70.30)
Steam flowrate (at 0.25% moisture) per SG, lb/hr (kg/hr)	8.975 x 10 ⁶ (4.070 x 10 ⁶)
Feedwater temperature at full power, °F (°C)	450 (232.22)
Moisture carryover, weight maximum, %	0.25

Table 1.3 Design Parameters of APR1400 Reactor Core

Parameters	Design Value
Number of fuel assembly	241
Maximum burn-up (MWD/MTU)	60,000
Refueling cycle	Above 18 months
Number of refueled fuel at equilibrium cycle	92

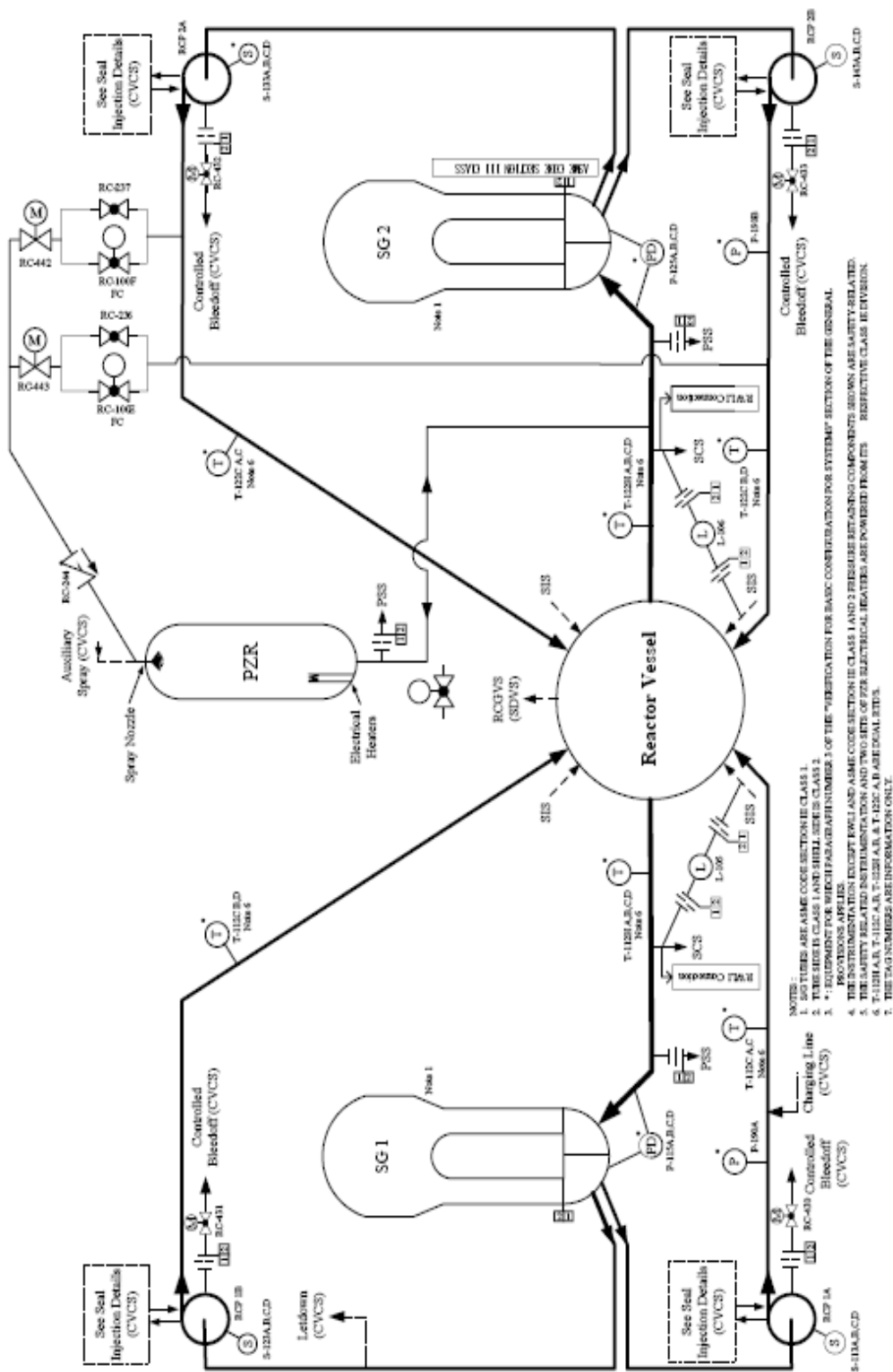


Figure 1.2 Schematic Diagram of Reactor Coolant System

Chapter 2. Safety Systems

2.1 Safety Injection System

In the highly unlikely event of a loss-of-coolant-accident (LOCA), the SIS injects borated water into the Reactor Coolant System. The APR1400 SIS incorporates a four-train safety injection configuration and an In-Containment Refueling Water Storage Tank (IRWST). The SIS utilizes four safety injection pumps to inject borated water directly into the Reactor Vessel (Refer to Figure 2.1). Each pump has 50% capacity against large break LOCA and 100% capacity for the other DBAs.

In addition, four safety injection tanks are provided in order to fill reactor vessel rapidly during large break LOCA. The passive internal device (Fluidic Device) is installed in the SIT tank (Refer to Figure 2.8). This device regulates discharged flow-rate passively depending on the water level. When the water level is above stand-pipe elevation, a large amount of water is discharged to fill reactor vessel during LOCA. When the water level is below stand-pipe elevation, small amount of water is discharged for long time (~200sec). By the adoption of the Fluidic Device, large capacity low pressure safety injection pumps are eliminated in APR1400 design.

The design of SIS is identical for all APR1400 plants.

2.2 Shutdown Cooling System

The Shutdown Cooling System (SCS) is used to reduce the temperature of the reactor coolant, at a controlled rate, from 350°F (176.7°C) to a refueling temperature of 120°F (48.9°C) and to maintain the proper reactor coolant temperature during refueling. This system utilizes the shutdown cooling pumps to circulate the reactor coolant through shutdown heat exchangers, returning it to the reactor coolant system. The component cooling water system supplies cooling water for the shutdown cooling heat exchangers.

The SCS for the APR1400 has a design pressure of 900 psig (63.28 kg/cm²G). This system pressure provides for greater operational flexibility and simplifies concerns regarding system over pressurization during inter-system LOCA.

The basic strategy of engineered safety feature design is enhancing safety by the adoption of four independent mechanical trains. For SCS design, APR1400 design has two different approaches.

The first approach is interchangeable design between Shutdown Cooling Pump (SCP) and Containment Spray Pump (CSP) using the identical design of SCP and CSP. In APR1400, SCS does not have emergency core cooling function during LOCA thus SCS does not operate with CSS concurrently. For this reason, CSP can back-up SCP, because CSP and SCP are identical. In these APR1400 plants, the SCS contains two heat exchangers and two pumps. If SCPs are out of service, CSPs can be manually aligned to perform the shutdown cooling function. APR1400 plants for Korea, USA and UAE adopt this approach. (Refer Figure 2.2(a))

2.3 Containment Spray System

The Containment Spray System (CSS) of APR1400 is designed to maintain containment pressure and temperature within design limits in the unlikely event of design basis mass-energy releases to the containment atmosphere.

The basic strategy of engineered safety feature design is enhancing safety by the adoption of four independent mechanical trains. For CSS design, APR1400 design has two different approaches.

The first approach is interchangeable design between Containment Spray Pump (CSP) and Shutdown Cooling Pump (SCP) using identical design of CSP and SCP. In these APR1400 plants, the CSS contains two identical containment spray pumps, two containment spray heat exchangers, two containment spray mini-flow heat exchangers, containment spray headers, and associated valves. Two containment spray pumps supply water through two heat exchangers to the upper region of the containment. Spray headers are used to provide a relatively uniform distribution of spray over the cross sectional area of the containment. The In-Containment Refueling Water Storage Tank (IRWST) is used as the water source for the system. If CSPs are out of service, SCPs can be manually aligned to perform the containment spray function. APR1400 plants for Korea, USA and UAE adopt this approach. (Refer Figure 2.3(a))

2.4 Auxiliary Feedwater System

The Auxiliary Feedwater System (AFS) is a part of Engineered Safety Feature System and provides an independent means of supplying secondary makeup water to the steam generators for the events when the Feedwater System is inoperable or unavailable.

The AFS consists of the following major equipment:

- A. Two (2) 100% capacity motor-driven AF pumps
- B. Two (2) 100% capacity turbine-driven AF pumps
- C. Four (4) motor-operated AF isolation valves with handwheel
- D. Four (4) solenoid-operated AF modulating valves

The AFS consists of two separate redundant trains. Each AF train consists of one (1) 100% capacity auxiliary feedwater storage tank (AFWST), one (1) 100% capacity motor-driven pump, one (1) 100% capacity turbine-driven pump, associated valves, cavitating venturi, piping, instrumentation and controls.

The design of AFS is identical for all APR1400 plants.

2.5 Over-Pressure Protection

In APR1400, there are two kinds of over-pressure protection systems. One is primary circuit over-pressure protection. The other is secondary circuit over-pressure protection.

For the primary circuit over-pressure protection, four POSRVs are installed on the top of the pressurizer in APR1400 plants for Korea, UAE, and US NRC. In these countries, POSRVs are treated as a passive component for which we do not need to apply single failure assumption.

For the secondary circuit over-pressure protection, 20 MSSVs and 4 MSADVs are installed on the main steam lines in APR1400 plants for Korea, UAE, and US NRC. APR1400 has 4 main steam lines. Therefore, 5 MSSVs and one MSADV are

installed on each main steam line. In these countries, MSSVs are treated as a passive component for which single failure assumption is not need to apply. The schematic diagram of secondary circuit over-pressure protection system is shown in Figure 2.5.

2.6 Support Systems

In APR1400, there are several safety related systems to support safety systems as follows;

- Component Cooling Water System (CCWS)
- Essential Service Water System (ESWS)
- Essential Chilled Water System (ECWS)
- MCR HVAC System
- MCR Emergency HVAC System

The design of the above systems is basically same for all APR1400 plants. Each system has two divisions. One or two EDG supplies electricity for the components in each division (See Subsection 4.2). The capacity of each system is as follows;

- CCWS : 2 × 100% CCW pumps / Div.
- ESWS : 2 × 100% ESW pumps / Div.
- ECWS : 2 × 100% Chillers / Div.
- MCR HVAC System : 1 × 100% AHU / Div. (APR1400 DC : 2 at 100% / Div.)
- MCR Emergency HVAC System : 1 × 100% AHU / Div.

Although APR1400 plants have same number of pumps or chillers, the capacities are different, because different environment conditions are applied. The dominant environment condition to determine the capacity of pumps or chillers is outdoor temperatures. Each APR1400 plant has different reference outdoor temperature depending on the site specific conditions as follows;

- SKN 3&4 : 38.6 ~ -16.7 °C
- BNPP : 50 ~ 0 °C
- NRC DC : 46.1 ~ -40 °C

The capacity of pumps or chillers of the support systems are summarized in Table 2.1.

Table 2.1 The Comparison of Pump/Chiller Capacity of the Support Systems

	CCW Pump L/min (gpm)	ESW Pump L/min (gpm)	ECW Chiller L/min (gpm)	MCR AHU* L/min (gpm)
SKN 3&4	70,030 (18,500)	64,352 (17,000)	3,331.2 (880)	716,417(189,257)
BNPP	75,708 (20,000)	75,708 (20,000)	7,154 (1,890)	836,767 (221,050)
NRC DC	94,635 (25,000)	75,708 (20,000)	8,710 (2,300)	826,833 (218,426)

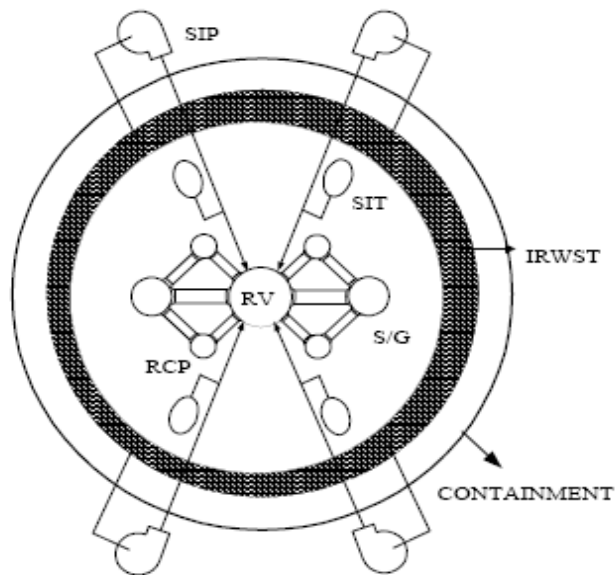


Figure 2.1 Schematics of Safety Injection System

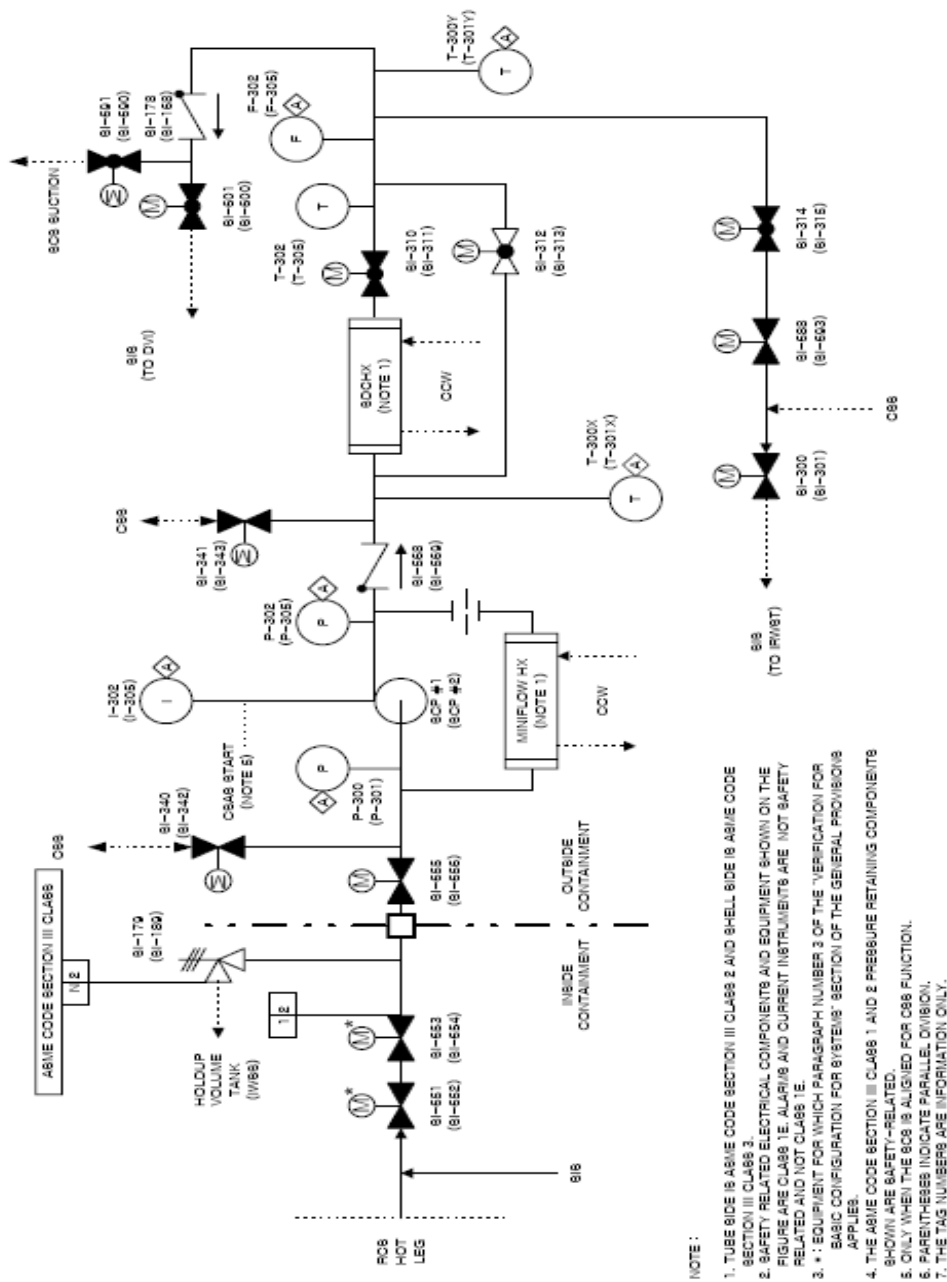


Figure 2.2 Schematics of Shutdown Cooling System

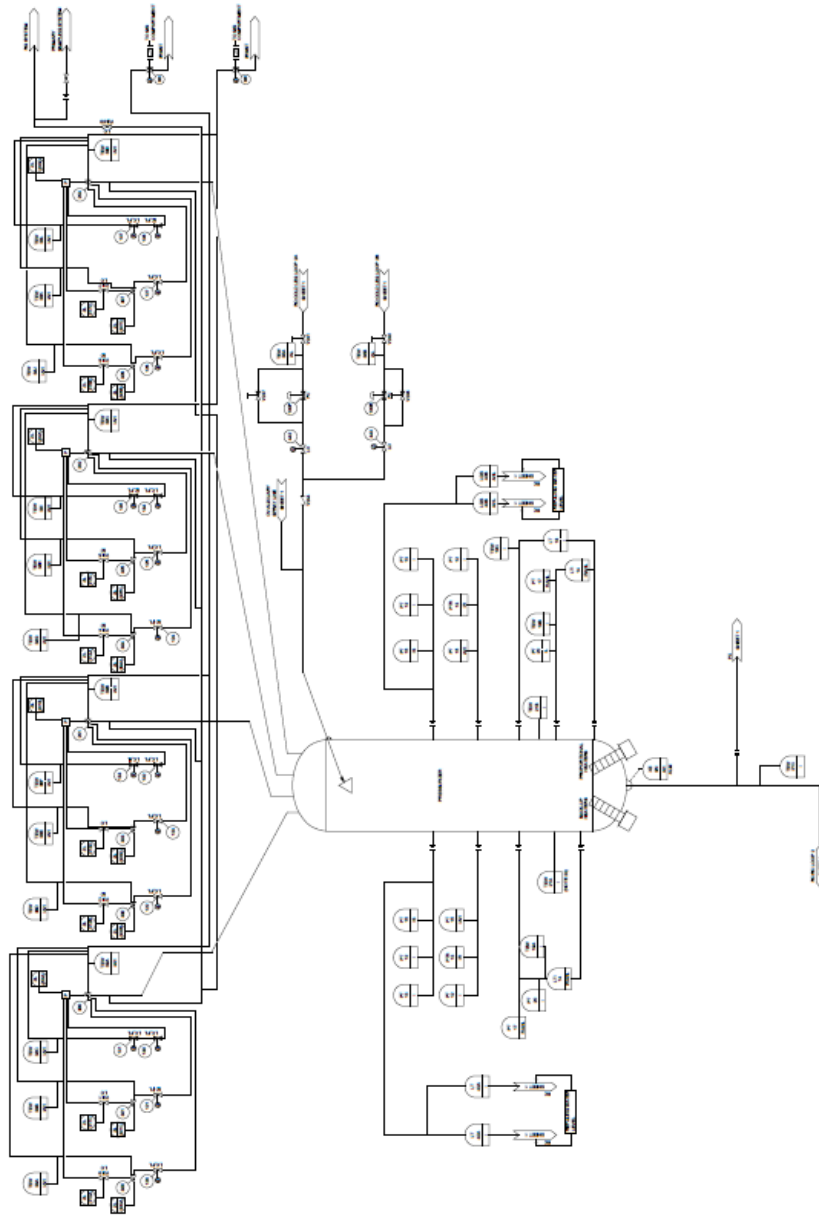


Figure 2.4 Schematics of Primary Over-pressure Protection System

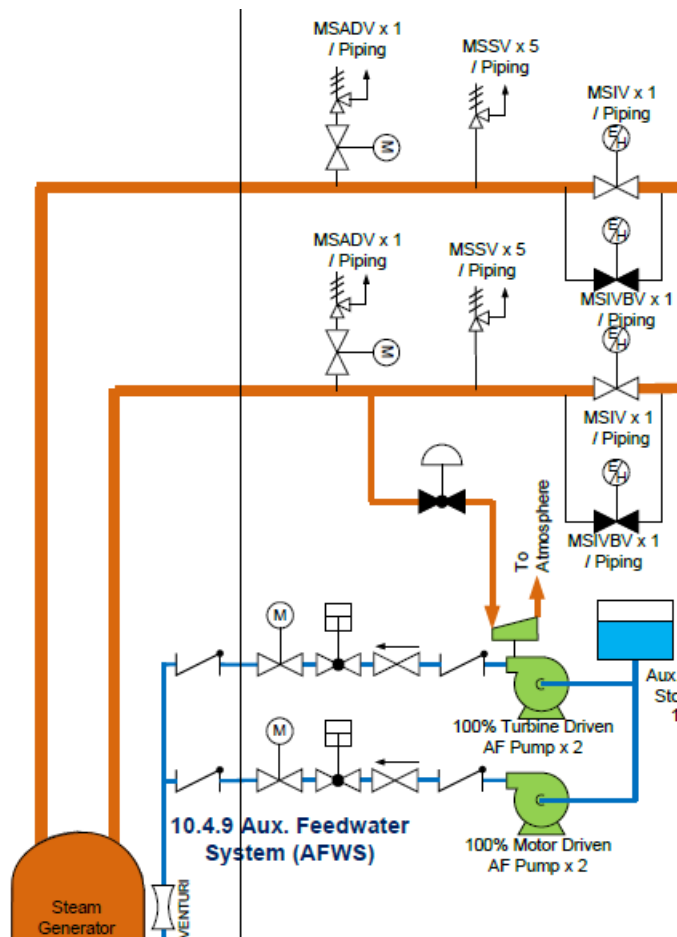


Figure 2.5 Schematics of Secondary Over-pressure Protection System

Chapter 3. Severe Accident Mitigation Systems

3.1 Rapid Depressurization during Severe Accident

The rapid depressurization feature of the Safety Depressurization System (SDS) serves an important role in severe accident mitigation. When a high pressure meltdown scenario develops and the feed portion of feed and bleed cannot be established due to unavailability of the SI pumps, the SDS can be used to depressurize the RCS to ensure that a High Pressure Melt Ejection (HPME) event does not occur, thereby minimizing the potential for the Direct Containment Heating (DCH) following a vessel breach.

In original APR1400 design, the rapid depressurization function is performed by POSRVs on the top of pressurizer, discharge lines from the pressurizer to the IRWST, and the IRWST which acts as a water reservoir to condense the steam effluent from the RCS and collect the discharged fluid as shown in Figure 3.1. After recognition of the initiation of severe accident by CET temperature, operator opens POSRVs to reduce RCS pressure below 200psig. Then discharged fluid is collected into the IRWST. During severe accident, hydrogen could be generated in the core by metal-water reaction, and thus, discharged fluid might be mixture of steam and hydrogen. Steam is condensed in the IRWST but hydrogen could be accumulated in the IRWST upper annulus. The possibility of hydrogen accumulation in IRWST has been identified and discussed between the Korean Regulatory Body and the Vendor.

As the results of the discussion to resolve the hydrogen accumulation, Shin-Kori 3&4, the first APR1400 plants, modified design to adopts 3-way valve on the discharge line from POSRVs. During feed-and-bleed operation under the total loss of feedwater accident, discharged fluid is directed to the IRWST by the 3-way valve. In case of severe accident, 3-way valve is directed to SG compartment to avoid accumulation of hydrogen in the IRWST by the operator. APR1400s in UAE and USA adopt this design features. This design feature requires manual action during severe accident. The discussion between the Korean Regulatory Body and the Vendor to remove manual action results in the design modification as shown in Figure 3.3.

As shown in Figure 3.3, the separate valve is added on the top of the pressurizer named Emergency RCS Depressurization Valve (ERDV) in Shin-Kori 5&6. It

discharges steam and hydrogen to steam generator compartment and containment atmosphere during severe accident.

3.2 Hydrogen Control System

During a degraded core accident, hydrogen is generated at a greater rate than that of the design basis LOCA. The containment hydrogen control system of APR1400 is designed to accommodate the hydrogen generated from the metal-water reaction of 100 percent of the active fuel cladding and limit the average hydrogen concentration in containment below 10 percent to meet the requirements in 10 CFR 50.34(f) and 10 CFR 50.44 (Reference 5) for a degraded core accident. These limits are imposed to preclude detonations in containment that might jeopardize containment integrity or damage essential equipment.

The containment hydrogen control system consists of a system of passive autocatalytic recombiners (PARs) complemented by glow plug igniters installed within the containment. The PARs are capable of controlling hydrogen in all accident sequences with moderate hydrogen release rates, and are located throughout the containment. The igniters supplement PARs for accidents in which rapid hydrogen release rates are expected, and are placed near anticipated source locations to enhance the combustion of hydrogen in a controlled manner.

The most APR1400s has 30 PARs and 10 igniters.

US-APR1400 has 30 PARs and 8 igniters. There is no igniter in the upper free volume of IRWST because POSRV discharges the steam and hydrogen to steam generator compartment and the hydrogen concentration is not high in the IRWST.

The PARs are strategically distributed so that the overall average concentration requirements are met. These locations are determined based on equipment and piping proximity as well as inspection and maintenance access. The PAR components and igniter assembly are designed to meet seismic Category I requirements.

3.3 Cavity Flooding System including Core Catcher

After reactor vessel failure, core debris is discharged into the containment and Molten Core Concrete Interaction (MCCI) begins, leading to erosion of the concrete in the reactor vessel cavity. This threatens the integrity of the containment pressure boundary due to the possibility of melt-through of containment liners and the concrete basemat.

The reactor cavity is configured to promote retention of, and heat removal from, the postulated core debris during a severe accident, thus serving several roles in severe accident mitigation. The large cavity floor area allows for spreading of the core debris, enhancing its coolability within the reactor cavity region. The containment liner plate in reactor cavity area is embedded in the concrete 0.91 m (3 ft.) below from the cavity floor at the minimum.

The cavity flooding system (CFS) provides a means of flooding the reactor cavity during a severe accident to cool the core debris in the reactor cavity and to scrub fission product. The water delivery from the IRWST to the reactor cavity is accomplished by means of active components.

The CFS takes water from the IRWST and directs it to the reactor cavity. The water flows first into the HVT by way of the two HVT spillways and then into the reactor cavity by way of two reactor cavity spillways. Flooding of the HVT progresses until the water levels in IRWST, HVT, and reactor cavity equalize at 6.4 m (21 ft.) from the reactor cavity.

In Original APR1400, the concrete composition of reactor vessel cavity wall and floor is the same as that of all containment building and it is a kind of basaltic concrete. In the SKN 3 and 4, one feet of Limestone Common Sand (LCS) concrete is added to the floor and wall of the cavity. It is known that LCS concretes produces more gaseous products and makes weaker crust following the interaction with molten corium than a basaltic or siliceous concrete does. The US APR1400 and SKN 5, 6 use only LCS concrete in the floor and wall of reactor cavity. Figure 3.4 shows the schematic design of reactor cavity and cavity flooding system.

3.4 Containment Depressurization during Severe Accident

According to the requirements for containment performance for the APR1400 design,

the containment is designed so that the containment meets the FLC requirement of ASME Section III, Division 2, Subarticle CC-3720. For a provision against a beyond-design-basis accident where either two SC pumps and two CS pumps or the IRWST is unavailable, the Emergency Containment Spray Backup System (ECSBS) is provided as an alternative to the CSS.

The ECSBS is designed to protect the containment integrity against overpressure and prevent the uncontrollable release of radioactive materials into the environment. The emergency containment spray flow path is from external water sources (the reactor makeup water tank, demineralized water storage tank, fresh water tank, or the raw water tank), through the fire protection system line via the diesel-driven fire pump, to the ECSBS line emergency connection located at ground level near the auxiliary building. Figure 3.5 shows the schematic diagram of ECSBS and containment spray.

ECSBS operation began 24 hours after the onset of core damage and is capable of controlling containment pressure and reducing containment atmospheric temperature for a period of 48 hours. The maximum pressure and temperature following the initial 24-hour period are enveloped by the maximum pressure and temperature during the initial 24-hour period. This prevents the uncontrolled release of fission products into the environment.

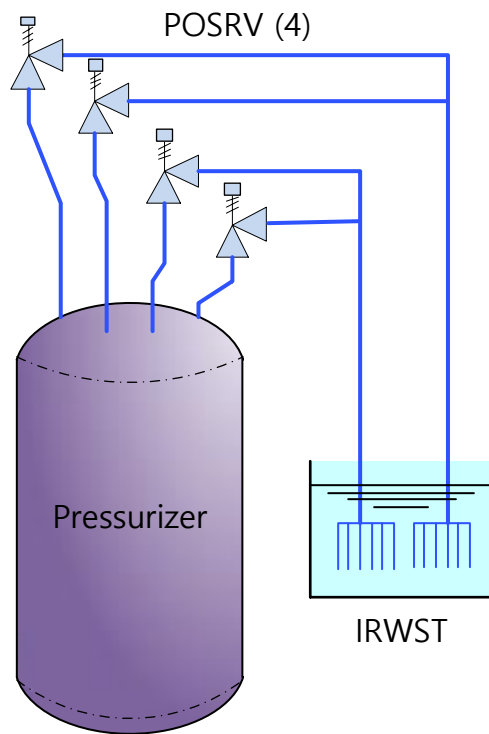


Figure 3.1 Schematics of Rapid Depressurization Features of original APR1400

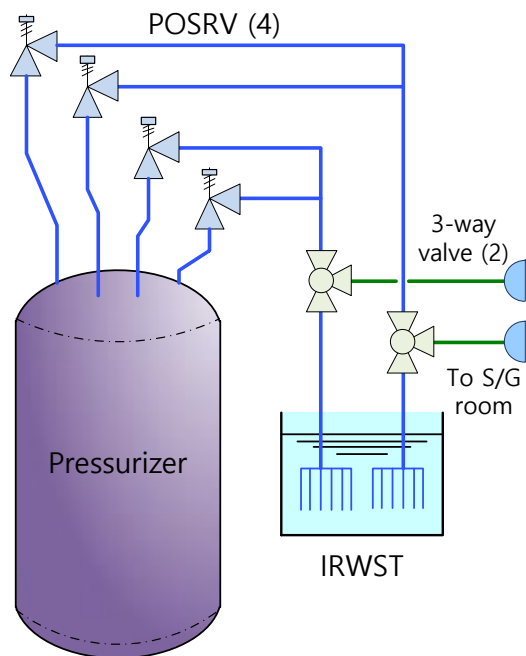


Figure 3.2 Schematics of Rapid Depressurization Features of Shin-Kori 3&4

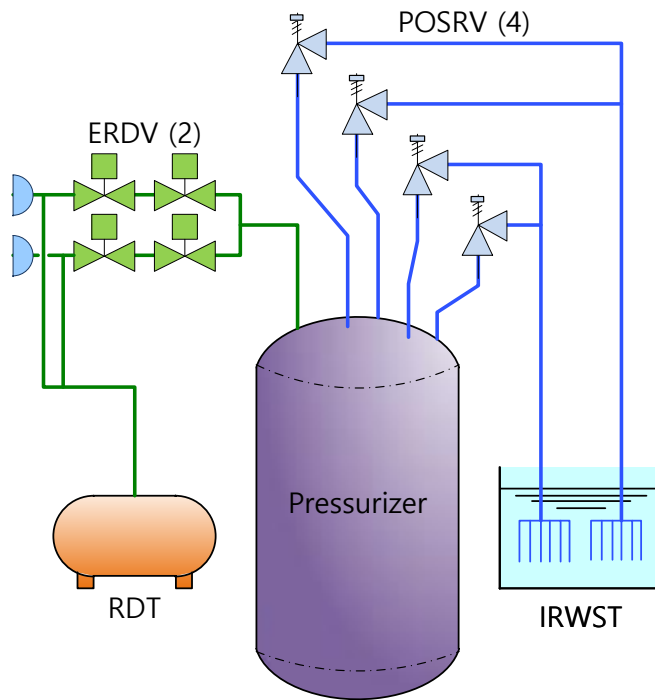


Figure 3.3 Schematics of Rapid Depressurization Features of Shin-Kori 5&6

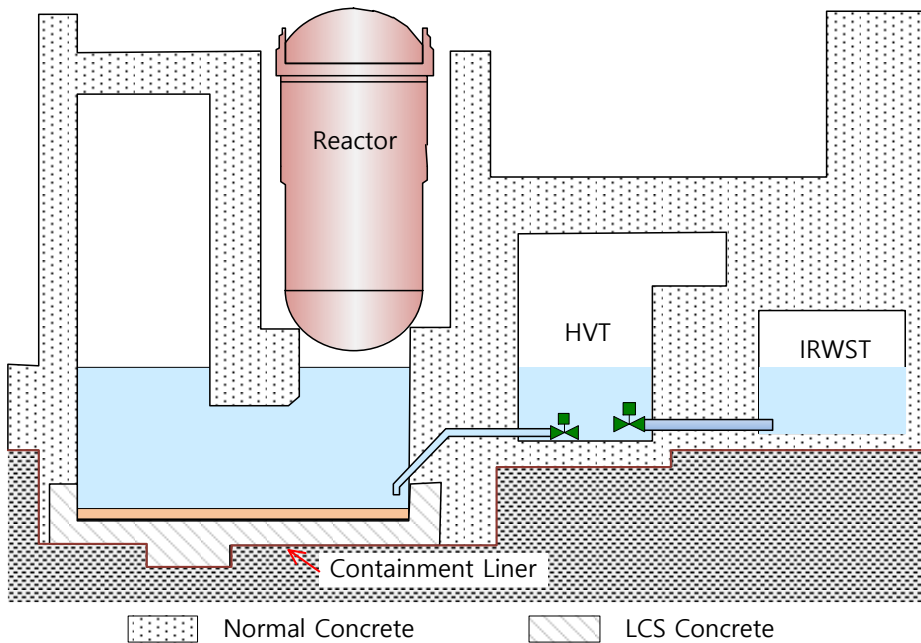


Figure 3.4 Schematics of Reactor Cavity and Cavity Flooding System (Shin-Kori 5&6, US-APR1400, etc.)

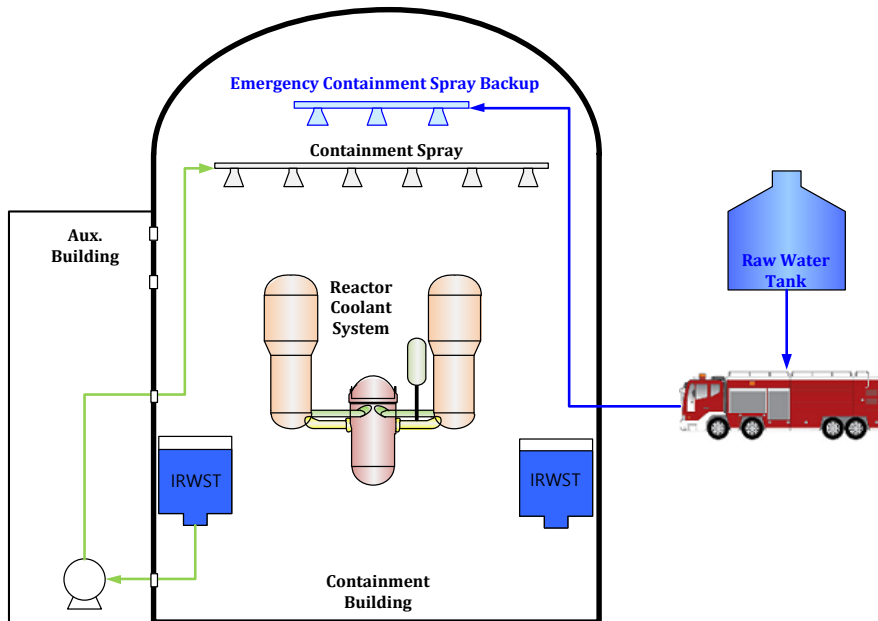


Figure 3.5 Schematics of ECSBS and CSS

Chapter 4. I&C and Electrical System

4.1 I&C System Platform

The APR1400 instrumentation and control (I&C) system consists of safety I&C system, non-safety control system, diverse actuation systems (DAS) and human-system interfaces (HSIs) in the main control room (MCR) and remote shutdown room (RSR).

The safety I&C systems, which are based on a common PLC platform, consist of plant protection system (PPS), core protection calculator system (CPCS), engineered safety features – component control system (ESF-CCS), and qualified indication and alarm system – P (QIAS-P) for accident monitoring.

Most of non-safety I&C systems are implemented in a distributed control system (DCS) which has been proven by operating experience from the nuclear industry as well as other industries. The non-safety control systems consist of plant control system (PCS) and the process-component control system (P-CCS). The PCS includes the reactor regulating system (RRS), the digital rod control system (DRCS), and the reactor power cutback system (RPCS). The P-CCS includes NSSS process control system (NPCS) and BOP control systems. The NPCS consists of the feed water control system (FWCS), the steam bypass control system (SBCS), the pressurizer pressure control system (PPCS), the pressurizer level control system (PLCS), and other miscellaneous NSSS control systems.

The Diverse Protection System (DPS) is designed to mitigate the effects of an anticipated transient without scram (ATWS) event characterized by an anticipated operational occurrence (AOO) followed by a failure of the reactor trip portion of the protection system. In addition, the DPS is designed to include functions to assist in the mitigation of the postulated software Common Cause Failure (CCF) of the digital safety I&C system coincident with a design base accident.

In original APR1400 design, DPS is implemented in a DCS which is independent from the safety I&C system. There are two platforms, PLC for safety system and DCS for non-safety control system including DPS. DPS consists of two channels and generates trip signal or ESF actuation signal by 2-out-of-2 logic to prevent spurious actuation.

In APR1400 design for NRC DC project, DPS is implemented in a Field Programmable Gate Array (FPGA) based logic controller which is diverse from the safety I&C system and non-safety control system. There are three platforms which are PLC, DCS and FPGA in APR1400 design for NRC DC project. Figure 4.1 shows the I&C system architecture of APR1400 for NRC DC. DPS is designed with four channels and use 2-out-of-4 logic to improve availability. For the implementation of four channel DPS, two reactor trip switchgear is added.

4.2 Emergency Diesel Generator System

In the event of a complete loss of off-site power (LOOP), the Emergency Diesel Generator System (EDG), in conjunction with the Class 1E Auxiliary Power System, provides an on-site standby source of AC electric power to the Class 1E loads, such as the Engineered Safety Feature System (ESF) and equipment required to 1) safely shutdown the reactor, 2) maintain the reactor in a safe shutdown condition so that off-site radiation doses are kept below the requirement of 10 CFR 100. In addition to these loads, the diesel generator also supplies power to selected critical Non-Class 1E loads, per NUREG-0737, only after the operator has determined that generator has enough spare capacity to feed these loads. The EDG is designed to attain rated voltage and frequency within 20 seconds. Once the EDG has reached rated voltage and speed, the diesel generator breaker closes and the sequencer generates proper signal to connect ESF equipment to the Class 1E bus in a programmed time sequence.

The EDG in SKN 3&4 consists of two identical and independent subsystems, one for each ESF division i.e. two identical EDG sets are provided for each plant unit. Each subsystem serves as a full-capacity independent on-site standby power source, thereby satisfying the single failure criteria. The two divisions are physically and electrically separated. Each diesel generator set consists of one generator rated for 8,000 kW continuous operation and one diesel engine.

The EDG in BNPP is identical to the EDG in SKN 3&4 except the generation capacity. The capacity of BNPP EDG is 8,700 kW. The main reason of this difference is electrical load increase by ambient conditions of BNPP described in Section 2.6.

As described in Section 2.6, the ambient conditions for APR1400s in USA and

Finland have very wide range to cover enveloped temperature condition of those countries. The capacities of pumps and chillers are related to outdoor temperature condition. EDGs for USA have much larger capacities than those of SKN 3&4 (Refer Table 2.1). The designer has discussed the possibility of supplying larger size EDGs, for example, larger than 15,000 kW. Finally, the designer has decided to install 4 EDGs instead of 2 EDGs for USA project. One division consists 2 EDGs with 9,100 kW (EDG A and B) and the other division consists 2 ECCS only EDGs with 7.500 kW.

4.3 Alternate AC Diesel Generator System

In the event of a loss of off-site power(LOOP), the Alternate AC (AAC) power supplier provide Non-Class 1E power to permanent non-safety loads such as turning gear oil pumps. The AAC power supplier is also used as an Alternate AC source to cope with Station Blackout (SBO).

In SKN 3&4 and BNPP, Diesel Generator is adopted as AAC. AAC Diesel Generator set including one diesel fuel oil storage tank for 3 days of operation plus 15% additional storage of fuel required to test the engine periodically, two diesel fuel oil transfer pumps and one fuel oil day tank, is provided for the APR1400. In the event of SBO, the AAC Diesel Generator automatically starts and is ready to accept load within 20 seconds from a start signal and the AAC will be available to supply power within 10 minutes of the SBO. The capacity and the number of AAC DG are as follows;

- SKN 3&4 : 7,200 kW, one AAC DG
- BNPP : 8,700 kW, one AAC DG

In APR1400 for NRC DC, the 4.16 kV Non-Class 1E AAC GTG is provided as an AAC source. The AAC GTG has sufficient capacity to operate the system necessary for coping with the SBO for the time required to bring and maintain the plant in a safe shutdown condition. The AAC GTG is designed to attain rated voltage and frequency within 2 minutes after the reception of a starting signal. The AAC GTG is manually connected to the designated Class 1E 4.16 kV switchgears (train A or train B) by the operator within 10 minutes from the beginning of the SBO event.

To minimize the potential for common-cause failures with Class 1E EDGs, the AAC GTG is provided with a gas turbine engine with a diverse starting and cooling system. The AAC GTG, including the related auxiliary equipment, is installed in a separate

building. Therefore, no single-point vulnerability exists in which a weather-related event or single active failure disables any portion of the onsite EAC sources or the offsite power sources and the AAC source.

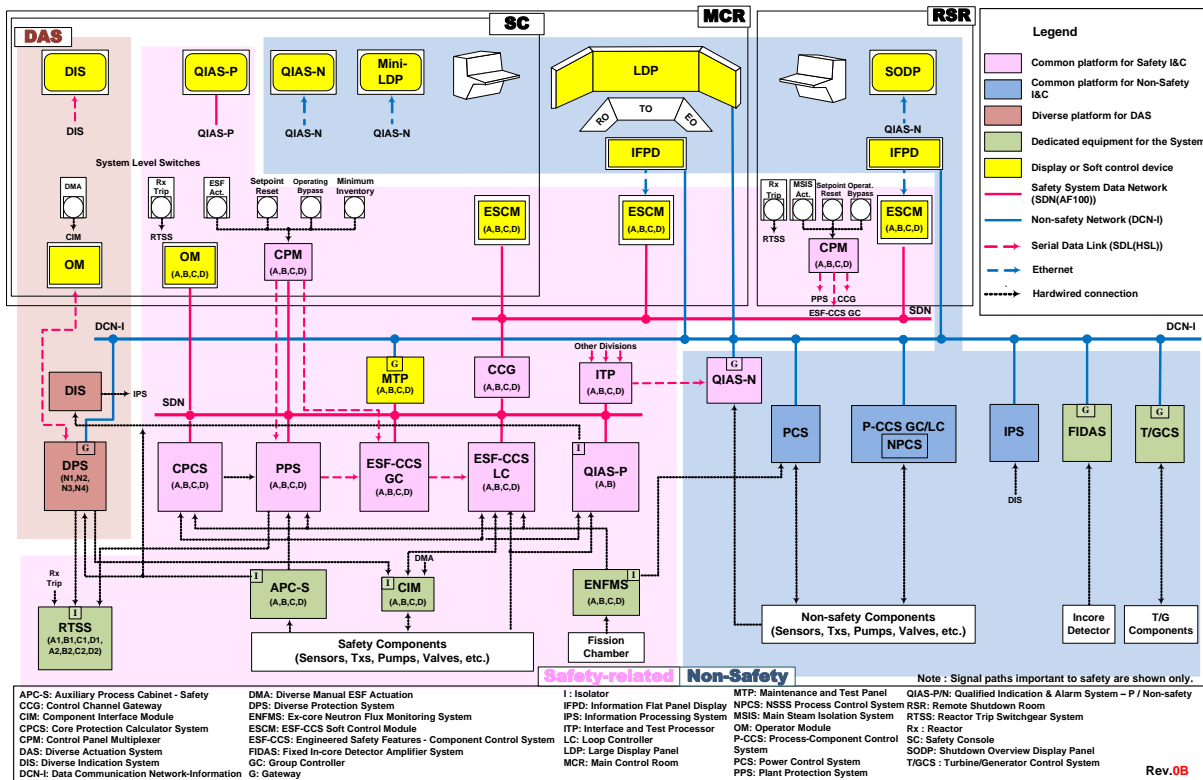


Figure 4.1 I&C System Architecture of APR1400 for NRC DC

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