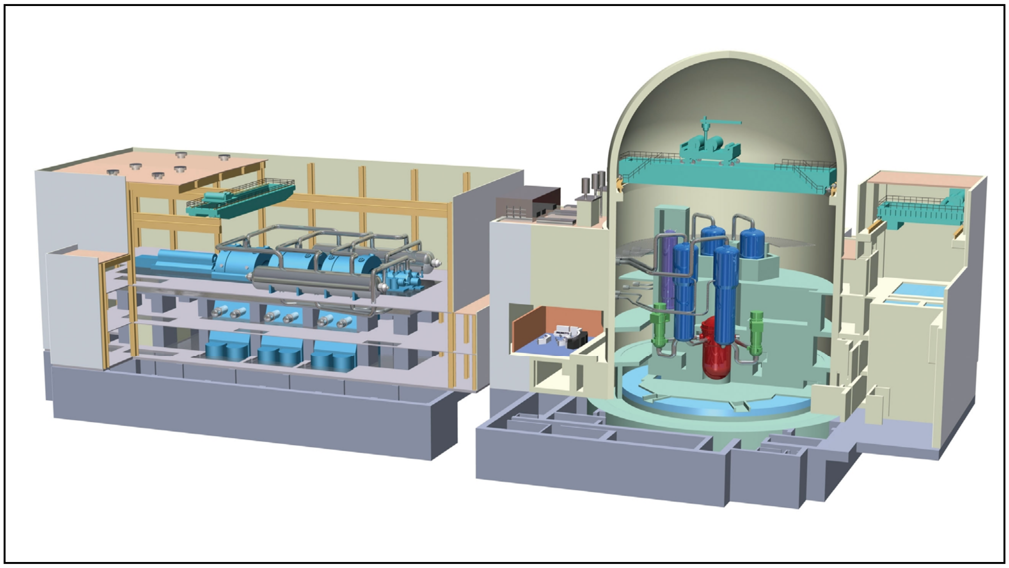
US-APWR DESIGN DESCRIPTION



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The design and engineering information has been prepared by MHI in connection with its request to the NRC for a pre-application review of the US-APWR.

All figures and numerical values used in this document are based on data available at the time of preparation of this document, and are subject to change as necessary. The final design information of US-APWR will be provided in the Design Control Document.

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Table of Contents

Table of Contents
(continued)

Table of Contents
(continued)

List of Tables

List of Figures

List of Figures
(continued)

List of Acronyms

A/B	Auxiliary Building
AC/B	Access Control Building
ACC	Accumulator
ACI	American Concrete Institute
AIC	Ag-In-Cd
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	As Low As Reasonably Achievable
ALR	Automatic Load Regulator
APWR	Advanced Pressurized Water Reactor
ASCE	American Society of Civil Engineers
ATWS	Anticipated Transient Without Scram
AVR	Auto Voltage Regulator
AWS	American Welding Society
BISI	Bypassed or Inoperable Status Indication
BRS	Boron Recycle System
CCW	Component Cooling Water
CCWS	Component Cooling Water System
CDF	Core Damage Frequency
CMF	Common Mode Failure
CRDM	Control Rod Drive Mechanism
CS/RHR	Containment Spray / Residual Heat Removal
CSS	Containment Spray System
CV	Containment Vessel
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DAS	Diverse Actuation System
DBA	Design Basis Accident
DCS	Data Communication System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio

List of Acronyms

(continued)

DVI	Direct Vessel Injection
ECCS	Emergency Core Cooling System
EFWS	Emergency Feed Water System
EHG	Electro Hydraulic Governor
ELD	Emergency Letdown System
EOF	Emergency Operations Facility
ERF	Emergency Response Facility
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
ESW	Essential Service Water
FA	Fuel Assembly
FP	Fission Products
FWS	Feed Water System
GLBS	Generator Load Break Switch
GT/G	Gas Turbine Generator
GT/B	Gas Turbine Building
GWMS	Gaseous Waste Management System
HDSR	Historical Data Storage and Retrieval
HEPA	High-Efficiency Particulate Air
HHSI	High Head Safety Injection
HVAC	Heat, Ventilation, and Air Conditioning
HSI	Human System Interface
HX	Heat Exchanger
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICIS	In-Core Instrumentation System
ICTS	In-Core Temperature System
I&C	Instrumentation and Control
IHP	Integrated Head Package
I/O	Input/Output

List of Acronyms
(continued)

ISB	Integral Shroud Blade
LBB	Leak Before Break
LCSP	Lower Core Support Plate
LDP	Large Display Panel
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Off-site Power
LWMS	Liquid Waste Management System
MCB	Main Control Board
MCP	Main Coolant Pipe
MCR	Main Control Room
MELTAC	Mitsubishi Electric Total Advanced Controller for Nuclear Application
MFW	Main Feed Water
MFWS	Main Feed Water System
MHI	Mitsubishi Heavy Industries, ltd.
MMF	Minimum Measured Flow
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSR	Moisture Separator and Re-heater
MSRV	Main Steam Relief Valve
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MTTR	Mean Time To Repair
NDL	Nuclear Data Link
NIS	Nuclear Instrumentation System
NPSH	Net Positive Suction Head
NR	Neutron Reflector
NSSS	Nuclear Steam Supply System
OLM	On-Line Maintenance
OPC	Overspeed Protection Controller
OPDMS	On-line Power Distribution Monitoring System

List of Acronyms
(continued)

OS	Operating System
PAM	Post-Accident Monitoring System
PAR	Passive Autocatalytic Recombiner
PCCV	Pre-stressed Concrete Containment Vessel
PCMS	Plant Control and Monitoring System
PGA	Peak Ground Acceleration
PIF	Power Interface Card
PMES	Primary Make-up Water System
POL	Problem Oriented Language
PORV	Power Operated Relief Valve
PORVIV	Power Operated Relief Valve Isolation Valve
PSMS	Protection and Safety Monitoring System
PZR	Pressurizer
R/B	Reactor Building
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RMS	Plant Radiation Monitor System
RPS	Reactor Protection System
RPI	Rod Position Indicator
RSC	Remote Shutdown Console
RTDP	Revised Thermal Design Procedure
RTP	Rated Thermal Power
RV	Reactor Vessel
RVCH	Reactor Vessel Closer Head
RVH	Reactor Vessel Head
RVI	Reactor Vessel Internal
RSS	Remote Shutdown Station

List of Acronyms
(continued)

RWSAT	Refueling Water Storage Auxiliary Tank
RWSP	Refueling Water Storage Pit
SA	Severe Accident
SBO	Station Black Out
SC	Steel Concrete
SDV	Safety Depressurization Valve
SFP	Spent Fuel Pit
SFPCS	Spent Fuel Pit Purification and Cooling System
SG	Steam Generator
SGBDS	Steam Generator Blow Down System
SGTR	Steam Generator Tube Rupture
SIP	Safety Injection Pump
SIS	Safety Injection System
SPDS	Safety Parameter Display System
SRV	Safety Relief Valve
SS	Sampling System
SSC	Structures, Systems, and Components
SSE	Safety Shutdown Earthquake
STA	Shift Technical Advisor
SWS	Essential Service Water System
T/B	Turbine Building
TC	Core Exit Thermo Couple
TCS	Turbine Building Closed Cycle Cooling water System
TD	Theoretical Density
TDF	Thermal Design Flow
T/G	Turbine/Generator
TSC	Technical Support Center
UHS	Ultimate Heat Sink
UMC	Unit Management Computer
UPS	Uninterruptible Power Supply

List of Acronyms
(continued)

VCT	Volume Control Tank
VDU	Visual Display Unit
V&V	Verification and Validation
WMS	Waste Management System

1.0 INTRODUCTION

This document provides overall design features of the US-APWR to assist in understanding the features of design concepts, fuel, main components, structures, fluid systems, and I&C and electrical systems. This document is provided to the NRC for information only, and is not intended as a formal submission for staff review.

1.1 Safety Design Philosophy

It is most important that both the public and workers are protected by installing in the US-APWR an effective barrier against radioactive materials and hazards. Main design concepts of US-APWR are utilization of the proven technology and well-balanced safety design. Significant experiences in the design, fabrication, installation, construction and operation of pressurized water reactors in Japan produce proven technologies of MHI. These proven technologies are used to the US-APWR. The US-APWR is a plant that consists of highly reliable prevention functions, well-established mitigation systems with active safety functions and passive safety functions and measures against beyond design basis accidents. These three functions are integrated in balance design.

In order to achieve this design, the deterministic design approach and the risk management technology using Probablistic Risk Assessment are applied. Furthermore the reliability of the physical barriers and the protection level are improved based on the concept of defense in depth, which is applied from normal operation to severe accidents.

The design of the US-APWR based on above procedures meets the U. S. Regulatory requirements.

Finally the following safety goal about core damage frequency and large release frequency are referred to evaluate safety of the US-APWR.

- Core Damage Frequency
  The target of Core Damage Frequency is less than 10^{-5}/reactor-year for internal and external events during all operation modes.
- Large Release Frequency
  The target of Large Release Frequency is less than 10^{-6}/reactor-year.

The following parts explain the basic policies of safety design and the measures. (Ref. Table 1.1-1)

Table 1.1 -1 Design classification and basic concept of measures corresponding to plant condition

Objective of defense	Basic concept	
	Essential means	Points of improvement
Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation	•Careful selection of materials and use of qualified fabrication processes •Adequate margins in the design of systems and plant components •Utilization of operating experience
Detection of failures, control of abnormal operation	•Protection systems	•Redundancy •Separation
Control accident within the design basis	•Engineered safety features including critical support systems	
Control of beyond design basis accident	Complementary measures and accident management	Diverse systems different from protection systems and engineered safety features

1.1.1 The Basic Policies of Safety Design against internal events

The objectives of each level of defense in depth are as follows.

- Prevention of abnormal operation and failure
- Detection of failures and control of abnormal operation
- Control accident within the design basis
- Mitigation of Beyond-Design-Basis-Accidents

1.1.1.1 Prevention of abnormal operation and failure

For prevention, the following measures are installed.

- Inherent safety

  The reactor is designed so that it has the inherent safety that does not result in anticipated transients or accidents by self-regulating characteristics, such as the Doppler effect and the moderator density effect.

- To design and manage so that the cause of abnormal operation is minimized by
  - Providing extensive margin
  - Improving equipment and control system reliability
  - Performing strict quality control during the manufacture of components
- Enhanced reliability of reactor coolant pressure boundary
  Following measures are applied to the US-APWR design.
  - Alloy 690 is used for vessel head nozzle
  - T_{cold} at vessel head plenum is achieved by increase of core bypass flow.
- Improvement of maintenance

Structures, systems, and components are designed so that occurrence of violations of Technical Specifications is reduced under On Line Maintenance by increasing redundancy of function such as four trains of Engineered Core Cooling Systems, electrical power systems, Instrumentation & Control systems, and plant cooling water systems.

- Improvement of safety during shut down

Based on Shutdown Probabilistic Risk Assessment, measures are provided to enhance plant safety by improving operation management.

- Reduction of operator's load and enhanced reliability of I & C system
  Advanced control room with enhanced operability and integrated digital technology with redundant architecture reduce operator load and increase reliability of I & C system

1.1.1.2 Detection of failures and control of abnormal operation

In the case of a certain system failure or human error occurring during operation, the abnormal conditions are detected at an early stage, and the following measures are taken for prevention of the further progress of the abnormal conditions.

- Signals
  Full four train protection systems with reactor trip breakers are provided to initiate a reactor trip when system failure or human error occur with phenomena capable of causing further deterioration in the plant status
- Shutdown system
  Reactor safety shutdown system is provided which consists of the control rods for reactor trip. Furthermore cold shutdown can be achieved by the following measures.
  - Emergency core cooling system
  - Emergency letdown line
  - Safety depressurization and vent system
- Design features that assure, if a malfunction does occur, protection measures will be taken.
  - Sufficient design margin
  - Use fail-safe design approaches where possible
  - Improve reliability of safeguard systems
  - Assure strict quality control in the manufacture of components
  - Make use of redundancy of design

1.1.1.3 Control accident within the design basis

The following measures are taken to prevent the progress of accidents, to mitigate effects of accidents and to protect both the public and site workers.

- Signals
  If accidents do occurs, signals initiate engineered safety systems.
- Safety systems
  To install high reliability emergency core cooling systems which cool the core in response to safeguards signals
  - Safety systems such as Accumulators, Safety Injection System, Containment Spray System and Emergency Feed Water System have four trains in separated divisions. Additionally Electrical Safety Systems, Emergency Power Generations and Service Water Systems have four trains.
  - The refueling water storage pit in the containment eliminates the switchover of suction of safety injection systems.

- The advanced accumulators with a vortex damper have two injection modes of large flow and small flow. The advanced passive accumulator substitutes for the low head safety injection pump.

- Sufficient time margin for operation
  If manual operation is needed, the operator has a sufficient time margin for making the required judgment, and actions are easily operated and virtually mistake-proof.

- Containment Vessel (CV)
  Build a containment vessel which is an effective pressure barrier to the environment, and the barrier to a diffusion of the radioactive materials

- CV spray system
  Install four train CV cooling spray system that assures intactness of CV

- Annulus
  Install an airtight circular space (annulus) between the containment vessel and the reactor building. The pressure of annulus is kept negative so that the release of the radioactive materials to the environment is controlled in the case of accidents.

- Emergency Power Generators
  Install high reliability gas turbine generator and relaxing start up time of emergency core cooling system by using advanced accumulators.

1.1.1.4 Mitigation of Beyond-Design-Basis-Accidents

The US-APWR establishes the following accident measures through the use of Probabilistic Risk Assessment. These measures are diverse compared to the above safety systems.

- Measures against Station Blackout
  Diversity of emergency power source to mitigate Station Blackout. The system is installed for achieving Safe Shut Down to cool down state after Station Blackout.

- Additional Protection against a System Interface LOCA
  The higher rated piping of residual heat removal systems reduces the occurrence of interfacing system LOCA. Even if isolation valves of residual heat removal system opened due to malfunction during normal operation, reactor coolant from main coolant pipe would flow to refueling water storage pit without pipe break outside containment.

- Measures against common mode failures in the digital safety system
  Diverse actuation system that consists of reactor trip, turbine trip and emergency feed water system initiation is installed to prevent common mode failures.

- Measures against severe accident after core damage
  Measures for prevention and/or mitigation of certain scenarios such as hydrogen combustion, molten core concrete interaction, high pressure melt ejection, etc.

1.1.2 The Basic Policies of Safety Design against fire and external events

1.1.2.1 Design for Natural Phenomena

Earthquake, tornado and hurricane are considered as natural phenomena.

	Explanation	Remarks
Design Requirement	10 CFR Part 50 Appendix 50 Criteria 2	SSC important to safety will be designed to withstand the effects of natural phenomena.
Considered Phenomena	Earthquake	Design response spectra are developed based on RG 1.60. Latest knowledge for design response spectra is considered. PGA of 0.3g is used to cover more than 70% of USA site. Design response spectra is ensured envelope potential site.
	Tornado	Maximum speed 300mph is taken based on RG 1.76 & SRP 2.3.1.
	Hurricane	ASCE 7-98 is used with basic wind speed of 145mph
Analytical Method	Earthquake	Equivalent static loads are computed time history analysis with 3D lumped mass stick model. Stress will be obtained employing shell model.
	Tornado/Hurricane	Static analysis is employed
Applied Regulations	ACI, AISC, AISI,AWS	Generally accepted US codes are applied.

1.1.2.2 Pipe Rupture Protection

The structures, systems, and components important to safety are protected against the dynamic effect associated with the postulated rupture of piping based on General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50.
LBB evaluation procedure will be applied to RCS boundary piping, etc so that the dynamic effect of pipe rupture is eliminated.

Table 1.1.2.2-1 LBB Evaluation Procedure (1/2)

Type of Piping	Leak Before Break		Rupture Configuration
High Energy Piping	LBB is applicable (RCS boundary piping, etc)	LBB is successfully demonstrated	Crack
		LBB is not successfully demonstrated	Break
	LBB is not applicable	-	Break
Moderate Energy Piping	LBB is not applicable	-	Crack

The evaluation and design is determined depending on the rupture configuration.

Table 1.1.2.2-2 LBB Evaluation Procedure (2/2)

Rupture Configuration	Evaluation	Design
Break	i) Pipe whip and Pipe internal load ii) Jet impingement from rupture pipe iii) Compartment pressurization iv) Flooding	i) Protective Enclosures ii) Physical Separation iii) Pipe whip restraint
Crack	i) The most severe environmental consequences	i) Protective Enclosures ii) Physical Separation

1.1.2.3 Missile Protection

The structures , systems, and components important to safety will be designed to withstand the effects of missiles based on General Design Criterion (GDC) 2 and 4 of Appendix A to 10 CFR 50.

Table 1.1.2.3-1 Design Process of Protection against Missile

Type of Missiles	Considered Missiles	Evaluation Method
Internally Generated Missiles	Pressurized components, high-energy piping and rotating equipment	i) Locating the system or component in a missile-proof structure ii) Separating redundant systems or components for the missile path or range (SRP 3.5.1.1, 3.5.1.2)
Turbine Missiles	Turbine disk (or internal structure) fragment	The probability of unacceptable damage from turbine missiles should be less than equal to 1 in 10 million per year for an individual plant. (SRP 3.5.1.3, RG 1.115)
Externally Generated Missiles	Tornado missiles	Establish the ability of seismic Category I structures and/or missile barriers to withstand the effect of tornado missiles. (SRP 3.5.1.4)

1.1.2.4 Fire Protection

a. General Description

The fire protection of US-APWR satisfies General Design Criterion (GDC) 3 of Appendix A to 10 CFR 50 and complies with BTP SPLB 9.5.1 attached to Section 9.5.1 of NUREG-0800.

b. Design Bases

The fire protection of US-APWR has three objectives based on a defense-in-depth concept. The objectives of the fire protection are as follows.

1) Fire prevention
2) Detection and extinguishing of fires
3) Mitigation of adverse effects of fires

Fire protection systems are installed to minimize the adverse effects of fires on structures, systems, and components important to safety. The fire protection systems consist of fire detection systems and fire suppression systems. Design descriptions of the fire protection systems are shown in Section 8.5.1.
Four train safety systems necessary for post-fire safe shutdown are separated each other so as to maintain safe shutdown capability even in the event of a fire during on line maintenance.

1.2 Overview of the US-APWR Design

1.2.1 Main Specifications

The main specifications of the US-APWR are shown in Table 1.2-1. The electrical output of the US-APWR is increased to 1,700MWe by improving the plant thermal efficiency approximately 10% with the same thermal output of 4,466MWt compared to the first APWR in Japan. This is achieved by improving the efficiency of the steam generators (SGs) and the turbine system.

Table 1.2-1 Main Specifications of the US-APWR

Item			US-APWR
Electric power NSSS power		(MWe) (MWt)	1,700 class 4,466
Number of loops			4
Coolant pressure Coolant temperature (Hot leg)		(psia) (deg.F)	2,250 617
Reactor	Number of fuel assemblies Fuel rod lattice Active core length Vessel height x diameter	 (ft) (inches)	257 17 x 17 14 535 x 203
Steam generator	Heat transfer area	(ft^2/SG)	91,500
Reactor coolant pump	Thermal design flow	(gpm/loop)	112,000
Turbine	LP last-stage rotating blades	(inches)	70 class

1.2.2 Primary System

a. Reactor Design

The US-APWR loads 257 assemblies of 17x17 type fuel, the same as the first APWR plant. But the US-APWR extends the effective fuel height from 12 ft (3,660 mm) to 14 ft (4,200 mm) to reduce the specific power of the core while keeping the same reactor vessel height. The low power density design enables a 24 month operating cycle with more than 2 regions of refuelling design strategy. This results in an approximate 16% reduction of the required uranium, compared with conventional PWRs while including the effect of improved thermal efficiency described above.

The neutron reflector (NR) will be applied to the APWR/US-APWR allowing structural simplification and improving reliability and neutron economy. The NR consists of stacked ring blocks made of stainless steel that replace the baffle plates, former plates, and neutron pads of the current PWR. The NR employs only about 50 parts including bolts and nuts, while the baffle former structure of a current PWR employs more than 2,000 bolts. This improved design not only increases the reliability of the structure but also reduces the inspection requirements for bolts located in the high fluence region. The NR also contributes to the reduction of neutron fluence to the reactor vessel by 60% from the conventional 4-loop PWR.

Additionally to ensure the reliability and integrity of the reactor and fuel, the coolant temperature is limited to 617 deg. F (325 deg. C) in the hot legs and less than 554 deg. F (290 deg. C) in the top plenum during normal operation. Furthermore, the US-APWR employs the top mounted in-core nuclear instrumentation system in order to improve reliability of the reactor vessel, while the first APWR in Japan employs the conventional bottom mounted system.

b. Steam Generator Design

The design of the SGs for APWR has been improved to attain high efficiency and reliability. Major improvements from the conventional design are highly effective smaller sized separators which had acquired excellent moisture carry-over performance (<0.1%), 3/4 inch (19.05 mm) outer diameter tubes made of TT690, and 9 points support anti-vibration bars. Owing to these designs, the weight of a steam generator is reduced by more than 10% when compared to the SG design with 7/8 inch (22.23 mm) tubing.

Furthermore, the SGs for the US-APWR were designed to have 30% more heat transfer area to achieve higher efficiency by adopting the tight triangular lattice. As the result, the diameter of the SG body of US-APWR became slightly smaller than that of the first APWR.

c. Reactor Coolant Pump Design

The US-APWR Reactor Coolant Pump has achieved larger capacity and high efficiency by remarkable improvement of the impeller and diffuser configuration. The advanced seal design contributes to its longer life.

1.2.3 Safety System Configuration

Improvement of safety is one of the most important design targets for the US-APWR. To obtain an effective, reliable safety system, the US-APWR employs the following advanced technologies. PRA results demonstrate a reduction in the core damage frequency to 1/10 prior value.

a. Four-train, Direct Vessel Safety Injection System

The US-APWR employs the 4-train direct vessel injection (DVI) system. Such system configuration increases redundancy and independency, and enhances safety and reliability. The 4-train DVI system brings about a simple and compact safety system and enabling it to reduce the capacity of each train from 100% to 50%. Inter-connecting piping between each train is also eliminated.

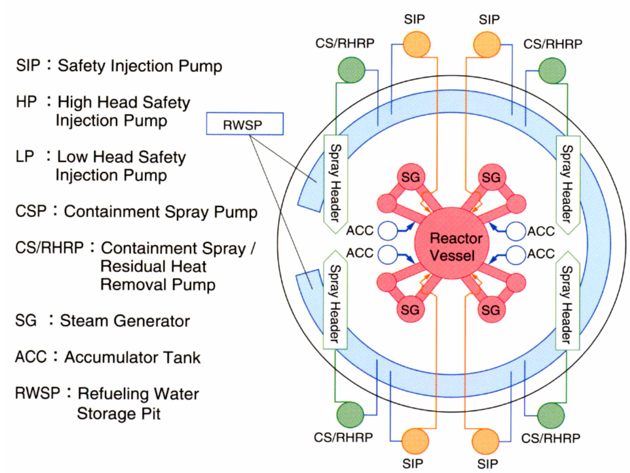


Figure 1.2-1 Configuration of Emergency Core Cooling System

The support system and the emergency AC power supply system also adopt a 4-train configuration to enhance the reliability of the system. The advanced accumulator described below allows for relaxing the start-up time requirement of the emergency generators, and thus gas turbine generators can be used for the emergency AC power supply system of the US-APWR.

b. Emergency Water Storage inside the Containment

The US-APWR eliminates the switchover operation of emergency water sources following a Loss of Coolant Accident (LOCA) by installing the Refuelling Water Storage Pit (RWSP) inside the containment as shown in Figure 1.2-1. The RWSP is formed with a lined concrete structure and works as the emergency water source. This design of significantly contributes to lowering core damage frequency.

c. Passive Low Head Injection Function

The safety system of the US-APWR consists of an optimized combination of active and passive components. The advanced accumulator is a passive component employed to enhance both safety and economics by its injection flow characteristics and displacing the low head injection system.

By adopting the vortex damper mechanism, the advanced accumulator supplies water with a large flow rate at the early stage of LOCA, and lower flow rate at the later stages.

1.2.4 I&C System

The advanced main control boards/consoles and the full digital I&C systems including the reactor protection system are applied to the US-APWR to improve man-machine interface and reliability.

An advanced alarm display system that dynamically prioritizes alarms was developed and applied to the US-APWR to avoid information overflow and to facilitate plant state identification. The prioritized alarms and their relevant process parameters are provided in the graphically presented plant systems on the large display panel with 3-level categorized colour coordination. The computerized operator support system for the US-APWR was developed and implemented also to support operators in abnormal situations that threaten plant safety.

Owing to the new systems, the physical and mental workload level was reduced by 25% and the estimated potential human error was also reduced by 25%.

1.2.5 Turbine System

MHI turbine achieves high reliability using integral low pressure rotor, reinforced rotating blade and blade grooves and material and structure highly resistant erosion. Also the MHI turbine achieves high efficiency and performance by adopting two stage reheat systems, three dimensional designed reaction blade and the integral shroud blade(ISB). The 54 inch low-pressure turbine last stage blade has been operating with high efficiency and performance. For the US-APWR, the length of the low-pressure turbine last stage blade will be extended to the 70 inches class.

1.3 Comparison with Current Operating PWRs

Comparison of principal parameters among the US-APWR, a current US four-loop plant, a Japanese four-loop plant and the Japanese APWR plant is shown in Table 1.3-1.

Table 1.3-1 Comparison of Principal Parameters

Parameter	US-APWR	Japanese APWR	US Current four-loop	Japanese four-loop
Gross electric output (MWe)	1,700 class	1,538	1,219	1,180
Core thermal output (MWt)	4,451	4,451	3,565	3,411
Operation pressure (psia)	2,250	2,250	2,250	2,250
Hot leg temperature (deg. F)	617	617	620	617
Thermal design flow (gpm/loop)	112,000	113,600	93,600	88,500
Number of fuel assembly	257	257	193	193
Fuel assembly lattice	17x17	17x17	17x17	17x17
Effective fuel length (ft)	14	12	12	12
Number of fuel rods per FA	264	264	264	264
Average linear heat rate (kW/ft)	4.6	5.3	5.7	5.5
Number of RCCA	69	69	53	53
Number of control rods per RCCA	24	24	24	24
SG heat transfer area (ft^2)	91,500	70,000	55,000	52,400
PZR volume (ft^3)	2,900	2,300	1,800	1,800
Design life (years)	60	60	40	40

2.0 BUILDING STRUCTURE AND LAYOUT

2.1 Outline

The main power block is comprised of the following buildings;

- Reactor Building (R/B)
- Gas Turbine Building (GT/B)
- Auxiliary Building (A/B)
- Access Control Building (AC/B)
- Turbine Building (T/B)

The outline and the arrangement of those buildings are shown in Figure 2.1-1.

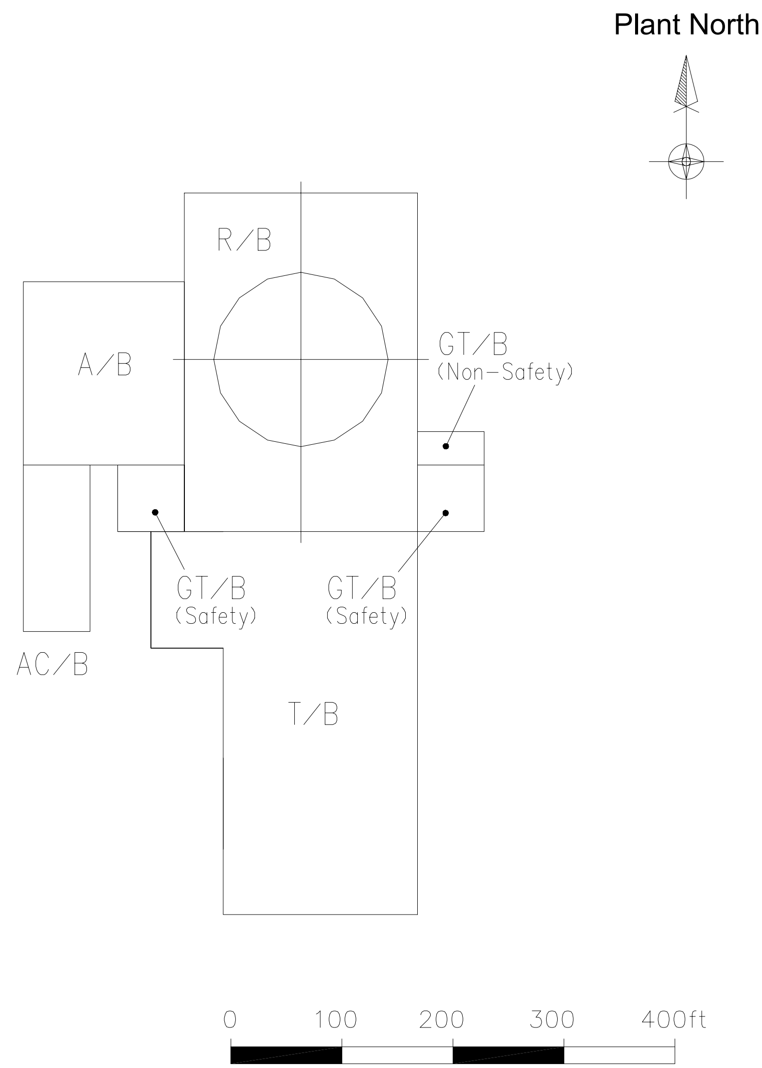
The layout of equipment within the buildings is aimed at providing ease of plant operation and maintenance and to minimize personnel radiation exposure. Provisions, including redundant train separation and segregation barriers, have been made to ensure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high energy pipe break events. Within the buildings, access control zonings are established to regulate access to radiation area and to define the required radiation shielding and monitoring during operation, shutdown, and accident conditions.

R/B and GT/B for safety-related gas turbine generator are designed and constructed as safety-related structures and seismic category is I (one). All safety-related systems, structures, and components are housed in these two buildings. These safety-related structures are designed for the effects of all applicable loads and their combinations, including the postulated seismic response loads, and is founded on the foundation bedrock.

Other buildings (A/B, AC/B, GT/B for non-safety-related gas turbine generator for Station Black Out and T/B) are designed as non-safety-related structures and seismic categories are non.

Each building is free standing on the separated concrete base mat and adopts the concrete structure for the primary structure type.

Figure2.1-2, 2.1-3 and 2.1-4 shows the general arrangement of primary system related buildings.



R/B : Reactor Building

A/B : Auxiliary Building

GT/B (Safety) : Gas Turbine Building for safety-related gas turbine generator

GT/B (Non-Safety) : Gas Turbine Building for non-safety-related gas turbine generator for Station Black Out

AC/B : Access Control Building

T/B : Turbine Building

Figure 2.1-1 The Outline of Power Block Buildings

- **Figure removed per 10 CFR 2.390**

Figure 2.1-2 Primary System Related Buildings Arrangement (Plan View)

Figure removed per 10 CFR 2.390

Figure 2.1-3 Primary System Related Buildings Arrangement (Section View)

Figure
removed
per
10 CFR 2.390

Figure 2.1-4 Primary System Related Buildings Arrangement (Section View)

2.2 Reactor Building

The reactor building has five main floors. The building contains the containment vessel at its center and is founded on a common mat.

The reactor building consists of the following five areas from their functions:

- Containment facility and inner structure
- Safety system pumps and heat exchangers area
- Fuel handling area
- Main steam and feed water area
- Safety-related electrical area

2.2.1 Containment Facility and Inner Structures

The containment facility is comprised of the containment vessel and the annulus enclosing the containment penetration area, and provides an efficient leak-tight barrier and environmental radiation protection under all postulated conditions, including a loss-of-coolant accident. The containment vessel is a prestressed concrete structure to endure the peak pressure in the LOCA conditions (Figure2.2-1 through 2.2-6 shows the general arrangement in the containment vessel, and Section 5.2 for specification of the containment systems). Accessible galleries are provided for periodic inspection and testing of circumferential and axial prestressing tendons.

To ease access for operation, maintenance, repair and refueling, the following accesses to the containment vessel are also provided:

- A normal personnel airlock, located at floor level below the operating floor
- An equipment hatch and emergency airlock, located at operating floor level

The annulus is located adjacent to the containment vessel and includes all penetration area to prevent from the direct release of containment atmosphere to the environment through the containment penetrations. The pressure in the annulus is kept at a slightly negative level following accident conditions to control the release of the radioactive materials to the environment.

The Refueling Water Storage Pit (RWSP) is located in the lowest part of a containment vessel. The RWSP provides a continuous suction source for both the safety injection pumps and the containment spray/residual heat removal pumps, thereby eliminating the switchover of suction source from the injection to the recirculation phase of accident recovery.
The RWSP has four recirculation sumps on the floor, these sumps are provided with the recirculation screen, and the wall and the floor of RWSP are lined with stainless steel liner plates.

The reactor vessel is located at the center of the containment vessel and is provided with a cylindrical concrete wall as a primary biological shield. There are four reactor coolant loops, each loop comprised of a steam generator, a reactor coolant pump, and loop piping. Concrete walls surrounding the loops are provided as supporting media and as secondary biological shields.

The pressurizer is located in its own compartment and is adjacent to the steam generators to minimize the length of the surge piping to the reactor coolant loop.

A refueling cavity with stainless steel liner is provided above the reactor vessel for refueling operations. The fuel transfer tube connects this cavity to the fuel handling area located outside the containment.

The steel concrete (SC) structures are adopted to the walls of inner structures.

The steam and feedwater pipe work which connect to the steam generators are routed within the containment with consideration of minimizing pipe run lengths, while providing sufficient flexibility to accommodate thermal expansion.

2.2.2 Safety System Pumps and Heat Exchangers Area

The safety system pumps (CS/RHR pumps and safety injection pumps), which require sufficient net positive suction head (NPSH) to draw water from the recirculation sumps inside the containment, are located at the lowest level of the reactor building to secure the required NPSH. Also, they are located adjacent to the containment to minimize pipe lengths. The safety system heat exchangers (CS/RHR heat exchangers) are located at the upper floor.

2.2.3 Fuel Handling Area

The fuel handling area is located on the northern side of the reactor building, and houses the following facilities:

- Spent fuel pit with the bridge crane (Spent Fuel Pit Crane)
- Fuel transfer system (Fuel transfer canal with the fuel transfer tube)
- Cask loading pit with the fuel handling area crane
- New fuel pit
- Decontamination pit

Fuel handling operations are performed on the top floor of the area at the same level as the containment vessel operating floor. The containment emergency airlock is located adjacent to the fuel handling area to facilitate easy access between the containment and fuel handling area when refueling procedures are in progress.

The bridge crane is located to span the spent fuel pit, the transfer canal, and the cask loading pit. The fuel handling area crane is capable of lifting the shipping cask from ground level to the operating floor. This crane is designed not to move over the spent fuel pit in order to avoid any possibility of the spent fuel in the pit being damaged by a potential load drop.

Design description for the refueling handling system is shown in Section 8.1.2

2.2.4 Main Steam and Feed Water Area

The main steam and feed water area is located on the southern side of the reactor building, between the containment and the turbine building. The main steam and feedwater piping room is located on the top floor of this area and contains the main steam and feedwater pipes where they pass between the turbine building and the containment.

2.2.5 Safety-related Electrical Area

The safety-related electrical area has two floors and is located on the southern side of the reactor building and under the main steam and feed water area.
It is normally a non-radioactive zone and completely separated from the radioactive zones of the reactor building.
This area houses the following safety-related facilities:

- Main control room
- Switchgear and batteries
- I &C cabinet room

2.2.6 Separation of Redundant System

Four redundant safety systems containing radioactive material are located in each zone of the four quadrants surrounding the containment structure. Each of the quadrant areas are separated by a physical barrier to ensure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high energy pipe break events.
Non-radioactive safety systems such as EFWS, CCWS and electrical system etc. are located in the southern area of containment structure. This area is also separated into four divisions by a physical barrier to ensure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high energy pipe break events.

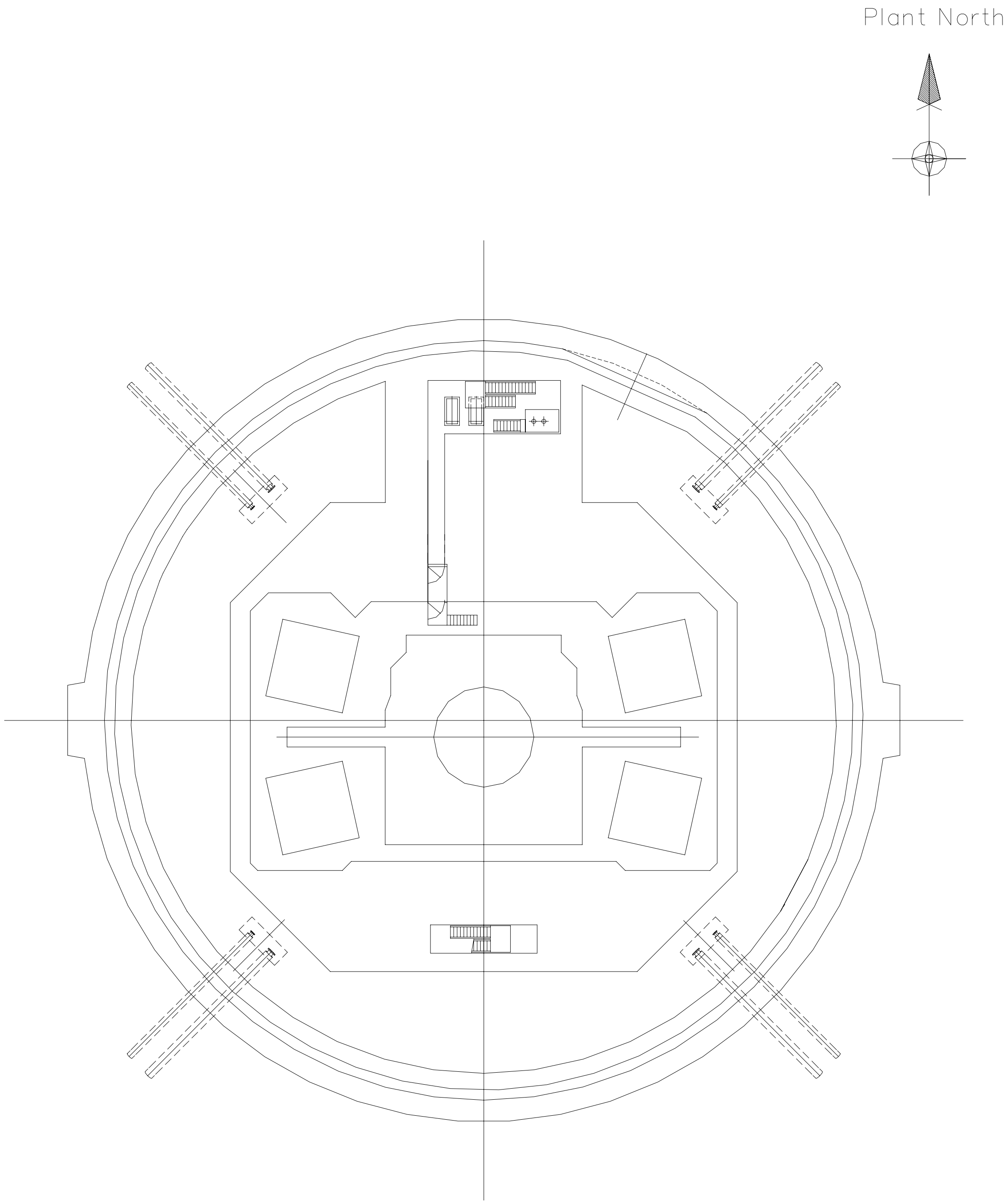


Figure 2.2-1 Plane View inside the Containment Vessel (1FL)

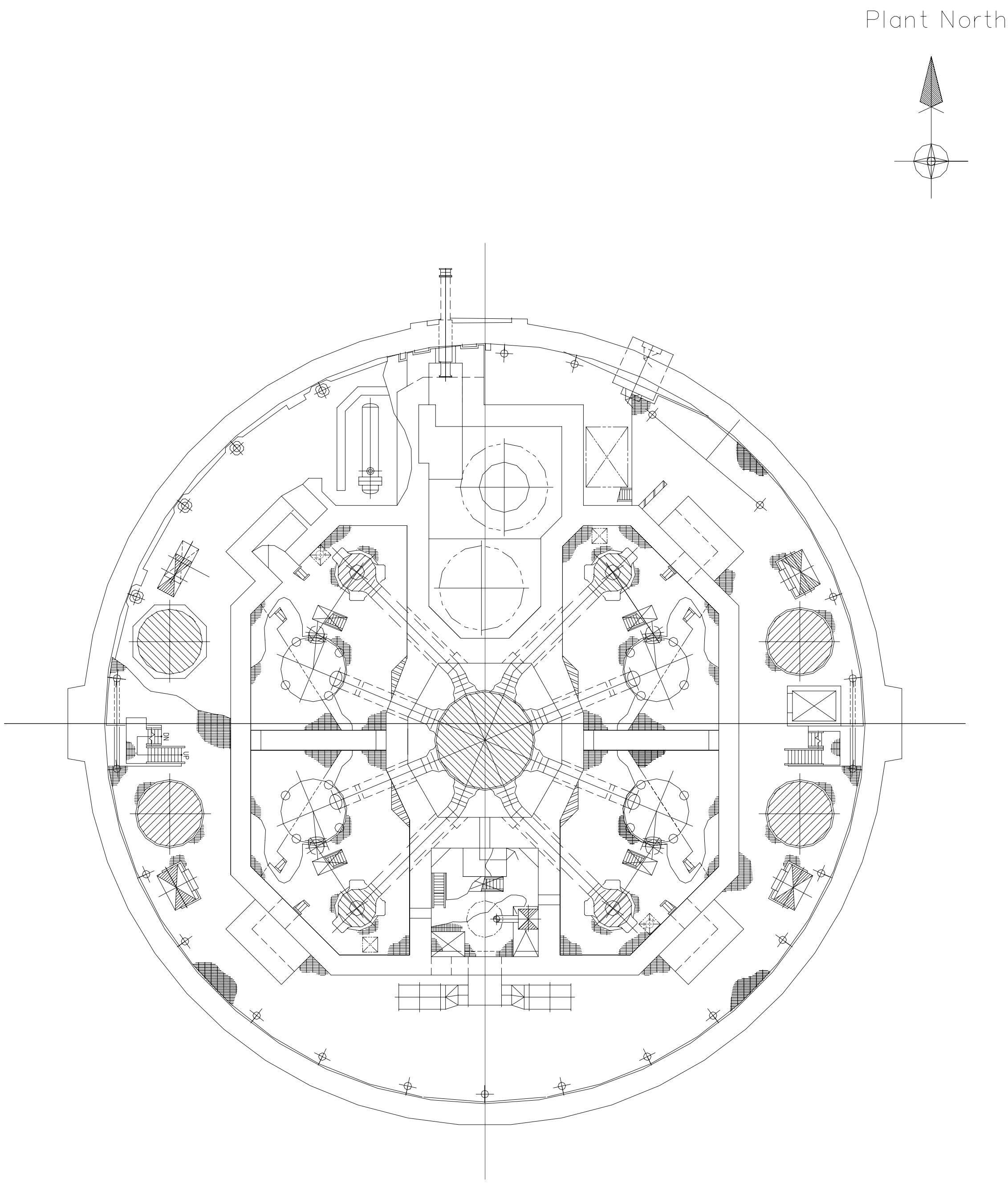


Figure 2.2-2 Plane View inside the Containment Vessel (2FL)

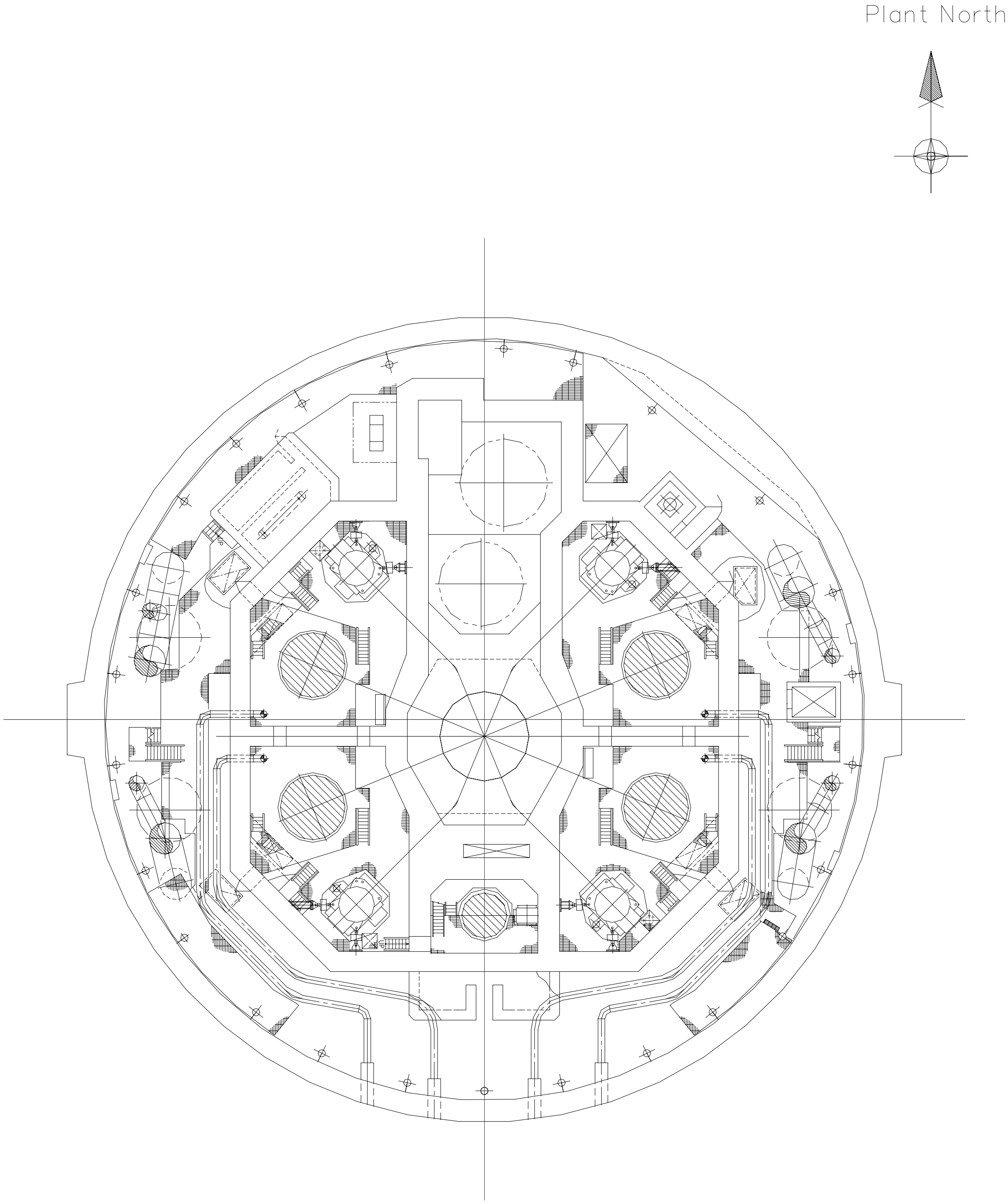


Figure 2.2-3 Plane View inside the Containment Vessel (3FL)

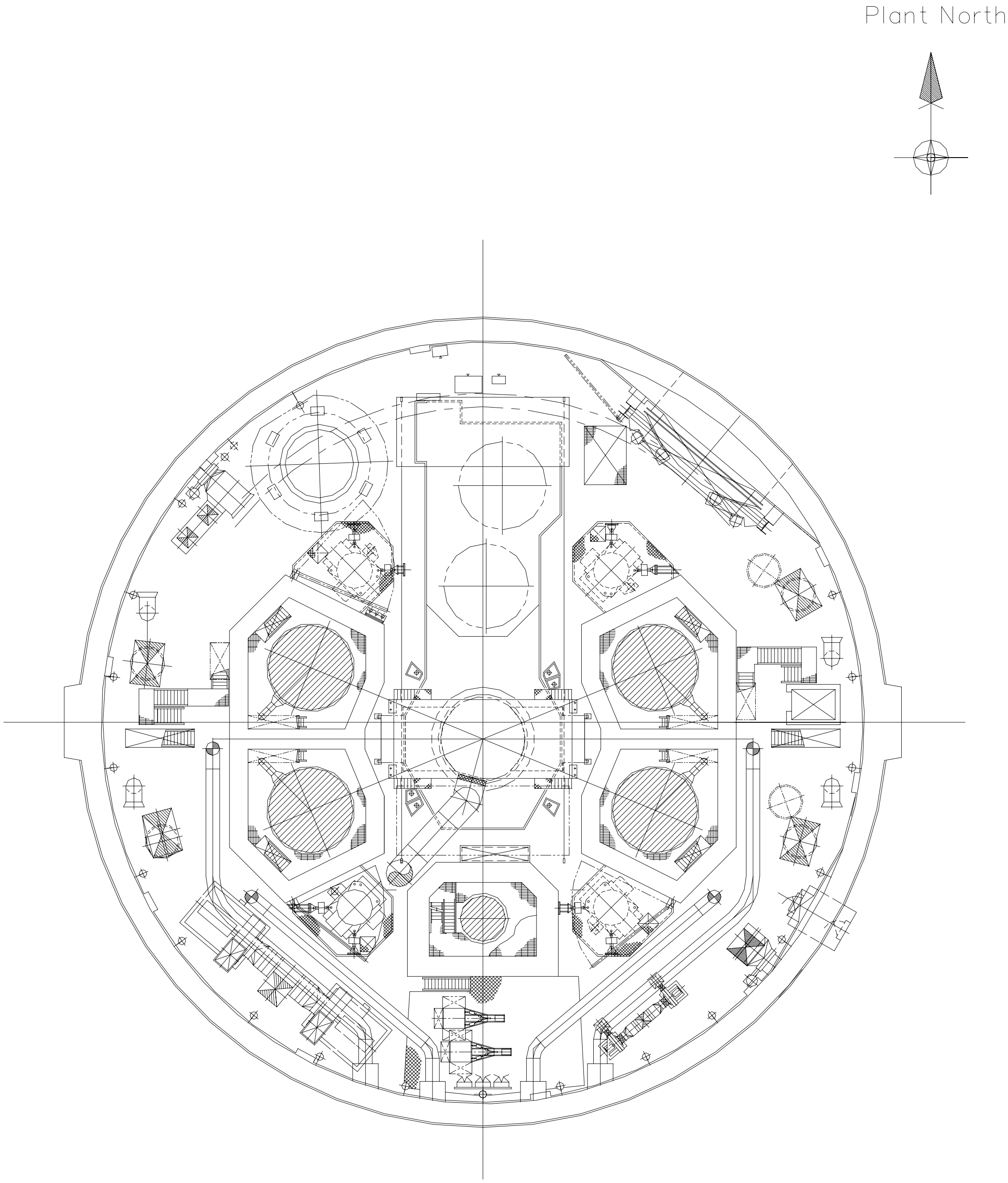


Figure 2.2-4 Plane View inside the Containment Vessel (4FL)

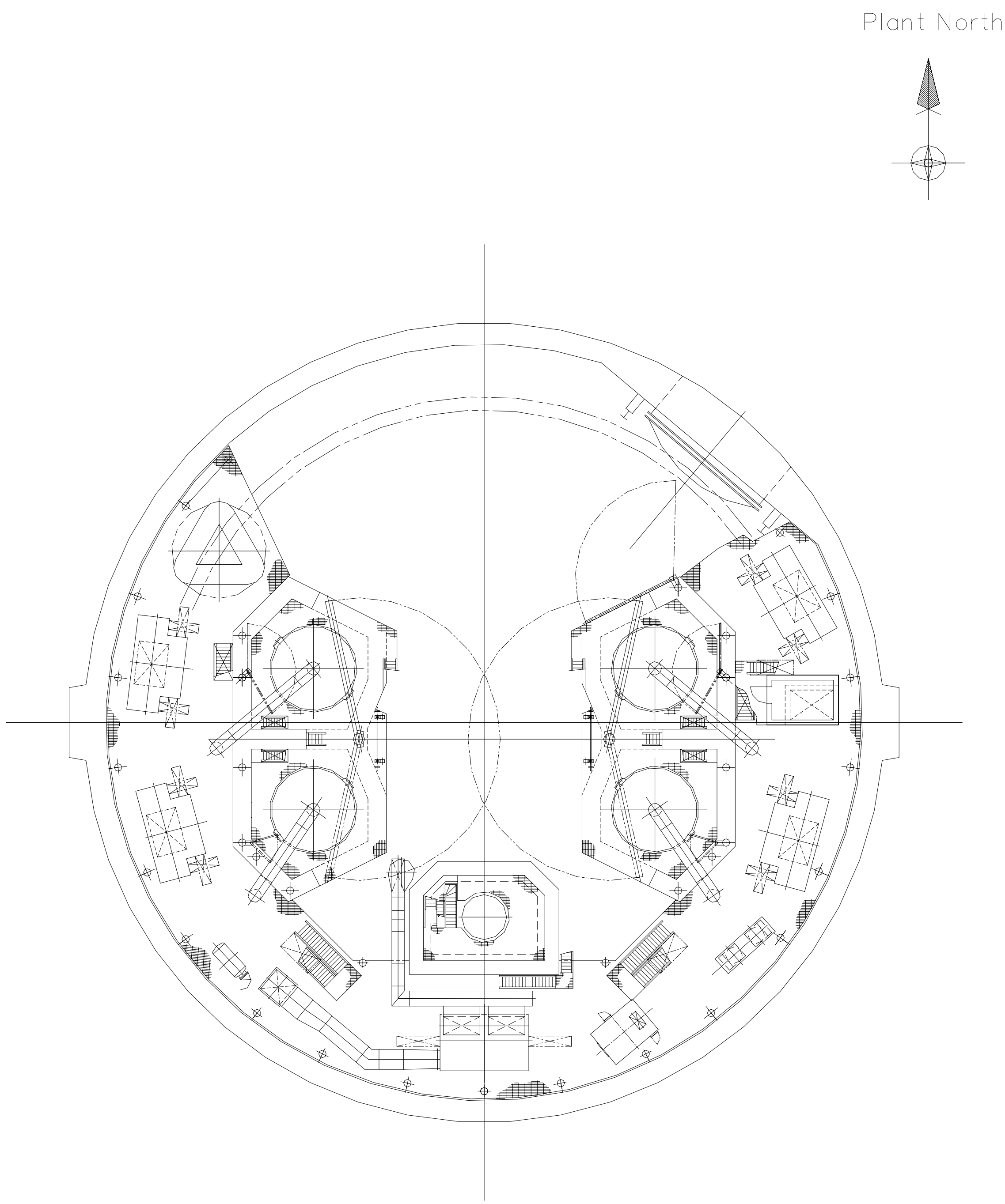


Figure 2.2-5 Plane View inside the Containment Vessel (5FL)

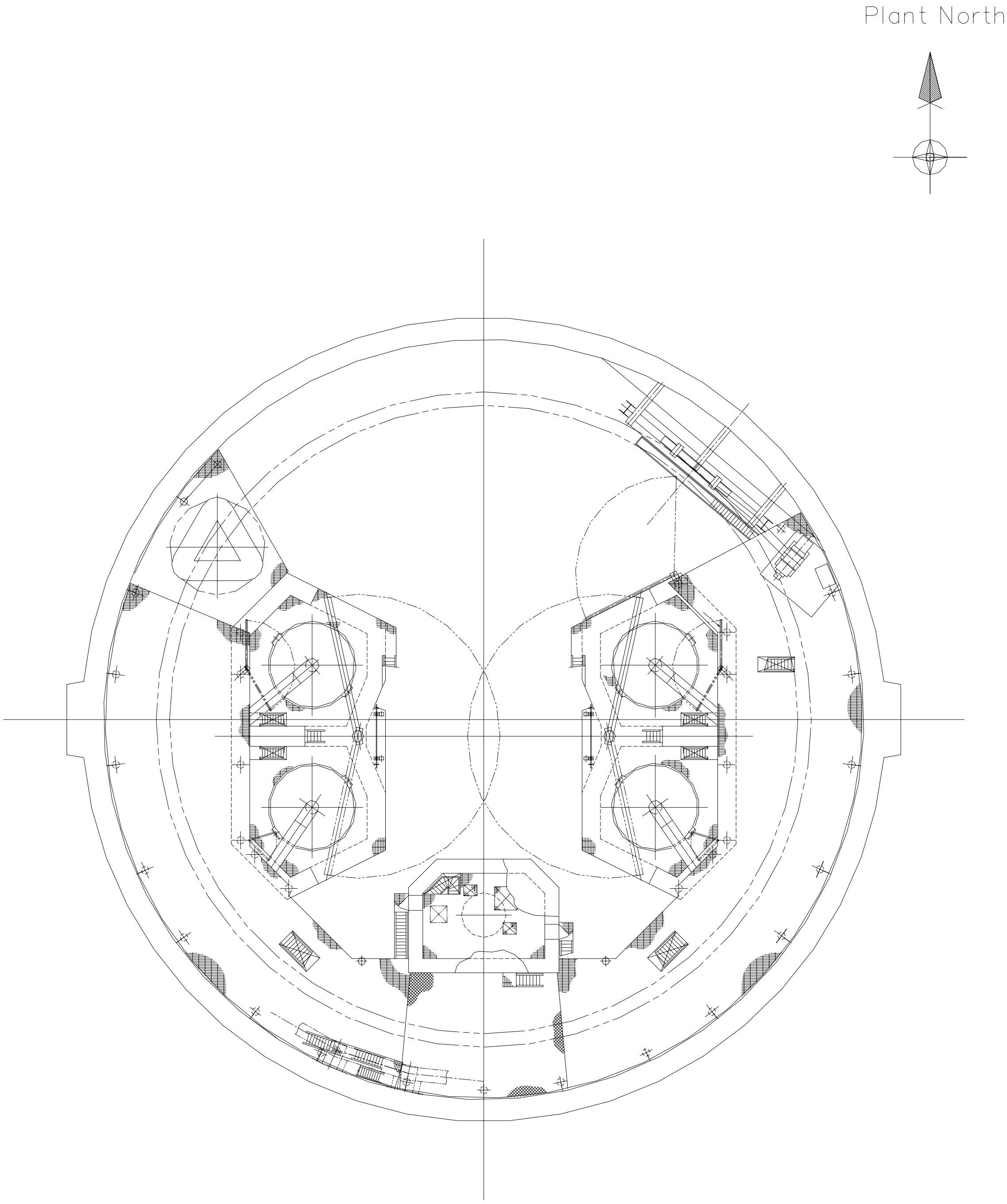


Figure 2.2-6 Plane View inside the Containment Vessel (Upper Platform)

2.3 Gas Turbine Generator Building

Two safety gas turbine buildings are arranged adjacent to reactor building.
These buildings are free-standing on a reinforced concrete mat and each building contains two identical gas turbine generators which are separated each other by physical barrier. In addition, the safety related chillers are also located in these building.

The gas turbine generator for Station Black Out is contained in the non –safety gas turbine building which is independent on safety-related gas turbine buildings.

The electrical, I&C and HVAC equipment related to the gas turbine generators are also contained in these buildings, and fuel storage tanks are arranged outside of buildings.

2.4 Auxiliary Building

The auxiliary building is located on the west side of the reactor building. This building contains main components of the waste disposal systems, the chemical and volume control system, and non safety-related electrical area. The non safety-related electrical area is normally a non-radioactive zone and is completely separated from the radioactive zones of the auxiliary building.

2.5 Access Control Building

The access control building is located adjacent to the south side of the auxiliary building.

This area houses the following facilities;

- Access control area and way
- Chemical sampling and laboratory area
- Non-safety chillers

2.6 Turbine Building

The turbine building contains the non safety-related equipment of the steam and power conversion system including the turbine-generator and its auxiliary system.

The building consists of a main building and a heater bay. The turbine-generator, main condenser with feedwater heaters, condensate pumps, turbine bypass valves and other equipment are installed in the main building. The deaerator with the feedwater storage tank, feedwater heaters, feedwater pumps and equipment are installed in the heater bay.

The turbine building is steel structure which is designed to withstand all loads including the load of the overhead traveling crane. The foundation of the building is made of concrete to reduce the thickness of the mat.

The building is designed based on the following:

- The turbine building is classified as Non Seismic Category I (consistent with the definition of Reg. Guide 1.29). The turbine generator building structure is designed to the provisions of the Uniform Building Code.

- The turbine building is oriented in such a way that any plane perpendicular to the turbine generator axis shall not intersect with the reactor building. This arrangement minimizes the probability of a turbine missile striking the reactor building (consistent with the guidance of Reg. Guide 1.115).

- The turbine building is independent of the reactor building to prevent internal hazards in the turbine building form spreading.

3.0 REACTOR AND CORE

3.1 Outline

The reactor of the US-APWR is designed to produce a thermal output of 4,451MW, which is inherited from the first Japanese APWRs (Tsuruga unit 3and 4). The extension of active fuel length to approximately 14feet, however, will provide a larger thermal margin to the US-APWR than that of the preceding units. It enables extended cycle operation up to 24 months, free from reduced fuel economy associated with a large reload batch now seen in existing PWRs. The utilization of the steel neutron reflector improves the fuel cycle cost.

The fuel assembly of the US-APWR has a 14 ft heated length with 11grid spacers. This design enables to keep the same grid spacing of approximately 18 inches as the 17x17 fuel assembly design in Japanese plant. This relatively shorter grid spacing provides greater margin to grid fretting and improves DNB performance in comparison with the widely used standard design of 14 ft heated length10 grid design.

DNB is the most important parameter of thermal & hydraulic design. DNB performance of the core is evaluated with VIPRE-01 sub-channel analysis code using WRB-2 DNB correlation. The DNB correlation was developed with a large amount of DNB data taken in Heat Transfer Research Facility of Columbia University. Statistical methodology is used to evaluate the DNBR for various operating conditions.

Despite the extended fuel length, the reactor vessel height remains the same as the first APWRs with the integration of the lower core plate and lower core support. This design enables use of the same reactor vessel and core barrel as the first APWRs. The metal neutron reflector also contributes to the enhancement of structural reliability.

3.2 Fuel System Design

3.2.1 Overall Features

The overview of the fuel assembly design for the US-APWR is shown in Figure 3.2.1-1 and in Table 3.2.1-1. The major design modifications of fuel assembly from the APWR to the US-APWR are fuel active length from 12 ft to14 ft and number of grids from 9 to 11. All the advanced technologies incorporated into the current 17x17 fuel assembly for higher burnup are applied for the US-APWR fuel assembly

The fuel assembly design for the US-APWR is based on the Mitsubishi 17x17 fuel assembly that has demonstrated high reliability through significant irradiation experience in Japan. This fuel design improves reliability, enhances fuel economy and enables flexible core operation.

a. Improving reliability

- Grid spring design and adoption of an 11 grid design result in high grid fretting resistance. This grid design concept has been proven with long term no-leakage records in Japanese PWRs for approximately13 years.
- ZIRLO™* cladding tube demonstrates high corrosion resistance under demanding conditions.
- Anti-debris bottom nozzle with a built-in filter enhances debris trapping capability.

The fuel assembly design for the US-APWR adopts the above features to ensure high reliability.

b. Enhance fuel economy

Pellet density of 97% TD (Theoretical Density) greater than the conventional 95% TD, improves fuel cycle cost by increasing the amount of uranium in the core.

c. Enable flexible core operation

Higher gadolinia content, maximum 10wt%, enables flexible core operation to higher burnup. Futhermore, larger rod plenum volume can produce potential margin for rod internal pressure increase caused by FP gas released, especially under high power operation at high burnup.

*ZIRLO™ : ZIRLO™ is trademark of Westinghouse Electric Corporation.

3.2.2 Fuel Assemblies

The fuel assembly consists of the 264 fuel rods arranged in a square array of 17x17, together with 24 control rod guide thimbles, an in-core instrumentation guide tube, 11 grid spacers, and top and bottom nozzles.

The fuel rod has a sealed structure, in which sintered uranium di-oxide pellets slightly enriched up to 5 wt % are inserted into a ZIRLO™ cladding tube. Some of the fuel rods contain uranium di-oxide pellets blended with maximum10 wt % content of gadolinia. The coil spring at the upper plenum is put in to prevent the pellets from moving during shipping and handling and Zircaloy-4 end plugs are welded onto both ends. For the US-APWR, lower plenum including stainless spacer is also provided to reduce internal pressure. The rod is pressurized with Helium gas.

The skeleton structure of the assembly consists of top/bottom nozzles, grid spacers, control rod guide thimble tubes and an in-core instrumentation tube, which is designed to withstand its own weight and external loads.
The fuel rods are restrained by 11 grid spacers located at almost equal intervals. The grid spacers are mechanically fixed to the 24 control rod guide thimbles. The control rod guide thimbles are symmetrically arrayed according to arrangement of the control rods in a rod cluster control assembly. The in-core instrumentation guide tube is located at the center of the square array of the fuel rods.

The grid spacers have lattice structure which is interlocked by thin straps made of nickel-chromium-iron Alloy 718 (Inconel 718) or Zircaloy-4. For the US-APWR fuel assembly, the upper and lower end grid spacers are made of Inconel 718, and the intermediate grid spacers are made of Zircaloy-4, improving neutron economy. The straps are brazed and welded using of Inconel 718 and Zircaloy-4, respectively.

The top nozzle assembly has hold-down springs made of Inconel 718 in order to prevent the fuel assembly from lifting due to the hydraulic force during normal reactor operation and abnormal operational transients. The top nozzle can be easily removed and reconstructed for repair, or maintenance.
The bottom nozzle, as shown in Figure 3.2.2-1, has a plate on which thin plates are placed and welded in grooved slits, providing a filter for debris passing through flow holes.

The guide thimbles are made of Zircaloy-4 and fixed by screws to the bottom nozzle. They guide in-core control components such as control rods, burnable absorber rods and neutron source rods into the fuel assembly. The bottom of the control rod guide thimble is small in diameter and is provided with several small holes, to provide a buffer to the dashpot effect at the end of a reactor trip motion.

The instrumentation tube is also made of Zircaloy-4, and both ends are inserted into the top and the bottom nozzles. The tube leads an In-core neutron detector into the fuel assembly from a center hole of the top nozzle plate.

3.2.3 Principal In-core Control Components

a. Rod Cluster Control Assemblies

The rod cluster control assembly has a structure consisting of 24 control rods arranged in symmetrical positions with a spider and coupling. Each control rod driven by a control rod driving mechanism inserts into a guide thimble. The control rod has a structure of silver-indium-cadmium alloy, as a neutron absorber, encased in a stainless steel tube. The both ends of the control rod are welded with end plugs, the top end being fixed to the spider.

The spider is coupled to the drive rod.

b. Burnable Absorber Assemblies

The burnable absorber assembly consists of stainless steel tubes filled with borosilicate glass. Arrangement of the tubes is the same as rod cluster control assembly. The tubes are inserted into the control rod guide thimbles of a fuel assembly.

Table 3.2.1-1 Fuel Assembly Specifications

Fuel Assemblies	
Fuel Rods Array	17 x 17
Number of Fuel Rods	264
Number of Control Rod Guide Thimbles	24
Number of In-core Instrumentation Guide Tube	1
Number of Spacer Grids	11
Fuel Rods	
Outside Diameter	0.374 in. (9.50mm)
Cladding Thickness	0.022 in. (0.57mm)
Active Fuel Length	165.4" (4,200mm)
Reload Fuel Enrichment	Max. 5 wt%
Gadolinia Content	Max. 10 wt%
Pellet Density	97 %TD
Materials	
Cladding	ZIRLO™
Top & Bottom Grids	Inconel 718
Middle Grids	Zircaloy-4
Guide Thimbles	Zircaloy-4
Nozzles	Stainless Steel
Holddown Springs	Inconel 718

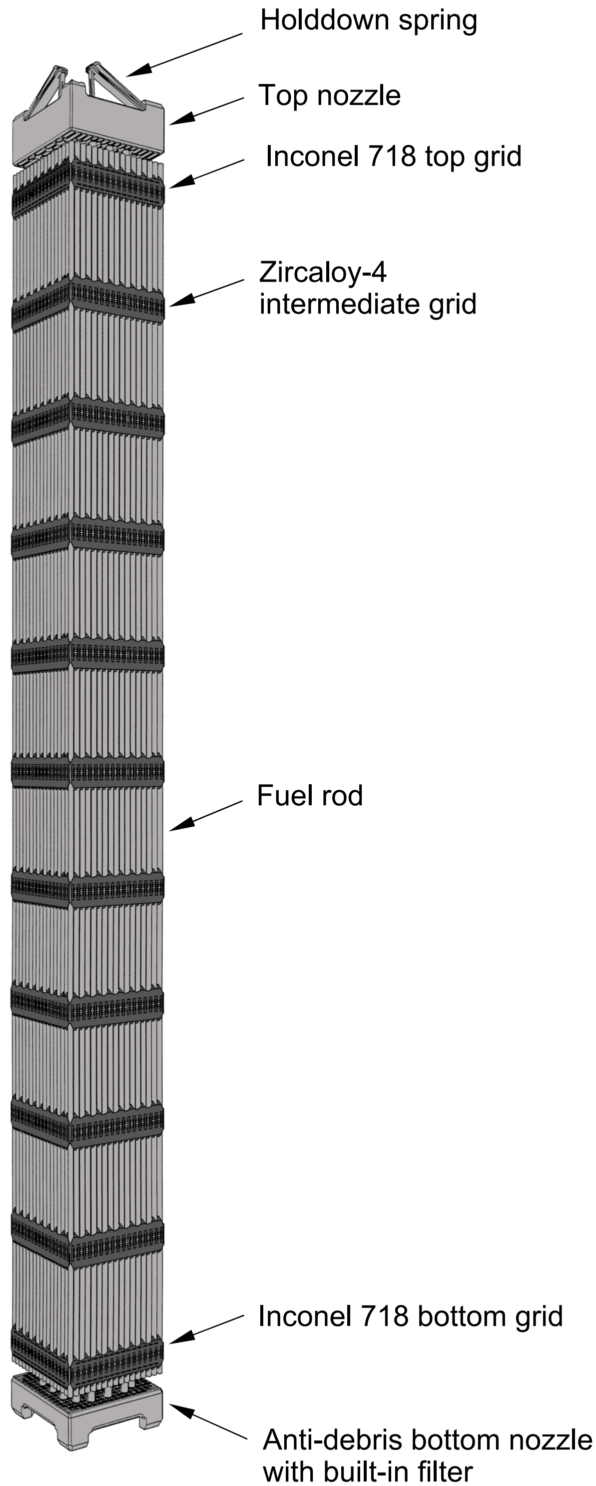


Figure 3.2.1-1 Mitsubishi US-APWR Fuel Assembly

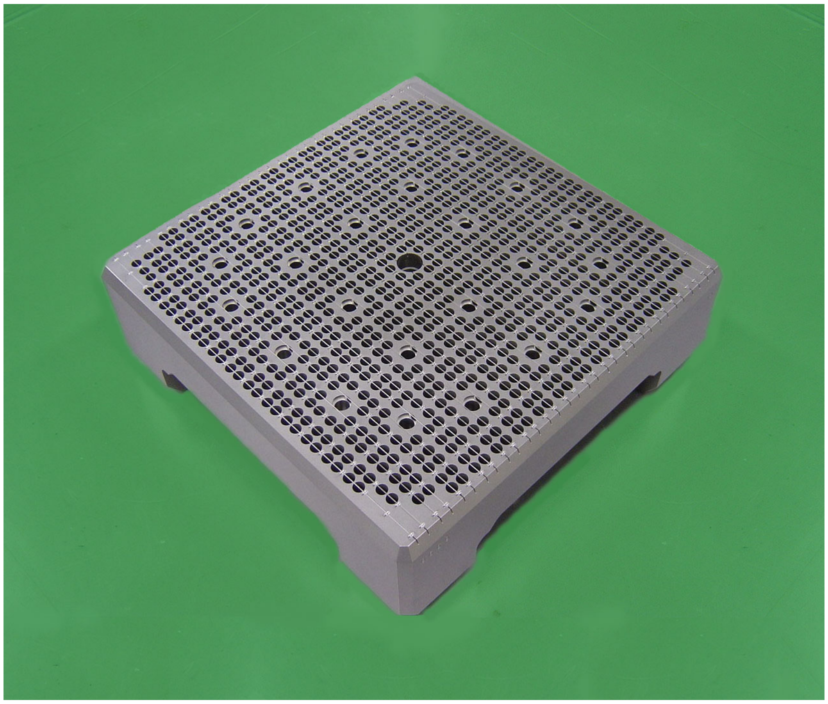


Figure 3.2.2-1 Anti-Debris Bottom Nozzle with Built-in Filter

3.3 Nuclear Design

3.3.1 Design Concept and Features

The US-APWR core design is based on the APWR core design which has been developed and planned to be built in Japan. To show the evolution of the core concept, the main nuclear design parameters of Japanese 4-loop PWR, APWR and US-APWR are shown in Table 3.3.1-1.

The concept of the APWR core developed from Japanese 4-loop PWR, which consists of 193 fuel assemblies with a 17x17 rod array. The number of fuel assemblies was increased to 257 in order to allow a larger thermal power of 4,451 MWt. The stainless steel radial neutron reflector design was employed to improve neutron utilization therefore reduced the fuel cycle cost and to obtain the additional benefit of reducing the reactor vessel irradiation. The US-APWR active fuel length is increased to approximately 14 feet. Since the US-APWR has the same thermal power as its predecessor, it has lower linear power density, approximately 4.6kW/ft, allowing flexible core and fuel management with improved thermal margins. Even under the constraints of fuel enrichment less than 5wt% and maximum fuel rod burnup of 62GWd/t, 24-month cycles with approximately 2-batch reload are feasible in the US-APWR.

The US-APWR fuel assembly has a 17x17 rod array, with 264 fuel rods, 24 control rod guide thimbles and an in-core instrumentation thimble. A cross sectional views of the 17x17 fuel assemblies are shown in Figure 3.3.1-1. In accordance with the fuel loading strategy, some fuel rods contain partial and/or full length gadolinia (Gd_2O_3) integral burnable poison and some of the control rod guide thimbles are filled by a borosilicate glass burnable poison. The control rod cluster assembly is composed of 24 control rods of Ag-In-Cd alloy, and clad in stainless steel. Figure 3.3.1-2 shows the fuel assembly and control rod assembly configuration in the US-APWR core.

3.3.2 Power Distribution Control and Monitoring

The large thermal margin provided by the large core volume, appropriate power distribution control and protection functions ensure fuel integrity during normal operation, including the effect of anticipated operational occurrences.

Therefore, in order to guarantee adequate DNB margin, the radial power distribution and thus $F_{\Delta H}^{N}$ are limited by appropriate selection of the fuel assemblies loading pattern in the core, the fuel enrichment and the use of gadolinia integral fuel rods as well as discrete burnable poisons. Even in the case of low leakage loading pattern, the $F_{\Delta H}^{N}$ can be limited below approximately 1.7.

The maximum linear power density can be limited by the appropriate fuel loading pattern and the power distribution control in conjunction with its monitoring thereby preventing the occurrence of unfavorable power shapes. Preliminary analysis shows that F_Q x P can be limited below approximately2.6 even with daily load follow operation, where “P” represents the fractional core thermal output to the rated thermal power (RTP).

The power distribution changes due to control rod maneuvering, xenon transient, fuel burnup and the anticipated operational occurrences. Those changes can be monitored by ex-core nuclear instrumentation system (NIS) and an optional on-line power distribution monitoring system (OPDMS). The ex-core detector output provides the axial offset data for the protection functions. Moreover, the measurements of power distribution with the in-core instrumentation system (ICIS) allow more precise knowledge of the 3-D power distribution.

3.3.3 Negative Reactivity Feedback and Stability

The core is designed to have negative reactivity feedback characteristics associated with fuel temperature and moderator temperature or density.

With these characteristics, power oscillations can be easily brought under control. Even a fast reactivity rise in the accident condition is immediately controlled by negative Doppler effect.

On the other hand, in order to have a negative moderator temperature coefficient at hot zero power and all control rods out condition, the critical boron concentration at the beginning of the cycle is limited using gadolinia integral fuel rods and/or burnable poisons.

3.3.4 Shutdown Margin and Reactivity Control

Control rods and soluble boron in the coolant are provided as two independent shutdown mechanisms and are also designed to control the reactivity during reactor operation.

The control rod system has enough reactivity to compensate fast reactivity fluctuation during operation and also the transition from full power to the hot zero power condition. In addition, the hot shutdown margin with the most reactive control rod stuck gives adequate subcriticality to minimize any consequences of over cooling events. In order to guarantee the shutdown margin, the control rod banks use insertion limits during operation.

Slow reactivity changes, such as fuel burnup and the transition from hot shutdown to cold shutdown, are compensated with soluble boron in the reactor coolant system. The negative reactivity insertion by soluble boron is rapid enough to overcome the reactivity rise due to the decay of built-up xenon. In addition, the boron concentration is controlled to maintain the subcriticality during refueling.

Table 3.3.1-1 Evolution of Nuclear Design Parameters

Core Parameters*	Stage of Evolution		
	US-APWR	APWR	Japanese 4-loop
Core thermal output (MWt)	4,451	4,451	3,411
Active core height (cold, ft)	14	12	12
Average linear power density (kW/ft)	4.6	5.3	5.5
Equivalent core diameter (cold, ft)	12.8	12.8	11.1
Fuel assembly pitch (cold, in)	8.46	8.46	8.46
Number of fuel assemblies / core	257	257	193
Number of control rod assemblies / core	69	69	53

* given as approximate figures except the integers for assemblies counting

Fuel Parameters	Parameters for US-APWR
Fuel rod array within assembly	17 X 17
Number of fuel rods / assembly	264
^{235}U enrichment (wt%)	< 5
Gadolinia content in Gd fuel rods (wt% Gd_2O_3)	up to 10
Cycle length (months)	up to 24
Maximum fuel rod burnup (GWd/t)	62

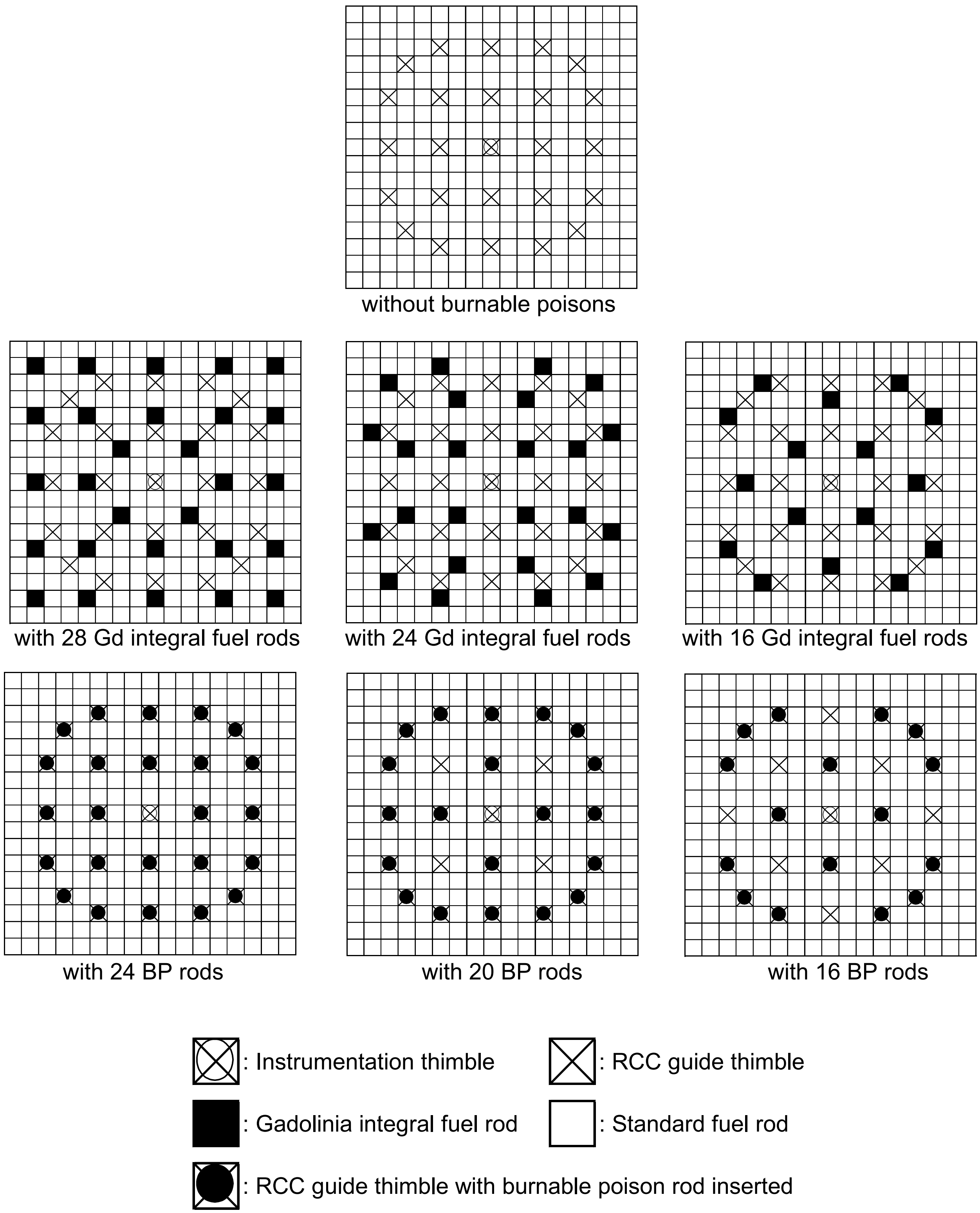


Figure 3.3.1-1 Arrangement of Fuel Assemblies and Burnable Poison Rods

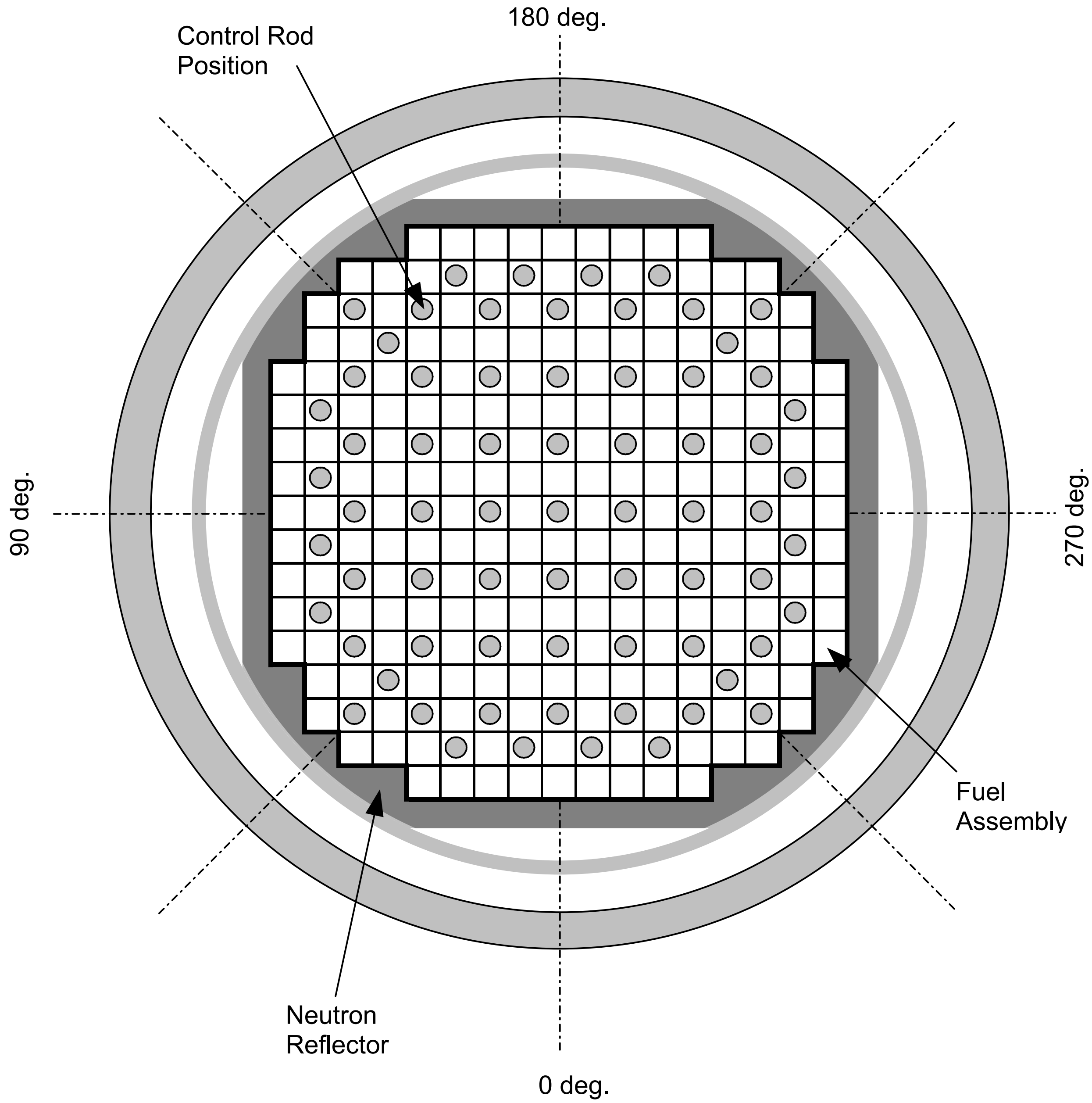


Figure 3.3.1-2 Arrangement of Fuel and Rod Cluster Control Assemblies

3.4 Thermal and Hydraulic Design

The objective of the thermal and hydraulic design is removing the heat generated in the core adequately to prevent fuel damage during normal operation and operational transients (Condition I events), and during transient conditions arising from abnormal occurrences of moderate frequency (Condition II events).
For the purpose, design bases for the US-APWR are determined as follows;

- To prevent fuel from Departure from Nucleate Boiling (DNB) during Condition I and Condition II events
- To prevent fuel from centerline melting during Condition I and Condition II events

Table 3.4-1 shows major thermal and hydraulic design parameters of US-APWR.

3.4.1 DNB

DNB design criterion for the US-APWR is that the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent at a 95 percent confidence level during Condition I and Condition II events.

DNB heat flux is predicted using empirical correlation and subchannel analysis code. WRB-2 DNB correlation and VIPRE-01 code can be suitably applied for Mitsubishi's fuel assembly design for the US-APWR. The margin for DNB occurrence is quantified by the DNB Ratio (DNBR), which is a ratio of DNB heat flux to local heat flux. A design limit value for the minimum DNBR in the core corresponding to the above criterion will be determined by using Revised Thermal Design Procedure (RTDP), which was developed by Westinghouse and approved by NRC. The uncertainties of the DNB correlation and input parameters are statistically combined in the method.

During the plant operation, the DNB design criterion is ensured by the safety protection systems of the US-APWR.
The Over Temperature ΔT protection system maintains the combination of core power, pressure and coolant temperature inside the DNBR limit conditions which is determined by thermal and hydraulic design analyses. Other significant variations of core power, pressure or coolant flow are limited by the protection systems related to the respective parameters, and therefore the core will be shut down sufficiently before reaching DNBR limit conditions.

Safety analyses will show those protection systems efficiently prevent the core from exceeding the DNB limits. Since the US-APWR has a lower power density core than the current plants, its DNBR analysis results will show relatively larger potential margin to the design limit value.

3.4.2 Fuel Centerline Temperature

The melting temperature of UO_2 fuel pellet is approximately 5070 °F(2800°C) for unirradiated fuel, and is assumed to decrease approximately 58 °F (32°C)for 10 GWd/MTU. The design limit for the fuel centerline temperature is determined lower than the melting temperature, considering model and fabrication uncertainties.

The lower linear power density than that of current plant design results in the US-APWR a relatively larger potential margin to fuel centerline melting.

During the plant operation, a fuel centerline temperature lower than the design limit is assured by limiting the local linear power density with the Over Power ΔT protection system.

3.4.3 Core Hydraulics

During the plant normal operation, Reactor Coolant System (RCS) flow is maintained to be greater than the Thermal Design Flow (TDF), which is evaluated as a lowest flow rate taking into account 10% Steam Generator (SG) tube plugging and measurement uncertainty. This conservative flow rate is used for the thermal design analyses, except that Minimum Measured Flow (MMF) is used as the nominal value of RCS flow in RTDP analysis. MMF is defined as the lowest flow rate with 10% SG tube plugging taking no account of the measurement uncertainty.

Some portion of RCS flow bypasses the core cooling path. Total amount of core bypass flow is approximately 7.5% of total RCS flow rate. It is relatively larger than that of a conventional plant design to allow enough cooling for the Reactor Vessel Head (RVH) and Neutron Reflector (NR). RVH temperature is designed to be same as the cold leg coolant temperature. The core bypass flow is excluded from the effective core flow which is used in thermal design analysis.

Table 3.4-1 Thermal and Hydraulic design parameters

Items	US-APWR	Japanese APWR	US Current four-loop
Reactor core heat output	4,451 MWt	4,451MWt	3,565MWt
Heat Generated in fuel	97.4 %	97.4 %	97.4 %
No. of Fuel Assemblies	257	257	193
Effective heated length	14 ft*	12 ft*	12 ft*
Average Linear Power Density	4.6 kW/ft*	5.3 kW/ft*	5.7 kW/ft*
System Pressure	2,250 psia	2,250 psia	2,250 psia
Thermal Design Flow (TDF)	448,000 gpm*	454,400 gpm*	374,400 gpm
Coolant Temperature Hot Leg RV average Cold Leg	 617 °F* 584 °F* 551 °F*	 617 °F* 584 °F* 552 °F*	 620 °F* 588 °F* 557 °F*
Core Bypass Flow	7.5 %**	7.0 %**	6.3 %
Minimum DNBR	>1.42***	>1.42	>1.55 (typical cell) >1.59 (thimble cell)

* given as approximate figures
** nominal value for DNBR analysis
*** temporarily assumed to be same as APWR

3.5 Functional Design of Reactivity Control Systems

The reactivity control functions are provided by the two independent mechanisms, i.e. maneuvering RCCAs (Rod Cluster Control Assemblies) and adjusting the boron concentration in the RCS (Reactor Coolant System).

The former is applied to control the fast reactivity transients during operation, such as the changes in reactor power demand or temperature transient. It also gives the sufficient negative reactivity to bring the reactor to the hot shutdown condition.

The latter is applied to the compensation of long term reactivity changes, such as the burn up of fuel and the cumulative effects of fission products. It also gives sufficient negative reactivity to bring the reactor to the cold shutdown condition.

3.5.1 Control Rod System

The US-APWR core has 69 RCCAs (See Figure 3.3.1-2).
Each RCCA has the 24 absorber rods made of Ag-In-Cd alloy with stainless steel cladding. These rods fill the guide thimbles in the single fuel assembly.

When the reactor is tripped, all RCCAs are inserted into the fuel assemblies. This insertion process is driven by gravity.

During normal operation, selected groups of RCCAs are maneuvered automatically or manually to control the reactor power to the load demand.

The RCCAs have their insertion limits to ensure the sufficient shut down margin at any time of the reactor operation.

The US-APWR control rod drive mechanism (CRDM) is based on a proven PWR design that has been used in many operating nuclear power plants. However, the structural reliability of the US-APWR CRDM housing is increased by eliminating canopy seals.

The CRDMs, located on the head of the reactor vessel, are coupled to RCCAs. The CRDM consists of the pressure housing, coil stuck assembly, latch assembly and drive rod (See Figure 3.5.1-1). The characteristics of RCCA and CRDM are shown in Table 3.5.1-1.

3.5.2 Boron Concentration Control System

The concentration of boron in the RCS is adjusted through the operation of the CVCS (chemical and volume control system). When increasing the boron concentration, the necessary amount of concentrated boric acid solution is injected into the RCS. When decreasing the boron concentration, pure water is added to the RCS to dilute the coolant water to the required boron concentration.

The concentration of boric acid in the reactor coolant is measured by sampling and chemical titration, as appropriate.

The boron concentration control method allows the reactor operation with the minimum insertion of RCCAs, to ensure that the power distribution is not excessively distorted.

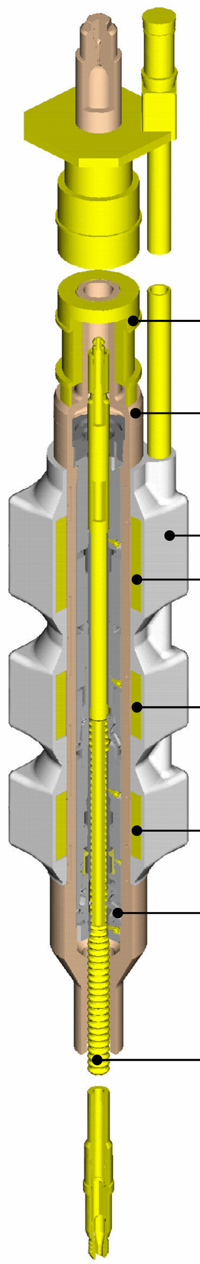


Figure 3.5.1-1
Control Rod Drive Mechanism

Table 3.5.1-1
Characteristics of the RCCAs and CRDMs

Characteristics	Data
RCCAs	
Quantity	69
Number of rods per assembly	24
Absorber (AIC)	
Weight composition (%): Ag, In, Cd	80, 15, 5
Absorber outer diameter	0.341 in
Cladding	
Material	AISI 304 stainless steel
Surface treatment (externally)	Chrome-plating
Outer diameter	0.381 in
Inner diameter	0.344 in
Control Rod Drive Mechanisms (CRDMs)	
Material	Housing : AISI 316 stainless steel
Quantity	69
Stepping speed	45 in/min

3.6 Reactor Pressure Vessel Internals

3.6.1 Design Arrangements

The reactor internals consist of two major assemblies, the lower core support assembly and the upper core support assembly. Figure 3.6-1 shows arrangements of the reactor internals.

The reactor internals support the core, maintain fuel assemblies alignment, limit fuel assemblies movement, maintain alignment between fuel assemblies and control rods, direct coolant flow past the fuel assemblies, transmit the loads from the core to the reactor vessel, provide radiation shielding for the reactor vessel and guide the in-core instrumentation.

The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the reactor vessel and then into a plenum at the bottom of the reactor vessel. The flow then turns and passes through the lower core support and into the core. After passing through the core, the coolant enters the upper plenum and then flows radially to the core barrel outlet nozzles. A small amount of coolant flows into the vessel head plenum to cool the vessel head area.

Both the upper and lower core support assembly are preloaded axially by a hold-down spring at the reactor vessel ledge location.

The fuel assembly alignment is provided by pins attached to the upper core plate and lower core support.

The material for the reactor internal structures is Type-304 stainless steel. The material of bolts and pins is strain hardened Type-316 stainless steels.

All reactor internals are removable from the vessel for inspection of the internals and of the vessel inner surfaces.

3.6.2 Lower Core Support Assembly

The lower core support assembly consists of the core barrel, the lower core support, the neutron reflector, the diffuser plate and the energy absorber. Figure 3.6-2 shows the lower core support assembly.

Vertical loads from weight, fuel assembly preload, control rod scram load, hydraulic loads, and seismic and LOCA loads are transmitted to the lower core support. The loads are then transferred to the core barrel flange, and transferred to the reactor vessel flange. Horizontal loads from coolant cross-flow, seismic and LOCA loads and flow induced vibration loads are transmitted to the core barrel flange and the lower radial supports and then transferred to the reactor vessel. Horizontal loads from the fuel assemblies are absorbed by fuel pins attached in the upper core plate and the lower core support and then transmitted to the core barrel.

3.6.2.1 Core Barrel

The core barrel is a long cylindrical structure which has four outlet nozzles, and the flange is welded at the top of the cylinder. The top flange is supported at a ledge of the reactor vessel. The bottom of the core barrel is horizontally supported by six radial supports welded to the reactor vessel. The neutron reflector is located inside the core barrel. The irradiation specimen guides are attached to the outside of the core barrel.

3.6.2.2 Lower Core Support

The lower core support is welded to the bottom of the core barrel. The lower core support supports all the fuel assemblies, the neutron reflector, the diffuser plate and the energy absorber. The alignment pins for the fuel assemblies are attached on the top of the plate.

Four flow holes are provided for each fuel assembly. Holes for the radial reflector cooling flow are also provided in periphery region of the lower core support.

3.6.2.3 Neutron Reflector

The neutron reflector is located between the core barrel and core, and forms the core cavity. The neutron reflector consists of ten thick stainless steel blocks. These blocks are aligned by alignment pins and fixed to the lower core support by tie rods and bolts. The top and bottom are supported and aligned by alignment pins which are welded to the core barrel.

The purposes of the neutron reflector are to improve neutron utilization and thus the fuel cycle cost, to reduce neutron irradiation of the reactor vessel, and to increase structural reliability by eliminating bolts in the high neutron flux region.

The neutron reflector is cooled by coolant through the cooling holes in the blocks to avoid boiling of coolant and prevent excessive stress and thermal deflections of the blocks due to the gamma heating.

Figure 3.6-4 shows the neutron reflector assembly.

3.6.2.4 Diffuser Plate

The diffuser plate is attached in the reactor vessel lower plenum. The purpose of the diffuser plate is to suppress flow vortices formed by the reactor coolant flow. The diffuser plate is supported by columns from the lower core support.

3.6.2.5 Energy Absorber

In the event of failure of the core barrel following hypothetical accident, the energy-absorber reduces the dynamic force imposed on the reactor vessel bottom head.
The energy absorber limits the vertical displacement of the core to prevent withdrawal of the control rods from the core.

3.6.3 Upper Core Support Assembly

The upper core support assembly consists of the upper core support, the upper core plate, the upper support columns and guide tubes.
Figure 3.6-2 shows the upper reactor internals assembly.

Vertical loads from seismic and LOCA loads, weight, hydraulic loads, and fuel assembly preload are transmitted to the upper core plate, then transferred to the upper core support , and then into the reactor vessel head. Horizontal loads from seismic and LOCA, coolant cross-flow, and flow-induced vibrations are transferred by the support columns to the upper core support and upper core plate.

3.6.3.1 Upper Core Support

The upper support plate is the major support structure of the upper internals, and consists of the flange, the cylindrical skirt and the thick forged plate.

The guide tubes and the upper support columns are supported by the upper core support.

3.6.3.2 Upper Core Plate

The upper core plate is connected to the upper core support by the upper support columns. The fuel assembly alignment pins are attached at the bottom of the upper core plate.

The upper core plate is positioned in its proper location, with respect to the lower support assembly, by the upper core plate guide pins in the core barrel.

3.6.3.3 Upper Support Column

The upper support columns connect the upper core support and the upper core plate. The main structure is fabricated from stainless steel tubes and the top and bottom are bolted to the upper core support and the upper core plate.

The upper support columns support the conduit tube of the thermocouple and the in-core instrumentation inserted from the reactor vessel head inside the columns.

3.6.3.4 Guide Tube

The guide tube assemblies guide the control rod drive shaft and RCCA’s.

The upper flange of guide tubes are fastened to the upper core support in the middle and the bottom of guide tubes are restrained by pins in the upper core plate. There are sixty-nine guide tubes.

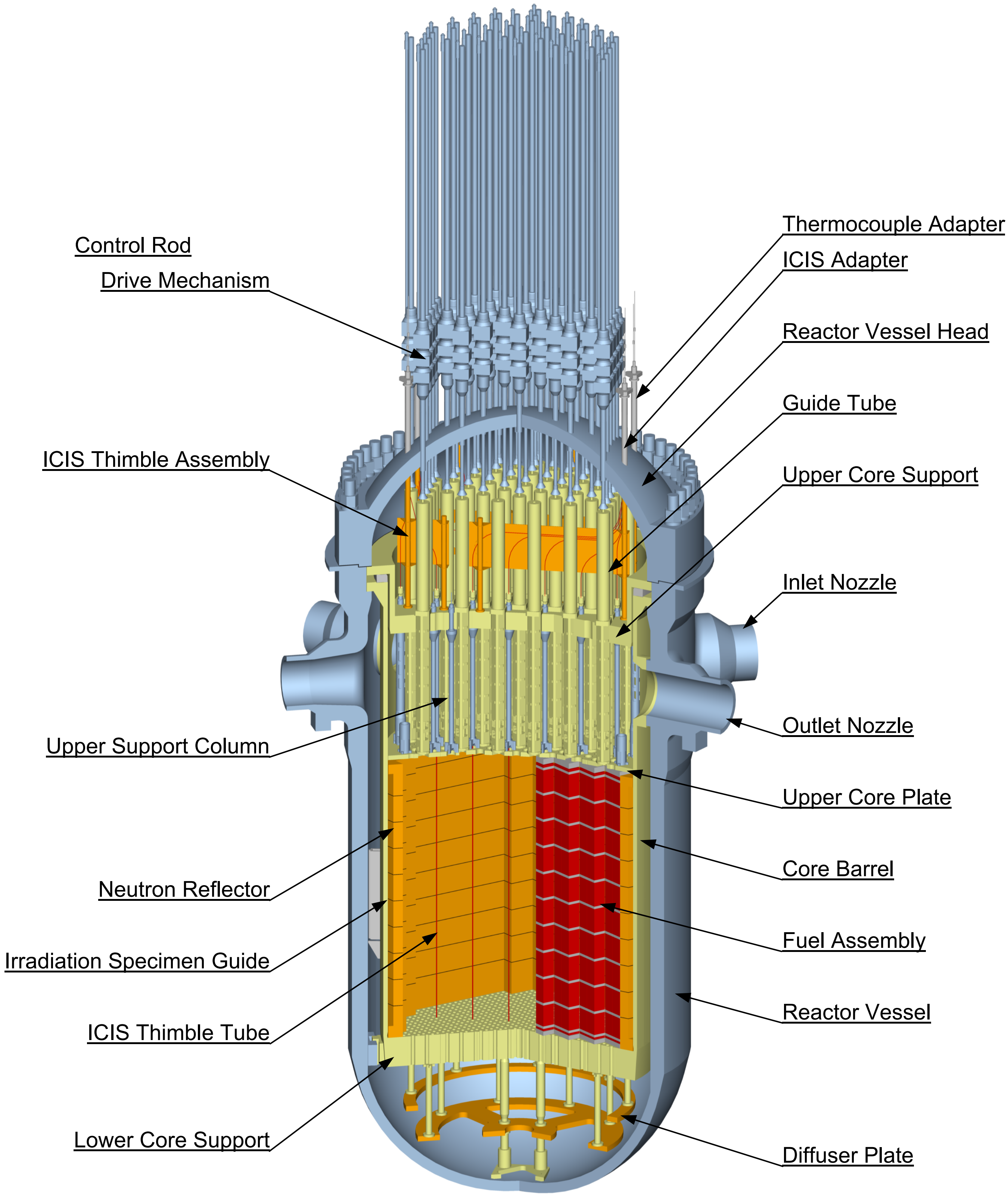


Figure 3.6-1 Reactor General Assembly

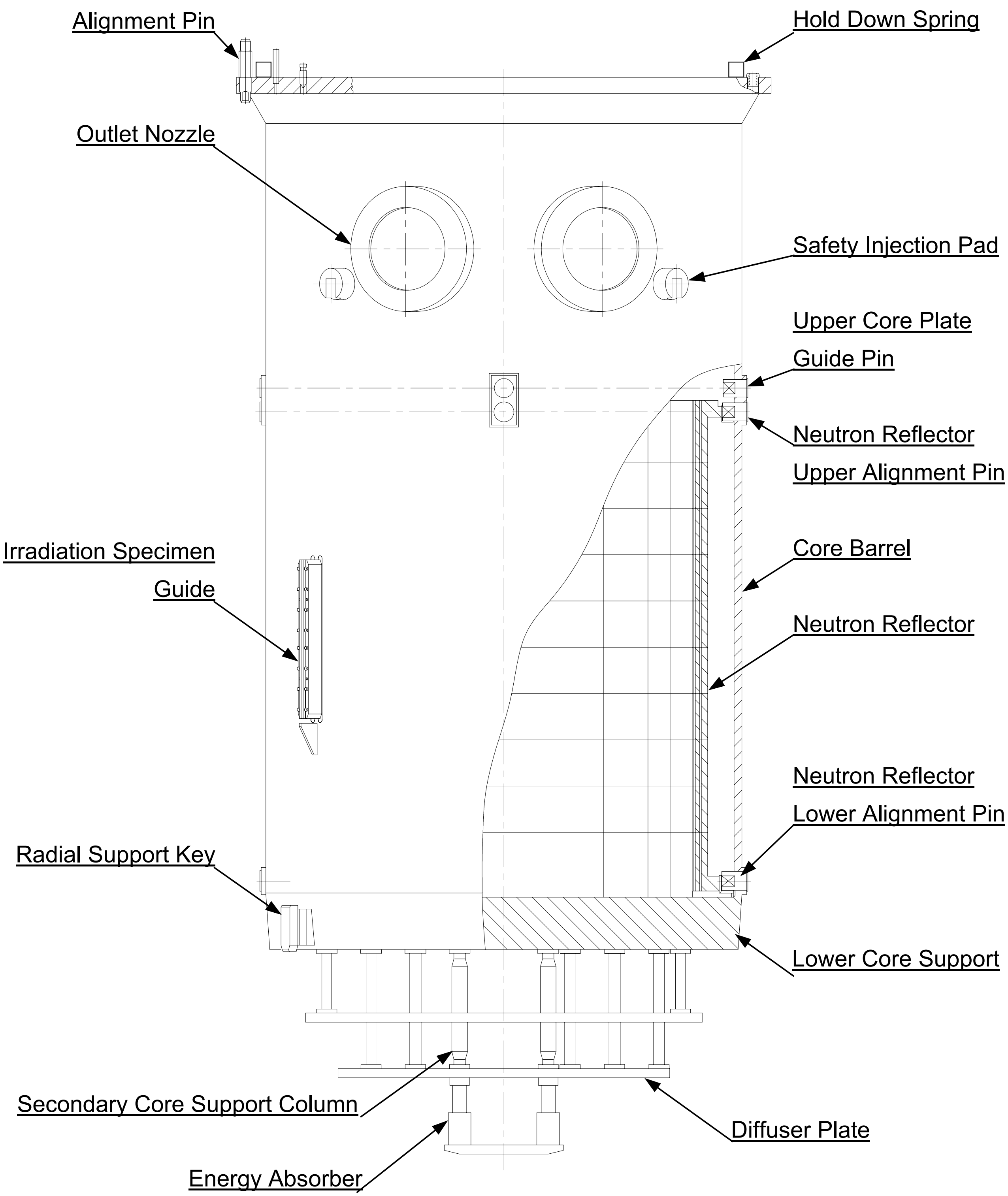


Figure 3.6-2 Lower Core Support Assembly

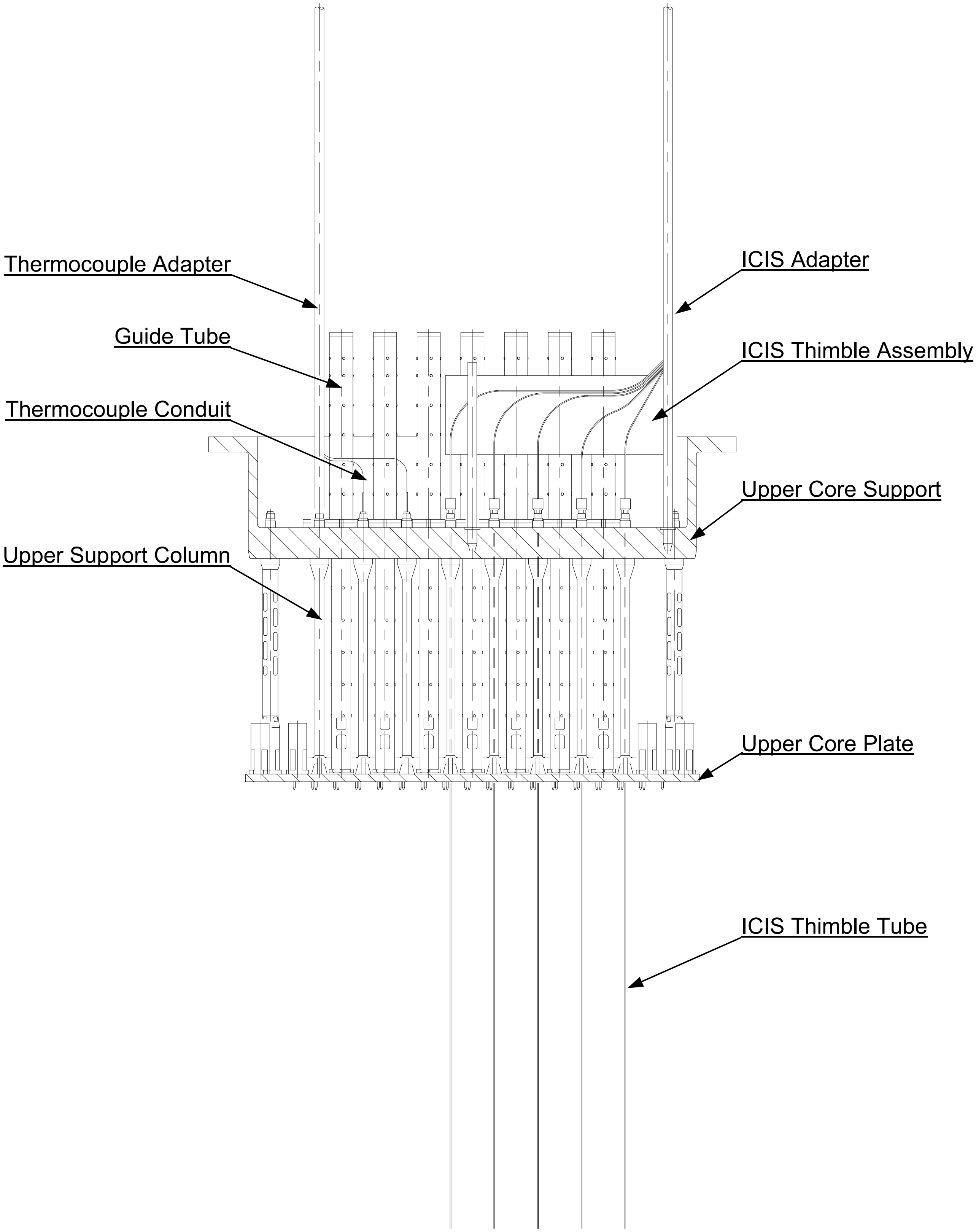


Figure 3.6-3 Upper Core Support Assembly

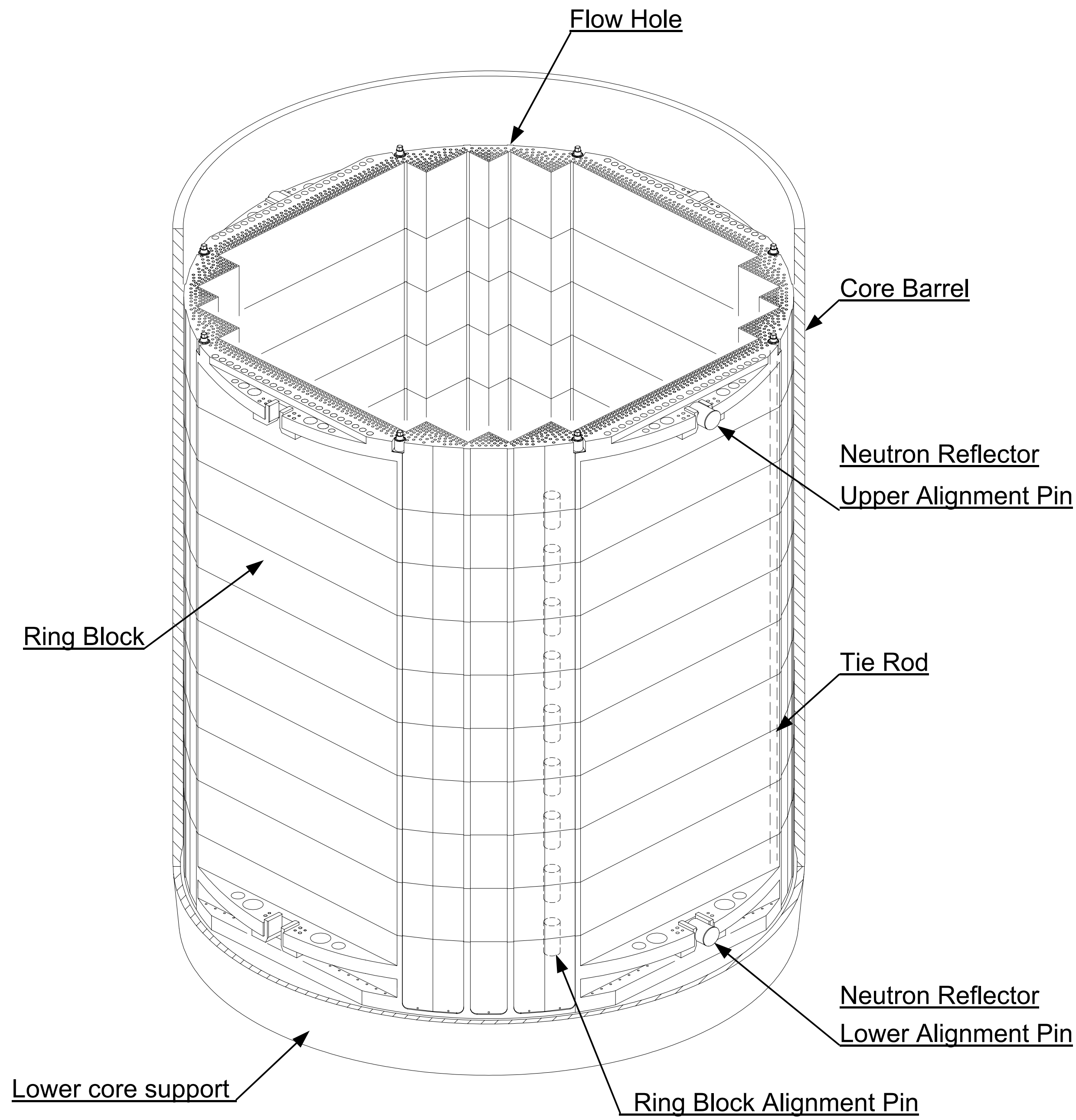


Figure 3.6-4 Neutron Reflector Assembly

4.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

4.1 Outline

The reactor coolant system (RCS) provides reactor cooling and energy transport functions. The RCS shown, in Figure 4.1-1, consists of the reactor vessel, the steam generators, the reactor coolant pumps, the pressurizer, the reactor coolant pipes, and valves.

The reactor coolant system, including connections to related auxiliary systems, constitutes the reactor coolant pressure boundary.

The reactor coolant system performs the following functions:

- Circulates the reactor coolant through the reactor core and transfers heat to the secondary system via the steam generators.
- Cools the core sufficiently to prevent core damage during reactor operation.
- Forms the reactor coolant pressure boundary, which functions as a barrier to prevent radioactive materials in the reactor coolant from being released to the environment.
- Functions as a neutron moderator and reflector and as a solvent for boron.
- Controls the reactor coolant pressure.

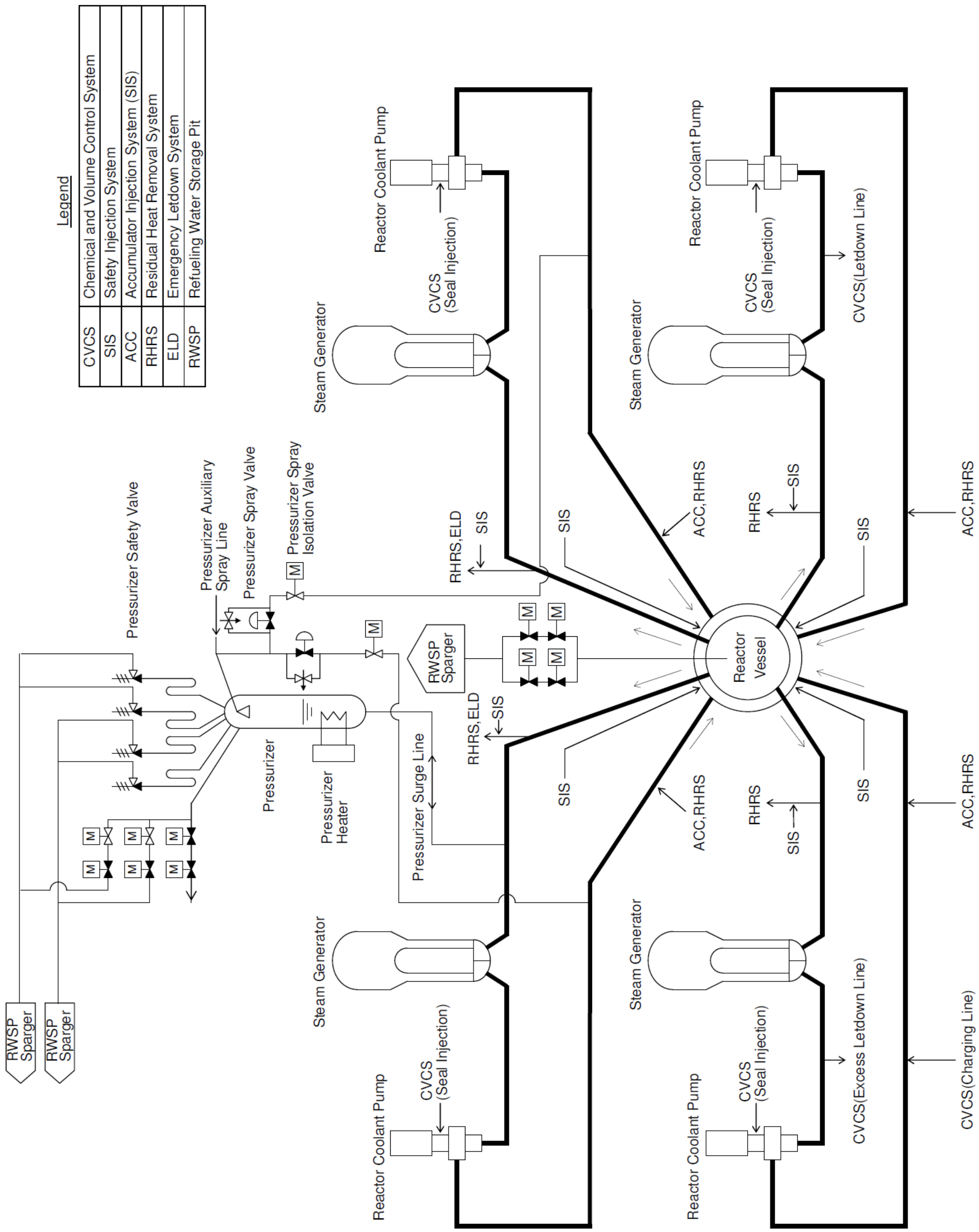


Figure 4.1-1 Reactor Coolant System

4.2 Design Principle

The reactor coolant system is the major portion of the reactor coolant pressure boundary and is important to safety, preventing accidents and controlling their consequences. Sufficient attention is paid to its design, material selection, and quality control to satisfy the following:

- The reactor coolant system is designed to provide core cooling during normal operation, transients, and accident conditions.
- The materials of the reactor vessel, steam generators, pressurizer, reactor coolant pumps, piping, valves, and other components which contact the reactor coolant, are selected to maximize corrosion resistance.
- Components that form the reactor coolant pressure boundary are designed and operated to prevent nil-ductility fracture. Special attention is paid to material selection, design, manufacture, and operation of ferritic steel components. This is done to assure that, under normal operation, transients, maintenance, testing, and accident conditions, the ferritic steel components behave in a non-brittle manner and the probability of rapidly propagating fracture is minimized. The operation of the reactor coolant system during startup and shutdown is controlled in accordance with heating and cooling limits that consider fast-neutron irradiation effects throughout the lifetime of the plant.
- Class I seismic design is applied to the reactor coolant system and its supporting structures.
- The design and arrangement of supporting structures, including concrete, are such that functions important to safety are not impaired by the impact resulting from a postulated rupture of the piping which forms the reactor coolant pressure boundary. Pipe whipping restraints are installed where necessary.
- Reactor coolant pressure boundary components are designed using conservative assumptions about future plant operating conditions. These include transient conditions such as variations in temperature and pressure and conservative estimates of the number of cycles for each transient.
- A leak monitoring system is used to provide early detection of leakage from the reactor coolant pressure boundary.
- The systems and components that form the reactor coolant pressure boundary are designed to allow periodic in-service inspections in accordance with ASME Code, Section XI.

4.3 Overpressure Protection

The reactor coolant system is designed so that the system pressure can be maintained at less than 1.1 times the design pressure by the pressure relief system.

The pressure relief system has the following design features:

- Spring-loaded safety relief valves (SRVs) are installed on separate relief lines at the top of the pressurizer
- An additional relief line has motor-operated relief valves for safety depressurization valves (SDVs). The valves are arranged in parallel and are driven by motor operators. A remotely controlled, motor-operated isolation valve is installed upstream of each the SDVs to allow isolation in the event of a leak.
- Discharged pressurizer steam is discharged through spargers installed in the refueling water storage pit (RWSP) inside the containment.
- Safety relief valves are installed in the each residual heat removal system to provide over-pressurization protection for unacceptable combinations of high reactor coolant system pressure and low reactor coolant system temperature.

4.4 Reactor Vessel

The Reactor Vessel (RV), shown in Figure 4.4-1, is a vertical cylindrical vessel with hemispherical top and bottom heads. The top head is a removable flanged closure head connected to the RV upper shell flange by stud bolts. Pads on the RV nozzles support the vessel.

The RV contains the fuel assemblies and reactor vessel internal core, including the core support structures, control rods, neutron reflector and other structures associated with the core.

The RV consists of four inlet nozzles, four outlet nozzles and four safety injection nozzles, which are located between the upper reactor vessel flange and the top of the core, so as to be able to maintain coolant in the reactor vessel in the case of leakage in the reactor coolant loop.

Coolant enters the vessel through the inlet nozzles, flows down the annulus between the core barrel and RV wall, turns at the bottom of the vessel and flows upwards through the core to the outlet nozzles.

Sealing between the upper closure head flange and RV upper shell flange is performed by two metallic O-rings. The O-rings are fitted into the grooves machined on the cladding of the bottom surface of the upper closure head flange. Seal leakage is detected by means of two monitoring tubes in the upper shell flange, one located between the inner and outer O-rings, and one located outside the outer O-rings. Piping and associated valves direct any leakage to the reactor coolant drain tank. Excessive leakage is indicated by a high temperature alarm signal.

The RV closure head consists of a hemispherical dome welded to the closure head flange, which is a thick ring forging.

The RV closure head is equipped with control rod drive mechanism nozzles to which the pressure housings for the control rod drive mechanisms are welded. The top closure head is also equipped with in-core instrumentation and thermocouple nozzles.

A vent line to the vent drain system is installed on the RV closure head and used during venting operations of the closure head prior to plant heat-up.

The main cylindrical shell of the RV consists of an upper and lower shell. Both shells are ring forgings and the upper shell includes an integrated flange. The use of one-piece forgings reduces the number of welds that need to be inspected during in-service inspections.

The bottom head consists of a transition ring and bottom hemispherical dome. The transition ring is a ring forging which connects the bottom dome to the RV lower shell. The bottom dome does not have any penetrations.

The main parts of the RV are made of low alloy steel with weld-deposited stainless steel cladding on all internal surfaces exposed to the reactor coolant. All other parts exposed to the coolant are either stainless steel or nickel base alloys.

Encapsulated test specimens are inserted between the core barrel and the reactor vessel. After being irradiated the test specimens are withdrawn at appropriate periods and destructively tested to monitor changes in material characteristics during the service periods.

Where irradiation by fast neutron can be relatively high in the RV, the vessel wall is designed so that there are no shape discontinuities that could cause stress concentrations.

The outer surface of the RV is covered with stainless steel insulation which can resist corrosion from boric acid solution.

The inner surfaces of the vessel walls can be inspected from inside the RV when the RV internals have been removed.

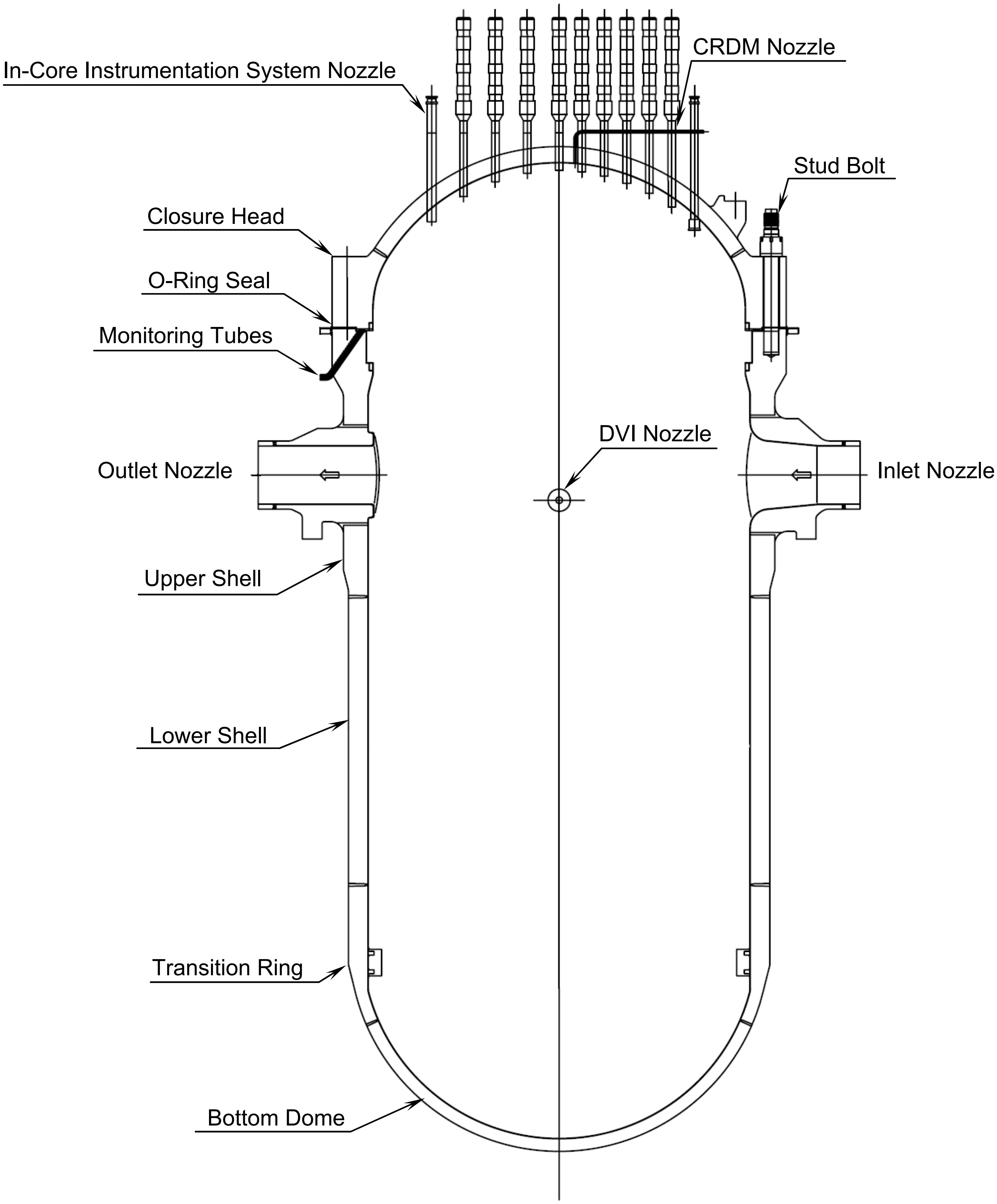


Figure 4.4-1 Reactor Vessel

4.5 Component and Subsystem Design

4.5.1 Reactor Coolant Pumps

The reactor coolant pumps, shown in Figure 4.5.1-1, are vertical single-stage centrifugal, shaft seal units, driven by three-phase induction motors. The shaft is vertical with the motor mounted above the pump. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The pump suction is located at the bottom, and the discharge is located on the side.

The reactor coolant that enters from the bottom of the casing is accelerated by the impeller attached on the pump shaft lower end, and is transformed to pressure through the diffuser which is then delivered through the discharge nozzle located on the casing side. The diffuser is located at the center of the casing in order to attain high hydraulic efficiency.

The US-APWR reactor coolant pump (Type 100A) has achieved larger capacity and higher efficiency by improving of the impeller and diffuser configuration.

The impeller is shrunk to the shaft with the interface of a conical shape. This shape is easy to maintain during assembly and disassembly. The diffuser has flange on the upper side. The diffuser flange, main flange and motor stand are jointly fitted to the casing with studs. The motor is installed on the top of the pump and is connected to the pump shaft by a rigid coupling.

The pump and motor shaft is supported by three radial bearings located at the upper and lower ends of the motor as well as at an interior point in the pump. The bearing located in the pump is water lubricated and is cooled by the component cooling water. The bearings located in the motor are oil lubricated.

Leakage along the reactor coolant pump shaft is normally controlled by three shaft seals, The No.1 seal is a hydrostatic seal and the No.2 or No.3 seal is a hydrodynamic seal, arranged in series so that any reactor coolant leakage to the containment is essentially zero. As for the No.1 seal, its characteristics and durability have been significantly improved.

A high pressure seal water, supplied by the chemical and volume control system (CVCS) is injected under the No.1 seal assembly to prevent the high temperature reactor coolant from entering into the shaft seal. Some part of this injected water flows upward through the controlled leakage seal. Most of this water then leaves the pump assembly and is returned to the CVCS. The remaining injected water, which cools the shaft and bearing, flows downward through the thermal barrier heat exchanger, and into the RCS.

The No.2 seal is provided as a backup for the No.1 seal. If one of these two seals fails, the other seal can fully perform the sealing function.

In addition, the No.3 seal prevents leakage through the No.2 seal from being released into the containment environment so that the containment environment is protected from contamination.

The pump shaft, seal housing, thermal barrier, main flange, and impeller of the RCP can be removed from the casing as a unit without disturbing the reactor coolant piping.

All parts of the pump in contact with the reactor coolant are stainless steel except for the seals, bearings, and special parts.

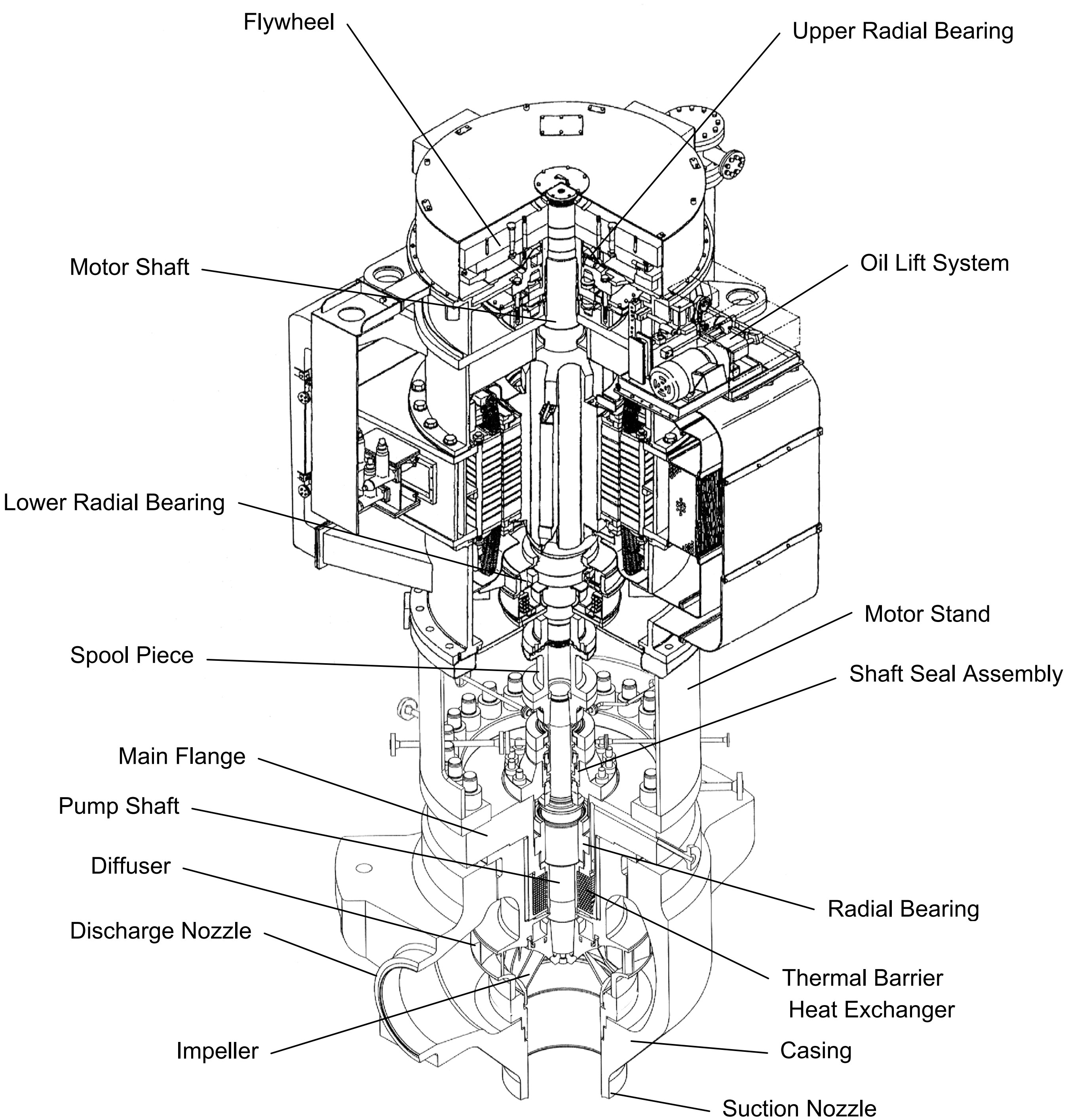


Figure 4.5.1-1 Reactor Coolant Pump

4.5.2 Steam Generators

The steam generators (SGs) are vertical shell U-tube evaporators with integral moisture separating equipments. Figure 4.5.2-1 shows the steam generator, with identification of several of its design features.

The reactor coolant enters the channel head via the hot side primary coolant nozzle, flows through the inverted U-tubes, transferring heat from the primary side to the secondary side, and leaves from the channel head via the cold side primary coolant nozzle. The channel head is divided into inlet and outlet chambers by a vertical partition plate extending from the apex of the head to the tube sheet.

The cladding on the primary side of the tube sheet is Ni Cr Fe alloy, and the cladding on the channel head is stainless steel.

The tube material is alloy 690 with thermal treatment, which is widely used in SGs throughout the world. The tube has good performance of corrosion resistance.

Steam generated on the shell side (secondary side), flows upward, and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes through a feedwater nozzle. The feedwater enters a feedring and is distributed through nozzles attached to the top of the feedring. Material of the nozzles and feedring is an alloy that is resistant to erosion and corrosion for the expected secondary water chemistry and flow rate through the nozzles and the feedring. After exiting the nozzles, the feedwater mixes with saturated water removed by the moisture separators. The flow then enters the downcomer annulus between the wrapper and the shell.

Supports of the tubes are provided by ferritic stainless steel support plates. The holes in the tube support plates are broached with flow area around tubes. Anti-vibration bars are installed in the U-bend portion of the tube bundle to minimize the potential for excessive tube vibration and wear.

When the water passes the tube bundle, it is converted to a steam-water mixture. The steam-water mixture from the tube bundle then rises into the primary separators and the secondary separators. Figure 4.5.2-2 shows the high performance of the primary separators. The high performance primary separators and the secondary separators make moisture carryover ratio less than 0.1% at the SG outlet.

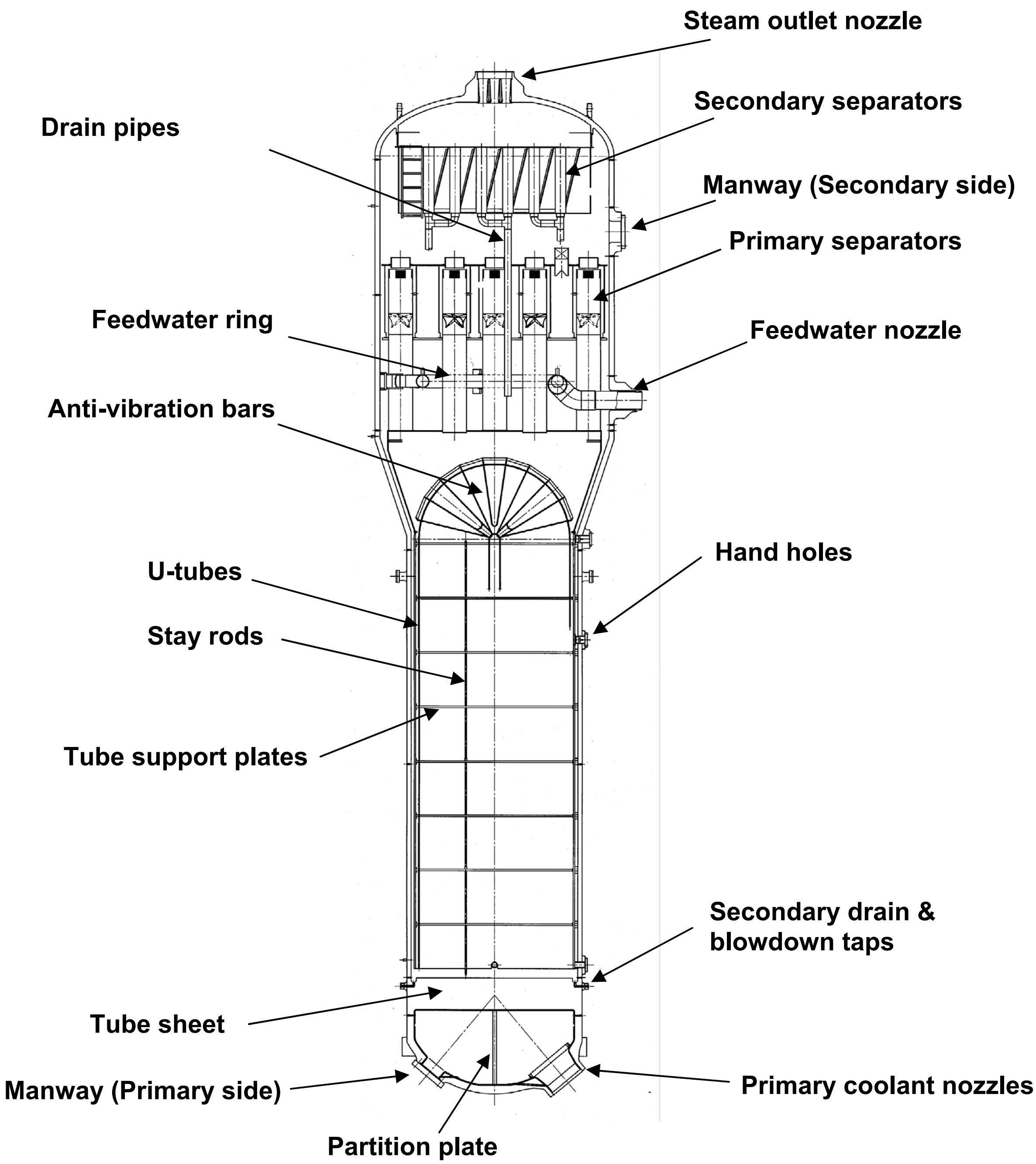


Figure 4.5.2-1 Steam Generator

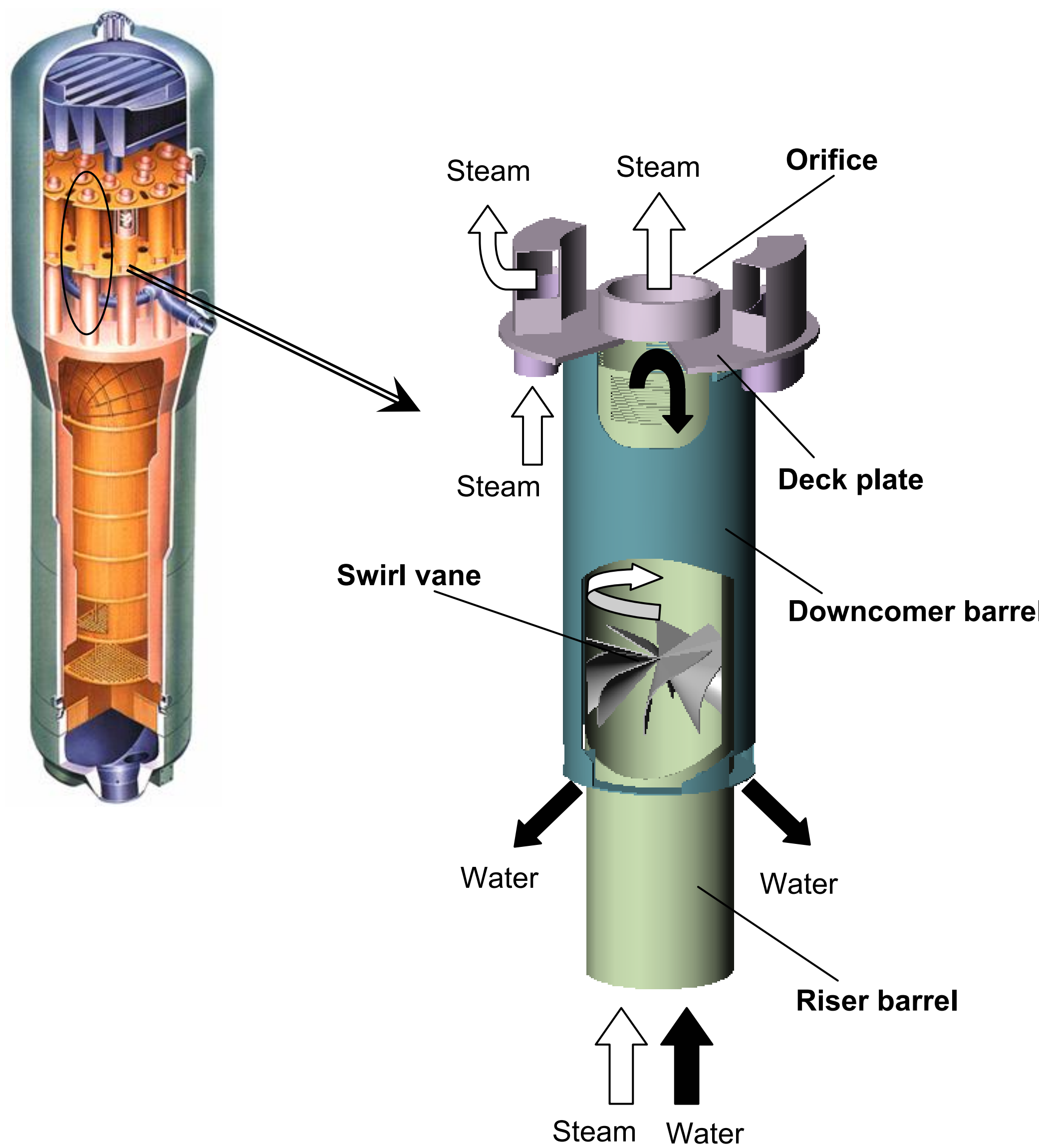


Figure 4.5.2-2 High Performance Primary Separator

4.5.3 Reactor Coolant System Piping

The reactor coolant pipe work consists of the pipes connecting the reactor pressure vessel, steam generators, reactor coolant pumps, and pressurizer, together with the various branches off the main pipe work up to the appropriate isolating valve. It also includes instrumentation connections to the reactor coolant system that provide for flow, temperature, pressure, chemical, and radiation monitoring.

The reactor coolant pipes and fittings are of austenitic stainless steel. Pipes and fittings are seamless and comply with the requirements of the ASME Code, Section II (Parts A and C), Section III and Section IX. All smaller piping that is part of the reactor coolant system, such as the pressurizer surge line, spray and relief line, loop drains, and connecting lines to other systems, are also austenitic stainless steel. All joints and connections are welded, except for the pressurizer safety valves where flanged joints are used.

The reactor coolant piping is designed using the Leak-Before-Break (LBB) concept.

4.5.4 Residual Heat Removal System

The residual heat removal system (RHRS), shown in Figure 4.5.4-1, consists of four independent subsystems, each having one containment spray/residual heat removal (CS/RHR) heat exchanger, one CS/RHR pump, and connecting piping and valves.

Because the emergency core cooling system uses the advanced accumulator design and an improved high head injection subsystem (Refer to section 5.3), the low head injection subsystem found in a conventional plant is not necessary. The residual heat removal function is transferred to the containment spray system, and the CS/RHR heat exchangers and the CS/RHR pumps are used for both RHRS and CSS.

The RHRS has the following functions;

- Removes reactor core decay heat and other residual heat from the reactor coolant.
- Transfers refueling water between the reactor cavity and the refueling water storage pit at the beginning and end of refueling operations.

The RHRS design is based on the following;

- The RHRS is designed to cool the reactor by removing decay heat and residual heat from the reactor coolant system after the initial phase of cooldown.
- The RHRS is designed with four independent subsystems.
- The CS/RHR pumps receive power from safety electrical buses so that the RHRS functions are maintained during a loss of offsite power.
- The RHRS is designed to provide the ability to reduce the reactor coolant temperature with only two of the four subsystems operating.
- The RHRS is designed to transfer borated water from the refueling water storage pit to the refueling cavity at the beginning of a refueling operation. After refueling, the reactor cavity is drained by pumping the water back to the refueling water storage pit or allowing it to return by gravity.

The RHRS is placed in operation when the pressure and temperature of the RCS are approximately 400 psi (2.76 MPa) and 350 °F (177 °C), respectively.

During system operation, each CS/RHR pump takes suction from one of the reactor coolant system hot legs by a separate suction line. The pumps discharge through the CS/RHR heat exchangers which transfer heat from the reactor coolant to the component cooling water system. The reactor coolant is returned to the four reactor coolant system cold legs.

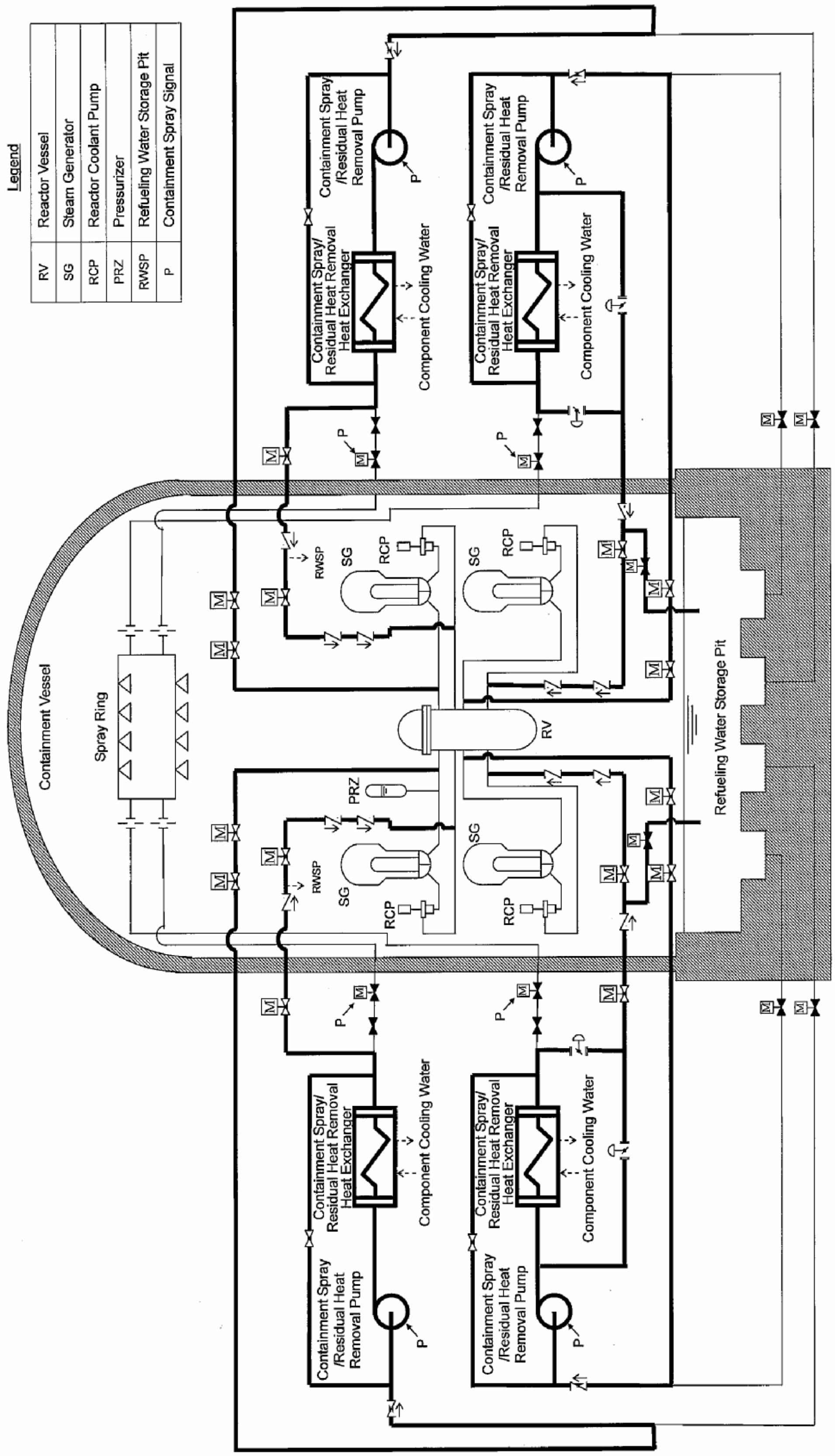


Figure 4.5.4-1 Residual Heat Removal System

4.5.5 Pressurizer

The pressurizer provides a point in the reactor coolant system where liquid and vapor can be maintained in equilibrium under saturated conditions for pressure control. The pressurizer, shown in Figure 4.5.5-1, is a vertical, cylindrical vessel with hemispherical top and bottom heads. It is constructed of low-alloy steel with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. Electrical immersion heaters are installed vertically through the bottom head of the vessel while the spray nozzle, and relief and safety valve connections are located in the top head of the vessel. A manway is also provided in the top head for access to the internal space for inspections and maintenance of the spray nozzle. The manway closure is a gasketed cover fixed with threaded fasteners.

The pressurizer is designed to accommodate positive and negative volume surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects to the hot leg of a reactor coolant loop.

A screen above the surge line is provided to prevent passage of foreign particles from the pressurizer to the reactor coolant system.

Baffles in the lower section of the pressurizer prevent in-surge of cold water from flowing directly to the steam/water interface and assist in mixing.

The baffles also provide support to limit vibration of the heaters.

The pressurizer is supported by a skirt welded to the bottom head.

Each spray line is provided with a separate, automatically-controlled, air-operated spray valve, with manual override and a spray block valve. A manual throttle valve is provided in parallel with each spray control valve. This throttle valve enables a small continuous flow to be maintained in the spray lines when the spray valves are closed. An auxiliary spray line is provided from the chemical and volume control system to ensure that the pressurizer spray is available to permit reactor cooldown should the reactor coolant pumps not be available. The pressurizer surge line connects the pressurizer to the hot leg of one of the reactor coolant loops.

Spring-loaded safety relief valves (SRVs) are positioned on separate relief lines from the pressurizer. Another relief line incorporates motor-operated relief valves as safety depressurization valves (SDVs) arranged in parallel. These valves are driven by the motor operators. All relief lines run into spargers, which carry pressurizer steam discharge to the refueling water storage pit (RWSP) inside the containment. Remotely controlled, motor-operated isolation valves are provided upstream of each of the SDV to allow isolation of a leaking SDV.

The safety relief valves provide overpressure protection of the reactor coolant system. The spray valves limit reactor coolant system pressure rises following less severe transients to prevent undesirable opening of the pressurizer safety relief valves. Other safety relief valves in the residual heat removal system provide the cold overpressurization protection against unacceptable combinations of high reactor coolant system pressure and low reactor coolant system temperature.

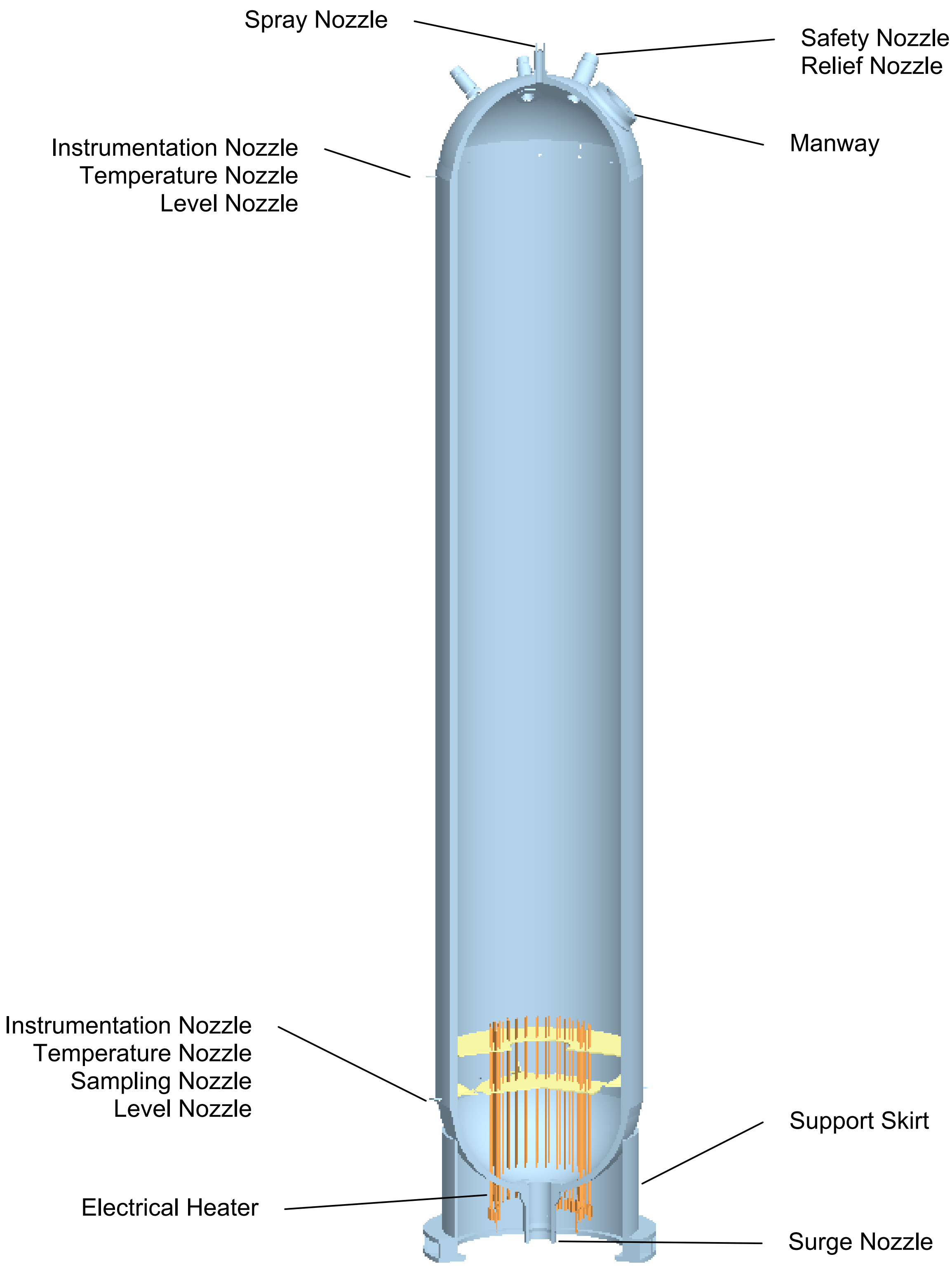


Figure 4.5.5-1 Pressurizer

4.5.6 Component Supports

a. Reactor Vessel

Figure 4.5.6-1 shows the support structures for the reactor vessel.
The reactor vessel is supported by 8 steel support pads which are one-piece with the inlet and outlet nozzle forgings. The support pads are placed on support brackets, which are supported by steel structure around the reactor vessel (base plate).
Radial movement, which results from the vessel expansion and contraction caused by temperature change, is accommodated by sliding surfaces between the shim plates and the support pads while the horizontal load in an earthquake is supported by the support brackets and the base plate, so that the center position of the vessel can always remain unchanged.
The support brackets, which are of box-shaped structure, are air cooled by the reactor vessel compartment cooling fans in order to minimize heat transfer from the reactor vessel to the concrete support portions through the support brackets.

b. Steam Generators

Figure 4.5.6-2 shows the support structures for the steam generators.
The steam generators are supported by an upper lateral support structure, a lower lateral support structure, and support columns.
The upper lateral support structure supports steam generator by using snubbers.
The lower lateral support structure is a structure made of steel.
The support structures for the upper shell and bottom shell are designed by considering the thermal expansion of piping.
At the same time, they can restrain the horizontal movement of the steam generator in the events of earthquake or accidents.
The support columns support vertical loads, and the upper and lower ends of the support pipe are pin-jointed, so as not to restrain the movement of the steam generators caused by thermal expansion of piping.

c. Reactor Coolant Pumps

Figure 4.5.6-3 shows the support structures for the reactor coolant pump.
The reactor coolant pump is supported by upper and lower support structures, and support columns.
The upper support structure supports reactor coolant pump by using snubbers while the lower support structure is a structure made of steel.
The upper and the lower support structures are designed by considering thermal expansion of piping.
At the same time, they can restrain the horizontal movement of the reactor coolant pump in the events of earthquake and accidents.
The support columns support vertical loads, and the support pipe upper and lower ends are pin-jointed in the same manner as the steam generator so as not to restrain the movement of the reactor coolant pump caused by thermal expansion of the piping.

d. Pressurizer

Figure 4.5.6-4 shows the support structures for the pressurizer.
The pressurizer is supported by upper support structure and lower support skirt.
The upper support structure supports pressurizer using steel structure, while the lower structure supports vertical load using a skirt welded to the bottom shell of the pressurizer.
The upper support structure does not restrain the movement of the pressurizer caused by thermal expansion, but restrains horizontal movements in the events of earthquake or accidents.

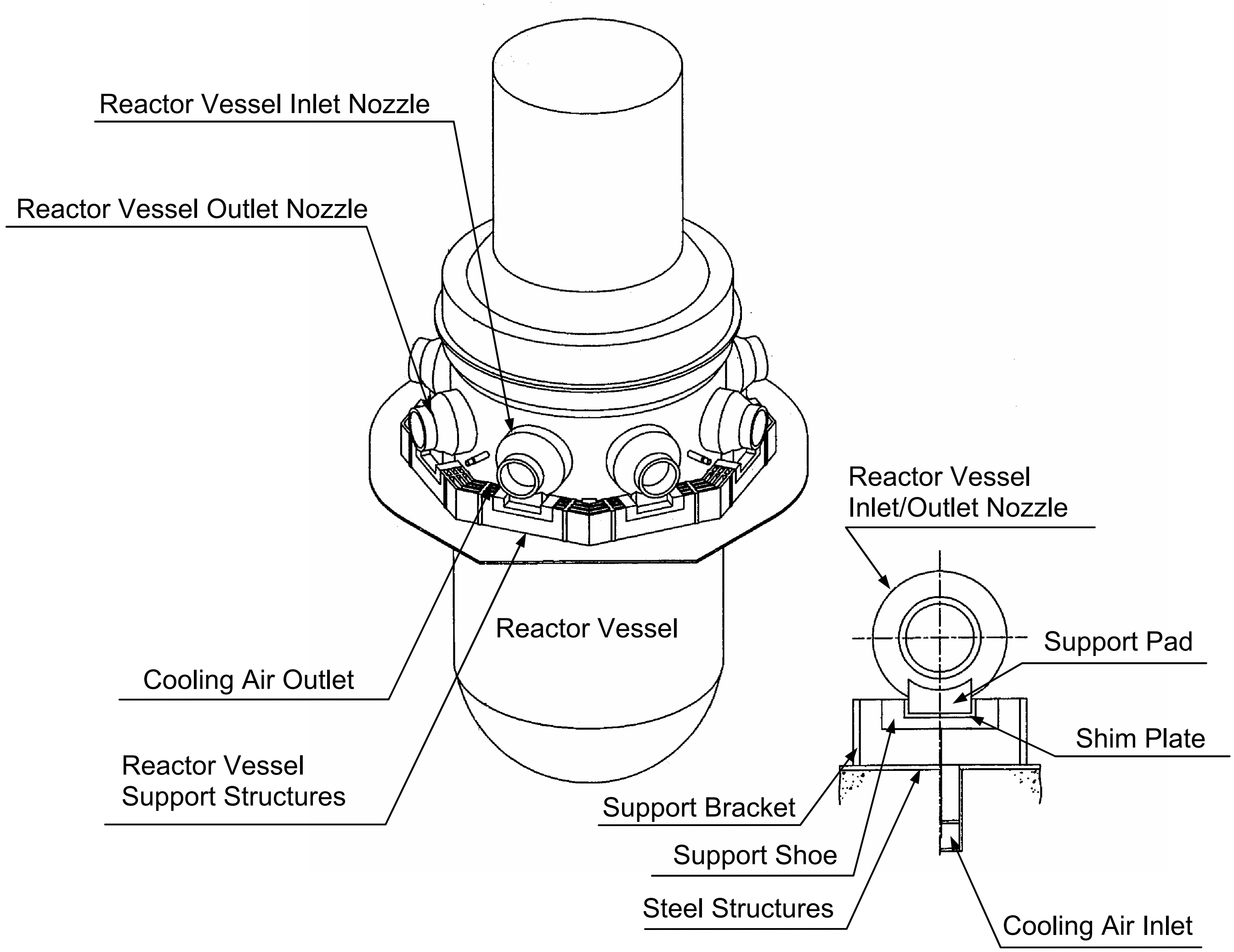


Figure 4.5.6-1 Reactor Vessel Support Structures

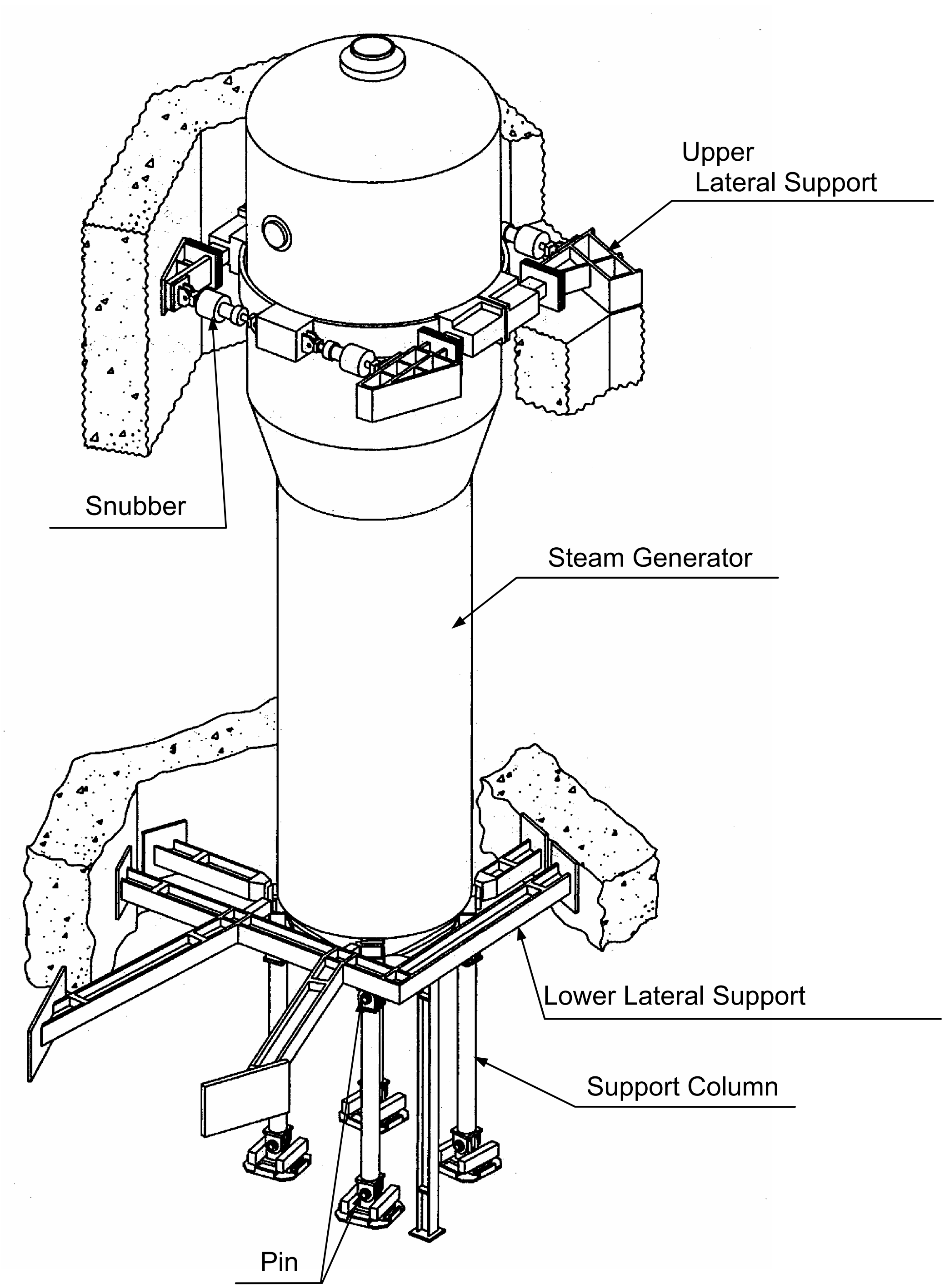


Figure 4.5.6-2 Steam Generator Support Structures

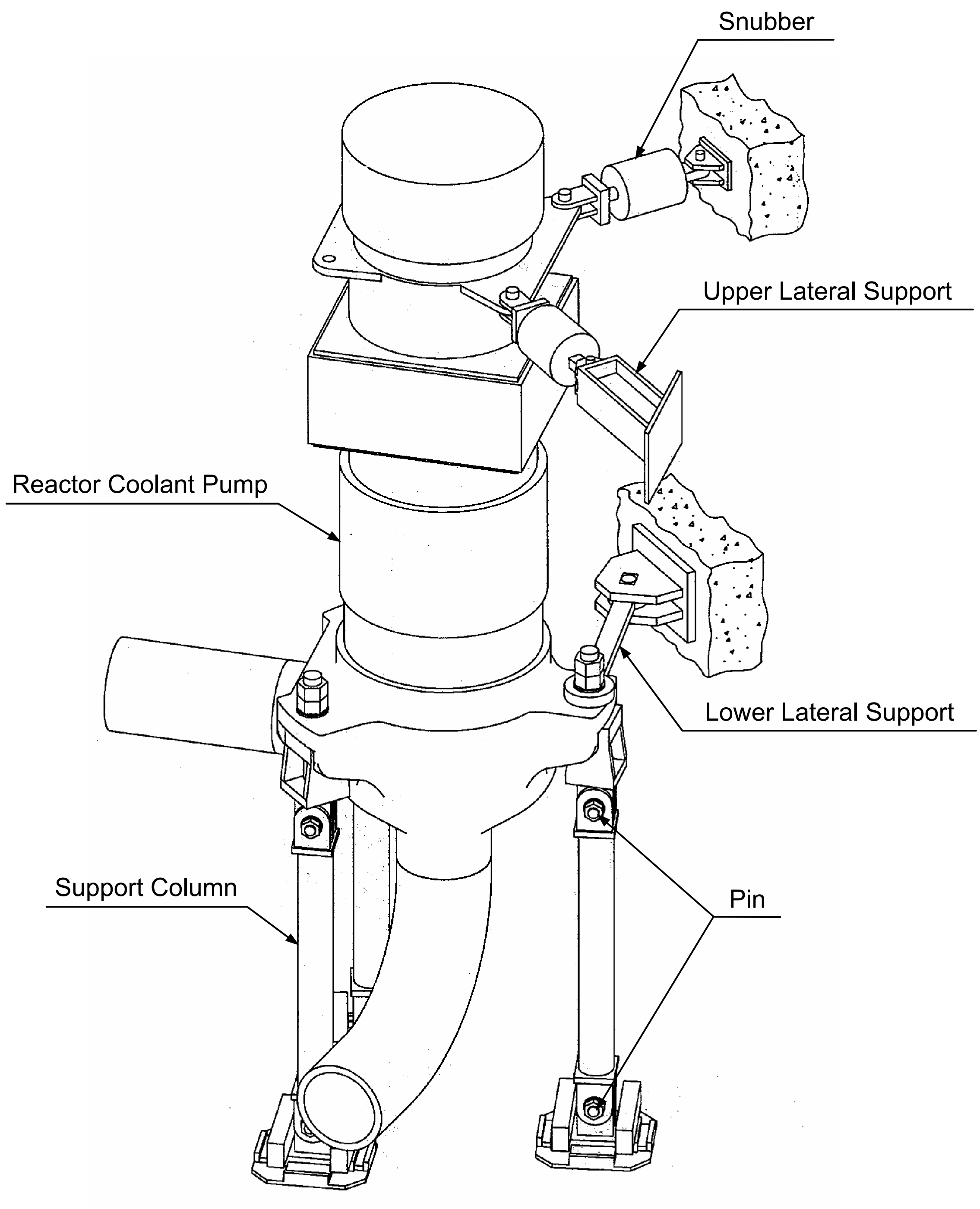


Figure 4.5.6-3 Reactor Coolant Pump Support Structures

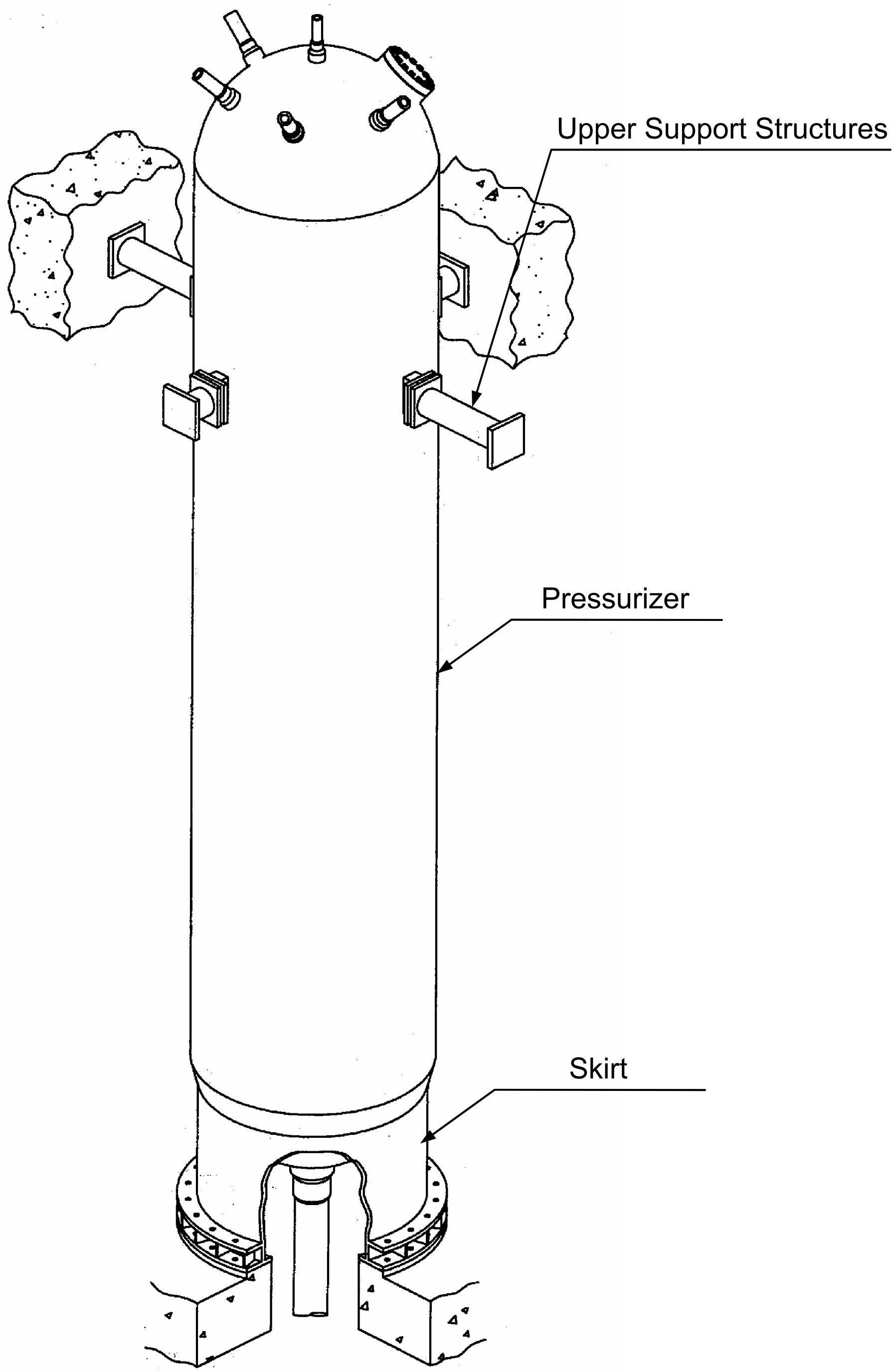


Figure 4.5.6-4 Pressurizer Support Structures

5.0 ENGINEERED SAFETY FEATURES

5.1 Outline

The engineered safety features (ESFs) serve to mitigate the consequences of a design basis accident in which radioactive fission products are released from the reactor coolant system.

The ESFs consist of the emergency core cooling system, containment systems, containment spray system, annulus air cleanup system and main control room HVAC system.

The descriptions of annulus air cleanup system and main control room HVAC system are provided in section 8.4.

The conceptual configuration of the ECCS and CSS/RHRS is shown in Figure 5.1-1.

Legend

(dark blue line)	Emergency Core Cooing System
(green line)	Containment Spray System and Residual Heat Removal System
SIP	Safety Injection Pump
CS/RHRP	Containment Spray / Residual Heat Removal Pump
CS/RHRHx	Containment Spray / Residual Heat Removal Heat Exchanger
ACC	Accumulator
RWSP	Refueling Water Storage Pit
GT/G	Gas Turbine Generator
R/V	Reactor Vessel
RCP	Reactor Coolant Pump
S/G	Steam Generator
PRZ	Pressurizer
S	Safety Injection Signal
P	Containment Spray Signal

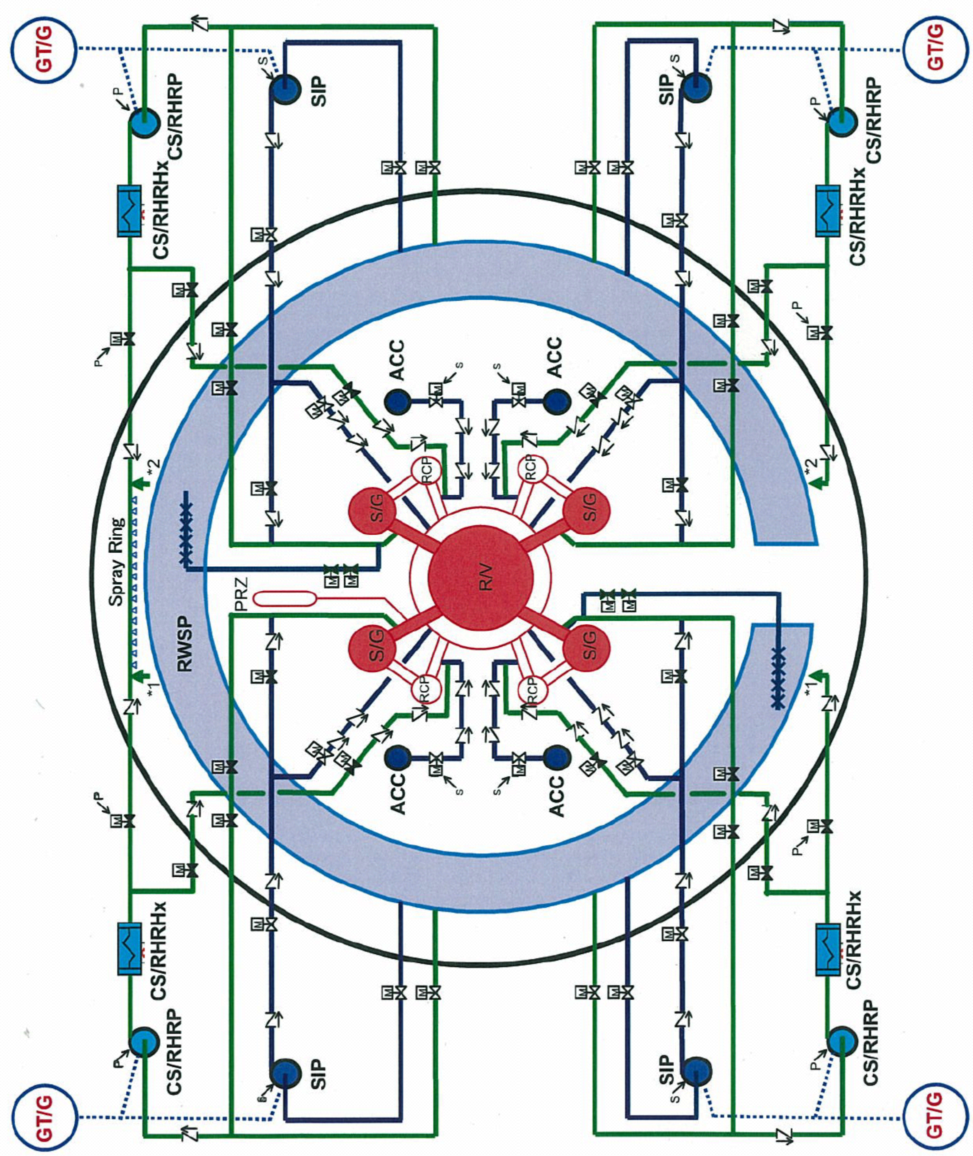


Figure 5.1-1 Conceptual Configuration of ECCS and CSS/RHRS

5.2 Containment Systems

5.2.1 Containment Vessel

The containment vessel is designed to completely enclose the reactor and reactor coolant system and to ensure that essentially no leakage of radioactive materials to the environment would result even if a major failure of the reactor coolant system were to occur.

The containment vessel consists of a prestressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat reinforced concrete foundation slab. The inside surface of the structure is lined with carbon steel.

The design pressure and temperature of the containment vessel are defined by the following postulated accidents;

- Loss- of-coolant accident (LOCA)
- Main steam line break (MSLB)

The containment vessel is designed to contain the energy and radioactive materials which result from a postulated loss-of-coolant accident, and for an 83 psia (68 psig) internal pressure to ensure a high degree of leak tightness during normal operation and under accident conditions.

A internal polar crane is supported by the containment vessel. A continuous crane girder transfers the polar crane loads to the containment vessel wall.

Hydrogen igniters are provided for protection against possible detonation following a core damage accident.

The major parameters of the containment vessel are shown in Table 5.2.1-1, and the configuration of the containment vessel is shown in Figure 5.2.1-1.

5.2.2 Containment Isolation

The lines which penetrate the containment vessel are in general provided with containment isolation valves.

Each line that is part of the reactor coolant pressure boundary or connected directly to the containment atmosphere and is not closed outside containment is provided with one containment isolation valve inside and outside containment.

Each line that is neither part of the reactor pressure boundary nor connected directly to the containment atmosphere is provided with one containment isolation valve outside containment.

The containment isolation valves are designed not fail open upon loss of actuating power after closing. In addition, the containment isolation valves that close automatically upon receiving an isolation signal are designed not to open automatically if the isolation signal is removed.

Containment isolation valves are designed to be tested for both functional and leakage.

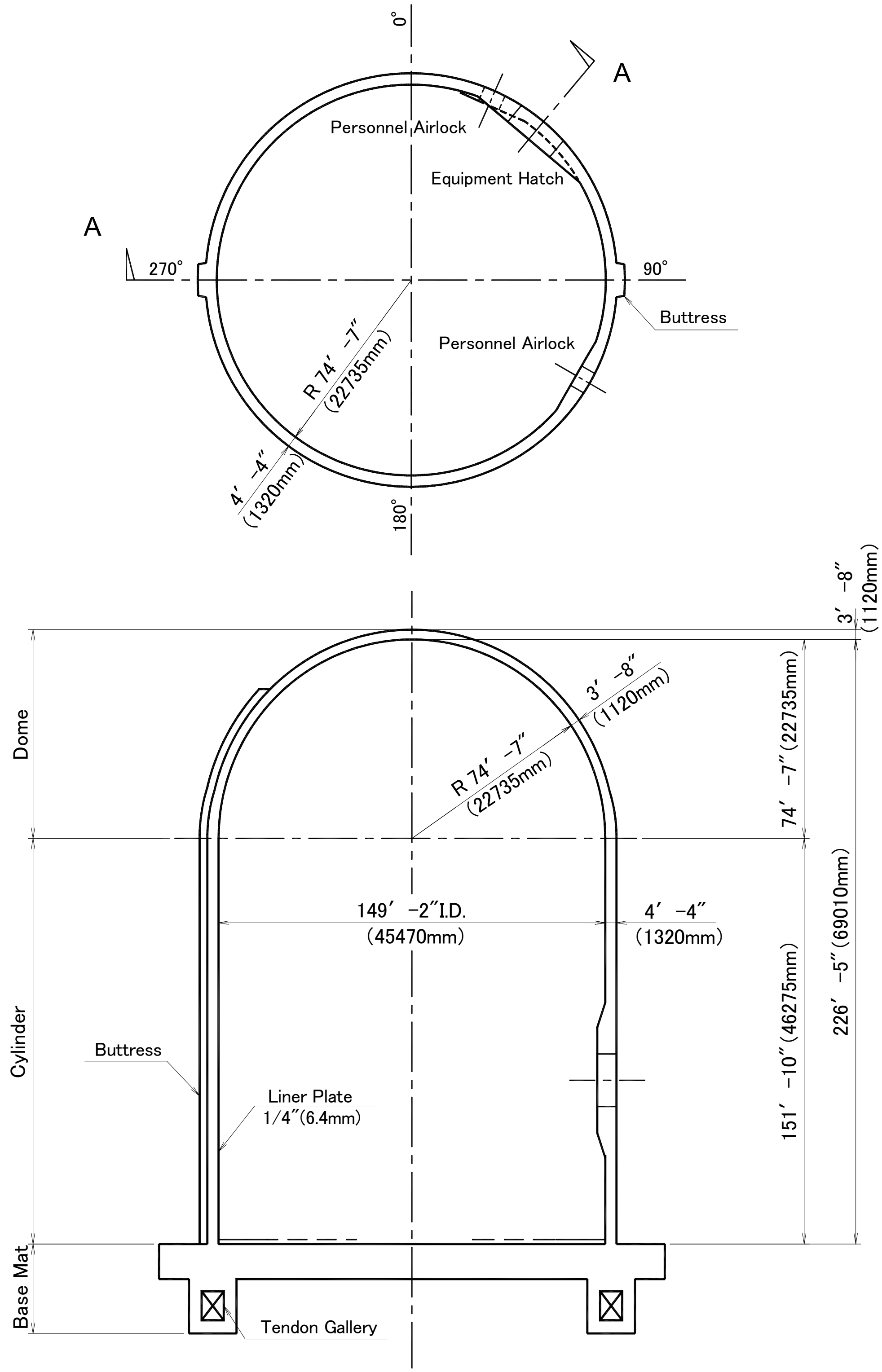
5.2.3 Annulus

The annulus is located adjacent to the containment vessel and includes all penetration areas to prevent from the direct release of containment atmosphere to the environment through the containment penetrations.

The annulus is kept a slightly negative pressure to control the release of the radioactive materials to the environment following an accident.

Table 5.2.1-1 Design Data for the Containment Vessel

Type	Prestressed Concrete Containment Vessel with Carbon Steel Liner
Number	1
Design Pressure	83 psia (68 psig)
Dimension	
Inside diameter of the cylinder	149' - 2"
Total height	226' - 5"
Thickness	
Dome	3' - 8"
Cylinder	4' - 4"
Accessories	
Equipment hatch	1
Personnel air locks	2
No. of electrical and piping penetrations	155
No. of containment isolation valves	235
Hydrogen Control System	Hydrogen igniters



A-A Vertical Section

Figure 5.2.1-1 Configuration of Containment Vessel

5.3 Emergency Core Cooling System

The emergency core cooling system (ECCS), shown in Figure 5.3-1, includes the accumulator system, high-head injection system and emergency letdown system. The ECCS injects borated water into the reactor coolant system following a postulated accident and performs the following functions;

- Following a loss-of-coolant accident (LOCA), the ECCS cools the reactor core, prevents the fuel and fuel cladding from serious damage, and limits the zirconium-water reaction of the fuel cladding to a very small amount.
- Following a main steam line break (MSLB), the ECCS provides negative reactivity to shut down the reactor.
- In the event that the normal CVCS letdown and boration capability is lost, the ECCS provides emergency letdown and boration of the RCS.

The ECCS design is based on the following;

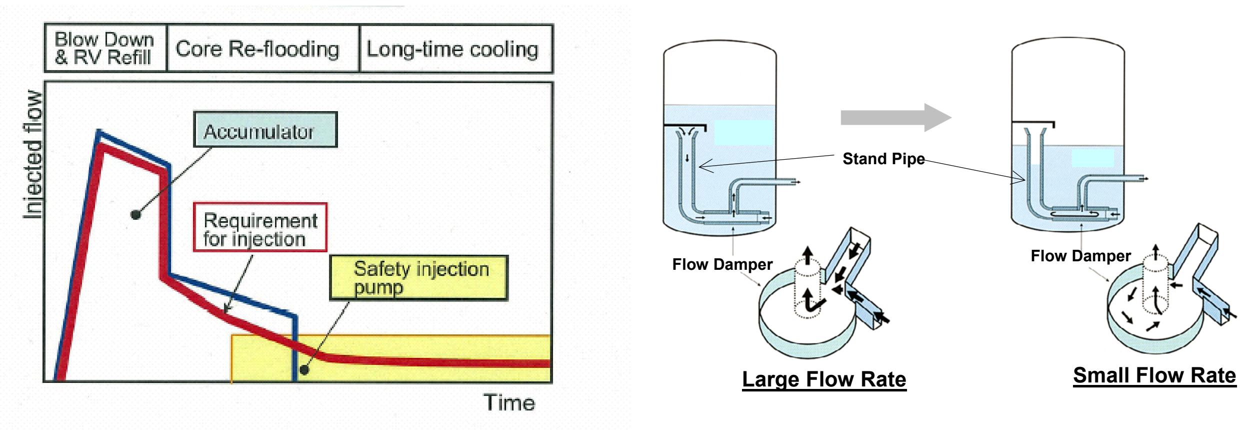
- In combination with control rod insertion, the ECCS is designed to shut down and cool the reactor during the following accidents;
  - Loss-of-coolant accidents ranging from a small break of a primary pipe to a double-ended guillotine break of a main pipe
  - Control rod ejection
  - Main steam line break
  - Steam generator tube rupture
- The ECCS is designed with sufficient redundancy (four trains) to accomplish the specified safety functions assuming a single failure of an active component in the short term following an accident with one train out of service for maintenance, or a single failure of an active component or passive component in the long term following an accident with one train out of service for maintenance.
- The gas turbine generators supply electrical power to the essential components of the ECCS, so the safety functions can be maintained during a loss of offsite power.
- The ECCS is automatically initiated by a safety injection signal.
- The ECCS design permits periodical tests and inspections to verify integrity and operability.

The accumulator system stores borated water under pressure and automatically injects it if the reactor coolant pressure decreases significantly.

The accumulator system consists of four accumulators and the associated valves and piping, one for each RCS loop. The system is connected to the cold legs of the reactor coolant piping and injects borated water when the RCS pressure falls below the accumulator operating pressure. The system is a passive. Pressurized nitrogen gas forces borated water from the tanks into the RCS.

The accumulators incorporate internal passive flow dampers which function to inject a large flow to refill the reactor vessel in the first stage of injection, and then reduce the flow as the water level in the accumulator drop. When the water level is above the top of the standpipe, water enters the flow damper through both inlets at the top of the standpipe and at the side of the flow damper and thus it injects water with a large flow rate. When the water level drops below the top of the standpipe, the water enters the flow damper only through the side inlet and thus it injects water with a relatively low flow rate.
The accumulators perform the large flow injection to refill the reactor vessel and the following small flow injection during core reflooding in association with the safety injection pumps. The combined performance of the accumulator system and the high head injection system eliminated need for a conventional low head injection system.

The safety system's performance for US-APWR is shown in below figure.



The safety system's performance for US-APWR

The high head injection system consists of four independent trains, each containing a safety injection pump and the associated valves and piping. The safety injection pumps start automatically upon receipt of the safety injection signal. One of four independent safety electrical buses is available to each safety injection pump.

The safety injection pumps are aligned to take suction from the refueling water storage pit and to deliver borated water to the safety injection nozzles on the reactor vessel. Two safety injection trains are capable of meeting the design cooling function for a large LOCA, assuming a single failure in one train and another train out of service for maintenance.

The refueling water storage pit in the containment provides a continuous borated water source for the safety injection pumps thus eliminating the conventional realignment from the refueling water storage tank to the containment sump.

The emergency letdown system consists of two emergency letdown lines from the RCS hot legs to the refueling water storage pit. In the event that the normal CVCS letdown and boration capability is not available, the feed and bleed emergency letdown and boration operation can be utilized to achieve a cold shutdown boration level in the reactor coolant prior to the safe shutdown operation. The emergency letdown directs reactor coolant to the refueling water storage pit. The safety injection pumps provide borated coolant to the RCS from the refueling water storage pit.

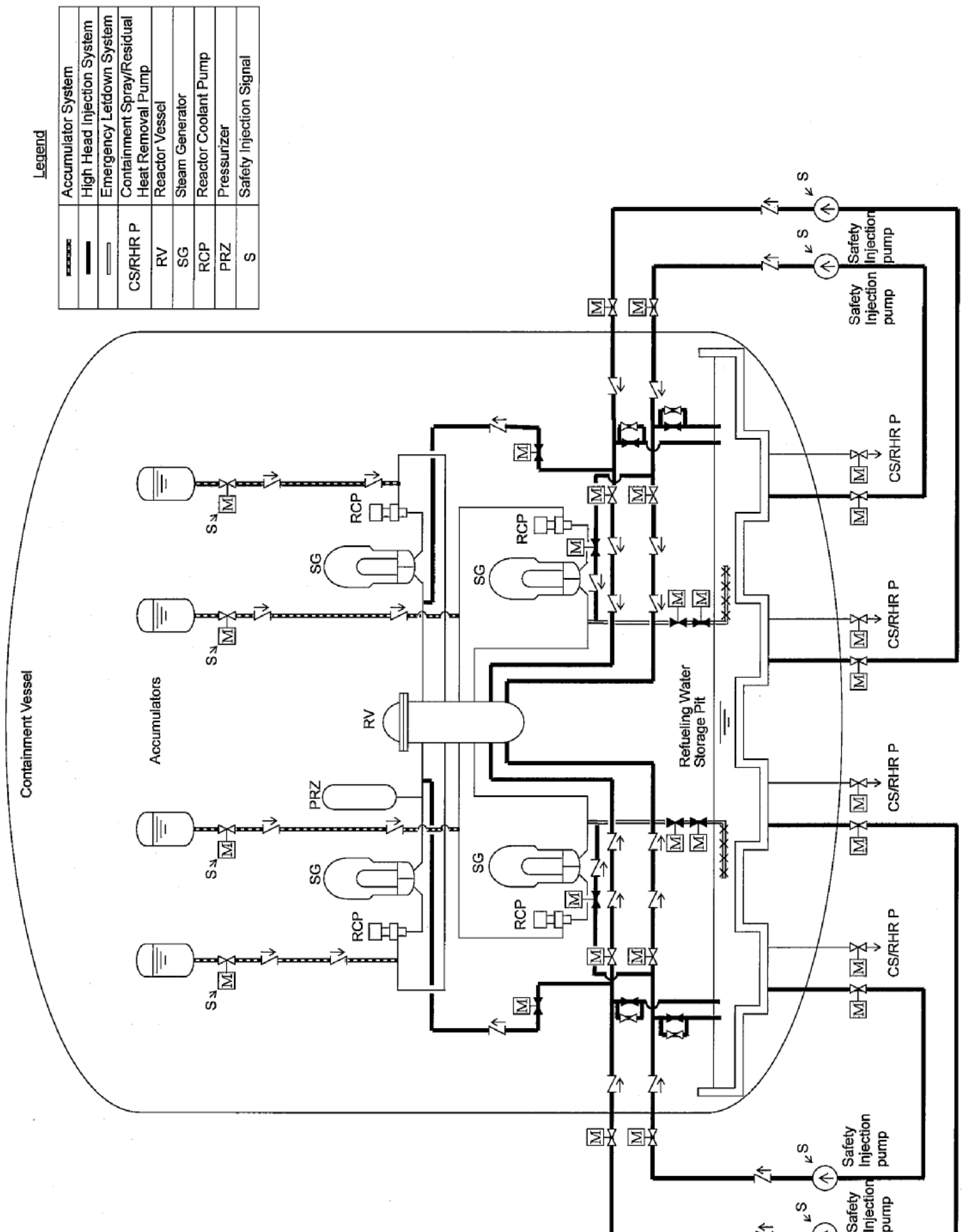


Figure 5.3-1 Emergency Core Cooling System

5.4 Containment Spray System

The containment spray system (CSS), shown in Figure 5.4-1, consists of four independent trains, each containing a containment spray/residual heat removal (CS/RHR) heat exchanger, a CS/RHR pump, spray nozzles, piping and valves. The CS/RHR heat exchangers and the CS/RHR pumps are used for both CSS and RHRS function. The CSS sprays borated water into the containment vessel in the event of a loss-of-coolant accident.

The containment spray system functions to

- Maintain the containment vessel internal peak pressure below the design pressure and reduce it to approximately atmospheric pressure in the event of a loss of coolant accident or a main steam line break.

The CSS is designed based on the following;

- The CSS is designed with sufficient redundancy (four trains) to accomplish the specified safety functions assuming a single failure of an active component in the short term following an accident with one train out of service for maintenance, or a single failure of an active component or passive component in the long term following an accident with one train out of service for maintenance.
- The gas turbine generators supply electrical power to the essential components of the CSS, so that safety functions can be maintained during a loss of offsite power.
- The CSS is automatically initiated by a containment spray signal.
- The CSS design permits periodical tests and inspections to verify integrity and operability.

The CSS includes four CS/RHR pumps and four CS/RHR heat exchangers, piping, spray nozzles and valves.

The CSS is automatically actuated on receipt of a containment spray signal. When the signal is received the CS/RHR heat exchanger outlet valves open and the CS/RHR pumps start. The CS/RHR pump motor is connected to a safety bus, so the gas turbine generator can supply electrical power in case of a loss of offsite power. The CS/RHR pumps take suction from the refueling water storage pit, and the stop valve on the inlet line is always open during reactor operation. The water in the pit is cooled by the CS/RHR heat exchangers and is delivered to the spray headers located in the top of the containment vessel.

The refueling water storage pit in the containment provides a continuous suction source for the CS/RHR pumps thus eliminating the conventional realignment from the refueling water storage pit to the containment sump.

The CSS has sufficient redundancy to perform its required safety functions following an accident assuming a single failure in one train with a second train out of service for maintenance.

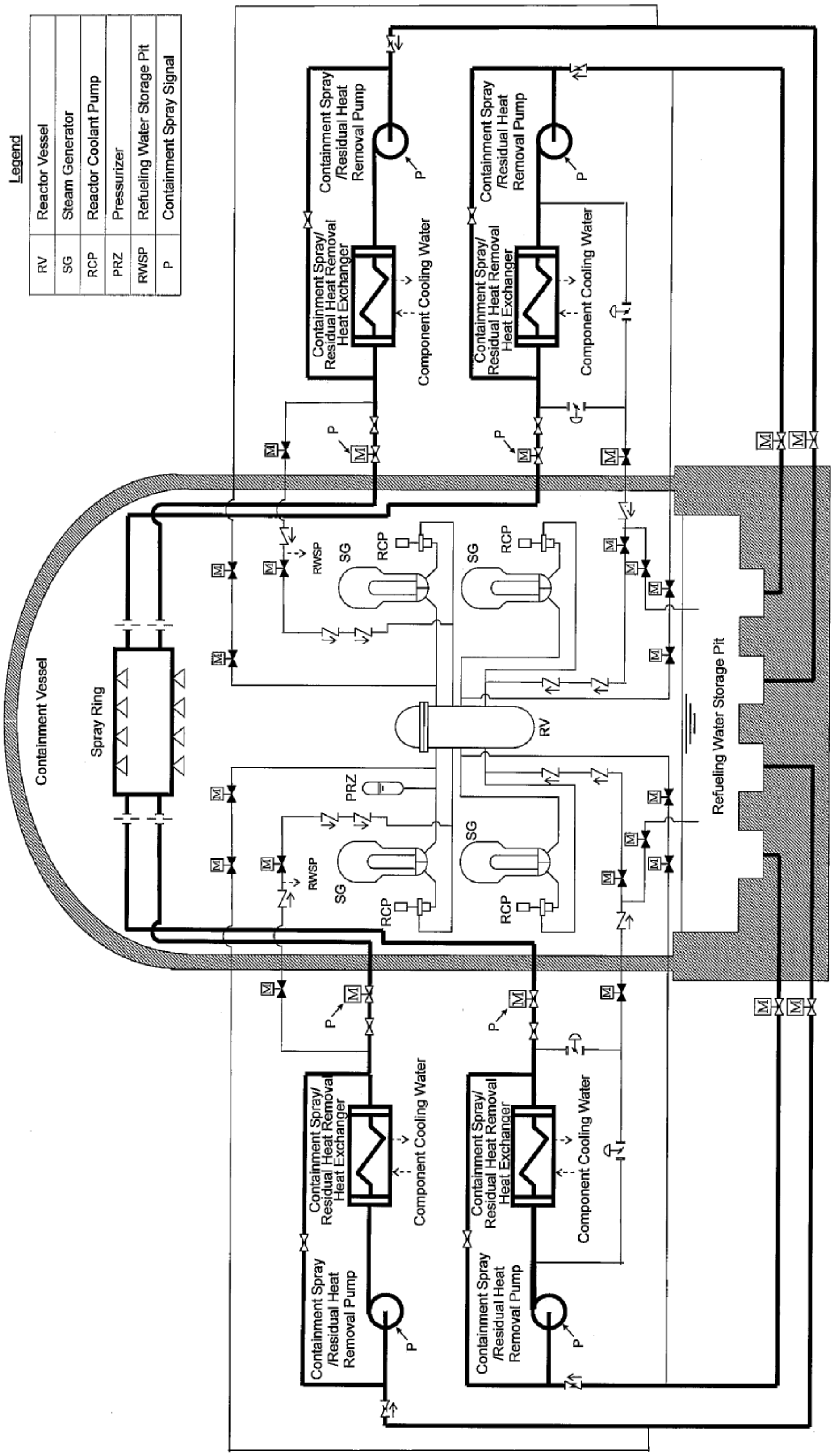


Figure 5.4-1 Containment Spray System

6.0 INSTRUMENTATION AND CONTROLS

6.1 Introduction

The instrumentation and control (I&C) systems presented in this chapter provide protection against unsafe reactor operation during steady-state and transient power operation. The primary purpose of the I&C systems is to provide automatic protection and to exercise proper control against unsafe and improper reactor operation during steady state and transient power operations and to provide initiating signals to mitigate the consequences of faulted conditions.

This chapter provides the functional performance requirements, design bases and system descriptions for those systems.

6.1.1 Identification of Safety Systems

Figure 6-1 illustrates the I&C overall architecture for the US-APWR. Safety related functions of the US-APWR are performed by several systems shown in Figure 6-1.

The design bases of the I&C systems for the US-APWR are described in Subsection 6.1.2. The overall I&C system consists of the following four categories;

- Human System Interface (HSI) System,
  including HSI portions of the Protection and Safety Monitoring System, the Plant Control and Monitoring System and the Diverse Actuation System
- Protection and Safety Monitoring System (PSMS)
- Plant Control and Monitoring System (PCMS)
- Diverse Actuation System (DAS)

A summary of all the safety functions is given in this section, while more detailed descriptions are given in Section 6.2 for reactor trip, 6.3 for engineered safety features actuation, 6.4 for systems required for safe shutdown, 6.5 for information systems important to safety and 6.6 for interlock systems important to safety.

Safety functions are those actions required to achieve the system responses assumed in the safety analyses, and those credited to achieve safe shutdown of the plant. For the US-APWR safe shutdown is defined as cold shutdown. Some safety functions are automatically initiated by the Protection and Safety Monitoring System (PSMS). For back-up purposes, these same safety functions may also be manually initiated and monitored by operators using the HSI System. The HSI System is also used to manually initiate other safety functions that do not require time critical actuation, and safety functions credited for safe shutdown. After manual initiation from the HSI System, all safety functions are executed by the PSMS. The HSI System also provides all plant information to operators, including critical parameters required for post accident conditions. The HSI System includes both Safety and Non-safety sections which are discussed in Section 6.5.

This report has grouped the main safety functions into two categories:

- Reactor Trip

- Engineered Safety Features (ESF) Actuation

6.1.1.1 Reactor Trip Function

The safety systems automatically trip the reactor and initiate ESF (if required) whenever predetermined limits are approached. The PSMS maintains surveillance on nuclear and process variables which are related to equipment mechanical limitations, such as pressure, and on variables which directly affect the heat transfer capability of the reactor, such as reactor coolant flow and temperature. When a limit is approached, the PSMS initiates the signal to open the reactor trip circuit breakers. This action removes power to the control rod drive mechanism coils permitting the rods to fall by gravity into the core. This rapid negative reactivity insertion will cause the reactor to shutdown.

6.1.1.2 Engineered Safety Features Actuation Functions

The occurrence of a limiting fault, such as a loss of coolant accident or a steam line break, requires a reactor trip plus actuation of one or more engineered safety features (ESF) in order to prevent or mitigate damage to the core and reactor coolant system components, and to ensure containment integrity. The PSMS will determine whether or not safety limits are being approached for selected plant parameters. If they are, the PSMS will combine the signals through logic functions which respond to combinations indicative of an accident situation. Once the required logic combination is generated, the PSMS will send signals to the appropriate ESF components for protective action.

6.1.2 Design Bases of I&C System

6.1.2.1 Defense in Depth and Diversity Concept

The architecture of the overall I&C system is based on the “defense in depth and diversity” concept.

The “defense in depth concept” consists of the following echelons of defense.

- Reactor Control System
- Reactor Protection System
- Engineered Safety Features Actuation System
- Operation and Monitoring System

Each echelon has different safety functions which relate to different functional design criteria.

Based on a beyond design basis common mode failure of the safety I&C systems (defined above), a diverse echelon of defense is also provided.

- Diverse Actuation System

6.1.2.2 Basic Design Requirement

The design of the I&C system meets all US codes and standards for safety systems including the following safety requirements.

a. Redundancy

The PSMS systems have appropriate redundancy to satisfy the single failure criterion during normal operation and during all planned on-line test/maintenance configurations. Moreover, physical separation and electrical isolation is provided between the redundant subsystems. Also, all equipment within the PSMS satisfies all general requirements such as environmental requirements, seismic requirement, testability, etc.

Redundancy within the PSMS is consistent with the unavailability target value required to achieve the total safety goal of the plant. The configuration of four trains with two-out-of-four voting logic is provided from sensors to trip breakers in the Reactor Protection System. The configuration of four trains with two-out-of-four voting logic is also provided in the Engineered Safety Feature Actuation System. In addition to train redundancy, the CPUs and the safety logic buses are also redundant within each train.

b. Isolation

Fault isolation devices are incorporated into data links and safety buses which connect redundant train sets, or which carry signals to or from non-safety systems. The isolation devices ensure that credible faults, such as physical damage, short circuits, open circuits, or the application of credible fault voltage do not propagate between systems. The isolation devices provide assurance that, where protection signals are used by non-safety systems, and non-safety signals are used by safety systems, credible single failures in the non-safety system will not degrade the performance of the safety system. For signals interfaced between redundant train sets, the isolation devices provide assurance that failures in one train set cannot degrade the performance of any other train set.

For most applications the I&C systems use fiber optic data communication links to provide fault isolation. Fiber optic cables provide inherent electrical fault isolation and allow required physical separation. For a few cases where electrical isolators are employed (such as, relays, transformers or photo-couplers, etc.), the isolator is qualified by testing. Physical separation is accommodated through equipment mounting and cable routing.

In addition to the above electrical and physical isolations, functional isolation between non-safety systems and safety systems is provided. The functional isolation is provided by priority logics in the safety systems, or by signal selector logic in the non-safety systems. The priority logic ensures that safety actuation signals, both automatic and manual (system level and component level), override all control signals from the non-safety systems. The signal selector logic used within the non-safety systems is discussed below.

Functional isolation is also provided for safety signals interfaced between train sets. The functional isolation is provided by two-out-of-four voting logic which ensures erroneous data from one train set does not cause adverse operation of any other train set.

In addition to electrical isolation, physical isolation and functional isolation (as described above), communication isolation is provided for all interface that use communication data links. Communication isolation ensures that all computers run asynchronously without any handshaking, interrupts or data exchange that may create operational dependencies between two computers. Communication isolation ensures that computers execute their internal logic functions on a predefined cyclical basis, regardless of the interaction, performance or failure of their communication data links.

Finally, all computer to computer interfaces employ data isolation. Data isolation ensures that computers exchange only predefined data sets that include only the information required for the predefined functional algorithms. Predefined data sets are independently maintained by both the sending computer and the receiving computer. Both data sets must match to exchange any information. Any mismatch in the data sets will result in no communication. Data isolation ensures there is no capability (and therefore no potential) to exchange any unexpected or malicious data or files (including viruses, trojans, etc.) that could change or corrupt the computers basic software, application software or memory.

c. Integrity of Software

Design principles shown below are adopted for the design of the software itself. These principles assure simplicity, enable high efficiency in design, and assure certainty of the verification and validation (V&V).

- A structured and modular architecture is applied.
- The basic Operating System (OS) software and the application software are separated.
- Early detection of failures is facilitated by self-diagnosis functions of the digital system.
- The basic OS software is implemented in a high level programming language. All functions execute with cyclical single task processing and no interrupts.
- The basic OS software performs only minimal necessary functions, such as initialization, periodic execution of required functions, error handling, etc.
- The application software is described in a graphically symbolized manner using the Problem Oriented Language (POL) so that functions can be easily understood.

The V&V program executed for safety systems conforms to all US requirements for high integrity Class 1E software.

Also, for the non-safety I&C systems, the efficiency and reliability of design, production, testing, maintenance, etc. is achieved by using the same basic OS software and application software design tools as in the safety system. This software is being used in numerous non-safety systems in Japanese PWR plants and has significant operating experience. Beyond design basis failure of this common safety and non-safety system software is discussed below.

d. EMI/RFI Compatibility

Since electromagnetic noise induced by actuation of large equipment, lightning, radio frequency emission etc., may commonly affect the I&C systems, sufficient noise/surge withstand capability is designed into the controllers, input/output devices, power supply circuits, etc. of every I&C subsystem.

Optical fiber is used for data transmission to provide electrical independence and to provide protection against electromagnetic noise and surge, etc

e. Signal Selector

To prevent adverse functional interaction between the PSMS and the Plant Control and Monitoring System (PCMS), a signal selector function is applied where process signals are transferred from a safety subsystem to a control subsystem. The signal selector, within the control system, receives all safety process trains but passes only the second highest process signal value to the control system's automation algorithms. This ensures the control subsystem can never take erroneous control actions due to a single failure of a safety system process sensor or safety signal processing equipment.

Preventing erroneous control actions ensures that a single failure, which degrades the safety system, does not also generate erroneous control actions that would result in a challenge to the safety system while it is degraded. When one train of the safety systems is out of service for testing or maintenance, the signal selector selects the second highest value from only the remaining operable trains.

f. Countermeasures against Common Mode Failure

The software applied for the PSMS has high integrity due to design simplicity and a comprehensive software quality program including independent V&V.

Based on BTP HICB-19, regardless of this high reliability, appropriate hard-wired back-up systems are provided to accommodate beyond design basis common mode failures that adversely affect all common safety and control systems. The operator can monitor critical safety functions and manually actuate safety process systems by back-up I&C systems that are diverse from the digital safety system. Per BTP HICB-19 diverse automatic back-up actuation functions are provided where the time required for manual operator action is insufficient for accident mitigation.

A best estimate analysis will demonstrate the effectiveness of these diverse back-up systems in coping with all design basis accidents with a concurrent software CMF that disables the safety and non-safety systems which use the same digital platform. The analysis assumes the CMF results in passive non-actuation of the safety functions and a fail-as-is mode of the control systems. CMFs that result in active failures (off-demand failure), such as spurious control systems actions, with a concurrent accident are considered incredible and are not analyzed.

6.1.2.3 Other System Design Features

Design features of the digital I&C systems have been developed based on significant plant experience with digital systems in Japanese NPP. The key points are as follows.

a. Unified Architecture

The following advantages are obtained through the application of the integrated digital I&C system.

- Components are compact, and reliability is enhanced through internal redundancy.

- Early detection of failures is facilitated by self-diagnostic features.
- Future upgrading and improvement of the I&C system are easily facilitated by the high flexibility of computer based technology.
- Reduction of cables is facilitated by applying plant-wide communication networks between the I&C subsystems.
- Maintainability is facilitated by a modular and standardized architecture.

In order to maximize the benefit of these advantages, a unified architecture is provided that applies the same platform to all the digital I&C systems. Also, to apply VDU (visual display unit) based operation, integrated digital technology is applied to all I&C systems plant-wide.

b. Redundant Configuration

To prevent disturbance of the plant caused by a failure of the I&C system, a redundant configuration is applied to all subsystems that may directly result in spurious plant trip or spurious system level ESF actuation. Failed components, including CPU, I/O cards and communication network, etc. are detected by self-diagnostic features which automatically switch the system to redundant stand-by components. This redundant fail-over system will continue uninterrupted control without causing any disturbance to the plant.

The AC power source for the I&C system is supplied from two different feeders, one from the inverter and the other from the back-up AVR transformer. The configuration of the power supplies within each I&C subsystem ensures no loss of function due to a single failure of the electric power source.

c. Bypass Mode Testing

The safety system may be placed in a bypass mode to allow testing and maintenance while the plant is on-line. During this bypass mode, a single failure in the safety system will not result in a spurious plant trip or spurious system level ESF actuation.

Automatic bypass management logic continuously checks for multiple bypassed conditions to ensure the minimum redundancy required by the Technical Specifications is always maintained. Indication is provided for bypassed or inoperable conditions in accordance with Regulatory Guide 1.47.

d. Operability

The main control board provides operation and monitoring capabilities necessary for normal operation, anticipated transients, and accidents. Automated supporting features are provided to reduce operator workload and potential for human errors during all operating modes.

It is easier to apply advanced control algorithms and new automation using the high capability and flexibility of the digital I&C system. Based on MHI experience, many kinds of functional improvements can be applied in comparison to conventional I&C system.

e. Self-diagnosis Function

The integrity of digital I&C components is continuously checked by self-diagnosis features. These self-diagnostic features allow early detection of failures, and allow easy and quick repair that improves system availability. Information about detected failures is gathered through networks and provided to maintenance staff in a comprehensive manner. Also, these self-diagnostic features control the redundant configuration to maintain all system functions even with failures.

Continuous self-diagnostic features allow elimination of most manual testing required for Technical Specification compliance. Manual testing and manual calibration verification will only be provided for functions with no self-diagnostics. Reliability analysis will demonstrate the need to conduct manual tests of the safety system no more frequently than once per fuel cycle. Uncertainty analysis along with continuous cross channel monitoring of redundant measurements will demonstrate the need to conduct manual calibration of safety sensors no more frequently than once per fuel cycle on a staggered channel basis.

f. Manual Testing

Manual test features are provided to allow periodic testing of all functions that are not automatically tested through self-diagnostics. This includes primarily manual initiation functions and final actuation of plant components. All manual tests may be conducted on-line without full system actuation and without plant disturbance.

Final actuation of plant components is tested individually or in small groups to allow most plant components to be tested on-line with no disturbance to plant operation. Since the reliability of the I&C equipment is significantly higher than the reliability of the plant components, the periodic test frequency is determined by the reliability of the plant components, not the reliability of the I&C equipment.

g. On-line Maintenance

Repair of the digital I&C system is achieved by easy on-line module replacement. Together with easier identification of a failure by the self-diagnostic features, the MTTR (mean time to repair) of the digital I&C system is considerably less than the MTTR of a conventional system.

h. Suitable Margin and Easier Modification

The digital I&C system has ample spare space for additional hardware and spare computing capacity for additional software. These features enable easy modifications or addition of new functions in the future.

Also, user friendly design features enable easier understanding of the system and planning of design modifications.

i. Local Distributed Allocation

Digital input/output (I/O) equipment, which provides interface to sensors and actuated plant components, is geographically distributed throughout the plant, so that the amount of cabling between the sensor or the actuator and the digital subsystem is reduced to the minimum.

I/O information is connected to upper level subsystems via fiber optic networks.

j. Standardization of Hardware and Software

Specification of the hardware modules, such as CPU and I/O cards, etc. used for each subsystem is basically the same, except for some modules which are designed for specific applications (e.g. rod position interface). This approach allows the total number of required spare parts to be minimized.

The configuration of the basic OS software, and the language and engineering tools for specification of the application software is the same in all I&C subsystems. Maintenance procedures and maintenance tools are standardized for all subsystems, so the training effort for the maintenance staff and the potential for human error during maintenance is also minimized.

6.1.2.4 Human System Interface (HSI)

The Main Control Room (MCR) is designed to perform centralized monitoring and control of the instrumentation and control systems that are necessary for use during normal operation, abnormal transients and accidents. Furthermore, the central control panels are also designed to reduce the potential for miss-operation and miss-judgment and to allow easy operation.

The HSI other than the MCR includes the Remote Shutdown Console (RSC), Local control stations such as Auxiliary Equipment Control Console, Technical Support Center (TSC) and Emergency Operations Facility (EOF).

The design features of the HSI system is described Section 6.5.

6.1.3 System Description of overall I&C System

6.1.3.1 Introduction of Mitsubishi Digital I&C System

The I&C system of the US-APWR is fully digital. It has been developed and applied in a step-by-step approach in Japanese PWR plants.

The digital I&C Systems have been developed and fully applied to non-safety grade systems in five Japanese PWR plants. These plants were constructed approximately 15 years ago. Based on excellent experience in these plants, digital safety grade I&C system and digital Human-System Interface (HSI), are being applied to the latest Japanese PWR plants.

The basic design of the latest safety and non-safety I&C System is complete and performance testing has been validated by full-scale prototype systems. The complete digital I&C system, including the safety function and the human system interface, for the latest Japanese plant, Tomari Unit-3, is under final factory acceptance testing and has an expected installation date of March 2007. This is a full scale integrated test with plant simulation.

The I&C system for the US-APWR is based on the I&C systems for Tomari Unit-3. The same basic architecture, basic data communication technology and basic HSI features are utilized.

The I&C platform will be based on the platform used for Tomari Unit-3, but it will reflect the continuing incremental evolution that is common in digital I&C technology.

a. General Description of I&C System

General specifications of the I&C system are summarized as below.

Main control board	Fully computerized Consists of safety VDU and non-safety VDU Minimal conventional switch and indicator, only for regulatory compliance (RG 1.62 and BTP-19)
Safety I&C	Fully digital Consists of Mitsubishi digital controller MELTAC-N plus Four train redundant Reactor Trip System Four train redundant ESF Actuation System Four train redundant Safety Logic System for component control
Non-safety I&C	Fully digital Consists of Mitsubishi digital controller MELTAC-N plus Duplex digital architecture for each control and process monitoring sub-system
Data communication	Fully multiplexed including class 1E signals Consists of multi-drop data bus and serial data link Uses fiber optics communication networks for noise immunity and required isolation

The overall architecture of the I&C system is shown in Figure 6-1.

6.1.3.2 Function of Each I&C System

The following sections summarize the function of each I&C system.

a. Human System Interface System

1) Operator Console

Plant information and controls (i.e. for all safety and non-safety divisions) are displayed and accessed on the non-safety VDU screens of the Operator Console. All operations from the Operator Console are available using touch screen on the non-safety VDUs. In addition, Safety VDUs on the Operator Console provide back-up access to information and controls for safety systems.

Priority logic in the safety system ensures that erroneous signals, that may result from non-safety VDU malfunctions, can be overridden by manual control signals from the safety VDU and by Class 1E automation signals within the safety systems. All operations are available from the VDUs with touch screens or other pointing device.

Some conventional hardwired switches for system-level-related operations, such as reactor trip, engineered safety features actuation, etc., are also installed on the Operator Console.

2) Large Display Panel

The Large Display Panel includes the necessary information, so that the total status of the plant can be easily accessed without requesting VDU screens on the Operator Console. Important information for normal operation and important information for emergency or accident conditions are displayed on the Large Display Panel.

Easy and reliable comprehension for all operating crew members is achieved from the information on this panel by displaying high level plant conditions.

3) Supervisor Console

The Supervisor Console is designed for use by the main control room supervisor. The Supervisor Console has the same operational capability as the Operator Console. However, normally the Supervisor Console has monitoring only capability. All operations are available from the VDUs with touch screens or other pointing devices.

4) Shift Technical Advisor Console

The Shift Technical Advisor Console is for the safety engineer. It is located in the Main Control Room (MCR). The Shift Technical Advisor Console has the same operational capability as the Operator Console. However, normally the Shift Technical Advisor Console has monitoring capability only.

All operations are available from VDUs with touch screens or other pointing devices.

5) Diverse HSI Panel

The Diverse Control Panel consists of some hardwired back-up switches and indicators. The Diverse Control Panel is used in the case of a common mode failure of the safety and non-safety digital I&C systems.

6) Process Recording Computer

The Process Recording Computer provides historical data storage and retrieval (HDSR) functions. The system records process trends and all binary transitions such as alarms, equipment state changes, etc. Historical data from the Process Recording Computer is accessible on all Consoles (described above). There is also an interface to the plant information technology network via the Unit Management Computer that makes this information available to personnel throughout the plant management organization. This is a one-way data communication interface with appropriate security to prevent unauthorized access or malicious intrusion.

7) Alarm Logic Processor

The Alarm Logic Processor receives the alarm signals from the I&C equipment. This processor executes alarm filtering logic, classifies these alarms according to their priority and their acknowledgement status, and transmits alarm status information to the Alarm VDU Processor and Large Display Panel Processor.

8) Unit Management Computer

The Unit Management Computer performs plant performance calculations, including core monitoring and fuel management applications. It also compiles data to create daily operations reports. Calculation results and reports are accessible on all Consoles (described above). There is also a one-way interface to the plant information technology network that makes this information available to personnel throughout the plant management organization.

9) Operation VDU Processor

The Operation VDU Processor manages information and graphic displays for the non-safety Operation VDUs located on the Operator Console, STA Console and Supervisor Console. It also receives operator commands such as screen navigation and soft control from the Operation VDUs.

10) Alarm VDU Processor

The Alarm VDU Processor manages the displays for the Alarm VDUs located on the Operator, STA and Supervisor Console. It also receives operator commands such as screen navigation and alarm acknowledgement from the Alarm VDUs.

11) Operation Procedures VDU Processor

The Operation Procedures VDU Processor manages the displays for the Operation Procedures VDU located on the Operator, STA and Supervisor Console. It also receives operator commands such as procedure navigation, from the Operation Procedure VDU.

12) Large Display Processor

The Large Display Panel Processor manages the displays on the Large Display Panel.

13) Safety VDU Processors

The Safety VDU Processors manage the displays on the Safety VDUs located on the Operator Console and the Remote Shutdown Console. They also receive operator commands such as screen navigation and soft control from the Safety VDUs.

14) Remote Shutdown Console

The Remote Shutdown Console is used for achieving and maintaining safe shutdown conditions in the event that the MCR is not available due to any conditions, including fire which results in catastrophic damage to I&C equipment located in the MCR. For the US-APWR safe shutdown is defined as Cold Shutdown.

The Remote Shutdown Console includes non-safety Remote Shutdown VDUs, which provide monitoring and control of process equipment in all safety and non-safety divisions. The Remote Shutdown Console also provides Safety VDUs as a back-up which provide control for only safety systems.

15) Remote Shutdown Console VDU Processor

The Remote Shutdown Console VDU Processor manages information and graphic displays for the non-safety Remote Shutdown VDUs located on the Remote Shutdown Console. It also receives operator commands such as screen navigation and soft control from the non-safety Remote Shutdown VDUs.

b. Technical Support Center (TSC), Emergency Operations Facility (EOF)

1) TSC Computer

The TSC Computer provides plant data displays to assist in the analysis and diagnosis of plant conditions, etc.

2) EOF Computer

The EOF Computer provides plant data displays to assist in the diagnosis of plant conditions and to evaluate the potential or actual release of radioactive materials to the environment, etc.

c. Protection and Safety Monitoring System

1) Reactor Protection System

The Reactor Protection System has a configuration of four redundant train sets, with each train set located in a separate I&C equipment room. Each train set receives process signals, including NIS (nuclear instrumentation system) and safety RMS (plant radiation monitoring system), from safety-related field sensors, and performs bi-stable calculations for reactor trip and engineered safety features actuation.

Each train set performs two-out-of-four voting logic for like sensor coincidence to actuate trip signals to the reactor trip switchgears and actuate ESF signals to the ESF Actuation Systems.

This is a microprocessor based digital system that achieves high reliability through segmentation of primary and back-up trip/actuation functions, redundancy within each train set and segment, failed equipment bypass functions, and microprocessor self-diagnostics, including data communications.

The system also includes features to allow manual periodic testing of functions that are not automatically tested by the self-diagnostics, such as actuation of reactor trip breakers. Manual periodic tests can be conducted with the plant on-line and without jeopardy of spurious trips due to single failures during testing.

2) ESF Actuation System

The ESF Actuation System has four redundant train sets, with each train set located in a separate I&C equipment room.

Each train set receives the output of the ESF actuation signals from the Reactor Protection System and manual system level actuation signals from redundant hardwired switches on the Operator Console. Each ESF Actuation System train set performs two-out-of-four voting logic for like system level coincidence or like manual actuation signal coincidence to actuate system level ESF actuation signals to the Safety Logic System for its respective train. The ESF Actuation Systems also provides automatic load sequencing for the safety Gas-Turbine Generators to accommodate the Loss of Offsite Power (LOOP) accident. Safety plant components are manually loaded on the Alternate Gas-Turbine Generator from the Safety Logic System for Station Blackout conditions.

This is a microprocessor based system that achieves high reliability through redundancy within each train set and microprocessor self-diagnostics, including data communications. The system also includes features to allow manual periodic testing of functions that are not automatically tested by the self-diagnostics, such as manual system level actuation inputs. Manual periodic tests can be conducted with the plant on-line and without jeopardy of spurious system level actuation due to single failures during testing.

3) Safety Logic System

The Safety Logic System receives automatic ESF system level actuation demand signals and LOOP load sequencing signals for the safety components from each redundancy within one train set of the ESF Actuation System. The Safety Logic System also receives manual component level control signals from the Operator Console (Safety VDUs and non-safety Operation VDUs), and manual components level control signals from the hardwired back-up switches on the Diverse Control Panel. This system performs the component-level control logic for safety actuators (e.g. motor-operated valves, solenoid operated valves, switchgear, etc.)

This system has a configuration of four redundant train sets. The controllers for each train set are located in separate I&C equipment rooms. To minimize field cabling, the I/O for each train set is remotely distributed throughout the plant in close proximity to safety actuators. The system has conventional I/O portions and these I/O portions with priority logic to accommodate signals from the Diverse Actuation System (which is discussed below).

This is a microprocessor based system that achieves high reliability through redundancy within each train set and microprocessor self-diagnostics, including data communications. The system also includes features to allow periodic testing of functions that are not automatically tested by the self-diagnostics, such as final actuation of safety components. Manual periodic tests can be conducted with the plant on-line and without jeopardy of spurious system level actuation due to single failures during testing.

d. Non-Safety System (Control and Monitoring)

1) Reactor Control System

The Reactor Control System receives field sensor signals, including non-safety ICTS (in-core temperature system). This system also receives status signals from the component and manual operation signals from the Operator Console to control and monitor the NSSS process components. This system provides component-level actuation logics and controls continuous control components, such as modulating air operated valves, and discrete state components such as motor-operated valves, solenoid-operated valves, pumps, etc.

This is a microprocessor based system that achieves high reliability through segmentation of process system groups (e.g. pressurizer pressure control, feedwater control, rod control, etc.), redundancy within each segment, and microprocessor self-diagnostics, including data communications.

2) Radiation Monitoring System

The Radiation Monitoring System is a microprocessor based system that monitors plant process radio-activity and area radiation level.

3) Rod Position Indication System

The Rod Position Indication System is a microprocessor based system that monitors control rod position. It detects dropped rods and misalignment of control rods. The system consist of processing equipment located in the I&C equipment room and remote I/O located inside the containment vessel.

4) CRDM Control System

The CRDM Control System is a microprocessor based system that receives control rod direction and speed demand signals from the Reactor Control System and manual operation signals from the Operator Console. This system outputs signals to control the electro-magnetic coil sequencing within the CRDMs.

5) In-Core Neutron Instrumentation System

The In-core Neutron Instrumentation System is a microprocessor based system that provides remote data acquisition for in-core detector signal monitoring.

The In-core Neutron Instrumentation is top-mounted. In-core detectors are inserted into the core through detector guide thimbles which lead to the fuel assemblies and cover the effective axial fuel length. The In-core detectors are horizontally distributed over the entire core at approximately 40 locations. The In-core detectors provide signals for the measurement of core power distribution. The In-core detector system consists of proven devices with significant experience in PWRs.

6) Diverse Actuation System

For coping with common mode failures (CMF) in the software of the digital safety I&C systems, the Diverse Actuation System (DAS) provides back-up actuation of the safety and non-safety components required to mitigate anticipated operational occurrences and accidents.

The DAS consists of hardwired or digital components which are diverse from the digital safety I&C systems, so that a postulated CMF in the digital safety system will not impair the DAS function.

The DAS is classified as a non-safety system. Interfaces to safety process inputs and the Safety Logic System outputs are isolated within the safety systems.

The DAS initiates selected safety functions manually based on the availability of time for manual operator action. Where the time is insufficient for the manual operator action, automatic actuation of DAS functions is provided.

7) Turbine Protection System

The Turbine Protection System receives signals regarding the turbine-generator and provides appropriate trip actions when it detects undesirable operating conditions of the turbine-generator.

This is a microprocessor based system that achieves high reliability through redundancy within the system and microprocessor self-diagnostics.

8) Turbine EHG Control System

The Turbine EH Governor System consists of redundant microprocessors and several hardwired logic parts (servo controller etc.). The cabinet has a speed control unit, a load control unit, an over-speed protection unit and an automatic turbine control unit. This system is used, either for control or for supervisory purposes.

This is a microprocessor based system that achieves high reliability through redundancy and microprocessor self-diagnostics.

9) Turbine Generator Control System

The Turbine Generator Control System receives process signals from the field. This system also receives manual operation signals from the Operator Console to control and monitor the T/G system. The Turbine Generator Control System outputs signals to control modulating control valves, and discrete state components such as motor-operated valves, solenoid-operated valves, pumps, etc.

This is a microprocessor based system that achieves high reliability through segmentation of process systems groups, redundancy within each segment, and microprocessor self-diagnostics, including data communications.

10) Turbine Supervisory Instrument System

The Turbine Supervisory Instrument system monitors important parameters of the turbine such as vibration, rotor position, etc.

This is a microprocessor based system that achieves high reliability through microprocessor self-diagnostics.

11) Electrical System Logic System

The Electrical System Logic System controls and monitors the electrical system and components, etc.

This is a microprocessor based system that achieves high reliability through redundancy within the system and microprocessor self-diagnostics.

12) Generator Transformer Protection System

The Generator Transformer Protection System provides a generator trip in case of receiving a turbine trip signal. This system also controls related components (breaker) in case of undesirable operating conditions of the generator and transformer.

13) AVR/ALR System

The AVR (Auto Voltage Regulator) /ALR (Automatic Load Regulator) System provides regulation of generator voltage, etc.

14) Auxiliary Equipment Control Console

The Auxiliary Equipment Control Console provides the operator interface for auxiliary equipment that is not controlled from the MCR or may be normally controlled from MCR, but also requires occasional local control (e.g. waste disposal, water treatment , etc.).

15) Auxiliary Equipment Control System

The Auxiliary Equipment Control System controls and monitors auxiliary systems (e.g. waste disposal system, water treatment, etc.).

This is a microprocessor based system that achieves high reliability through redundancy within the system and microprocessor self-diagnostics.

6.2 Reactor Trip System

6.2.1 Description

The reactor trip system automatically prevents operation of the reactor in an unsafe region by shutting down the reactor whenever the limits of the safe region are approached. The safe operating region is defined by several considerations such as mechanical/hydraulic limitations on equipment, and heat transfer phenomena. Therefore the reactor trip system maintains surveillance of process variables which are directly related to equipment mechanical limitations, such as pressure, pressurizer water level (to prevent water discharge through safety valves) and also on variables which directly affect the heat transfer capability of the reactor (e.g. flow and reactor coolant temperatures). Still other parameters utilized in the reactor trip system are calculated from various process variables. In any event, whenever a direct process or calculated variable exceeds a setpoint the reactor will be shutdown in order to protect against either gross damage to fuel clad or loss of system integrity which could lead to release of radioactive fission products into the containment.

The equipment involved in reactor trip system is listed below and is shown in figure 6-1. Section 6.1.3 provides a description of the equipment.

The following systems make up the reactor trip system:

- Sensors and manual inputs
- Reactor Protection System
- Reactor Trip Switchgears
- Safety VDU Processors and Safety VDU's

Normally, four redundant measurements using four separate sensors are made for each variable use for reactor trip. Selected analog measurements are converted to digital form by analog-to-digital converters within the Reactor Protection System (RPS). Signal conditioning may be applied to selected inputs following the conversion to digital form. Following necessary calculations and processing, the measurements are compared against the applicable setpoint for that variable. A partial trip signal for a given parameter is generated if one train's measurement exceeds its limit. Processing on all variables for reactor trip is duplicated in each of the four redundant segments of the RPS called train sets. Each train set sends its train's partial trip status to each of the other three train sets over isolated serial data links. Each train set will generate a reactor trip signal if two or more of redundant trains of a single variable are in the partial trip state.

Each train set of the RPS consists of two electrically separated digital subsystems to achieve defense-in-depth through functional diversity. Functional diversity provides two separate methods of detecting the same abnormal plant condition. Each functionally diverse digital subsystem can initiate a reactor trip. Also, each subsystem consists of a duplex architecture using dual CPUs, to enhance reliability.

The reactor trip signal from each of the four RPS is sent to a corresponding reactor trip actuation train. Each of the 4 reactor trip actuation trains consists of two reactor trip breakers. The reactor is tripped when two or more actuation trains receive a reactor trip signal. This automatic trip demand initiates the following two actions: 1) it de-energizes the under-voltage trip attachments on the reactor trip breakers, and 2) it energizes the shunt trip devices on the reactor trip breakers. Either action causes the breakers to trip. Opening of the appropriate trip breakers removes power to the rod drive mechanism coils, allowing the rods to fall into the core. This rapid negative reactivity insertion shuts down the reactor.

The Single Failure criterion is met even when one train set and one trip actuation train are bypassed. Therefore, bypass of a train set used to generate reactor trip signals and bypass of a reactor trip actuation train are both permitted concurrently and for an unrestricted time period. Technical Specifications establish time limits for bypassing two train sets or two actuation trains. The reactor is automatically tripped if three or four trains are attempted to be bypassed.

6.2.2 Functional Description

A list of the reactor trip signals and resulting functions are shown in Table 6-1.

6.3 Engineered Safety Features Actuation System

In addition, to the requirements for a reactor trip for anticipated abnormal transients, adequate instrumentation and controls are provided to sense accident situations and initiate the operation of necessary engineered safety features (ESF). The occurrence of a limiting fault, such as a loss of coolant accident (LOCA) or a steam line break, requires a reactor trip plus actuation of one or more of the ESF in order to prevent or mitigate damage to the core and reactor coolant system (RCS) components, and ensure containment Integrity.

In order to accomplish these design objectives, the engineered safety features actuation system (ESFAS) has proper and timely initiating signals which are supplied by the sensors, transmitters, and logic components making up the various instrumentation trains of the ESFAS.

6.3.1 Description

The ESFAS uses selected plant parameters, determines whether or not predetermined safety limits are being exceeded and, if they are, combines the signals into logic matrices sensitive to combinations indicative of primary or secondary system boundary ruptures (ANSI Condition III or IV faults). Once the required logic combination is completed, the system sends actuation signals to the appropriate ESF components.

The ESFAS is a functionally described in this section. The equipment involved in the ESFAS is listed below and is shown in Figure 6-1. Section 6.1.3 provides a description of the equipment. The following systems make up the ESFAS:

- Sensors and manual inputs
- Reactor Protection System
- Engineered Safety Features Actuation System
- Safety Logic System
- Safety VDU Processors and Safety VDU’s

Four sensors normally monitor each variable which is used for an engineered safety feature (ESF) actuation. (These sensors may be monitoring the same variable for a reactor trip function as well.) Analog measurements are converted to digital form by analog-to-digital converters within each of the four trains of the RPS. Following required signal conditioning or processing, the measurements are compared against the setpoints for the ESF to be generated. When the measurement exceeds the setpoint, the output of comparison results in a train partial actuation condition. Each RPS train set sends its train’s partial actuation status to each of the other three RPS train sets over isolated serial data links. Each RPS train set will generate a system level ESF actuation signal if two or more redundant trains of single variable are in the partial actuation state.

The system level ESF actuation signal from each of the four RPS train sets is transmitted over isolated data links to all four ESFAS trains. Two-out-of-four coincidence voting logic is performed twice within each division through redundant ESFAS subsystems. Each subsystem generates an actuation signal if the required coincidence of ESFAS actuation signals exists at its input. Within each redundant ESFAS subsystem, the signals are combined through ESF logic sensitive to accident situations to generate a system-level ESF signal. System-level manual actuation signals are also processed by the logic in each division.

The ESFAS provides all system level ESF actuation logics including the automatic load sequence for the safety Gas-Turbine Generators. Whether automatically or manually generated, system level ESF actuation signals are transmitted to the Safety Logic System.

Within the Safety Logic System, the system-level ESF actuation signals are then broken down to individual actuation signals to actuate each component associated with a system-level ESF. For example, a single safety injection signal must start pumps, align valves, start diesel generators, etc. The interposing logic within each safety logic cabinet accomplishes this function and also performs necessary interlocking to ensure that components are properly aligned for safety. Component-level manual actions from the Operational VDUs and Safety VDUs are also processed in the interposing logic.

The power interface also transforms the low level signals to voltage and currents commensurate with the actuation devices (such as, motor starters, switchgear, etc) which they must operate. The actuation devices, in turn, control motive power to the final ESF component. The four trains of safety logic cabinets thus interface the PSMS to the four trains of the plant process equipment.

6.3.2 Functional Description

A list of the ESF actuation signals and resulting functions are shown in Table 6-2.

6.4 Systems Required for Safe Shutdown

6.4.1 Safe Shutdown

The systems necessary for safe shutdown perform two basic functions. First, they provide the necessary reactivity control to maintain the core in a sub-critical condition. Boration capability is provided to compensate for xenon decay and to maintain the required core shutdown margin. Second, these systems must provide residual heat removal capability to maintain adequate core cooling.

6.4.2 Safe Shutdown Using Safety-related System

For the US-APWR safe shutdown is defined as cold shutdown. The Reactor Protection and the Engineered Safety Features Actuation systems are designed to mitigate accident conditions and achieve immediate stable hot shutdown conditions for the plant.

Manual controls through the Safety VDUs allow operators to maintain longer term hot shutdown conditions and transition to and maintain cold shutdown conditions for the plant. All manual and automatic operation of plant safety systems is via the Safety Logic System. Non-safety systems are not required for safe shutdown of the plant.

6.4.3 Safe Shutdown from Outside the Main Control Room

The Remote Shutdown Console, located outside the Main Control Room fire zone, is installed so that safe shutdown can be achieved in the case that the operators can not stay within the Main Control Room.

In order to achieve and maintain the reactor in the cold shutdown condition (safe shutdown state), it is necessary to remove excess heat to control the temperature, pressure and volume of the reactor coolant, and to supply boric acid, etc. Therefore, the operating controls, of those plant systems necessary for the above mentioned operations, can be operated from the Remote Shutdown Console. The Remote Shutdown Console provides the equivalent functions of the Operation VDUs and the Safety VDUs in the Main Control Room.

These controls are switched over from the Main Control Room to the Remote Shutdown Console by MCR/RSC Transfer Systems. These Transfer Systems are installed in the four separate Safety I&C equipment rooms (one for each PSMS train to ensure fire protection and separation).

Redundant Transfer Switches (each with four separated contacts) to control each of the four Transfer Systems are located just outside of the Main Control Room or near the Remote Shutdown Room. When the transfer actions from the Main Control Room to Remote Shutdown Console are initiated from these switches, the selecting signals for the Remote Shutdown Console are logically latched. Activating these Transfer Switches blocks HSI signals from the MCR and enables HSI signals at the RSC for all PSMS trains and the PCMS. Any subsequent damage to these Transfer Switches or the MCR HSI devises, caused by the fire in the Main Control Room, does not affect the functions of the Remote Shutdown Console. Transfer from the RSC back to the MCR is activated separately for each of the four Transfer Systems from each of the Safety I&C equipment rooms.

Access to the Remote Shutdown Console, the MCR/RSC Transfer Systems and the Transfer Switches is administratively controlled through closed areas with key access.

6.5 Information Systems Important to Safety

This section describes those instrumentation and control (I&C) systems that provide information to the plant operators for: (1) assessing plant conditions and safety system performance, and making decisions related to plant responses to abnormal events; and (2) preplanned manual operator actions related to accident mitigation. The information systems important to safety also provide the necessary information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences.

6.5.1 Post-Accident Monitoring (PAM)

The purpose of displaying post-accident monitoring (PAM) parameters is to assist main control room personnel in evaluating the safety status of the plant. PAM parameters are direct measurements or derived variables representative of the safety status of the plant. The primary function of the PAM parameters is to aid the operator in the rapid detection of abnormal operating conditions. As an operator aid, the PAM variables represent a minimum set of plant parameters from which the plant safety status can be assessed.

The PAM parameters are normally displayed continuously on the dedicated Class 1E Safety VDUs on the operator console in the main control room. There is one Safety VDU for each train. The parameters are selected based on R.G. 1.97 and at least two channels of each parameter are available.

The safety VDUs for each train are isolated from each other. The Safety VDUs are also isolated from each RPS train by suitable isolation devices to prevent any influence from a safety VDU failure on the RPS.

The PAM parameters are not visible on the Safety VDU when the Safety VDU is being used for back-up control functions. However, a summary of plant safety status is always continuously displayed on the Large Display Panel and detail information of the PAM parameters can be displayed on the Operation VDUs.

6.5.2 Bypassed or Inoperable Status Indication (BISI)

If a safety function of the protection system is inoperable, this is continuously indicated on the Large Display Panel. As a minimum BISI is provided for the following systems:

- RPS and ESFAS
- Interlocks for isolation of low-pressure systems from the reactor coolant system
- ECCS accumulator isolation valves
- Controls for changeover of RHR from injection to recirculation mode
- Operability of ESF process systems

All BISI information is provided in the MCR as fixed indication. The indications are provided for each train.

The most critical BISI information is displayed on the Large Display Panel (LDP) in the main control room as alarm information. The alarm information on the LDP is designed to be specially-dedicated and continuously visible. The LDP system is not Class 1E. Isolation for inputs from the data-network interface ensures independence and separation of safety systems.

6.5.3 Plant Annunciator (alarm) System

The primary purpose of the alarm system is "to alert operators that the plant is in an abnormal status." Alarms are used not only to draw operator's attention, but also to identify the extent (such as where and what degree) of the abnormal status. The main purposes of alarms can then be summarized as following.

- Alert operators that the plant is in abnormal status.
- Provide operators with information relating to the abnormal status (where and what degree)
- Help operators in making judgments and taking countermeasures

The computers and datalinks of the alarm system are redundant. The data links to the safety cabinets (RPS, ESFAS, etc.) are physically and functionally isolated to not influence the safety system in case of failure of the alarm system.

The alarm system is also designed taking into consideration functional and ergonomic aspects, thereby ensuring appropriate fulfillment of operator roles at the time of an alarm.
The main features of the alarm system are as follows;

- Adequate display to acknowledge and recognize alarm information
- Application of alarm prioritization to avoid alarm avalanche

- Request function from alarm display to relevant system display and alarm response procedures.

These functions help operators to identify and diagnose transients.

6.5.4 Safety Parameter Display System (SPDS), Information Systems associated with the Emergency Operations Facilities (EOF) and Nuclear Data Link (NDL).

a. Safety Parameter Display System (SPDS)

The safety parameter display system (SPDS) provides a display of plant parameters from which the safety status of operation may be assessed in the main control room, TSC, and EOF. The primary function of the SPDS is to help operating personnel in the main control room make quick assessments of plant safety status. Duplication of the SPDS displays in the TSC and EOF improves the exchange of information between these facilities and the control room and assists corporate and plant management in the decision-making process. The SPDS is operated during normal operations and during all classes of emergencies. The SPDS has the flexibility to allow future modifications to be incorporated, such as the capability to handle operator interaction and diagnostic analysis.

The functions and design of SPDS in the main control room are realized as a part of the overall HSI design

b. Technical Support Center

The onsite technical support center (TSC) provides the following functions:

- Provide plant management and technical support to plant operations personnel during emergency conditions.
- Relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations.
- Prevent congestion in the control room.
- Perform EOF functions for the Alert Emergency class and for the Site Area Emergency class and General Emergency class until the EOF is functional.

The TSC has facilities to support plant management and technical personnel who are assigned there during an emergency. The TSC is the primary onsite communications center for the plant during an emergency. The facility consists of a plant data display system by VDUs and Large Display Panel (LDP), data communication system, tele-communication system of telephones and facsimiles by multiple methods of transmission including private and public lines, satellite communications and ample working areas for all personnel.

The TSC is close to the Main Control Room, located within the same building (Reactor Building). The walking time from the TSC to the control room will not exceed 2 minutes. Working space, without crowding, for the personnel assigned to the TSC at the maximum level of occupancy is approximately 75 sq ft/person. The TSC working space is sized for a minimum of 25 persons, including 20 persons designated by the licensee and five NRC personnel.

The TSC need not meet seismic Category I criteria or be qualified as an engineered safety feature (ESF). The well-engineered structure of the Auxiliary Building provides adequate capability to withstand earthquakes.

The TSC ventilation system functions in a manner comparable to the control room ventilation system. The TSC ventilation system need not be seismic Category I qualified, redundant, instrumented in the control room, or automatically activated to fulfill its role. A TSC ventilation system is provided that includes high-efficiency particulate air (HEPA) and charcoal filters.

The HSI display design is basically the same as that of the MCR.

c. Emergency Operations Facilities (EOF) and Nuclear Data Link (NDL).

The emergency operations facility (EOF) is a nearsite support facility for the management of overall licensee emergency response (including coordination with Federal, State, and local officials), coordination of radiological and environmental assessments, and determination of recommended public protective actions.

The EOF has appropriate technical data displays and plant records to assist in the diagnosis of plant conditions and to evaluate the potential or actual release of radioactive materials to the environment. A senior licensee official in the EOF shall organize and manage licensee offsite resources to support the TSC and the control room operators.

The nuclear data link (NDL), a data transmission system, is designed to send a set of variables from the plant to the NRC Operations Center. This data may be used for analyses by the NRC headquarters technical support groups and NRC Executive Team. The NDL transmits information that will aid NRC in its role of providing advice and support to the nuclear power plant licensee, State and local authorities, and other Federal officials.

The SPDS function of the HSI system of the the plant provides data communication functions to the EOF and the NDL. It also provides adequate fire-wall function to prevent cyber invasions form outside of the plant.

6.6 Interlock Systems Important to Safety

This section describes interlock systems which operate to reduce the probability of occurrence of specific events or to verify the state of a safety system. These include interlocks to prevent overpressurization of low-pressure systems and interlocks to ensure availability of engineered safety features.

The typical examples of the interlock systems important to safety are as follows;

- Interlocks for Residual Heat Removal Heat Exchanger Inlet Isolation Valve
- Component Cooling Water Isolation for Non-safety Component
- Interlocks for Accumulator Isolation Valve

The Safety Logic System establishes the interlock systems important to safety for the plant. Non-safety systems are not required for the interlock systems important to safety for the plant.

The I&C functions necessary for the interlock systems important to safety are available through the Safety Logic System. This system automatically actuates the protective functions provided be the safety systems. Manual actuation of the associated safety systems is also provided.

6.7 Control Systems not Required for Safety

The general design objectives of the plant control systems are:

- To establish and maintain power equilibrium between primary and secondary systems during steady state unit operation.
- To constrain operational transients to preclude unit trip and re-establish steady state unit operation.
- To provide the reactor operator with monitoring Instrumentation that indicates all required input and output control parameters of the systems and provides the operator with the capability of assuming manual control of the systems.

The plant control systems are designed to assure high reliability during any anticipated operational occurrence. Equipment used in these systems is designed and constructed with a high level of reliability.

In some cases, it is advantageous to employ control signals derived from individual protection channels through isolation data links contained in the protection and safety monitoring system channel. As such, a failure in the control circuitry does not adversely affect the protection channel. Test results have shown that a short circuit or the application of worst case credible AC/DC voltage on the isolated output portion of the circuit (in transverse or common mode) will not affect the input (protection) side of the circuit.

The impact of the plant control and monitoring system on unavailability or inoperability is minimized. The probability of failure in the plant control and monitoring system resulting in plant unavailability during the power operation is reasonably small. The plant is defined to be unavailable if the failure(s) result in the plant being unable to achieve and/or maintain all steady state operating points within the warranted plant output.

Both automatic and manual controls are provided for each of the control functions identified here. Manual control will override automatic control. While in the manual mode, the automatic system tracks the manual system so that upon transfer to the automatic mode, the transfer is bumpless. (A bump-less transfer is defined as a transfer between modes which maintains a continuous and smooth control signal.) Similarly, transfer from the automatic mode to the manual mode is also bump-less. Sufficient information is supplied to the operator to allow manual operation in a safe and efficient manner.

The plant control and monitoring system design minimizes the time required for startup following a reactor trip or load rejection, it also minimizes any requirements for setpoint changes due to changes in plant operation.

The plant control and monitoring system reduces the requirements for complex operator actions during normal operation. Automatic control systems are used where practical.

No single failure at the component level within the plant control and monitoring system or its supporting systems will initiate an event so rapid that an operator can not reasonably intervene to prevent reactor trip or ESF actuation. This criterion is not applicable during maintenance of the plant control and monitoring system.

6.8 Diverse Actuation System

The Diverse Actuation System (DAS) is designed based on the requirements of U.S. NRC Standard Review Plan (Branch Technical Position HICB-19). DAS is classified as a non-safety system, so the single failure criterion is not applied.

DAS consists of diverse components from the digital safety system so that CMF in the digital safety system will not also impair DAS functions. (Hardwired device or diverse digital system)

Spurious actuation of the DAS functions might adversely affect plant availability. So the design ensures the DAS can sustain one random component failure without spurious actuation. Safety or non-safety sensors selected by the plant design are connected to the DAS (with no reliance on the digital I&C Systems) so that a postulated CMF within the digital safety or non-safety I&C system will not affect the DAS function.

Safety or non-safety sensors selected by the plant design are connected to DAS so that CMF within the digital safety system will not affect the DAS function.

The DAS initiates safety functions independent of the output from the digital safety system. Manual actuation is provided for all functions. Automatic actuation is also provided for functions where time for manual operator action is inadequate.

6.9 Data Communication System

6.9.1 Data Signal Transmission of Digital I&C Equipment

6.9.1.1 The Principle of Data Signal Transmission between the Systems

Multiplex transmission is used for most interfaces between systems. The Main data communication interfaces between the systems are described as follows.

6.9.1.2 Unit Bus

The Unit Bus provides non-safety data communication between all I&C systems. The main signals transmitted through the Unit Bus are;

- Manual operation signals transmitted from the Operator Console to the safety and the non-safety system.
- Process and alarm signals transmitted from the safety and the non-safety systems to the Operator & Supervisor Console, etc., and to the computer systems such as Process Recording Computer System, Alarm Processor System, etc.
- Shared signals, such as Tavg signal etc., are transmitted from safety systems to non-safety systems.

- Signals, such as interlock signals, are transmitted between non-safety and safety systems, and between multiple non-safety systems.

6.9.1.3 The Data Transmission between the Safety-related Systems

- Unidirectional fiber optic data links are used between the Reactor Protection System and ESF Actuation System.
- Unidirectional fiber optic data links are used between the four redundant Reactor Protection System train sets.
- There is one Safety Bus for each train. Each Safety Bus is used only within the same train. The Safety Bus is used between the ESF Actuation System and the Safety Logic System, and between subsystems of the Safety Logic System. Each train's Safety Bus is a master-less peer-to-peer multi-drop communication network.
- Within each train of the Safety Logic System is a Remote I/O bus. This bus is used to communicate signals to/from remote I/O modules to the microprocessors in each train. Each Remote I/O Bus is used only within one train.

6.9.1.4 The Interface by using I/O (Input/Output)

The main signals transmitted through I/O are described below.

- Between the digital I&C system and the local-related devices (Such as sensor, control valve, limit switch etc.)
- Between dedicated systems and the digital I&C system.
- Others (To satisfy functional requirements)

Digital output cards in the digital I&C system provide solid state outputs for controlling plant components (i.e., there are no interposing electromechanical relays). Digital outputs cards are used in both safety and non-safety applications.

The Power Interface (PIF) card, which also has solid state outputs, is used in safety applications, where testability is required for the output to the controlled plant safety component. The PIF to plant component interface can be manually tested using pulse signals.

The PIF is also used for plant components that must be actuated by both the Safety Logic System and the DAS. In these applications, priority logic within the PIF ensures that either system can actuate and maintain the component in the predetermined safe state (e.g., the on/open state for components relied upon to supply cooling water or the off/closed state for components relied upon for isolation).

The PIF is a very simple module that can be fully tested to ensure there is no potential for common mode hardware or software failure. Therefore it is used for compliance to HICB-19 and 10CFR50.62.

6.9.1.5 Hardwired Interfaces

Some signals in the digital I&C system scope are transmitted through hardwired circuits. The main interfaces that utilize hardwired transmission are described as follows:

- Between the Reactor Protection System and the Diverse Actuation System (Process signals for Diverse Actuation)
- Between ESF Equipment Control switches and Back-Up Operation Panel (Manual operation signals in case of CMF)
- Between Reactor Protection System and Back-Up Operation Panel (Process signals for CMF monitoring)

6.9.2 Data Communication with TSC, EOF and NDL

A data communication system establishes the interface and link with the TSC, EOF and NDL and allows data exchange with the plant. Several support centers may be foreseen. One would be the "master" and would allow the connection of the others to the site.

In order to maintain cyber security, each HSI computer system is not connected to external networks, only to full digital I&C equipment. Therefore, invading from outside of the plant is physically impossible.

The following countermeasures are applied to prevent cyber security threats;

- The plant I&C and HSI systems do not link to external networks. An exception is the link from unit management computer (UMC) to the Station Bus.
- Communication from UMC to the Station Bus is restricted one direction. A dedicated transmission protocol is used which is not general-purpose, such as TCP/IP, UDP, etc.
- Communication between the Station Bus and the TSC, EOF or NDL (NRC) is also one direction and uses a dedicated transmission protocol.
- If a computer system which has a general-purpose LAN network is connected to the Station Bus, an adequate gateway processor with firewall function is inserted.
- The firewall program currently used is MISTY®, which uses 128 bit code key. This firewall program is safer than DES code which is more typically used in the U.S. Alternate firewall programs may be used in the future, as the security features of new technology evolves.

Table 6-1 US-APWR PSMS Reactor Trip Signals

- Source Range Neutron Flux High
- Intermediate Range Neutron Flux High
- Power Range Neutron Flux High
- Power Range Neutron Flux Rate High
- Emergency Core Cooling System Actuation Signal
- Over Temperature Delta-T High
- Over Power Delta-T High
- Pressurizer Pressure High
- Pressurizer Pressure Low
- Reactor Coolant Flow Low
- Reactor Coolant Pump Speed Low
- Steam Generator Water Level Low
- Steam Generator Water Level High
- Pressurizer Water Level High
- Manual

Table 6-2 US-APWR PSMS ESF Actuation Signals

<u>Emergency Core Cooling System (ECCS) Actuation Signals</u>
- Pressurizer Pressure Low-Low
- Main Steam Line Pressure Low
- Containment Vessel Pressure High
- Manual

<u>Main Steam Line Isolation Signals</u>
- Containment Vessel Pressure High-High
- Main Steam Line Pressure Low
- Main Steam Line Pressure Rate High
- Manual

<u>Containment Vessel Spray Actuation Signals</u>
- Containment Vessel Pressure High-High-High
- Manual

<u>Containment Vessel Isolation Signals</u>
- ECCS Actuation Signal
- Containment Vessel Spray Actuation Signal
- Manual

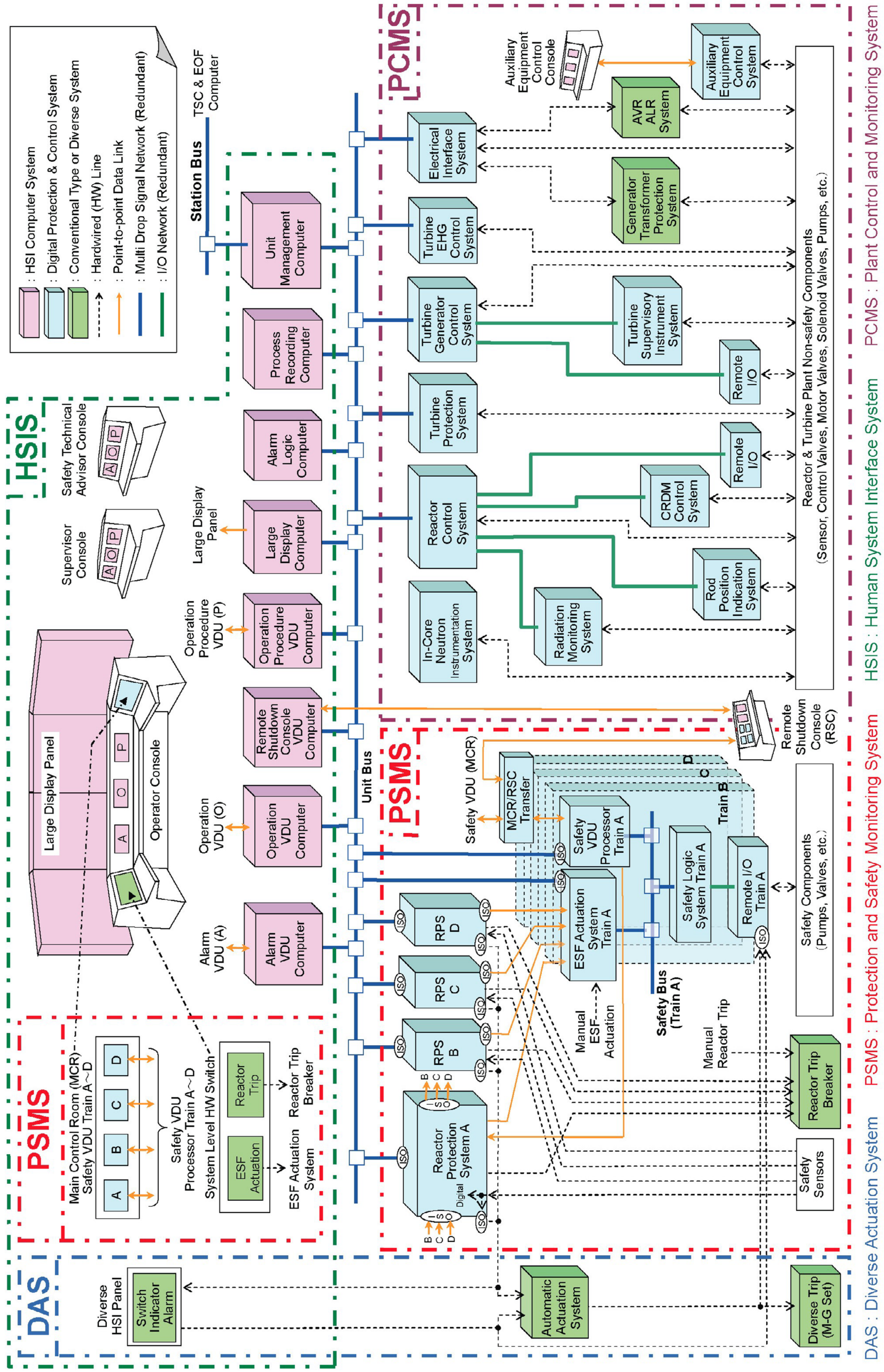


Figure 6-1 US-APWR I&C Overall Architecture

7.0 ELECTRIC POWER

7.1 Basic Design Concept

The basic design of the electrical systems is as follows.

- Figure 7.1-1 and 7.1-2 show the system configuration of the electrical system. The onsite electrical power system is supplied from two-offsite transmission systems. The onsite electrical power system consists of Class1E safety electrical systems and non-safety electrical systems. All plant loads are supplied offsite power from two House Service Transformers and a Reserve transformer.

- The safety electrical systems consist of four separate distribution systems. There are four AC medium voltage buses, four AC low voltage systems (Load Center, Motor Control Center), four DC power systems and four instrumental & control power systems. All safety loads are supplied offsite power from these buses under the condition of “Start-Up/Shut-Down”, “Normal Operation”, “Plant Outage”, and “Accident”.

- The US-APWR design basis allows on-line maintenance of each of the four Safety Gas-Turbine Generators (GTG). The four train safety system loads are connected with each train bus. There are also some two train safety system loads. These two train safety system loads are connected to buses that can be powered from either of two power sources. During maintenance of an Safety GTG, the two train loads are manually switched to the alternate train feeder.

- The four Safety GTGs and the four safety batteries are Class 1E emergency power sources. When a Loss of Offsite Power (LOOP) occurs, a Safety GTG is automatically started and reaches its rated voltage within 60 seconds from receiving an under voltage signal from the safety medium voltage bus. The design and analysis of the ECCS assumes more than 100 seconds for startup, allowing 40 seconds startup margin for the Safety GTG. Each Safety GTG circuit breaker is closed automatically, and the circuit breakers of safety loads are closed in sequence automatically. The Safety GTG has enough capacity for safety loads needed for a concurrent DBA and LOOP.

- The Gas-Turbine is a very simple rotary engine with few components. The Safety GTG system consists of a GTG package, a fuel transport system, a starting system and a control/instrumental system. It does not need any cooling system. The Safety GTG is started with a DC starting motor. This DC motor is supplied power from a safety battery which is part of the safety DC power supply system.

- If a station black out (SBO) occurs, the Safety GTGs are assumed to be inoperable. To provide the necessary power to cope with the SBO, a non-safety Alternate Gas-Turbine Generator (GTG) is provided which is diverse from the Safety GTGs (different specification and different manufactures). The Alternate GTG is normally isolated from the safety systems, but it is started automatically from the under voltage signal of any one of the medium voltage safety buses. And the circuit breaker of Alternate GTG is manually closed. After the bus is reenergized, loads required for SBO are manually started to allow the plant to achieve and maintain a safe shutdown condition.

- The safety battery supplies DC power to loads for two hours under the condition of loss of AC power.

7.2 Offsite Power System

Design conditions and specifications of offsite power are as follows.

- Offsite power is provided from two offsite transmission systems. The two offsite transmission systems are physically separated.
  - Main offsite power is received through the Main Transformer and two House Service Transformers
  - Auxiliary offsite power is received through the Reserve Transformer

- The unit output power is about 1,700MWe. The output voltage of the main turbine generator is connected to the main transmission system through the Main transformer. During normal operation, the Generator Load Break Switch (GLBS) is closed and output power of the main turbine generator feeds the offsite transmission line. When the main turbine-generator is not operating, the GLBS is open and the onsite electrical power system is supplied with offsite power through the Main Transformer and two House Service Transformers.

- The needed onsite power of the unit during start-up/shutdown mode is supplied from the Main Transformer and two House Service Transformers with the GLBS opened.

7.3 AC Power System

Design conditions and specifications of AC onsite power system are as follows.

- The onsite AC electrical power systems consist of Class 1E safety electrical systems and non-safety electrical systems.

- The safety electrical system consists of four separate trains. Each safety train consists of one 6.9kV AC medium voltage bus and 480V AC low voltage buses (Load Centers, Motor Control Centers). Each AC medium voltage bus connects to a Safety GTG.

- The plant is designed to allow on-line maintenance of each Safety GTG. The two train safety system loads are connected to buses that can be powered from either of two Safety GTG power sources.

- There are four non-safety electrical systems. Each non-safety system consists of 6.9kV AC medium voltage bus and 480V AC low voltage systems (Load Centers, Motor Control Centers). Non-safety AC loads are supplied power from these systems.

- There is one non-safety Alternate GTG for coping with SBO. The Alternate GTG is diverse from the Safety GTGs (different specification and different manufacturers). The Alternate GTG connects to four safety medium voltage buses via circuit breakers. All of these circuit breakers are normally open.

- When a LOOP occurs, each Safety GTG is started automatically from an under voltage signal of its respective medium voltage bus. And an Safety GTG reaches the rated voltage within 60 seconds. The circuit breaker of each Safety GTG is closed automatically and safety loads are automatically started in sequence.

7.4 DC Power System

Design conditions and specifications of DC onsite power system are as follows.

- The DC power system consists of four safety 125V DC power systems and the two non-safety 125V DC power systems.

- Each DC power system consists of a battery charger, a battery and a DC motor control center. Even if the in-feed power or battery charger are lost, DC loads can be supplied uninterrupted power from the battery.

- The DC power systems supplies power to DC motor valves, control circuit of switchgear, inverters, emergency lighting, etc.

- A safety battery supplies power to loads for two hours under a loss of AC power condition. A non-safety battery supplies power to loads for one hour under a loss of AC power condition. Batteries are continuously reacharged when AC power is provided from normal AC sources or from the Safety GTG’s.

7.5 Instrumentation & Control (I&C) Power System

Design conditions and specifications of the I&C power system are as follows.

- The I&C power systems consist of four safety 115V AC power systems and the four non-safety 115V AC power systems. These systems supply power to I&C system cabinets that are important from the view points safety and operation of the plant.

- The inverter unit is the main power source of each system. Even if there is fluctuation of in-feed power, the voltage and frequency of the inverter’s output power are regulated. The inverter has one in-feed AC power source and one in-feed DC power source. Even if one in-feed power source is lost, the inverter supplies uninterrupted power to its loads.

7.6 Countermeasures against Station Black Out

Design conditions and specifications of power to cope with SBO are as follows.

- If a SBO occurs, all Safety GTGs are assumed to be inoperable. A Alternate GTG is provided which is diverse from the Safety GTGs (different specification and manufacturers). The Alternate GTG supplies power to the loads needed to safely cope with the SBO condition.

- The Alternate GTG is started automatically by under voltage signals from any one of the medium voltage safety buses. The Alternate GTG reaches a rated voltage within 5 minutes and connects manually to the medium voltage safety buses which power loads required to achieve and maintain a safe condition during SBO.

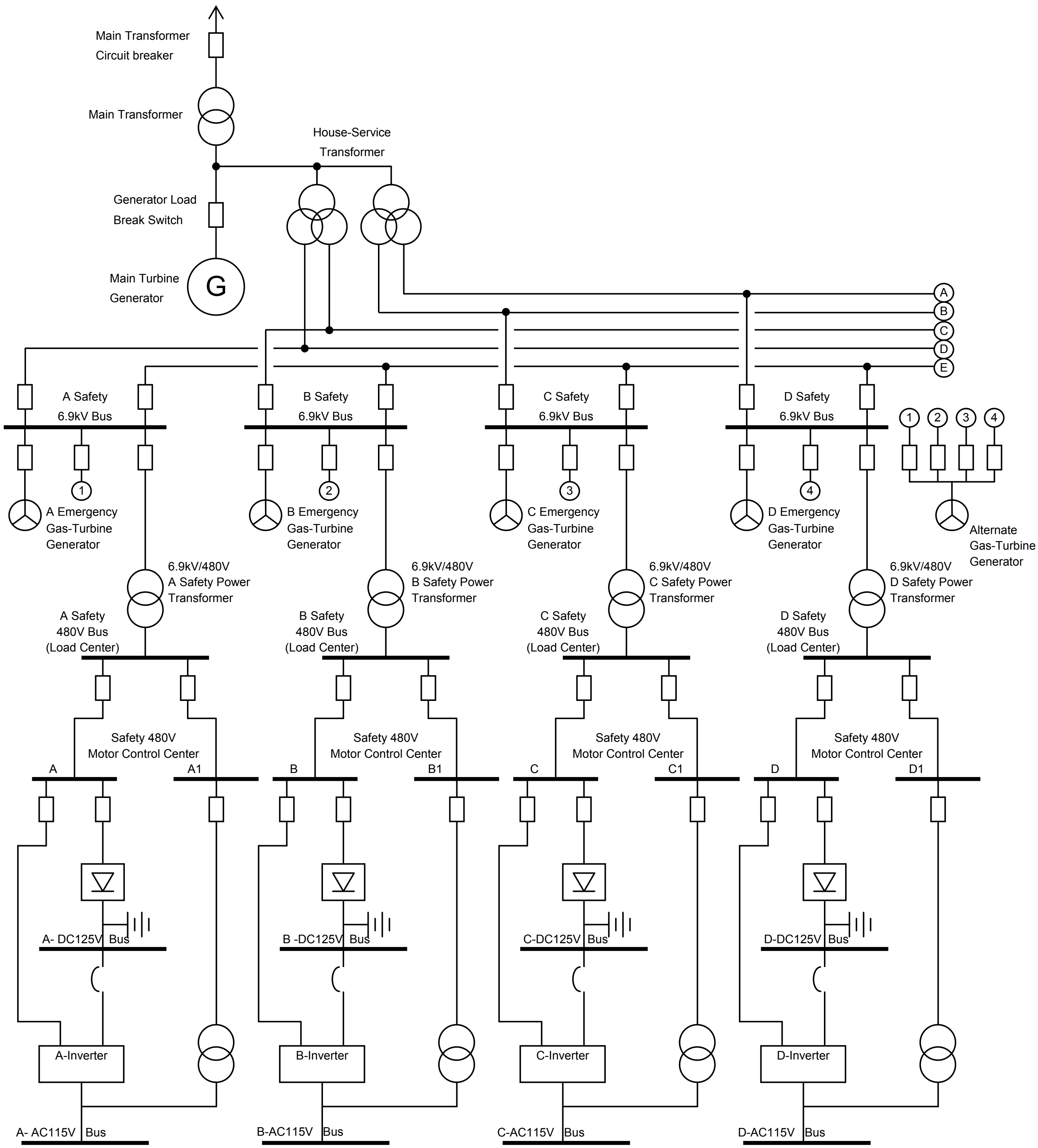


Figure 7.1-1 Safety System Single-line Diagram

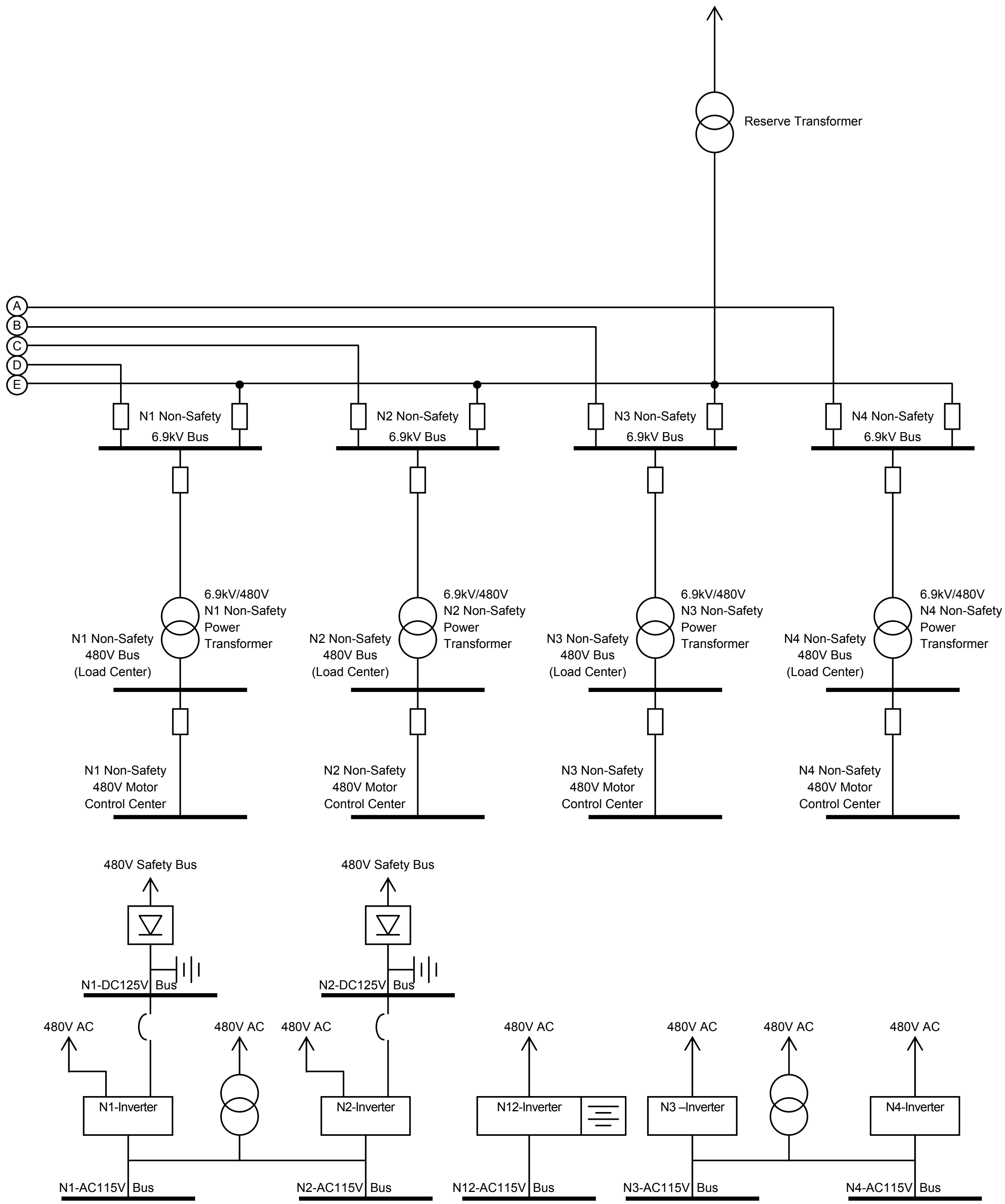


Figure 7.1-2 Non Safety System Single-line Diagram

8.0 AUXILIALLY SYSTEMS

8.1 Fuel Storage and Handling

8.1.1 Spent Fuel Pit Purification and Cooling System

The spent fuel pit (SFP) purification and cooling system (SFPCS), shown in Figure 8.1.1-1, consists of a closed circuit that includes the spent fuel pit heat exchangers (HXs), spent fuel pit pumps, spent fuel pit demineralizers, spent fuel pit filters, piping, and valves. The SFPCS has the following functions:

- Removal of decay heat from spent fuel in the SFP
- Purification of the borated water in the SFP, refueling water storage pit (RWSP), refueling water storage auxiliary tank (RWSAT), and reactor cavity.

The SFPCS design is based on the following;

- The SFPCS removes decay heat released by spent fuel stored in the SFP by cooling the SFP water.
- Demineralizers and filters remove particulate and ionic impurities from the SFP water.
- The gas turbine generators can supply electrical power to the SFP pumps, so that SFP cooling functions can be maintained during a loss of offsite power.
- Water can be added to the system using the supply line from the secondary make up water pump. In an emergency, replenishment of borated water can be accomplished using the supply line from the RWSP.
- The system is designed to maintain the water level of the SFP to prevent uncovering of stored fuel even if there is leakage due to failure of the piping.

The piping connected to the SFP is equipped with siphon breakers to prevent uncovering stored fuel in the event there is leakage in the system.

During decay heat removal operation, SFP water flows from the SFP to the SFP pump suction, through the SFP heat exchanger, transferring heat from the SFP water to the component cooling water, and returns to the SFP.

A portion of the SFP water is diverted through the demineralizers and the filters in the purification part of the system to maintain water purity.

During normal decay heat removal operation, one train can be used to purify the reactor cavity, the RWSP, and the RWSAT.

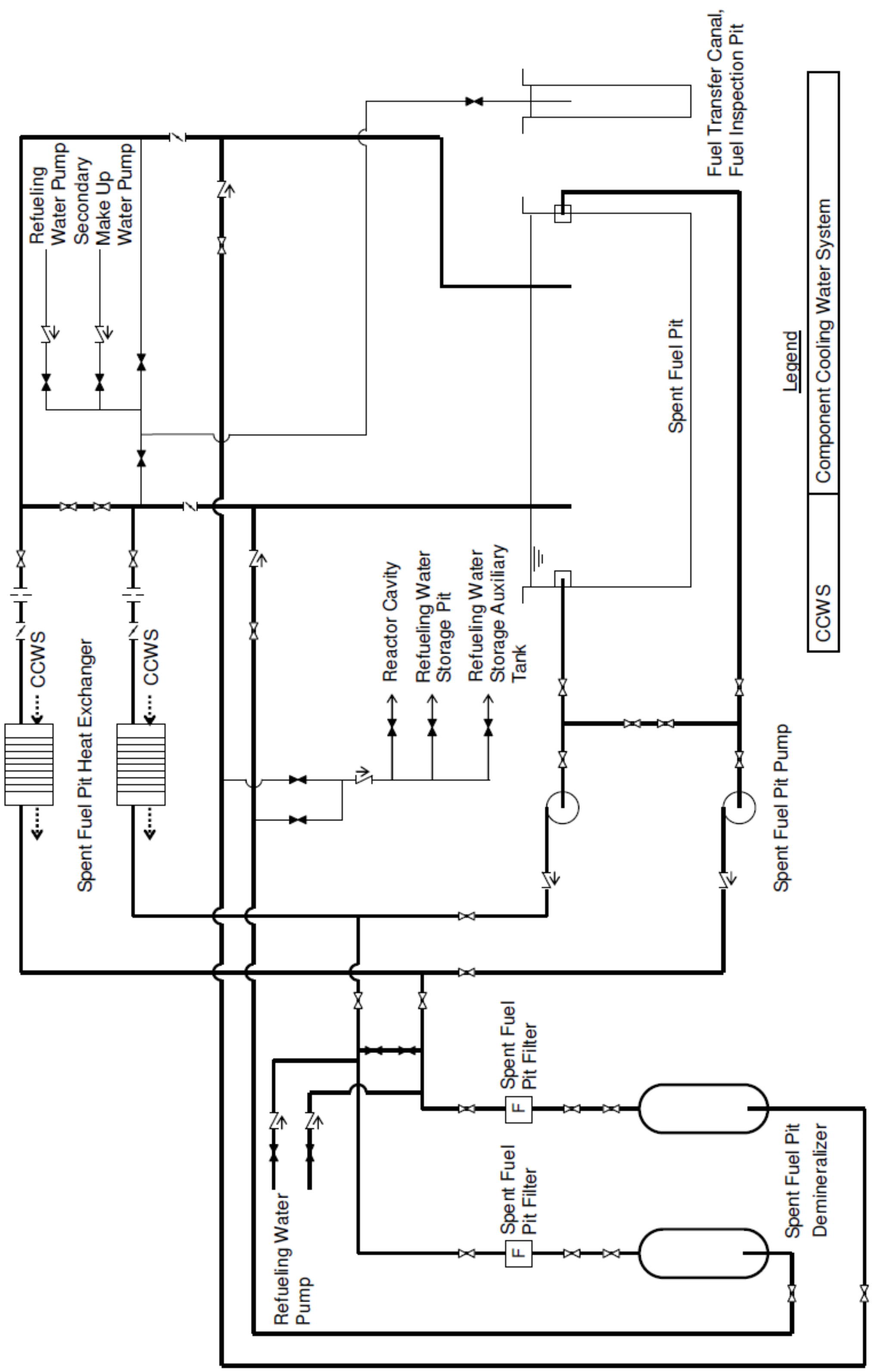


Figure 8.1.1-1 Spent Fuel Pit Purification and Cooling System

8.1.2 Fuel Storage and Handling

The fuel storage and handling system is to carry out fuel storage and handling safely and securely from the time when new fuel is brought into the power plant to the time when spent fuels are brought out to the outside of the plant.
The new fuels brought into the power plant are stored in the new fuel pit in the reactor building.
After reactor shutdown, the spent fuel in the reactor is transferred to the spent fuel pit through the reactor cavity, refueling canal and fuel transfer tube with borated water filled, by using refueling crane, fuel transfer equipment, spent fuel pit crane, etc.
All of these spent fuel transfer works are carried out in the borated water, and the water performs functions of shielding and cooling.
The spent fuel is stored in the spent fuel pit.
After completing the cooling, the spent fuel is charged in the water into the spent fuel transport packaging by using the spent fuel pit crane, etc., and are transported to the outside of the plant.

The major equipments of fuel storage and handling are as follows.

- New Fuel Pit
- Spent Fuel Pit
- Cask Loading Pit
- Decontamination Pit
- Reactor Cavity and Refueling Canal
- Refueling Crane
- Spent Fuel Pit Crane
- Fuel Handling Area Crane
- New Fuel Elevator
- Fuel Transfer System

8.2 Water Systems

8.2.1 Essential Service Water System

The essential service water system (SWS) consists of the service water pumps, piping, valves, and instrumentation. The system provides service water for the component cooling water heat exchangers.

The SWS transfers heat from the CCWS to the Ultimate Heat Sink (UHS).

The flow diagram for the system is shown in Figure 8.2.1-1

The SWS is designed based on the following;

- The service water pumps and the related piping are designed to take service water from the UHS and deliver it to the component cooling water heat exchangers.
- The whole system consists of four independent trains. Each train has one service water pump.
- The SWS is designed to provide sufficient cooling capacity for normal operation, transients, and accidents such as the loss-of-coolant accident and loss of offsite power.
- The service water pumps can be powered from the safety buses so safety-related functions are maintained during a loss of offsite power.
- The SWS is designed to perform safety-related functions assuming a single failure in a one train and another train out of service for maintenance.
- The SWS is automatically initiated by a safety injection signal.

Each service water pump takes water from the UHS, pumps it through the component cooling water heat exchangers and discharges it to the discharge pit. In an accident situation, the necessary safety functions can be performed by two of the four trains.

The system configuration for each operating mode is as follows:

- During normal operation, two trains are used to supply service water to two trains of the component cooling water system.
- During hot shutdown or plant cool down by the steam generators, a similar configuration is employed. In hot weather, if the outlet temperature of the component cooling water heat exchangers increase to near 100 °F, three service water trains are used.
- During the residual heat removal operation, after cool down by the steam generators, all four trains of the component cooling water system trains and the service water trains are operated.
- During the refueling, the number of service water pumps and component cooling water trains is determined by the decay heat to be removed.

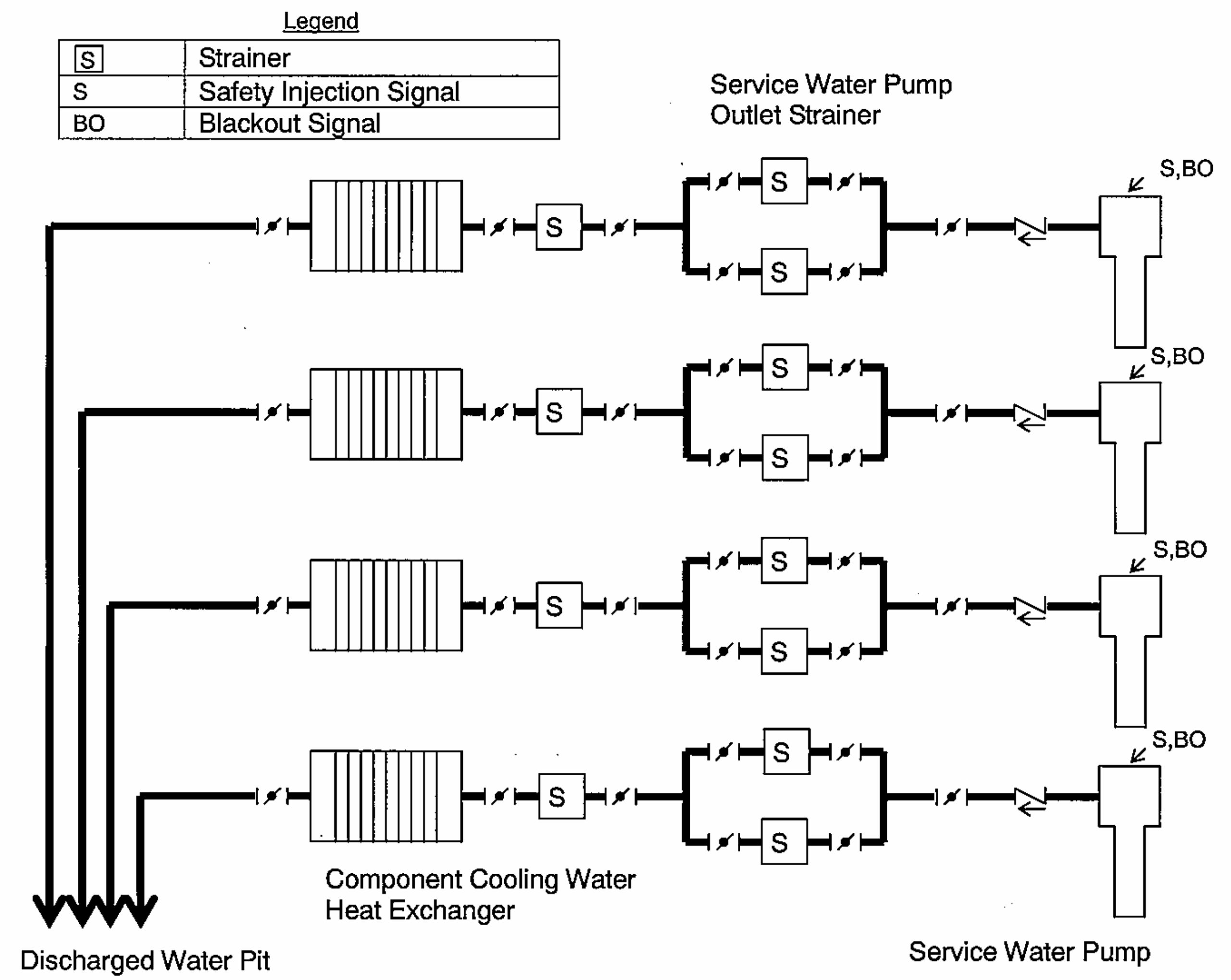


Figure 8.2.1-1 Essential Service Water System

8.2.2 Component Cooling Water System

The component cooling water system (CCWS) provides cooling water for the components of the primary systems during normal operation, plant shutdown, and after an accident. It also serves as an intermediate system between the reactor coolant and the essential service water system (SWS) to prevent leakage of radioactive material into the environment.

The flow diagram for the system is shown in Figure 8.2.2-1.

The CCWS has four trains. Each train has one pump and one heat exchanger. The trains are arranged in two subsystems, each of which has a suction and discharge header and a surge tank (CCWT). Electrical power to all trains is supplied by the safety-related buses, which are backed up by gas turbine generators.

The CCWS provides cooling water for essential equipment such as the containment vessel spray and residual heat removal (CS/RHR) heat exchangers, the spent fuel pit heat exchangers, the chiller cooling coils, the safeguard pump coolers and other normal operating loads, i.e. the chemical and volume control system coolers, the radwaste management system coolers, and the reactor coolant pump coolers etc.

The surge tanks accommodate the thermal expansion and contraction of the cooling water and potential leakage from the thermal barriers in the reactor coolant pumps.

The CCWS is designed based on following;

- The design is based on the service water maximum design temperature.
- The system is designed to ensure that any leakage of radioactive fluid from the cooled components is held within the plant.
- The component cooling water supply to safety-related components is designed to supply component cooling water from the independent subsystems. Each subsystem includes two trains and each train contains a heat exchanger and a pump.
- The CCWS is designed to provide the sufficient cooling capacity not only for the components required during normal operations such as plant power operation and residual heat removal operation, but also for those important to the safety in the event of an accident such as the loss-of-coolant accident or the abnormal operational transient condition involving the loss of offsite power.
- The component cooling water pumps can be powered from the safety-related buses so safety functions can be maintained during a loss of offsite power.
- The CCWS is designed to perform safety functions assuming a single failure in the one train and another train out of service for maintenance.
- The CCWS is automatically initiated by the safety injection signal.
- The radiation monitors are installed to detect the leakage of radioactive materials into the CCWS.

The system has cooling lines to the various components to be cooled, the associated piping, the valves and the instrumentation. The component cooling water flows from the component cooling water pumps, through the component cooling water heat exchangers, to the components to be cooled and returns to the pumps.

The component cooling water surge tanks are connected to the suction side of the component cooling water pumps. Makeup water is also supplied to the suction side. The component cooling water surge tank accommodates expansion and contraction of the system water due to temperature change or leakage. In case of a small leak in the system, the tank can supply makeup water until the leak is isolated. Isolation valves are installed for each component.

Two trains are used during normal operation. The other trains are on stand-by.

During residual heat removal operation, all four trains are used. The failure of one train may increase the time for cooldown, but does not affect the safe operation of the plant.

The component cooling water surge tanks are installed at an elevation high enough to provide the component cooling water pumps with sufficient suction head. The component cooling water heat exchangers are installed with sufficient space for maintenance.

The system configuration for each operating mode is as follows:

- During the normal power operation, the CCWS is operated with two pumps and two heat exchangers, one train in each subsystem. Should a running pump fail, the other pump in the same subsystem starts automatically.
- During the early stage of normal cooldown, the CCWS is operated in the same way as during normal power operation.
- Approximately four hours after shutdown, when the second stage of reactor coolant system cooldown occurs, the residual heat removal system is initiated. At this time, the CCWS isolation valves to the CS/RHR heat exchangers are opened.
- During the later phase of cooldown, all CCWS trains are operated.
- During the refueling, the CCWS is operated with two or three pumps and heat exchangers. The system is aligned the same as it is for the latter phase of normal cooldown.
- Following the receipt of a safety injection signal or the loss of offsite power, all four CCWS pumps receive a start signal. At the receipt of the safety injection signal, the CCWS isolation valves to the CS/RHR heat exchangers are opened.

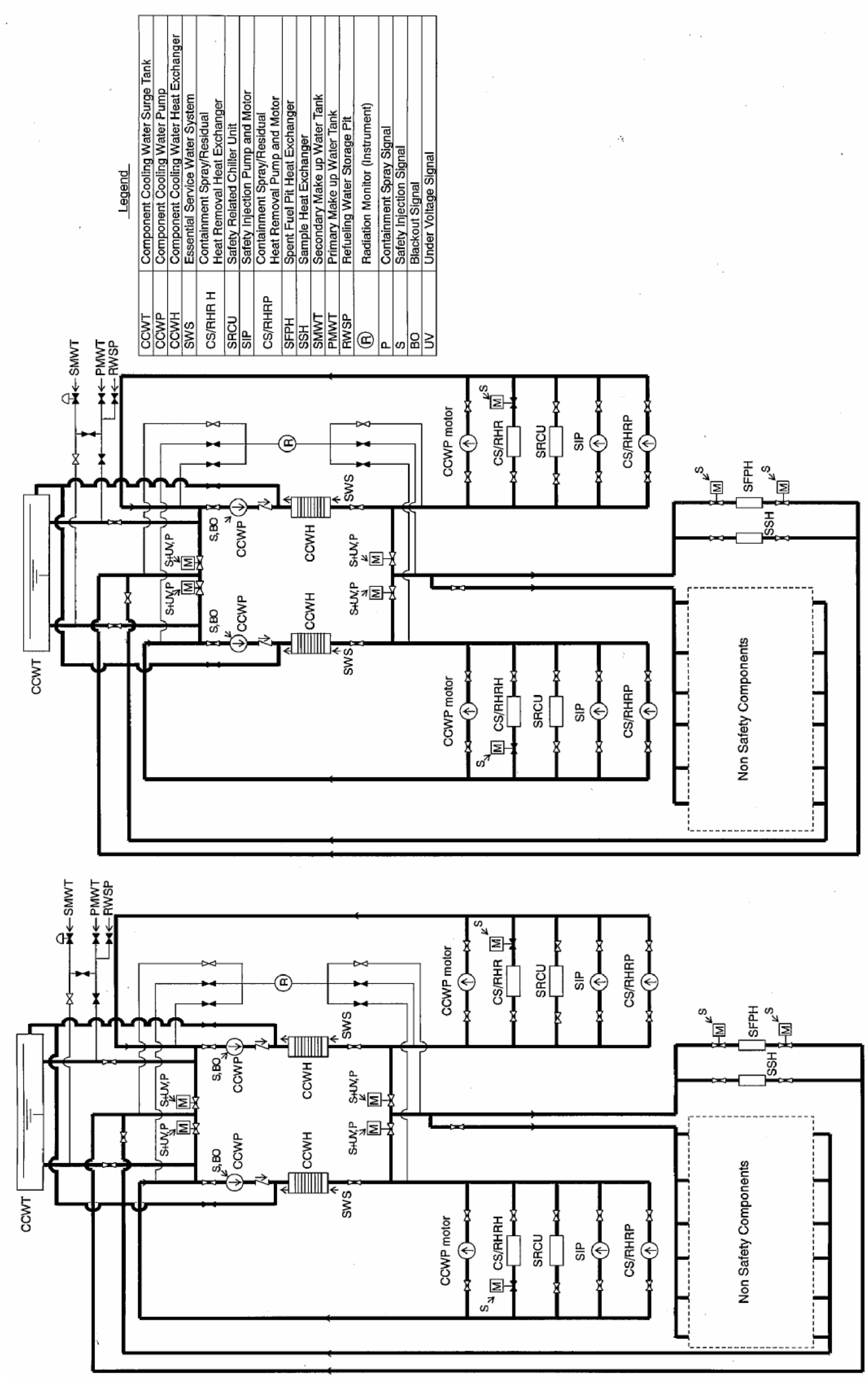


Figure 8.2.2-1 Component Cooling Water System

8.2.3 Primary Make-up Water System

The primary make-up water system (PMWS) consists of primary makeup water storage tanks, primary makeup water pumps, piping, valves, and instrumentation.
The PMWS has the following functions:

- Supplies make-up water to primary system equipment
- Containment isolation

The PMWS is designed to store and provide degasified, demineralized make-up water, and has no safety function except for containment isolation.
The primary makeup water storage tank is designed to store sufficient water to meet the demands of the primary system components. The primary makeup water pumps provide makeup water to several locations, including:

- The boric acid blender (makeup to the reactor coolant system).
- The chemical mixing tank (as a solvent).
- The component cooling water surge tanks (emergency makeup).
- The demineralizers, the spent resin discharge header, and the spent resin storage tanks

The PMWS receives water from the demineralized water system and also receives the recycled water from the boron recycle system.

8.2.4 Chilled Water Systems

The heating, ventilation, and air conditioning (HVAC) systems require chilled water to maintain the ambient conditions for operating personnel and equipments. The chilled water systems provide chilled water to the cooling coils of the air handling units and unit coolers of the HVAC systems.

The chilled water systems consist of safety chilled water system and non safety chilled water system. The safety chilled water system is the primary system to provide chilled water to safety-related HVAC systems. The non safety chilled water system provides chilled water to non safety-related HVAC systems.

The safety chilled water system is designed based on following;

- The safety chilled water system provides chilled water to the cooling coils of safety-related HVAC systems during normal operation, plant shutdown, and after an accident.
- This system is arranged in the four trains and each train is composed of one pump and one chiller. This system has two expansion tanks that each tank is shared by two trains.
- The safeguard buses supply electrical power to the safety-related components of this system, so that safety functions are maintained during a loss of offsite power.
- This system is designed so that the safety function is not affected by a single active component failure.
- This system is automatically initiated by a safety injection signal.

The non safety chilled water system is designed based on following;

- The non safety chilled water system is a closed loop system that provides chilled water to the cooling coils of non safety-related HVAC systems during normal operation.

8.3 Process Auxiliaries

8.3.1 Instrument Air System

The Instrument Air System (IAS) supplies clean, dry, and oil-free compressed air to the following equipment and facilities.

- Air-operated valves
- Heating, ventilation and air conditioning (HVAC) air dampers
- Pneumatic instruments and controls
- Measuring instruments
- Other equipment which is not safety related.

The IAS is designed based on followings;

- Compressed air is not used for the safety function
- The IAS comprises the redundant compressor packages.
- The IAS compressor package consists of an inlet air filter/silencer, a compressor, an after-cooler, an air reservoir and a drier.

The instrument air compressors are of an oil-free type so that the clean compressed air can be provided. The two instrument air compressors, each with a capacity of 100 percent, are installed.

8.3.2 Sampling System

The sampling system (SS) consists of sample heat exchangers, a containment atmosphere gas sample heat exchanger, a sample hood, a sample sink, sample pressure vessels, a containment atmosphere gas sample compressor, and associated piping and valves.
The SS provides the following functions;

- Collect liquid samples from the reactor coolant system and auxiliary systems and gaseous samples of the containment atmosphere.
- Containment and RCPB isolation.
- Provide gaseous samples of containment atmosphere and liquid samples of reactor coolant. Monitor hydrogen concentration and radioactivity in gaseous samples, and boron and radioactivity in liquid samples.
- Provide protection against exposure and contamination during collection of samples, and send spilled water and wash water to the Liquid Waste Management System.
- Collect gaseous samples of containment atmosphere to monitor the hydrogen content and radioactivity of the containment atmosphere after an accident.
- Collect reactor coolant to monitor the boron concentration and radioactivity of the reactor coolant after an accident.

The SS design is based on the following:

- Cool and depressurize samples collected at high temperature and high pressure.
- The system is designed so that containment isolation is not violated while collecting samples of the reactor coolant and the containment atmosphere after an accident.
- Collect liquid samples from the RCS, the CVCS, the RHRS for the purpose of monitoring the reactor coolant during normal plant operation and after an accident.
- Collect gaseous samples of the containment atmosphere following an accident.

Sampling points and the purpose of each is as follows:

- During normal plant operation,
  - The liquid sample from pressurelizer is used to monitor boron concentration.
  - The liquid samples from the RCS, CVCS and RHRS are used to monitor boron concentration, radioactivity, dissolved oxygen, hydrogen, halogens, pH, and conductivity to assure integrity of the fuel, and the quality of the reactor coolant.
  - The liquid samples at the inlet and outlet of the demineralizers and filters in the CVCS are analyzed for radioactivity and halogen concentration in order to monitor the performance of the demineralizers and filters.

- After an accident,
  - Liquid samples of the reactor coolant are analyzed for boron concentration and radioactivity in order to determine the extent of damage, if any, and conditions in the radiation barrier.
  - Gaseous samples of the containment atmosphere are analyzed for hydrogen concentration and radioactivity in order to determine the extent of the damage, if any, and conditions in the radiation barrier.

8.3.3 Chemical and Volume Control System

The chemical and volume control system (CVCS), including the boron recycle system, is shown in Figure 8.3.3-1. The CVCS includes the regenerative heat exchanger, letdown heat exchanger, letdown orifices, purification filters and demineralizers, volume control tank, boric acid tanks and transfer pumps, charging pumps, seal injection filters, piping, valves, and instrumentation.

The CVCS has the following functions:

- Maintain the coolant inventory in the RCS for all normal modes of operation, including startup, full-power operation, and cooldown.
- Provide the makeup capability for small leaks from the RCPB.
- Remove fission and activation products from the reactor coolant.
- Regulate the boron concentration in the reactor coolant during normal operation.
- Borate the reactor coolant system for cold shutdown.
- Provide a means to add chemicals to the RCS to control coolant pH, scavenge oxygen from the coolant during startup, and counteract the production of oxygen in the reactor coolant.
- Supply cool filtered water to the shafts seals and bearings of the reactor coolant pumps.

Letdown flow comes from the RCS and flows through the regenerative heat exchanger, where its temperature is reduced by transferring heat to the incoming charging flow. The coolant is then depressurized as it passes through the letdown orifices and is further cooled in the letdown heat exchanger. A second stage of pressure reduction is performed by the low-pressure letdown valve, which maintains upstream pressure to prevent flashing downstream of the letdown orifices. The letdown rate is controlled to suit various plant operating requirements by selecting a combination of letdown orifices.

The letdown flow then flow through the mix-bed demineralizers purification. The flow may also pass through the cation-bed demineralizer, which is used intermittently when additional purification is required. The deborating demineralizers are used near the end of core life for removal of boron.

The coolant then flows to the volume control tank (VCT) through a spray nozzle in the top of the tank. Hydrogen is supplied to the VCT, where it mixes with fission gases that are stripped from the reactor coolant. Contaminated hydrogen is vented to the gaseous waste management system. The partial pressure of hydrogen in the VCT is controlled to establish the concentration of hydrogen dissolved in the reactor coolant.

Two centrifugal charging pumps are provided to take suction from the VCT and return the cooled, purified reactor coolant to the RCS. The charging flow is pumped to the RCS through the regenerative heat exchanger, and injected into a cold leg of the reactor coolant system. A portion of the flow is directed to the reactor coolant pumps through a seal water injection filter.

An auxiliary pressurizer spray provides a means of cooling and depressurizing the pressurizer near the end of plant cooldown, when the reactor coolant pumps, which normally provide the driving head for the pressurizer spray, are not operating.

An excess letdown path is provided in the event that the normal letdown path is inoperable. The excess letdown flow path is also used to provide additional letdown capability during the final stages of plant heatup.

Changes in the reactor coolant system inventory due to load changes are accommodated primarily in the pressurizer. The VCT provide surge capacity for reactor coolant expansion not accommodated by the pressurizer. If the water level in the volume control tank exceeds the normal operating range, a three-way valve downstream of the reactor coolant filter diverts a portion of the letdown fluid to the boron recycle system (BRS).

The boron recycle system recycles reactor coolant for the reuse of boric acid and primary makeup water. The system decontaminates the coolant by means of demineralization and gas stripping, and uses evaporation to separate and recover the boric acid and primary makeup water.

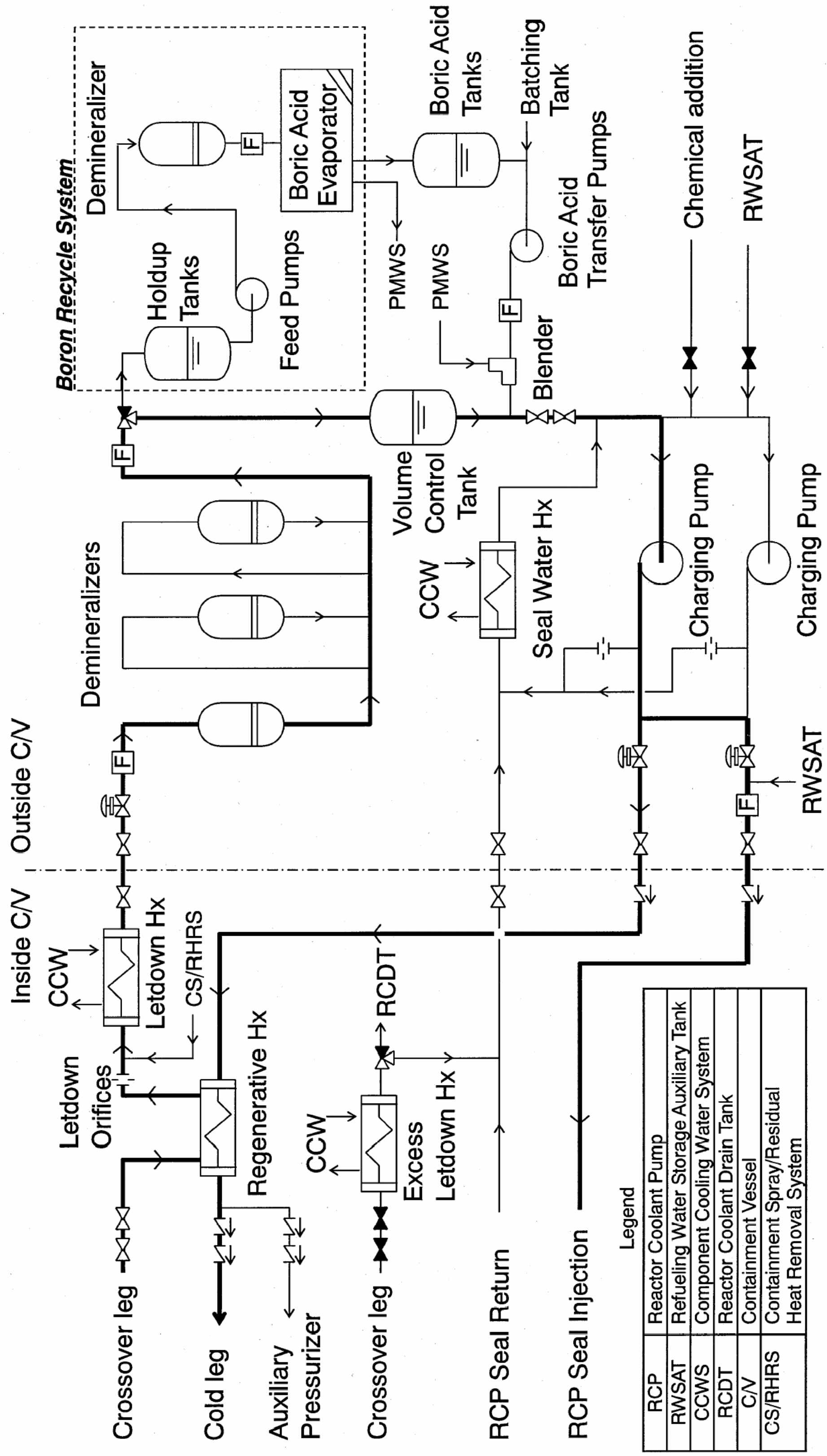


Figure 8.3.3-1 Chemical and Volume Control System

8.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

The air conditioning, heating, cooling, and ventilation systems (HVAC systems) supply fresh air to personnel working in the plant during normal plant operation, remove radioactive materials and restrict radioactive releases to the environment.

The HVAC systems maintain suitable ambient conditions for equipment during normal operation, plant shutdown and accident conditions.

The HVAC systems are designed based on following;

- The HVAC systems are adequately divided according to the function of each area served.
- The HVAC systems are designed to supply intake air for clean areas and exhaust from areas with higher radioactivity. The contaminated exhaust air is filtered and discharged through the vent stack.
- The capacity of each HVAC subsystem is sufficient to ventilate and remove heat in the area served.
- The HVAC systems which support the safety-related areas and systems are designed to perform the specified functions assuming a single active component failure.
- Passive components such as ducts and filters which have a small probability of failure are not redundant. If a failure should occur, the system can be restored within a period which will not adversely affect safety.
- The safeguard buses supply electrical power to the safety-related components of the HVAC systems, so that safety functions are maintained during a loss of offsite power.

The main HVAC systems are;

a. Annulus Air Cleanup System

The annulus air cleanup system is one of the engineered safety futures.

The annulus air cleanup system maintains the annulus and the safeguard component area pressure at a slightly negative pressure following an accident. The contaminated exhaust air is filtered by High Efficiency Particulate Air (HEPA) filters and charcoal absorbers and discharged through the vent stack.

The components of this system are designed to be powered from the safeguard buses so the specified safety functions are maintained during a loss of offsite power.

The annulus air cleanup system is designed to perform the specified safety functions assuming a single active component failure.

The annulus air cleanup system is automatically initiated by a safety injection signal.

b. Main Control Room HVAC System

The main control room HVAC system is one of the engineered safety futures.

The main control room HVAC system provides cooling, ventilation, and heating to maintain design conditions for the main control room during normal operation, plant shutdown and accident conditions.

The control room is automatically isolated and the main control room HVAC system automatically shifts to emergency mode, if radioactive material is detected in the control room or upon receipt of a safety injection signal. The main control room emergency supply fan and electric heating coils that are installed in the main control room emergency supply filter unit are initiated in emergency mode, and recirculation air to main control room is filtered by HEPA filter and charcoal absorbers.

The components of this system are designed to be powered from the safeguard buses so the specified safety functions are maintained during a loss of offsite power.

The main control room HVAC system is designed to perform the specified safety functions assuming a single active component failure.

This system will purge smoke in the event of a fire inside the main control room and isolates the main control room if smoke is detected in the normal outside air intake ducts.

c. Class 1E Electrical Room HVAC System

The class 1E electrical room HVAC system provides cooling, ventilation, and heating to maintain design conditions for safety-related electrical equipment rooms including the class 1E safety switchgear room, the relay room and the battery room during normal operation, plant shutdown and accident conditions.

The class 1E electrical room HVAC system is designed as safety-related system, and is automatically initiated by a safety injection signal.

d. Safeguard Component Area Cooling System

The safeguard component area cooling system provides cooling to maintain design conditions for safeguard component area during accident conditions.

The safeguard component area cooling system is designed as safety-related system, and is automatically controlled by the atmosphere temperature in the area served.

e. Emergency Feed Water Pump Area Cooling System

The emergency feed water pump area cooling system provides cooling to maintain design conditions for emergency feed water pump area during accident conditions.

The emergency feed water pump area cooling system is designed as safety-related system, and is automatically controlled by the atmosphere temperature in the area served.

f. Emergency Generator Area Ventilation System

The emergency generator area ventilation system provides cooling and ventilation to maintain design conditions for emergency generator area during accident conditions.

The emergency generator area ventilation system is designed as safety-related system, and is automatically controlled by the atmosphere temperature in the area served.

g. Containment Ventilation System

The containment ventilation system is provided to reduce containment airborne radioactivity to an acceptable level prior to personnel access for plant maintenance and refueling operations. It supplies intake air to the containment, exhausts containment air through HEPA filters and discharges the filtered air through the vent stack.

h. Containment Recirculation System

The containment recirculation system maintains a suitable containment temperature by circulating and cooling the containment air during normal operation.

i. Control Rod Drive Mechanism Cooling System

The control rod drive mechanism cooling system maintains the ambient air temperature of the control rod drive mechanisms in an acceptable level. Containment air is drawn into the control rod drive mechanism cooling shroud and after cooling, is discharged to the containment atmosphere.

j. Reactor Compartment Cooling System

The reactor compartment cooling system cools the neutron detectors and the primary shield concrete. Cooling air is drawn from the discharge of the containment recirculation system.

k. Containment Depressurization System

The containment depressurization system removes containment air to lower containment pressure when it reaches a defined upper limit during normal operation. It discharges to the annulus air cleanup system.

l. Auxiliary Building Ventilation System

The auxiliary building ventilation system provides cooling, ventilation, and heating to maintain design conditions for controlled access areas such as the fuel handling area and safeguard component area during normal operation. Exhaust air from controlled areas is discharged through the vent stack.

8.5 Other Auxiliary System

8.5.1 Fire Protection Systems

The fire protection systems are installed to minimize the adverse effects of fires on structures, systems, and components important to safety. The fire protection systems consist of fire detection systems and fire suppression systems. The fire suppression systems include water supply and mains, automatic fire suppression and manual fire suppression.

The fire protection systems are designed based on following;

- The fire protection systems are installed for early detection and effective control of fires.
- The fire protection systems are designed so that they do not adversely affect structures, systems or components important to safety due to damage, inadvertent action or improper operation of the systems.

a. Fire Detection

The fire detection systems are installed in all plant areas that contain or present a fire hazard to structures, systems, or components important to safety. Actuation of the fire detection systems presents an alarm in the control room.

b. Water Supply and Mains

The water supply system has two fire water tanks. The capacity of each tank is equal to the largest expected flow rate for two hours, but not less than 300,000 gallons. Two redundant 100-percent capacity fire pumps are installed to supply fire water in the event of a failure of one pump or a loss of offsite power. An underground yard fire main is installed to furnish water to various locations.

c. Automatic Fire Suppression

Automatic fire suppression requirements are determined by a fire hazard analysis and as necessary for protection of safety-related systems and components. Water-based or gaseous fire suppression systems are selected based on the type of fire.

d. Manual Fire Suppression

Manual firefighting capability is provided to limit the extent of fire damage. Standpipes, hydrants, hoses, nozzles, and extinguishers are provided throughout the plant.

9.0 STEAM AND POWER CONVERSION SYSTEMS

9.1 Outline

The steam and power conversion system, shown in Figure 9.1-1, consists of the turbine-generator, main steam system, main feedwater system, emergency feedwater system, steam generator blowdown system, and other systems.

The functions of the steam and power conversion systems are as follows.

- The steam generated in the steam generator is taken to the turbine and electricity is generated.
- Water is fed to the steam generators.
- When the load on the turbine is reduced, residual heat generated in the reactor is still removed.
- When the reactor is shutdown, decay heat and other residual heat of the fission products in the reactor core are removed.
- If a main feedwater pipe is accidentally broken, water is fed to the steam generators even if the function of the normal feed water system is lost.

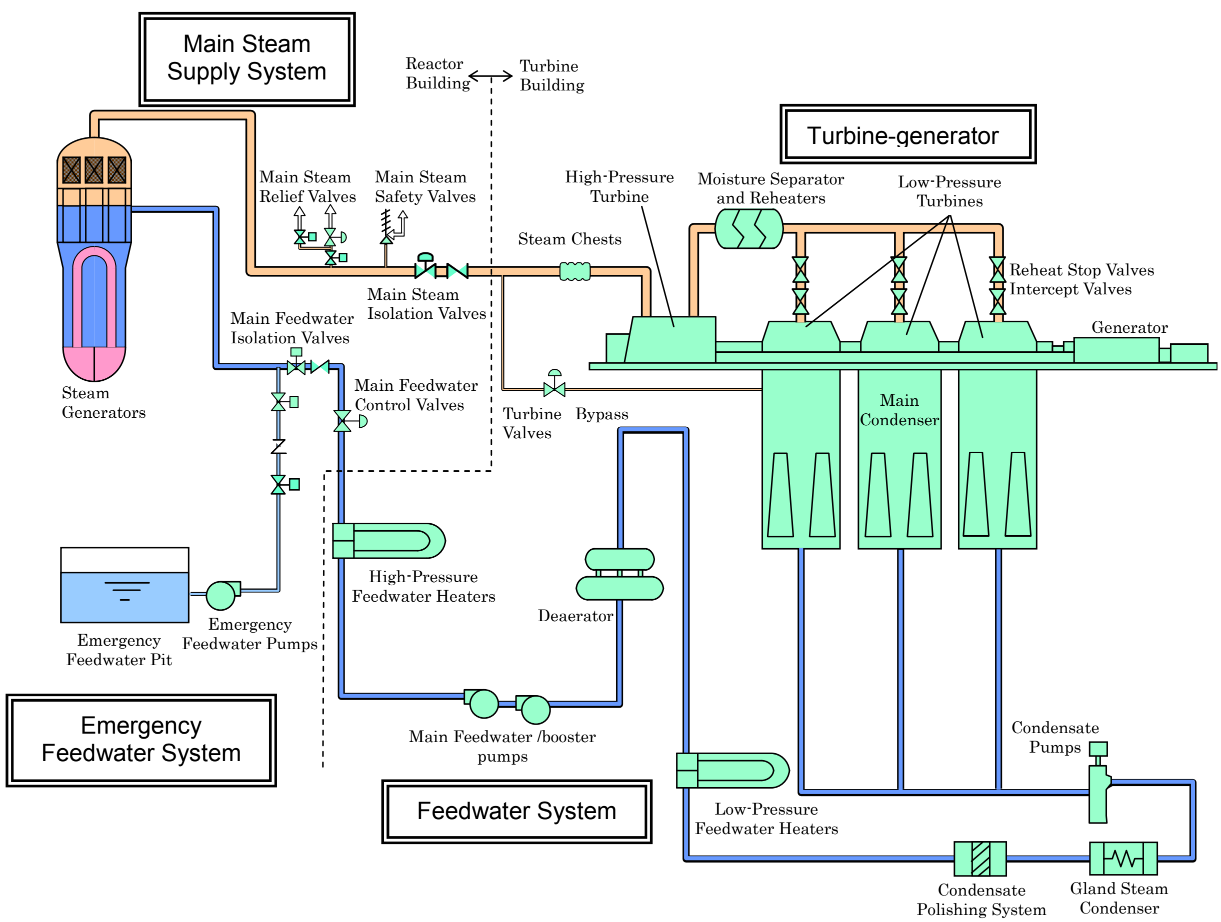


Figure 9.1-1 The Steam and Power Conversion System

9.2 Turbine-Generator

The turbine-generator consists of the turbine, generator, moisture separator and reheaters, steam valves, and their auxiliary systems (See Figure 9.2-1). The turbine is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work, which rotates the generator to generate electrical power.

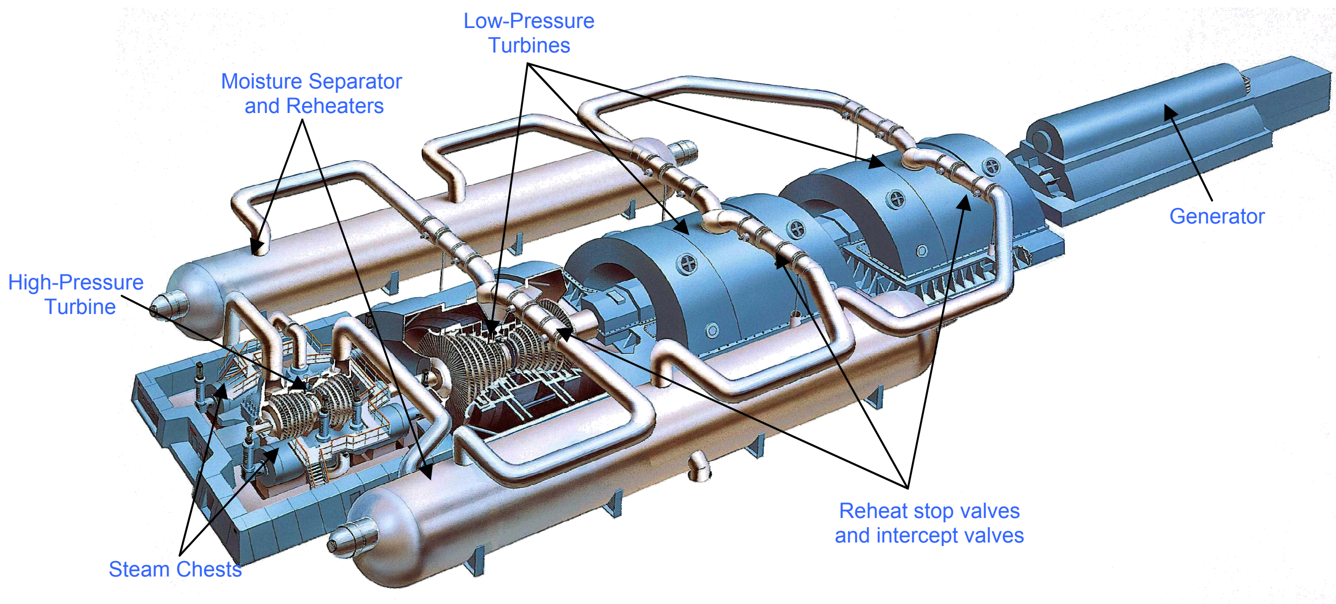


Figure 9.2-1 The View of the Turbine-Generator

The turbine is of a tandem compound type, 1,800 rpm machine and consists of a double-flow high-pressure turbine and three double-flow low-pressure turbines. One-piece designed low-pressure turbine rotor is used to improve the resistance against stress corrosion cracking and corrosion fatigue. The generator is of a four-pole water cooled type and is directly coupled to the turbine. Two moisture separators and reheaters are installed between the high-pressure turbine and low-pressure turbines to improve the thermal efficiency.
Steam flow from the steam generators is introduced to the two floor-mounted steam chests and then to the high-pressure turbine. Each steam chest contains two main steam stop valves and two governing valves which control steam flow. After leaving the high-pressure turbine, the steam flow is led to three low-pressure turbines through six reheat stop valves and intercept valves.

The turbine-generator is designed based on the following:

- The turbine-generator does not perform and support any safety-related function and therefore has no nuclear safety design basis.
- The turbine-generator is designed, manufactured and inspected including the measures for prevention of turbine failures under the proper quality control.

- The turbine-generator is designed in consideration of the following items so that safe operation can be achieved by various protective, monitoring, and control devices.
  - Full consideration is given to prevent vibration of the turbine-generator shaft and also if vibration occurs the turbine-generator is designed so that alarms are raised by vibration monitoring devices.
  - Steam valves, governor, and etc. are designed with redundancy so that extreme over speed of the turbine-generator can be prevented.
  - The turbine-generator is considered to account for anticipated abnormal (off-design) operating conditions.
  - The turbine-generator is designed to trip automatically under events of unsafe condition for the turbine-generator.
  - Measures are taken to protect the turbine-generator from damage to prevent the occurrence of turbine missiles.
  - Full consideration is given to the structures of piping, bearings, and etc. so that turbine-lubricating oil does not leak out. If the lubricating oil leaks it and may cause fire, a fire fighting system is provided not to spread the fire.

- The turbine-generator is designed to be able to perform periodical operation tests for the valves essential for overspeed protection, and other protective devices.

9.3 Main Steam System

The main steam system includes the main steam pipes from the steam generator outlets to the turbine inlet steam chests and equipment and piping connected to the main steam pipes.

A main steam isolation valve is installed on each main steam line from the four steam generators.

Branch connections downstream of the main steam isolation valves include the supply lines to the moisture separators and reheaters, the turbine gland seals, and the deaerator.

The turbine bypass subsystem takes steam directly from the steam generators to the condenser, bypassing the turbine. The subsystem has the following functions:

- Controlling the temperature and pressure of the reactor coolant within an acceptable range, without tripping the reactor, following a rapid decrease in load.
- Removing residual heat from the reactor coolant system and controlling the average temperature of the reactor coolant system to the no-load temperature following a reactor trip.
- Maintaining the reactor at hot standby or hot shutdown. Then cooling the reactor coolant system at the specified cooling rate.

The main steam relief valves and main steam safety valves are installed upstream of the main steam isolation valve. Those prevent excessive steam pressure and maintain cooling of RCS if the turbine bypass is not available.

A stop valve is installed to isolate the main steam relief valves if they behave abnormally.

The total capacity of the main steam safety valves exceeds 100% of the rated main steam flow.

Branch pipes for driving the turbine-driven emergency feed water pumps are connected upstream of the main steam isolation valves.

9.4 Other Steam and Power Conversion Systems

9.4.1 Main Feedwater System

The main feedwater system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The system is composed of the condensate subsystem, the feedwater subsystem, and portions of the steam generator feedwater piping.

The condensate subsystem takes water from the condenser and pumps it to the deaerator, using the condensate pumps. The feedwater subsystem takes suction from the deaerator and delivers it to the steam generators using the main feedwater pumps. The feedwater subsystem includes the safety-related piping and valves that deliver feedwater to the steam generators. The condensate and feedwater subsystems are located within the turbine building and the feedwater subsystem is located in the reactor building and containment.

The condensate subsystem consists of the condensate pumps, gland steam condenser, condensate polishing system, low-pressure feedwater heaters, and deaerator. The feedwater subsystem consists of the main feedwater and booster pumps and high pressure feedwater heaters.

The feedwater control valves, the feedwater bypass control valves, the steam generator water charging control valves, the feedwater isolation valves, and the associated instrumentation are installed on the feedwater piping to the steam generators.

The isolation of the main feedwater lines is secured under the any active single failure condition.

9.4.2 Emergency Feedwater System

The emergency feedwater System (EFWS), shown in Figure 9.4.2-1, consists of two motor-driven pumps, two steam turbine-driven pumps, two emergency feedwater pits, and associated piping and valves

The EFWS removes reactor decay heat and RCS residual heat through the steam generators following transient conditions or postulated accidents such as reactor trip, loss of main feedwater, steam or feedwater line breaks, steam generator tube rupture, and unavailability of the main feedwater system.

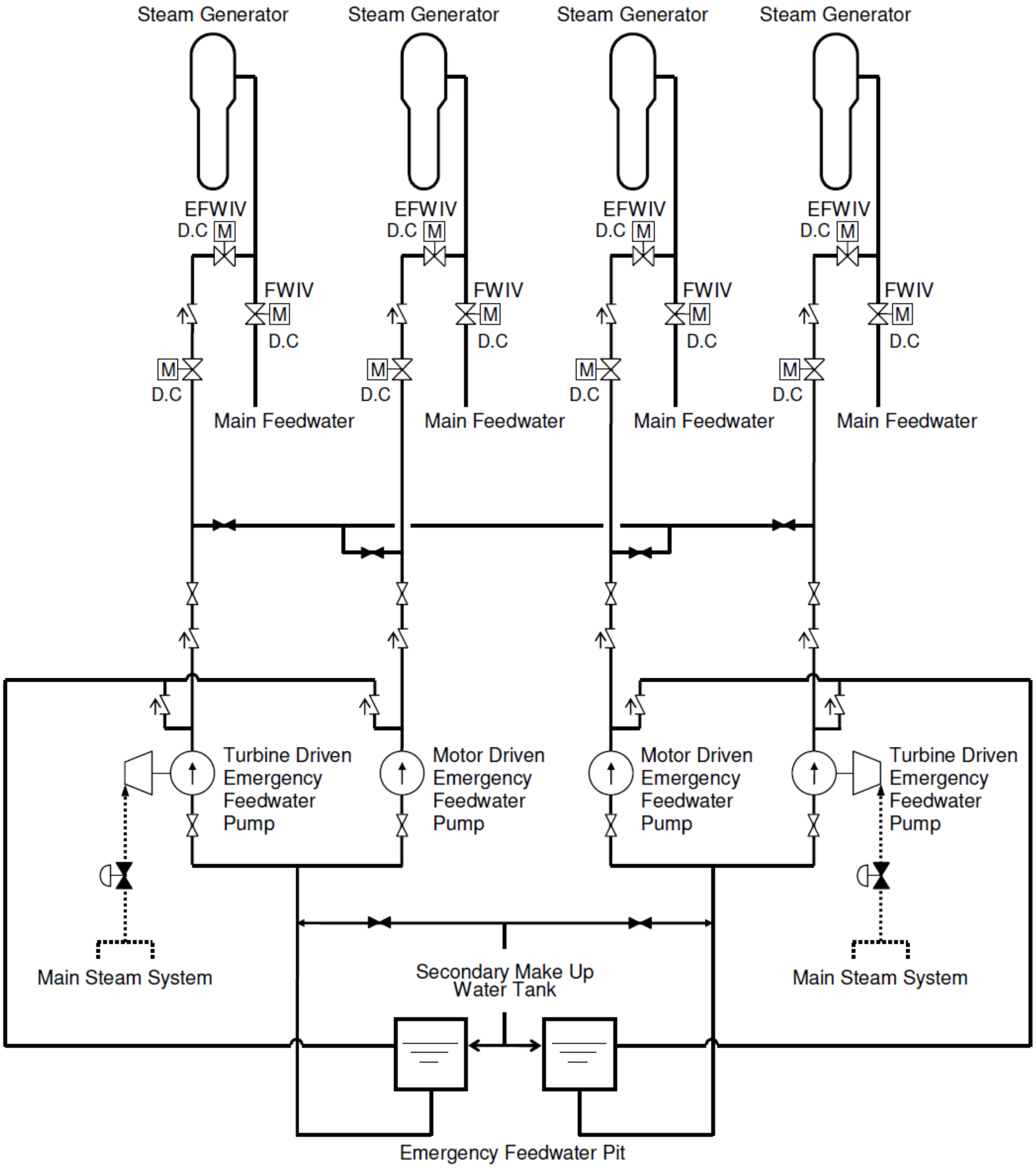
The EFWS design is based on the following;

- The EFWS maintains the capability of the steam generators to remove reactor decay heat and other RCS residual heat by converting the feedwater to steam, which is then discharged to the condenser or to the atmosphere.
- The EFWS satisfies the requirement that the pumps be powered by diverse power sources.
- The EFWS can perform all safety-related functions assuming a single failure in one train and a second train out of service for maintenance.
- The EFWS is automatically initiated by a steam generator water level low signal, etc.

The four pump configuration and the cross-connected discharge header allow the system to meet the single failure criteria with one pump out of service for maintenance.
Both motor-driven and turbine-driven pumps are used to satisfy the requirement that the pumps be powered by diverse power sources. Turbine-driven pumps are available at SBO condition.

The four emergency feedwater pumps take suction from two emergency feedwater pits.

Each pump is provided with a recirculation line, including a minimum flow line and a full flow line, leading back to the emergency feedwater pit. The minimum flow line ensures a minimum recirculation flow whenever the pumps are operating.



Legend

--------------------	Steam Line
FWIV	Feedwater Isolation Valve
EFWIV	Emergency Feedwater Isolation Valve
D.C	Direct Current

Figure 9.4.2-1 Emergency Feedwater System

9.4.3 Steam Generator Blowdown System

The steam generator blowdown system (SGBDS) is provided to control the steam generator secondary side water quality and to detect a leak or failure of a steam generator tubes.
The SGBDS includes blowdown lines and blowdown sample lines.

The system includes blowdown sample heat exchangers, pressure reducing valves, a radioactive process monitor, instruments, piping and valves.

The SGBDS has the following functions:

- Maintain acceptable secondary coolant water chemistry with monitoring through use of the blowdown sampling system
- Detect primary to secondary steam generator tube leakage

The SGBDS is designed based on the following:

- Provide continuous blowdown at a flow rate adequate to monitor and maintain impurity levels in the secondary water in the SGs within predetermined limits during normal plant operation.

- The blowdown lines are automatically isolated in the event of abnormal conditions within the blowdown system, the reactor coolant system, or the main steam system.

During normal operation, the blowdown water is purified by the polishing system.

Blowdown water is sent to the industrial waste or waste liquid management system when disposal is required instead of recovery.

The blowdown sample water is passed through the blowdown sample heat exchangers to be cooled and the pressure reducing valves to be depressurized for sampling.

The blowdown samples are used periodically to check the conductivity and pH of the steam generator secondary water to detect leakage or failure of a steam generator tube by radiation monitoring.

10.0 RADIOACTIVE WASTE MANAGEMENT

10.1 Liquid Waste Management System

The primary function of the liquid waste management system (LWMS) is to collect, segregate, and process various liquid wastes generated during normal, refueling, and maintenance operations. The liquid waste treated adequately is collected and monitored prior to discharge.

The LWMS is designed based on following:

- Liquid wastes are classified and segregated according to these feature and each liquid wastes is adequately treated.
- The concentration of radioactive materials is reduced as-low-as-reasonably-achievable by treating the waste with filtration and demineralization.
- Liquid waste is stored, reused, or held for release after treatment. The quantity of radioactive materials released to the environment is controlled in accordance with applicable regulations.
- Leakage and uncontrolled release of radioactive liquids is prevented.

The reactor coolant drains collected include the reactor coolant drains and leaks inside the containment. The liquid waste management system transfers them to the holdup tanks in the boron recycle system for processing and reuse.

The equipment and floor drains include auxiliary building drains and a chemical drain (except for strong acid drains) from the laboratory. These waste liquids are collected in the containment building sump and the auxiliary building sump tank and transferred to the waste holdup tanks. Prior to being discharged, processed waste is treated by an ion exchanger, and the radioactivity is determined to be within the acceptable limits. A radiation monitor is installed in the discharge line. In order to reduce the amount of borated water discharged to the environment, an evaporator system in the CVCS can be used.

The detergent drains, including personnel decontamination waste, and the strong acid drains are collected separately, treated if necessary, and monitored prior to discharge.

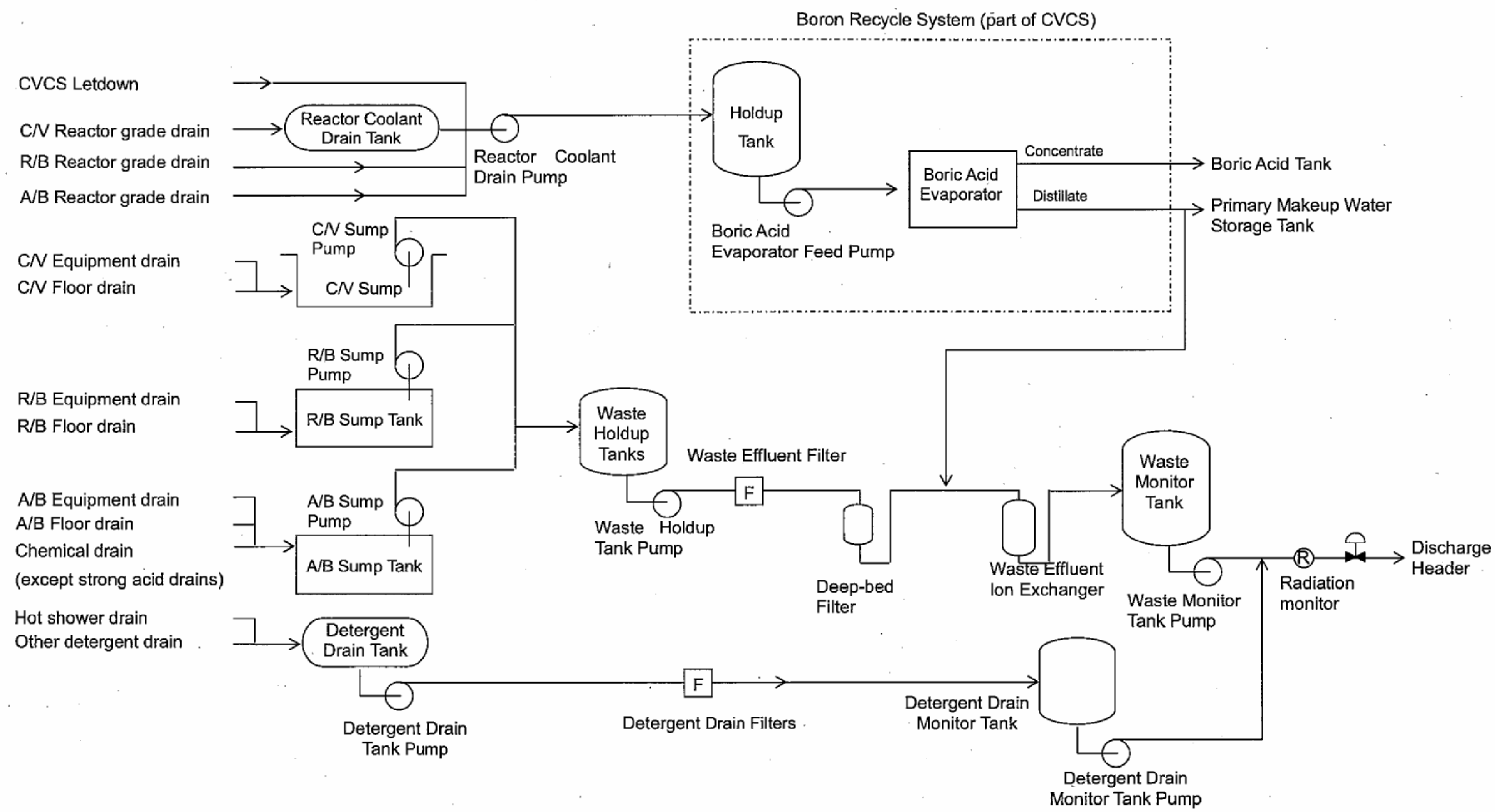


Figure 10.1-1 Liquid Waste Management System

10.2 Gaseous Waste Management System

The gaseous waste management system (GWMS) consists of waste gas compressor packages, gas surge tanks, and a noble gas holdup system.

The GWMS has the following functions:

- Treat nitrogen and hydrogen waste gases.
- Maintain the pressure of the noble gas holdup system slightly positive to reduce in-leakage.

The GWMS design is based on the following;

- Gaseous waste radioactivity is reduced by a noble gas holdup system using a charcoal absorption bed.
- The radioactivity and quantity of gaseous waste released to the environment is reduced to as-low-as-reasonably-achievable.
- The leakage of radioactivity out of the system and leakage of air into the system is minimized.

Nitrogen waste gas is compressed by the waste gas compressor packages in order to decrease the volume. It is then sent, intermittently, after passing through the gas surge tank, to an active carbon noble gas holdup system. After decay, the nitrogen waste gas is released from the vent stack through the charcoal filters of the ventilation system.

Hydrogen waste gas from the volume control tank containing fission products is released from the vent stack through an active carbon type noble gas holdup system and the charcoal filters of the ventilation system. The radioactivity level is monitored before release.

The treatment of the hydrogen waste gas prevents leakage which might lead to a hydrogen explosion. The atmosphere of each room where components containing hydrogen waste gas are located is continuously ventilated.

The waste gas compressor packages allow nitrogen waste gas to be temporarily stored in the gas surge tanks. Holdup towers of the active carbon noble gas holdup system have a quantity of active carbon sufficient to maintain released noble gas activity below permissible limits.

A dehumidifier system is used to prevent the active carbon holdup capacity from deteriorating.

Gas analyzers are provided to monitor hydrogen and oxygen concentration. The oxygen gas analyzer includes two oxygen gas concentration measurement devices for redundancy.