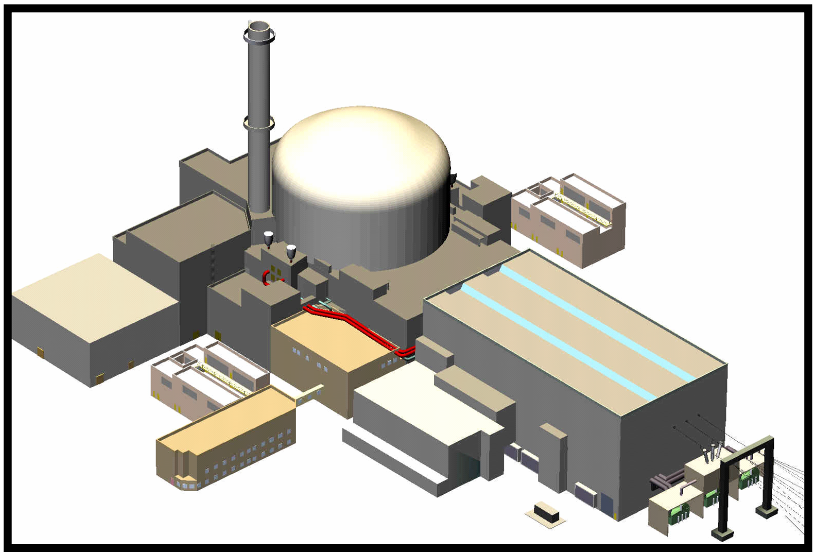

EPR Design Description

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AREVA

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Disclaimer

The design and engineering information contained in this document has been prepared by Framatome ANP, Inc., in connection with its request to the NRC for a pre-application review of the EPR nuclear power plant design.

The figures and numerical values used in this document are representative only, for discussion purposes, and should not be considered final. The information herein is subject to change and the final design information will be provided in the application for design certification.

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List of Acronyms

Acronym	Description
ALARA	As Low As Reasonably Achievable
ATWS	Anticipated Transient Without Scram
AVS	Annulus Ventilation System
CCW(S)	Component Cooling Water (System)
CDF	Core Damage Frequency
CGCS	Combustible Gas Control System
CHF	Critical Heat Flux
COTC	Core Outlet Thermocouple
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
DNB(R)	Departure from Nucleate Boiling (Ratio)
EBS	Extra Borating System
EDG	Emergency Diesel Generator
EDS	Electrical Distribution System
EFW(S)	Emergency Feedwater (System)
EPSS	Emergency Power Supply System
EPW	Explosion Pressure Wave
ESW(S)	Essential Service Water (System)
FDS	Fire Detection System
FPCS	Fuel Pool Cooling System
FPPS	Fuel Pool Purification System
GWPS	Gaseous Waste Processing System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilation, Air Conditioning
I&C	Instrumentation and Control
IRWST	In-Containment Refueling Water Storage Tank
LBB	Leak-Before-Break
LBLOCA	Large Break LOCA
LCO	Limiting Conditions for Operation
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LRF	Large Release Frequency
LSSS	Limiting Safety System Setting
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MFW(S)	Main Feedwater (System)
MHSI	Medium Head Safety Injection
MMI	Man-Machine Interface
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSRIV	Main Steam Relief Isolation Valve
MSRT	Main Steam Relief Train
MSRV	Main Steam Relief Valve
MSS	Main Steam System
MSSV	Main Steam Safety Valve

List of Acronyms
(continued)

Acronym	Description
NPSH	Net Positive Suction Head
NPSS	Normal Power Supply System
OPP	Overpressure Protection
PACS	Priority and Actuator Control System
PAMS	Post-Accident Monitoring System
PAR	Passive Autocatalytic Recombiner
PAS	Process Automation System
PDD	Power Density Detector
PDS	Pressurizer Depressurization System
PICS	Process Information and Control System
PRT	Pressurizer Relief Tank
PS	Protection System
PWR	Pressurized Water Reactor
PZR	Pressurizer
RBWMS	Reactor Boron and Water Make-Up System
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RCSL	Reactor Control, Surveillance and Limitation
RHR(S)	Residual Heat Removal (System)
RPV	Reactor Pressure Vessel
RSS	Remote Shutdown Station
SAHRS	Severe Accident Heat Removal System
SAS	Safety Automation System
SBLOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SGTR	Steam Generator Tube Rupture
SICS	Safety Information and Control System
SIS	Safety Injection System
SPND	Self-Powered Neutron Detector
SSE	Safe Shutdown Earthquake
SSS	Start-Up and Shutdown System
SSSS	Standstill Seal System
TSC	Technical Support Center
UCP	Upper Core Plate
UPS	Uninterruptible Power Supply
USP	Upper Support Plate
VCT	Volume Control Tank

1.0 INTRODUCTION

Framatome ANP, Inc. is planning to submit an application for EPR design certification under 10 CFR 52. The purpose of this document is to provide a summary of the design features of the EPR as part of the pre-application review for design certification. Additionally, it provides functional descriptions of the major systems, components and structures of the EPR. This document is for information only and is not intended for formal review and approval.

1.1 Design Philosophy

The EPR is an evolutionary Pressurized Water Reactor (PWR) designed by Framatome ANP, Inc., a jointly-owned subsidiary of AREVA and Siemens. It is a four-loop plant with a rated thermal power of 4,500 MWt. The primary system design, loop configuration, and main components are similar to those of currently operating PWRs, thus forming a proven foundation for the design.

The EPR is a global product with a basic set of common design features adaptable to the specific regulatory and commercial requirements of each country in which it is offered. The U.S. version of the EPR shares the basic set of design features such as four redundant trains of emergency core cooling, a containment and Shield Building, and a core melt retention system for severe accident mitigation, and it is adapted to meet applicable U.S. regulatory and commercial requirements.

The EPR design is based on the combined design and operating experiences of Framatome ANP and Siemens; French and German utilities and safety authorities; and with roots in the U.S. PWR fleet.

The EPR design philosophy is based on the following objectives related to the current generation of PWRs:

- Increase redundancy and separation
- Reduce core damage frequency (CDF)
- Reduce large release frequency (LRF)
- Mitigate severe accidents
- Protect critical systems from external events
- Improve man-machine interface (MMI)
- Extend response times for operator actions

A cornerstone of the EPR design philosophy, the principle of "defense-in-depth," has been improved on all levels, resulting in:

- Reductions in radiological consequences and accident initiator frequencies
- Favorable transient plant behavior
- Simplification of the safety systems and functional separation
- Elimination of common mode failures by physical separation and diverse back-up safety functions
- Increased redundancy and arrangement of the redundant trains into separated divisions
(The divisional separation is also extended to supporting features such as cooling water, power supply and Instrumentation and Control (I&C). The divisions are without interconnections, except for some normally-closed headers, up to the connection to the primary or secondary circuit. In the event of a

loss of one division by an internal hazard, the remaining divisions provide at least one full system capacity, taking into account a single failure.)

- Low sensitivity to failures, including human errors, by incorporation of adequate design margins; automation and extended times for operator actions; high reliability of the devices in their expected environment; and protection against common mode failures
(Response times for operator actions are increased by the larger Steam Generator (SG) inventory and Pressurizer (PZR) steam volume to ameliorate transients.)
- Less sensitivity to human errors by optimized digital I&C systems and information supplied by state-of-the art operator information systems
- Consideration of operating concerns in the design phase to simplify and optimize operation

The safety design of the EPR is based primarily on deterministic analyses complemented by probabilistic analyses. The deterministic approach is based on the “defense-in-depth” concept which comprises four levels:

1. A combination of conservative design, quality assurance, and surveillance activities to prevent departures from normal operation

2. Detection of deviations from normal operation and protection devices and control systems to cope with them (This level of protection is provided to ensure the integrity of the fuel cladding and of the Reactor Coolant Pressure Boundary (RCPB) in order to prevent accidents.)

3. Engineered safety features and protective systems that are provided to mitigate accidents and consequently to prevent their evolution into severe accidents

4. Measures to preserve the integrity of the containment and enable control of severe accidents

Low probability events with multiple failures and coincident occurrences up to the total loss of safety-grade systems are considered in addition to the deterministic design basis. Representative scenarios are defined for preventing both core melt and large releases in order to develop parameters for risk reduction features. A probabilistic approach is used to define these events and assess the specific measures available for their management. Consistent with international and U.S. probabilistic safety objectives, the frequency of core melt is less than 10^{-5}/reactor-year including all events and all reactor states.

The mean CDF design objective for the EPR is less than 10^{-5}/reactor-year considering the contribution from internal and external events (excluding seismic and sabotage) and for all operational modes.

The overall mean LRF of radioactive materials to the environment from a core damage event will be less than 10^{-6}/reactor-year.

Innovative features result in the low probability of energetic scenarios that could lead to early containment failure. Design provisions for the reduction of the residual risk, core melt mitigation, and the prevention of large releases are:

- Prevention of high pressure core melt by high reliability of decay heat removal systems, complemented by primary system Overpressure Protection (OPP)
- Primary system discharge into the containment in the event of a total loss of secondary side cooling
- Features for corium spreading and cooling

- Prevention of hydrogen detonation by reducing the hydrogen concentration in the containment at an early stage with catalytic hydrogen recombiners
- Control of the containment pressure increase by a dedicated Severe Accident Heat Removal System (SAHRS) consisting of a spray system with recirculation through the cooling structure of the melt retention device
- Collection of all leaks and prevention of bypass of the confinement, achieved by a double-wall containment

External events such as an aircraft hazard, Explosion Pressure Wave (EPW), seismic events, missiles, tornado, and fire have been considered in the design of Safeguard Buildings and the hardening of the Shield Building.

1.2 Overview of the EPR Design

The EPR is furnished with a four-loop, pressurized water, Reactor Coolant System (RCS) composed of a reactor vessel that contains the fuel assemblies, a PZR including control systems to maintain system pressure, one Reactor Coolant Pump (RCP) per loop, one SG per loop, associated piping, and related control and protection systems. The RCS is described in detail in Chapter 3.

The RCS is contained within a concrete containment building. The containment building is enclosed by a Shield Building with an annular space between the two buildings. The pre-stressed concrete shell of the Containment Building is furnished with a steel liner and the Shield Building wall is reinforced concrete. The Containment and Shield Buildings comprise the Reactor Building. The Reactor Building is surrounded by four Safeguard Buildings and a Fuel Building (see Figure 1-1). The internal structures and components within the Reactor Building, Fuel Building, and two Safeguard Buildings (including the plant Control Room) are protected against aircraft hazard and external explosions. The other two Safeguard Buildings are not protected against aircraft hazard or external explosions; however, they are separated by the Reactor Building, which restricts damage from these external events to a single safety division.

Redundant 100% capacity safety systems (one per Safeguard Building) are strictly separated into four divisions. This divisional separation is provided for electrical and mechanical safety systems. The four divisions of safety systems are consistent with an N+2 safety concept. With four divisions, one division can be out-of-service for maintenance and one division can fail to operate, while the remaining two divisions are available to perform the necessary safety functions even if one is ineffective due to the initiating event.

In the event of a loss of off-site power, each safeguard division is powered by a separate Emergency Diesel Generator (EDG). In addition to the four safety-related diesels that power various safeguards, two independent diesel generators are available to power essential equipment during a postulated Station Blackout (SBO) event—loss of off-site AC power with coincident failure of all four EDGs.

Water storage for safety injection is provided by the In-containment Refueling Water Storage Tank (IRWST). Also inside containment, below the Reactor Pressure Vessel (RPV), is a dedicated spreading area for molten core material following a postulated worst-case severe accident.

The fuel pool is located outside the Reactor Building in a dedicated building to simplify access for fuel handling during plant operation and handling of fuel casks. As stated previously, the Fuel Building is protected against aircraft hazard and external explosions. Fuel pool cooling is assured by two redundant, safety-related cooling trains.

1.3 Comparison with Currently Operating PWRs

Key EPR design parameters are compared with those of a typical U.S. four-loop plant, a French N4 plant, and a German KONVOI plant in Table 1-1.

Table 1-1
Comparison of Key EPR Design Parameters

Parameter	EPR	Current U.S. 4-Loop	French N4	German KONVOI
Design life (yrs)	60	40	40	40
Thermal power (MW)	4,500	3,411	4,250	3,850
Net electrical power (MW)	1,600	1,100	1,450	1,365
Efficiency (%)	36	32	34	35
Hot leg temp (°F)	624	617	626	619
Cold leg temp (°F)	564	559	558	558
Best Estimate RCS flow per loop (gpm)	125,000	98,000	109,420	100,450
Primary system pressure (psia)	2,250	2,250	2,250	2,290
Steam pressure (psia)	1,118	990	1,030	935
Steam flow per loop (Mlb/h)	5.1	3.8	4.9	4.2
Total RCS volume (ft^3)	16,245	12,600	14,830	14,210
PZR volume (ft^3)	2,650	1,800	2,120	2,300
SG inventory (lbm)	182,000	106,000	134,500	101,000
Number of fuel assemblies	241	193	205	193
Fuel lattice	17 x 17	17 x 17	17 x 17	18 x 18
Active fuel length (ft)	13.78	12.00	14.01	12.80
Rods per assembly	265	264	264	263
Average linear heat rate (kW/ft)	5.01	5.58	5.46	5.09
Best estimate peak operating linear heat rate (kW/ft)	13.72	13.95	13.65	12.77
Number of control rods	89	53	73	61
Primary volume/power (ft^3/MW)	3.61	3.69	3.49	3.69
Secondary mass/power (lbm/MW/SG)	40.4	31.1	31.6	26.2
PZR steam-to-RCS liquid volume	0.070	0.061	0.061	0.065

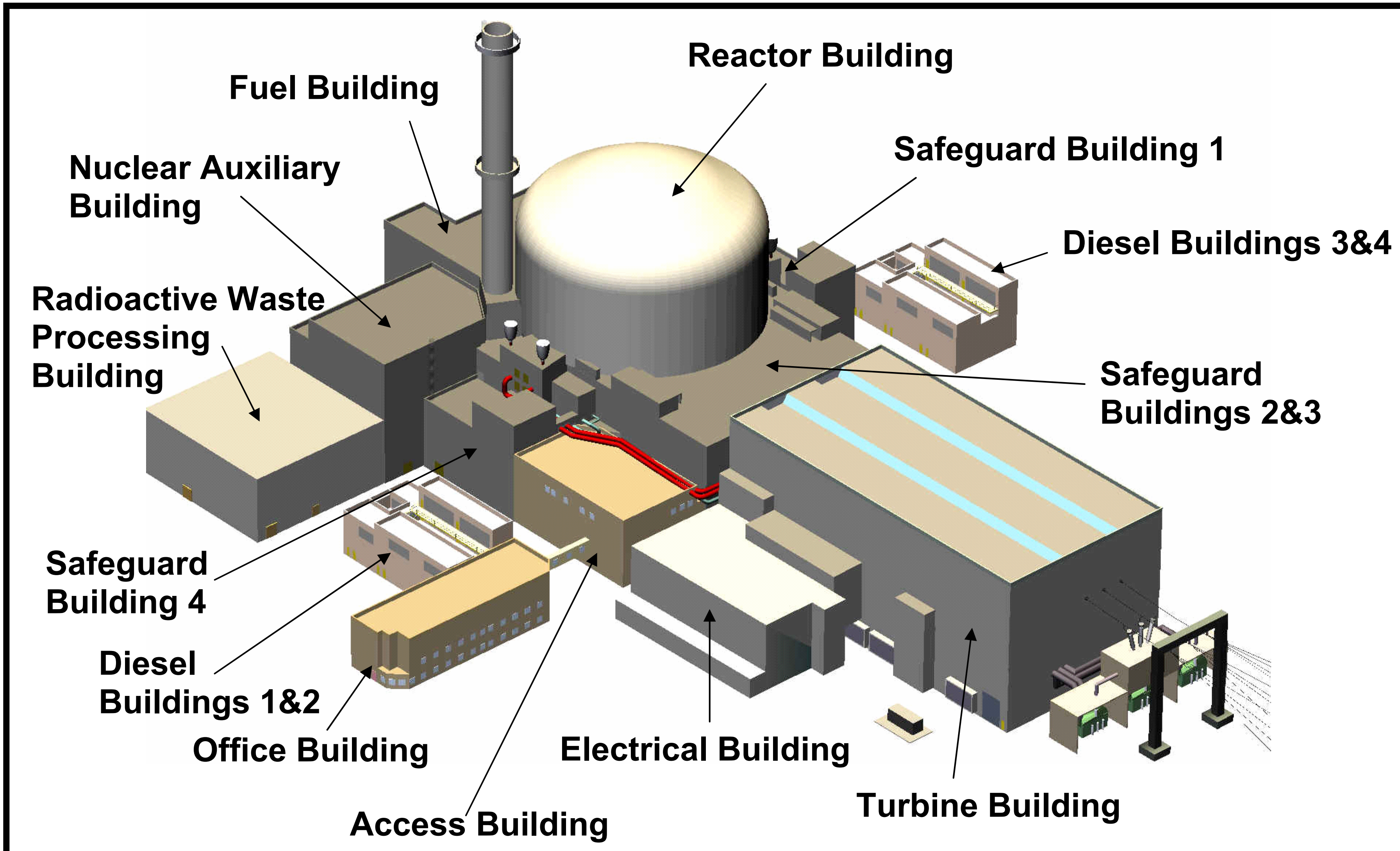


Figure 1-1
Plant Configuration

2.0 CORE DESIGN

2.1 Overall Features

The EPR has a rated core thermal power of 4,500 MWt. The main features of the core and its operating conditions result in a high thermal efficiency of the plant, low fuel cycle costs, and flexibility for extended fuel cycle lengths.

The reactor core consists of an array of 241 fuel assemblies with the following characteristics:

- A 17 x 17 lattice composed of 265 fuel rods mechanically joined in a square array
- Optimized and proven fuel rod design parameters
- Enrichment of up to 5 wt% ^{235}U
- Gd_2O_3 integral burnable poison with Gd concentration of 2 wt% to 8 wt%
- Highly corrosion-resistant and low-growth M5™ cladding and tubing
- Monobloc™ guide thimbles to increase structural strength
- Low growth M5™ intermediate spacers
- Alloy 718 end spacers providing improved fuel rod support and flow-induced fretting resistance
- Debris-resistant robust FUELGUARD™ bottom nozzle
- Removable top nozzle for ease of assembly repair

Preliminary values of the key core parameters are given in Table 2-1.

2.2 Fuel Assemblies

The fuel rods are mechanically restrained axially and radially in the fuel assembly structure by eight M5™ intermediate HTP grids and two Alloy 718 end HMP spacer grids. The grids have integrated curved flow channels for promoting the mixing of the coolant and improving the heat transfer between the cladding and the coolant. The intermediate grids are axially constrained with welds to the guide thimbles. The end grids are axially constrained by sleeves that are welded to the guide thimbles above and below the grids.

Twenty-four positions in the 17 x 17 array are equipped with M5™ Monobloc guide thimbles, which are joined to the grids and the top and bottom nozzles. The guide thimbles are used as locations for Rod Cluster Control Assemblies (RCCAs) and stationary core component assemblies such as thimble plug assemblies and neutron source assemblies. The guide thimbles are also used as locations for the movable or fixed in-core instrumentation. Each guide thimble also utilizes a quick-disconnect connection for efficient removal and replacement of the top nozzle.

The top nozzle of the fuel assembly is the structural element that interfaces with the top core plate. The top nozzle also supports the holddown springs of the fuel assembly, which are used to prevent hydraulic lift-off of the fuel assembly during operation. The holddown springs are Alloy 718 leaf springs that are bolted in two diagonal corners of the top nozzle. The top nozzle also incorporates an appropriate interface for the fuel handling equipment, RCCA, and stationary core components. The 180° rotational symmetry of the fuel assembly provides additional flexibility for fuel management.

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

Illustrations of the fuel assembly are provided as follows:

- Figure 2-1: Radial cross section of a fuel assembly
- Figure 2-2: Fuel assembly in full length view
- Figure 2-3: Photographs of the top nozzle and bottom nozzle of the fuel assembly

Key fuel assembly characteristics are given in Table 2-2.

2.3 Rod Cluster Control Assemblies and Reactivity Control

The core has a fast shutdown system consisting of eighty-nine RCCAs. All RCCAs are of the same type, consisting of twenty-four individual and identical absorber rods fastened to a common spider assembly. These rods are constructed of stainless steel tubing that contains neutron absorbing materials. When inserted, they cover nearly the complete active fuel assembly length. The material of the rods is a hybrid Ag-In-Cd (AIC) alloy and B_4C design, with AIC in the lower part and B_4C in the upper part.

The characteristics of the RCCAs and control rod drive mechanisms are given in Table 2-3.

The core is cooled and moderated by light water at a pressure of 2250 psia. The coolant contains boron (^{10}B enriched) as a neutron absorber. The boron concentration in the coolant is varied to control slow reactivity changes necessary for compensating xenon poisoning or burn-up effects during power operation and for compensating large reactivity changes associated with large temperature variations during cool down or heat-up phases.

2.4 Nuclear and Thermal-Hydraulic Designs

The major features of the EPR core (e.g., the type of fuel assembly, the size of the core, the heavy reflector, and the key operating parameters) are designed to maximize plant efficiency, margins, and flexibility of the fuel cycle length, while minimizing fuel cycle cost by improving the neutron economy and increasing the fuel burn-up.

The EPR core data are the result of nuclear and thermal-hydraulic design analyses performed with the following boundary conditions:

- Batch burn-ups of at least 55 GWd/MTU are expected, and the licensed limit for a single rod is 62 GWd/MTU.
- Margins are available to provide flexibility for in-core fuel management. The most likely fuel shuffle scheme used will be in-in-out; however, other schemes are possible (i.e., out-in-in and in-out-in)
- Cycle lengths from 18 months to 24 months
- Possible end-of-cycle coastdown operation of up to 70 effective full power days

Under these boundary conditions, the general safety requirements and more specific core-related design criteria are discussed below.

For both Condition I events (normal operation) and Condition II events (anticipated operational occurrences—once per year during plant operation), there is no loss of integrity of the fuel. For Condition II events and events of lower probability of occurrence that result in a plant shutdown, shutdown capabilities will bring the plant to a subcritical condition and maintain it in a safe shutdown state through the use of safety-related equipment.

For Condition I events, the controls, surveillance, and limitation systems automatically maintain the plant within Limiting Conditions for Operation (LCO) postulated for accident analyses and thus well below the integrity limits of the fuel cladding. These systems rely on efficient, accurate and reliable instrumentation concepts inherent in the design of the EPR.

For Condition II events (anticipated operational occurrences), automatic countermeasures (limitations) are actuated to terminate abnormal transients at an early stage and return the plant to Condition I without a reactor trip when possible. The protection trip function relies on the accurate monitoring of essential core parameters and is actuated only in the absence of operator response or when automatic control actions do not succeed in terminating the transient.

Safety analyses for Condition I and II events are performed up to a safe state. Two states are defined: the controlled state and the safe shutdown state. For each Condition I, II, and III event, it must be demonstrated that the controlled state can be reached. (Condition III events are events that are expected to occur infrequently, if at all, during the lifetime of the plant.) For the transition from the controlled state to the safe shutdown state (if required for Condition II and III events) one analysis per set of similar Condition II and III events with respect to transient behavior is performed.

Additional core-specific design criteria are defined below.

- Condition III events shall not cause more than a small fraction of the fuel elements in the reactor to be damaged (i.e., less than 1% of the rods shall enter Departure from Nucleate Boiling (DNB) for infrequent events and less than 10% of the rods shall enter DNB for events that are expected to occur less than once during the life of the reactor).
- Condition IV events (events that are not expected to occur but which are evaluated to demonstrate the adequacy of the design) shall not cause a release of radioactive material that exceeds the guidelines of 10 CFR 100.
- For Condition III and IV events, the fuel melting at the hot spot shall not exceed 10% in volume. This criterion translates to a 10 % area limit at the axial elevation of the power peak.
- For Condition II events, the maximum linear heat generation rate shall be limited to meet the fuel clad, fuel rod, and fuel centerline temperature specified acceptable fuel design limits. These limits are typically a function of the fuel rod burnup with a safety analysis accounting for irradiation-induced changes.
- For fast reactivity transients, the fuel enthalpy shall be limited to 220 cal/g and 200 cal/g for unirradiated and irradiated fuel, respectively.

2.4.1 Neutronic Design

With respect to power distribution, the nuclear design basis is described below.

- The maximum linear heat generation rate remains below the limit, which is ensured by a limitation function (High Linear Power Density LCO [$HLPD_{LCO}$]).

- The maximum local power under abnormal conditions, including the maximum overpower condition, does not cause local fuel melting.
- The fuel will not operate with a power distribution that violates the DNB design basis for Condition I and II events, including the maximum overpower condition.
- The power histories resulting from the fuel management are consistent with the assumptions for mechanical fuel rod design.

Table 2-1 through Table 2-4 provide data for key core components for the neutronic analysis.

Figure 2-4 and Figure 2-5 illustrate the available margins based on preliminary calculations. The margins between the best estimate values for F_q and $F^N_{\Delta H}$ and the LCO setpoints are derived from the corresponding LCO-limit-values (maximum peak power density and minimum DNBR) considered in accident analyses where $F_q = 3.0$ (Figure 2-4) and $F^N_{\Delta H} = 2.2$ (Figure 2-5). Note that the value of $F^N_{\Delta H}$ is roughly derived from the $DNBR_{LCO}$ value, but the $F^N_{\Delta H}$ derivation must conservatively consider a wide range of axial power distributions. The available margins are conservatively reduced to address the provisions needed to cover the range of normal operating conditions (e.g., a range of possible axial distribution, control rod insertions, and xenon redistribution effects).

Preliminary studies indicate significant margins exist to allow accommodation of various fuel management schemes for operating at a power level of 4,500 MWt. The slightly modified core characteristics resulting from those management schemes will be encompassed by the enveloping data used for the accident analyses.

Boron Concentration, Reactivity Coefficients, Shutdown Efficiency

The core design meets the safety objectives of providing stabilizing reactivity coefficients and an effective shutdown system.

The number of Gd rods and concentrations are adjusted to limit the critical boron concentration at beginning-of-cycle, hot zero power, all rods out; and to ensure that the moderator temperature coefficient under these conditions remains negative (including consideration of uncertainties).

Design of the Shutdown System

In the event of an accidental cooldown, the shutdown system maintains subcriticality assuming one stuck rod after actuation of the reactor trip until conditions are reached for automatic boration with safety-related systems. The criterion is to reach subcriticality after reactor trip and cool down the RCS to 500°F with consideration of an additional 500 pcm margin for fuel management flexibility.

The shutdown system is sized to meet the initial conditions of accidents in accordance with the defined LCO as well as uncertainties in the design tools and measurement systems. LCO considered for this purpose include:

- The reactor power
- The initial insertion of control rods (maximum required negative reactivity inserted for control purposes)
- The initial axial power and Xe distributions

A possible RCCA pattern is shown in Figure 2-6. Calculating minimum shutdown margins at bounding end-of-cycle conditions demonstrated that fewer RCCAs would be sufficient. The smallest shutdown margins are found for out-in-in types of loadings. (Note: U.S. fuel management practices typically do not use out-in-in loadings for economic reasons.) For in-in-out loadings, which may be utilized for cost saving reasons, the shutdown margins would be improved.

The shutdown and boration systems are designed to satisfy long-term subcriticality requirements after reactor trip for all types of shutdown conditions, including operational and accidental scenarios.

The plant is designed to operate with up to ~28.5 wt% of ^{10}B for UO_2 cores with the highest ^{235}U enrichment, allowing operation at coolant boron concentrations <1,400 ppm at BOC, HZP without Xe.

The theoretical boron concentration of the IRWST is in the range of 2,600 ppm natural boron and ~1,600 ppm based upon the use of enriched boron. The actual IRWST boron concentration will be somewhat higher to account for uncertainties and partial dilutions.

2.4.2 Thermal-Hydraulic Design

The objective of the thermal-hydraulic design is to provide adequate heat transfer to the fuel rods and control components. This ensures that the heat removal by the RCS or by the Safety Injection System (SIS) meets the operational targets for available margins and that safety design targets are met.

The key thermal-hydraulic design parameter is DNB. DNB depends on parameters such as local geometry of the heated component; mean heat generation; radial and axial power distributions; coolant temperatures and pressures; and local flow. The DNB ratio (DNBR) is based on an empirical correlation that is a surface fit to experimentally-derived Critical Heat Flux (CHF) data. For the EPR, the data base for the CHF correlation will include uniform and non-uniform axial power distributions for both unit cell and guide tube configurations. A unit cell is the region formed by four fuel rods. A guide tube configuration is formed by three fuel rods and a control rod guide tube.

Thermal Margin Design Basis

The thermal margin design basis provides a 95% probability (at a 95% confidence level) that DNB will not occur on the limiting fuel rods during normal operation and anticipated transients (Condition I and II events).

By preventing DNB, adequate heat transfer between the fuel cladding and the reactor coolant is ensured. The prevention of DNB relies on appropriately defined limitation and protection functions based upon on-line DNBR calculations. These functions use fixed in-core flux measurements to reconstruct the local thermal hydraulic conditions and calculate the minimum DNBR (MDNBR).

The reconstruction uncertainties, as well as uncertainties related to the fuel geometry and thermal-hydraulic model, are considered for determining the setpoints. The setpoint criterion is a 95% probability at a 95% confidence level that the DNB will not occur when the on-line calculated DNBR threshold is reached or when other protective functions have been actuated.

The methodology used for determining setpoints with respect to DNB depends on the type of reactor protection channels used to protect the core. Three types of transients are considered as described below.

- **Transients for which the DNBR protection is sufficient (Type 1) --** These transients are relatively slow and DNB is avoided by setting the DNBR threshold of the DNBR protection channel at a limit that guarantees avoidance of DNB.

- **Transients that occur at power, but for which the DNBR protection channel is not sufficient (Type 2) --** These transients exceed the response time of the protection channel. For these transients, the protection is based on specific event detection (e.g., “low pump speed” for detection of loss of RCS flow). Once the setpoints for these specific protections have been defined, the MDNBR during the corresponding transient(s) depends only on the initial condition(s) at which the event occurs. LCOs that define the worst initial conditions for these events are defined by appropriate accident analyses, thus preventing DNB limits from being exceeded during the transient. A surveillance/limitation function ensures that the actual DNBR always exceeds the DNBR threshold fixed for initial conditions of accidents (also called $DNBR_{LCO}$ -- mainly with regard to the loss of flow event). This setpoint takes into account all the uncertainties linked to the fuel geometry and those related to the surveillance/limitation functions.
- **Transients occurring at very low power or at subcritical conditions or leading to re-criticality at low temperature conditions (Type 3) --** For these events, the methods previously defined do not apply and specific protection functions and safety systems must intervene. The focus of the corresponding accident analyses for these transients is to characterize the protection and safety systems to ensure that the minimum DNBR limits that guarantee the integrity of the fuel are not violated during the accident.

This global approach along with some preliminary representative results of these sizing analyses are presented in Figure 2-5. The distance between nominal DNBR and the $DNBR_{LCO}$ illustrates the thermal-hydraulic margins.

Fuel Temperature Design Basis

For Condition I and II events, there is at least a 95% probability (at a 95% confidence level) that the fuel melting temperature is not exceeded in any part of the core. The melting temperature of UO_2 corresponds to ~5,100°F for unirradiated UO_2, decreasing by ~10°F per 10,000 MWd/MTU.

Precluding fuel melting preserves the fuel geometry, thus eliminating the possible adverse effects of molten fuel interacting with the fuel cladding.

Core Flow Design Basis and Thermal-Hydraulic Main Characteristics

Core cooling is ensured by the primary coolant flow. The coolant flow rate and the primary coolant temperature are optimized for maximum heat transfer to the secondary side, while ensuring acceptable thermal hydraulic conditions in the core.

Three different flow rates are considered for the design:

- A “thermal” flow rate, minimizing the flow entering the core, is used for the core’s thermal design. This flow rate considers all the uncertainties or allowances in a conservative manner for core cooling.
- A “best estimate” flow rate is used to predict the secondary side pressure under best-estimate assumptions and to design the RCPs.
- A “mechanical” flow rate, maximizing the flow entering the core, is used for the mechanical design of the components. This mechanical flow rate considers the uncertainties and allowances in a conservative manner.

A portion of the flow bypasses the fuel rods through the thimble tubes of the fuel or is used to cool the heavy reflector and establishes mean temperature conditions in the upper dome of the RPV. Evaluation of the thermal-hydraulic design flow considered a total conservative bypass between 5 and 6% of system flow.

The main thermal and hydraulic characteristics of the EPR are shown in Table 2-5.

Core Control Principles/Control Rods/Control Rod Maneuvering

The reactivity of the core is controlled at power by changing the boron concentration and inserting RCCAs.

As a general rule, slow reactivity variations resulting either from changes of the xenon concentration (e.g., following daily load variations) or from the evolution of the burn-up are compensated by adjusting the boron concentration. Faster reactivity changes necessary for adjusting the power level are obtained by modifying the RCCA insertion.

Core control relies on three main closed-loop controls:

- Reactor coolant temperature control
- Axial power distribution control
- Control of RCCA position with respect to shutdown efficiency

The essential features of the core control systems are listed below.

- All the RCCAs are "black" rods (i.e., highly neutron absorbing).
- At rated power, no control rods are deeply inserted.
- The power defect reactivity due to power level variations is essentially covered by movement of control rod groups (temperature control).
- The axial power distribution is primarily controlled at high power levels by moving slightly inserted control groups. Axial power distribution at lower power levels is controlled by adjusting the overlap of control groups that have been previously inserted.
- To ensure sufficient shutdown margins, the amount of negative reactivity inserted by RCCAs is automatically adjusted by modifying the boron concentration in the coolant.
- Rod dropping is used for fast power reductions in the event of perturbed operating conditions.
- The core control is fully automated.
- The operator may optimize plant operation by adjusting the setpoints appropriately.

2.5 Core Instrumentation

The safety approach for the protection of the core relies partly on the capacity to predict and measure the nuclear power level (or level of neutron flux) as well as the three-dimensional power distribution. The measurement of the nuclear power level (or neutron flux level) is performed by loop temperature instrumentation and wide range excore flux (power) instrumentation, as is classically done on PWRs. The capacity to predict and measure the three-dimensional power distribution relies on two types of in-core instrumentation:

- Movable instrumentation, also called the reference instrumentation. This instrumentation is used principally for validating the core models used for the core design and for calibrating the other sensors that are used for core surveillance and core protection purposes.

- Fixed instrumentation used for delivering the necessary on-line information to the different core surveillance and core protection systems. This instrumentation initiates the appropriate actions or countermeasures when anomalies are detected or when predefined limits are exceeded.

Excore Instrumentation

During power operation, the nuclear power level is measured principally by a four-fold redundant primary heat balance that relies on temperature measurements in the cold and hot legs of the RCS loops. This primary heat balance is used with excore neutron flux measurements (power range), which have a short response time, to provide an efficient system for fast and slow core power change detection.

The core is also monitored and protected when operated at very low power levels or in subcritical conditions. The appropriate surveillance and protective functions rely on redundant excore neutron flux channels covering approximately 9 to 10 decades of the total neutron flux range below the nominal power.

In-Core Instrumentation

In-core instrumentation is top-mounted and consists of:

- An aeroball measurement system as the movable reference core instrumentation
- A quantity of fixed in-core detector fingers containing axially distributed Self-Powered Neutron Detectors (SPNDs), used in four-fold redundant channels for core surveillance and core protection purposes during power operation
- A quantity of Core Outlet Thermocouples (COTCs) having the same radial locations as the SPNDs, used mainly for measuring the margin-to-saturation in post-accident or degraded thermal hydraulic conditions

The Aeroball System

The system for power distribution assessment is an aeroball system. Stacks of vanadium alloy steel balls, inserted from the top of the reactor vessel, are pneumatically transported into the reactor core inside guide thimbles of the fuel assemblies. This system is simple and reliable. The guide tubes for the balls have a small inner diameter (0.0079 in); their bend radii are small (only a few inches); and there are no major constraints for locating the measurement room and routing the tubing. The time periods necessary for a flux measurement are 3 minutes for activation followed by 5 minutes for activity measurements. This system therefore allows flux-mapping measurements in time intervals of 10 to 15 minutes.

Figure 2-7 shows a schematic of the aeroball system.

The aeroball probes are carried and distributed over the core by 12 in-core lances. Each in-core lance bears four aeroball fingers and one SPND finger. Aeroball probes are distributed radially throughout the core. Figure 2-8 shows the radial distribution arrangement of in-core instrumentation fingers. Figure 2-9 shows the in-core instrumentation inside a longitudinal cross section of the RPV.

After activation, the activity of the ball-columns is measured in a measuring table by means of thirty-six surface-barrier semi-conductors per column. These detectors are equally distributed over the active length and integrate the activity over a length of ~2.4 in.

The electronic part of the measuring system consists of pulse counters. This technique, combined with the short half-life of the V^{52} isotope (3.7 min.) that serves as the indicator, restricts the range of power over which accurate three-dimensional flux mapping is possible. In practice, acceptable two-dimensional flux maps can be obtained at ~5% reactor power and accuracy necessary for three-dimensional flux maps is reached at approximately 30% reactor power.

Fixed In-Core Instrumentation

The fixed in-core instrumentation consists of SPNDs and COTCs. The SPNDs have a fast response time. At twelve radial locations, six SPNDs are placed in a Power Density Detector (PDD)-finger to cover the core. Each of the yokes of the aeroball system contains one PDD-finger that is replaceable should a detector become defective. The number and the distribution of the SPNDs within the core allow the system to detect and assess local power density increases caused by flux and power redistributions that occur under non-steady-state conditions. The in-core detector system design also makes allowance for a proper “functional” signal redundancy. As core burnup progresses, the power-to-signal ratio and the reference power distribution changes. Therefore, calibration of the SPNDs to reference conditions is performed at regular intervals. Reference values for this calibration are local power and hot channel power densities within a section of the core volume assigned to each SPND available from the aeroball system. Under perturbed conditions, SPNDs change in line with the neutron flux at the detector location. Consequently, the calibrated SPND signals are able to accurately follow or track the highest linear heat generation rates distributed over the core. These signals are used for core control, limitation, and protection purposes. They are processed together with other selected process variables to yield continuous monitoring signals representative of core conditions.

Specific core surveillance and core protection systems are based on digital equipment that calculates the relevant limiting core parameters. These systems rely on models or data processing (e.g., three-dimensional core model for core surveillance and simplified algorithms for core protection) for calculating the safety-relevant state parameters of the core, such as peak power, DNBR, Linear Heat Generation Rate (LHGR) margin, axial offset, and core power tilt.

These systems operate properly for relatively slow core-related Condition I events and are introduced in order not to overly penalize the operation of the plant. As a consequence, they are able to operate without significant loss of accuracy under conditions such as dropped control rods, RCCA misalignments, and single failure.

Core surveillance and core protection systems are also equipped with sufficient redundant and diverse information including:

- RCCA positions
- Temperature, flow rates, power level
- Axial power distributions
- Radial power distributions

This allows these systems to monitor Condition I events, even if there is a partial unavailability of sensors.

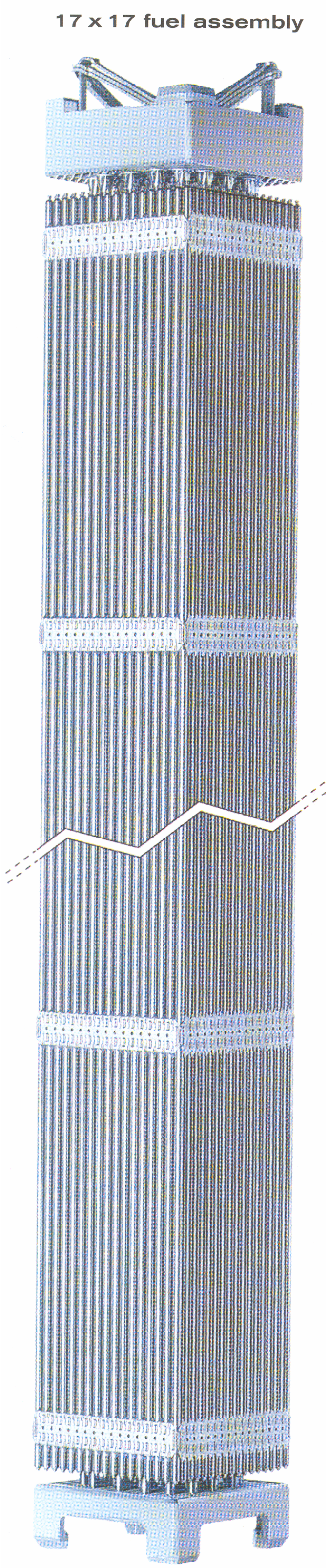
The available information necessary for characterizing the power distribution and thermal-hydraulic conditions is distributed to all the surveillance and protective channels simultaneously. This is possible due to the intrinsic system redundancy; the overlapping of information; the diversity of sensors; and the independence of their calibration procedures. This allows the detection of degraded operating conditions and failed sensors and differentiates between the two.

The COTCs are located in the top nozzle of the instrumented fuel assemblies and are radially distributed over the core in the same way as the SPND fingers. At every location, there is space for three thermocouples. These sensors are primarily used for post-accident measurement purposes, but they may also be used for obtaining additional information relative to radial power distribution and local thermal hydraulic conditions.

Table 2-1
Preliminary Core Parameters

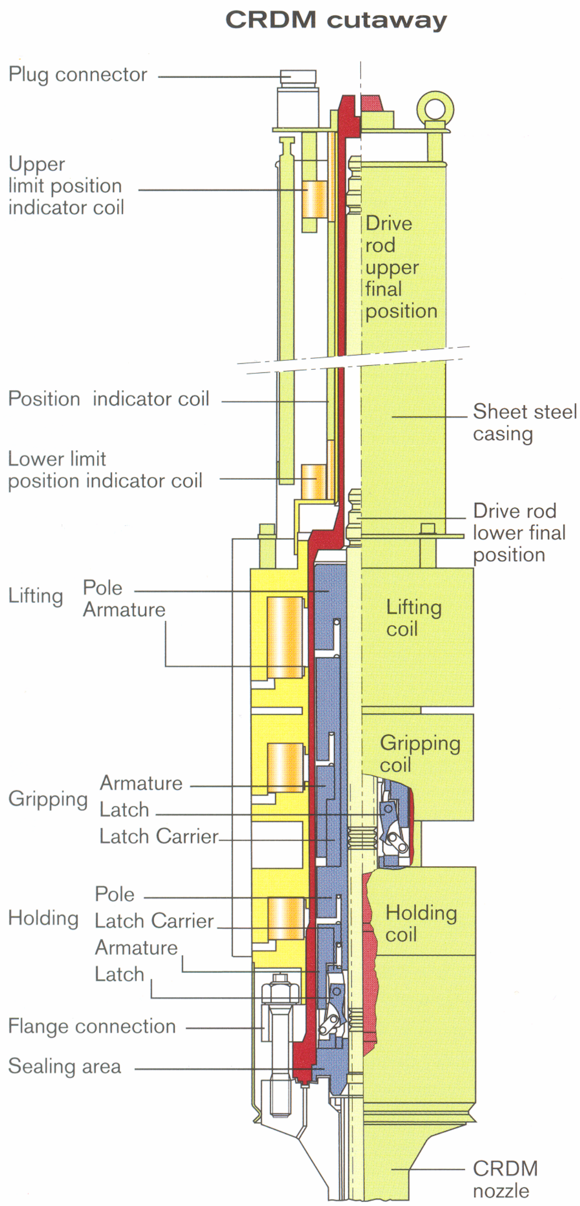
	Best Estimate Conditions
Nuclear power	4,500 MWt
Number of fuel assemblies	241
Fuel assembly pitch in core (cold)	8.466 in
Number of fuel rods/fuel assembly	265
Fuel assembly length (cold, without springs)	15.76 ft
Active length (cold)	13.78 ft
Average linear power (at rated power)	5.0 kW/ft
Diameter of fuel rods	0.374 in
Number of guide tubes per assembly	24
Guide tube outer diameter	0.490 in
Number of spacer grids per assembly	10
Fuel rod pitch	0.496 in
Total fuel rod length (cold)	14.93 ft
Total fuel assembly mass	1,731 lb
Assembly UO_2 mass	1,338 lb

Table 2-2
Key Fuel Assembly Characteristics



Characteristics	Data
Fuel Assemblies	
• Fuel rod array	17 x 17
• Lattice pitch	0.496 in
• Number of fuel rods per assembly	265
• Number of guide thimbles per assembly	24
Materials	
• Mixing spacer grids	
– Structure	M5™
– Springs	Inconel 718
• Top & bottom spacer grids	Inconel 718
• Guide thimbles	M5™
• Nozzles	Stainless steel
• Holddown springs	Inconel 718
Fuel Rods	
• Outside diameter	0.374 in
• Active length	13.78 ft
• Cladding thickness	0.022 in
• Cladding material	M5™

Table 2-3
Characteristics of the RCCAs and CRDMs



Characteristics	Data
RCCAs	
• Quantity	89
• Number of rods per assembly	24
Absorber	
• AIC part (lower part)	
– Weight composition (%): Ag, In, Cd	80, 15, 5
– Density	635 lb/ft^3
– Absorber outer diameter	0.301 in
• B4C part (upper part)	
– Natural boron	19.9% ^{10}B
– Density	112 lb/ft^3
– Absorber diameter	0.294 in
Cladding	
• Material	AISI 316 stainless steel
• Surface treatment (externally)	Ion-nitriding
• Outer diameter	0.381 in
• Inner diameter	0.304 in
Fill Gas	Helium
Control Rod Drive Mechanisms (CRDMs)	
• Quantity	89
• Stepping speed	14.76 in/min or 29.53 in/min
• Maximum scram time allowed	3.5 s
• Materials	• Forged 304 stainless steel • Magnetic 410 stainless steel • Non-magnetic stainless steel

Table 2-4
Neutronic Core Data

Active Core	
• Equivalent diameter	12.35 ft
• Core average active fuel height, first core (cold dimensions)	13.78 ft
• Height-to-diameter ratio	1.115
• Total cross-section area	119.96 ft^2
Fuel Rods	
• Number	63,865
• Outside diameter	0.374 in
• Diametrical gap	0.329 in
• Clad thickness	0.022 in
• Clad material	M5™
Co-Mixed Burnable Poison (Typical)	
• Material	$Gd_2 0_3$
• Gadolinium enrichment (wt%)	2 – 8
• UO_2 carrier enrichment (wt% ^{235}U)	0.7 host enrichment*

*Host enrichment is the non-gadolinium fuel pin enrichment.

Table 2-5
Thermal Hydraulic Design Data

Total core heat output	4,500 MWt
Number of loops	4
Nominal system pressure	2,250 psia
Coolant flow:	
• Core flow area	63.6 ft^2
• Core average coolant velocity	15.8 ft/s
• Core average mass velocity	2.8 $Mlb_m/h\text{-}ft^2$
• Vessel flow rate (Best Estimate)	184 Mlb_m/h
• Thermal design flow	~479,000 gpm
• Best estimate flow	~499,000 gpm
• Mechanical design flow	~539,000 gpm
Coolant temperature (°F):	
• Nominal inlet	564
• Average rise in vessel	62
• Average rise in core	65
• Average in core	597
• Average in vessel	595
Heat transfer:	
• Heat transfer surface area	86,170 ft^2
• Average core heat flux	178,200 $Btu/h\text{-}ft^2$
• Maximum core heat flux (nominal operation)	~535,000 $Btu/h\text{-}ft^2$
• Average linear power density	5 kW/ft
• Peak linear power for normal operating conditions w/uncertainty	~15 kW/ft
• Peak linear power protection threshold	~18 kW/ft
DNB ratio:	
• Minimum DNBR under nominal operating conditions with $F_{\Delta H}$ ~1.63	~2
Fuel assembly:	
• Number of fuel assemblies	241
• Fuel assembly pitch	8.466 in
• Active fuel height	165.35 in
• Lattice pitch	0.496 in
• Number of fuel rods per assembly	265
• Number of control rod assembly or instrumentation guide thimbles per assembly	24
• Outside fuel rod diameter	0.374 in
• Guide thimble diameter	0.490 in

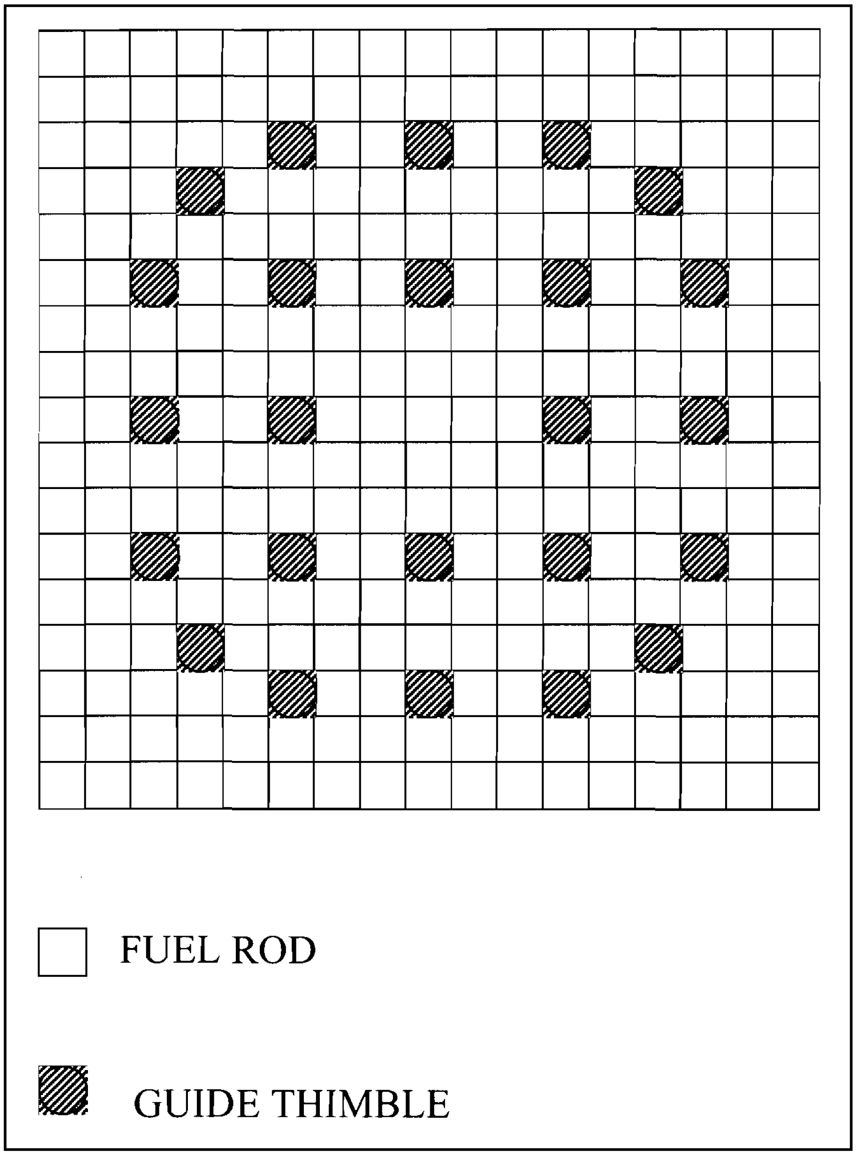


Figure 2-1
Radial Cross Section of a Fuel Assembly

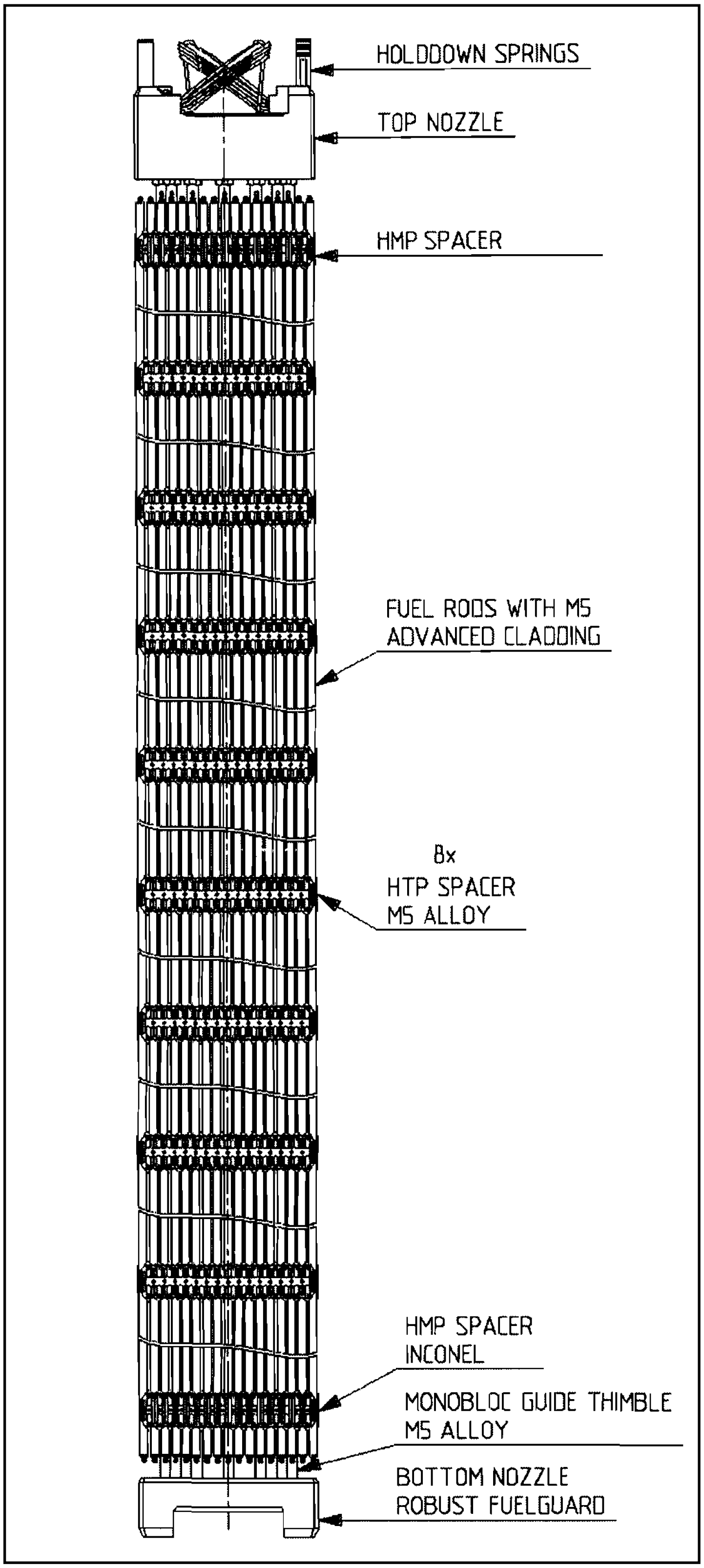
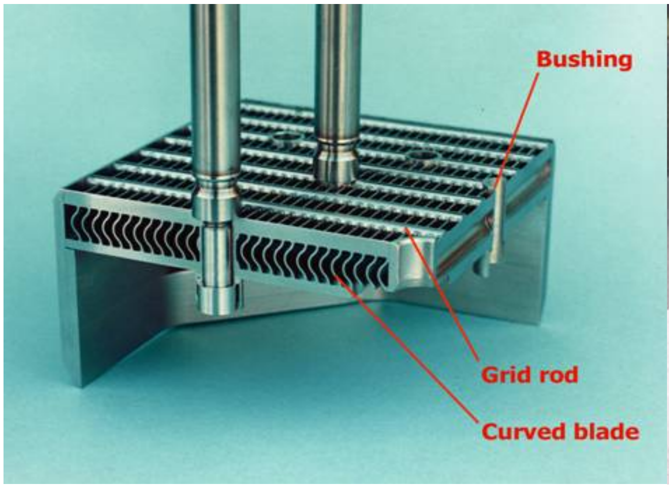


Figure 2-2
Fuel Assembly

Top Nozzle



Robust FUELGUARD™ Bottom Nozzle

Figure 2-3
Fuel Assembly Top and Bottom Nozzles

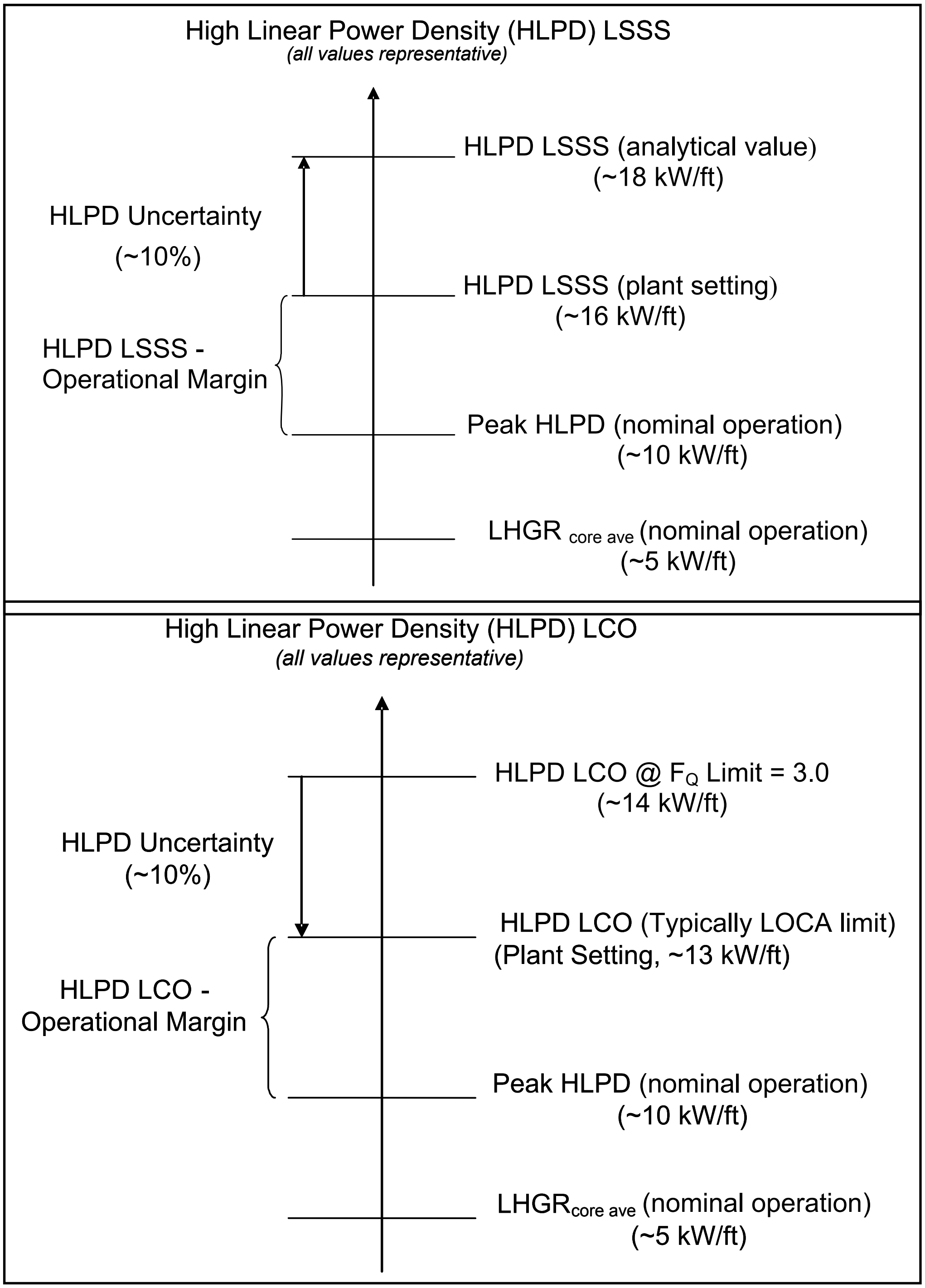


Figure 2-4
High Linear Power Density

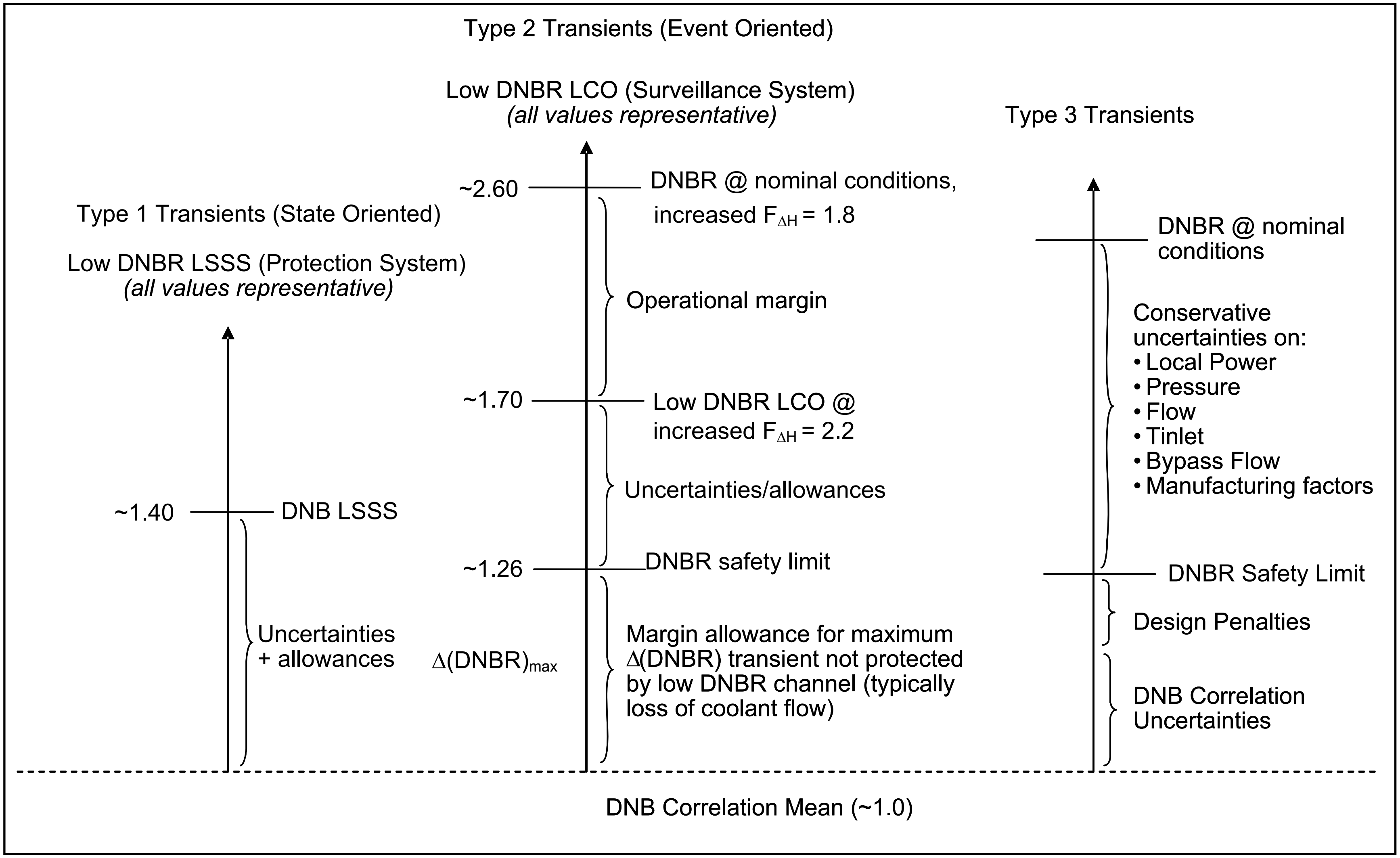


Figure 2-5
Transient Analysis Method – DNBR Criterion

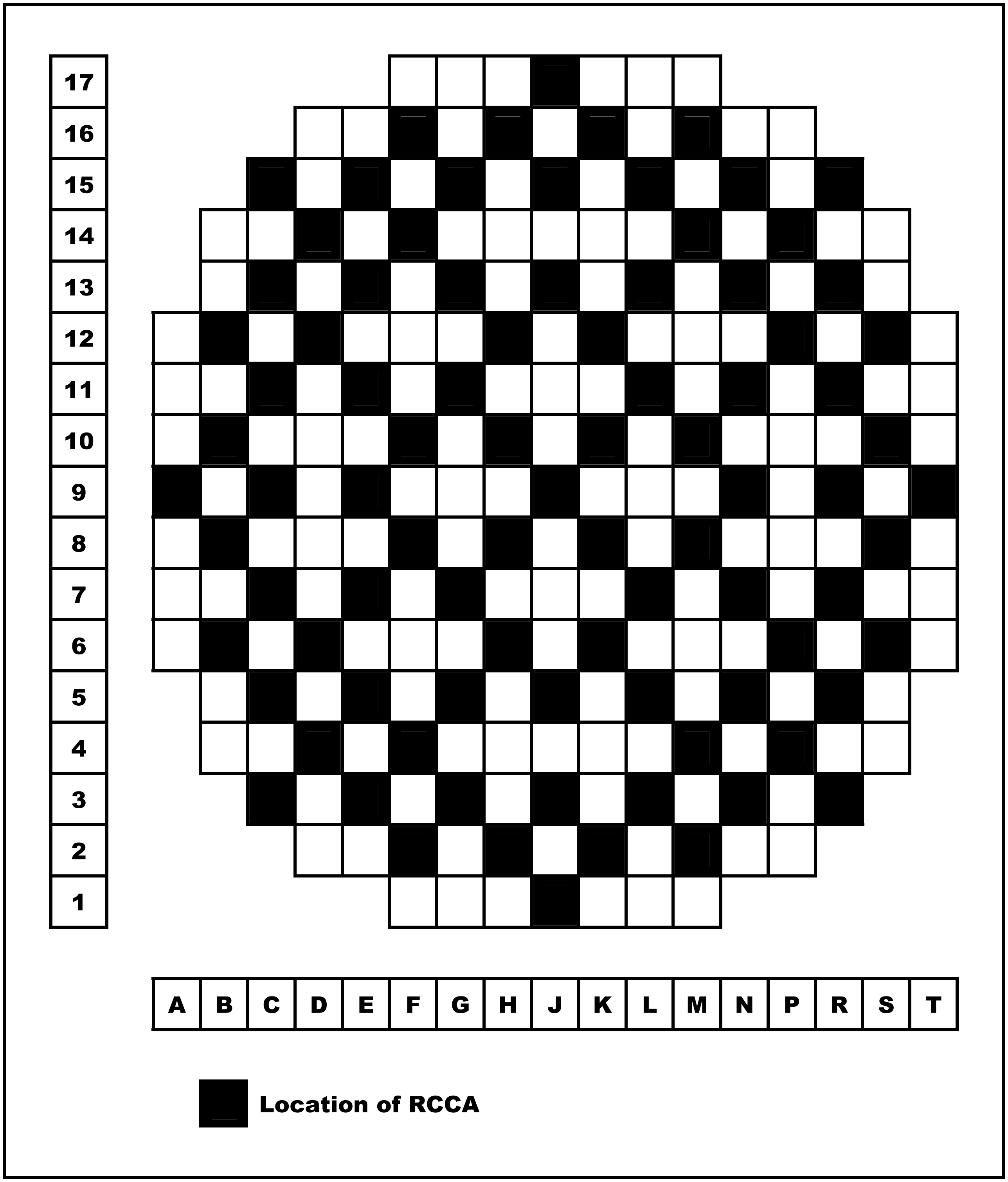


Figure 2-6
Possible RCCA Pattern Covering
Enveloping Requirements

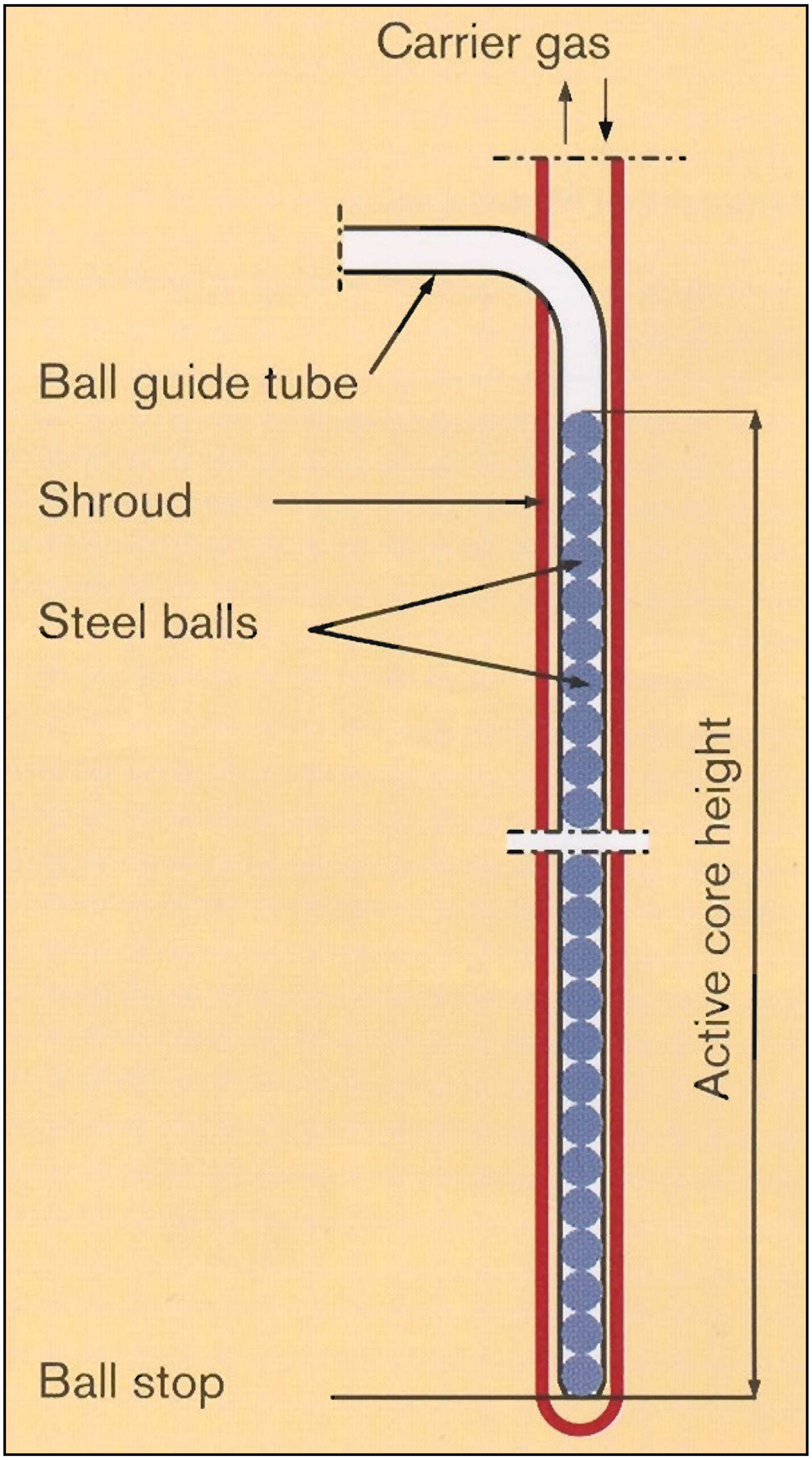


Figure 2-7
Aeroball System

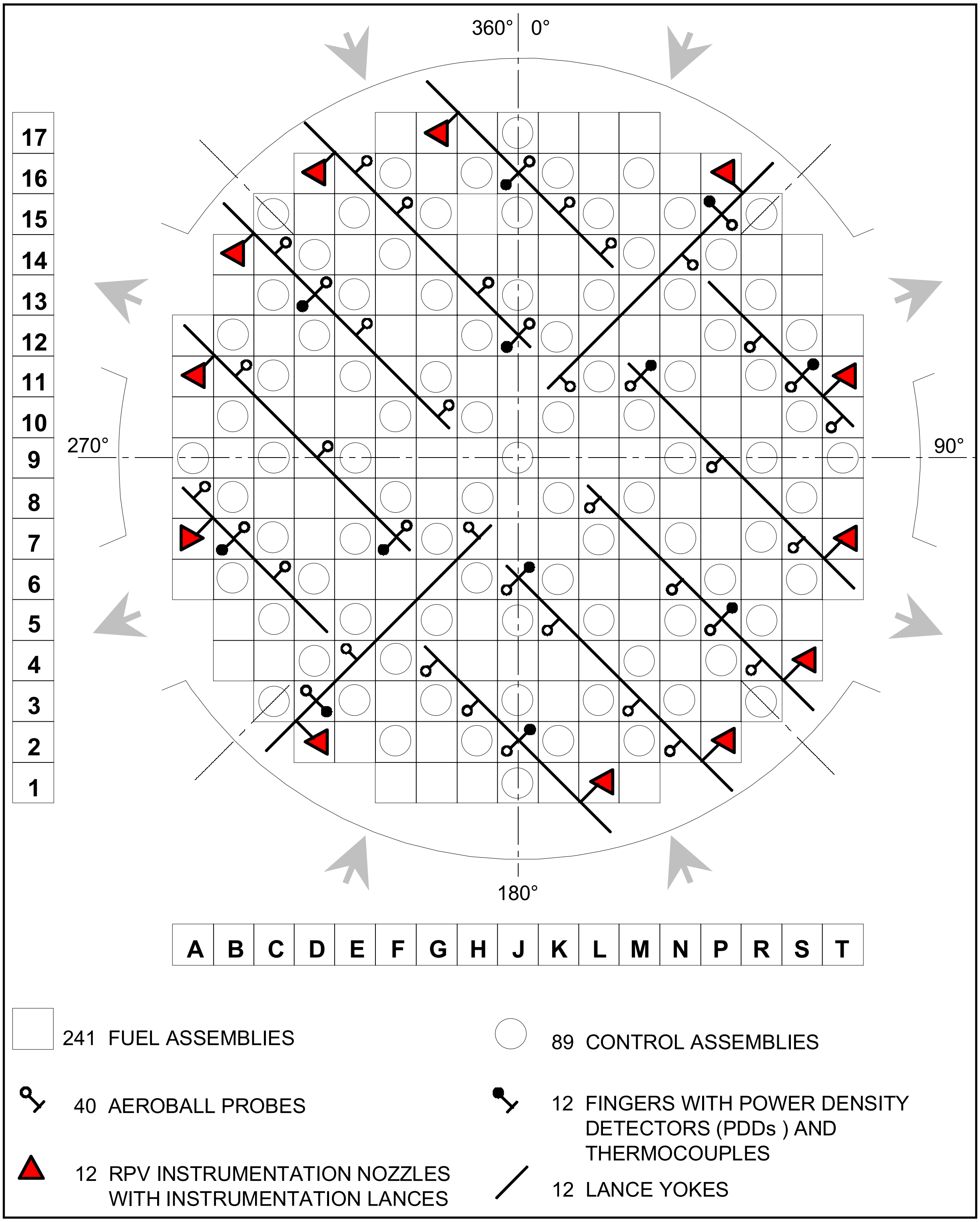


Figure 2-8
Arrangement of In-Core Instrumentation Fingers

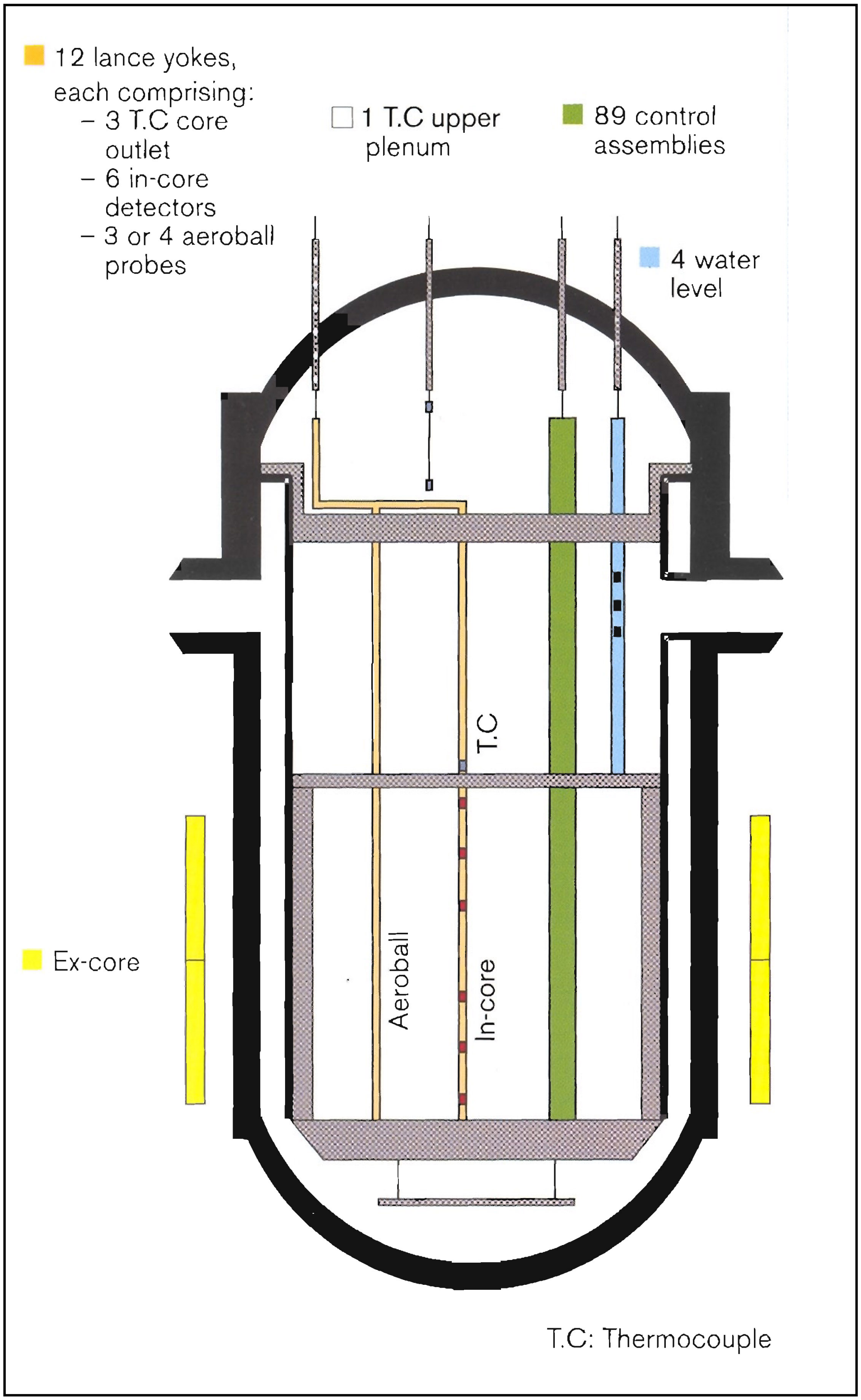


Figure 2-9
In-Core Instrumentation

3.0 REACTOR COOLANT SYSTEM

3.1 General Characteristics

The RCS configuration is a conventional four-loop design. The RPV is located at the center of the Reactor Building and contains the core with fuel assemblies. The reactor coolant flows through the hot leg pipes to the SGs and returns to the RPV via the cold leg pipes by the RCPs. The PZR is connected to one hot leg via the surge line and to two cold legs by the spray lines.

Figure 3-1 shows a flow schematic of the RCS and Figure 3-2 shows the layout of the RCS.

The RPV, PZR, and SGs have relatively large volume-to-core power ratios and are individually discussed below. For the RPV, the volume between the elevation of the RPV nozzles and the top of the active core is larger to improve the mitigation of Small Break Loss of Coolant Accidents (SBLOCAs) by prolonging the period until beginning of core uncovery or minimizing the core uncovery depth, if any. The increase in volume also contributes to an improvement in the mitigation of accidents during shutdown conditions, particularly in mid-loop operation (e.g., with loss of RHR), by extending the period for operator action.

For the PZR, a larger volume provides the following benefits:

- An increase of both water and steam volume with associated pressure and level scaling is favorable in handling many types of transients. A single countermeasure actuated at one limit can more easily become fully effective before the next limit is reached (e.g., in certain load reductions one PZR spray valve, instead of two, is sufficient to stop the pressure increase). The result is a reduction of loads on relevant systems and components (i.e., a reduced number of load cycles).
- For normal operating transients, parameter changes (e.g., pressure, water level) are mild and thus the potential for reactor trips is minimized.
- For events such as loss of condenser, the actuation of PZR safety valves can be avoided altogether.
- For SBLOCAs, the time until core uncovery is prolonged.

For the SG, the larger volume of the secondary side provides the following advantages:

- For normal operating transients, smooth parameter changes (e.g., pressure and water level) are obtained and thus the potential for unplanned reactor trips is further reduced.
- For mitigation of Steam Generator Tube Rupture (SGTR) scenarios, a large steam space results in a significant time delay for a mitigating response prior to filling of the SG.
- The secondary side water inventory of the SG at full load satisfies the most limiting requirement of a total loss of feedwater supply (including emergency feedwater). In this scenario, the time between reactor trip and a loss of heat removal is greater than 30 minutes and is sufficient for operating personnel to recover feedwater supply or initiate other countermeasures, such as primary side feed and bleed cooling.

The EPR design for a non-isolable Main Steam Line Break (MSLB) considers the increased water inventory in the SGs and resultant higher potential of mass and energy release into the containment. The large containment volume accommodates the pressure response.

Table 3-1 shows the approximate thermal-hydraulic parameters for the RCS.

3.2 Safety Concepts

A summary of the main safety concepts of the RCS is given below.

Actuation of Safety Systems

Actuation of safety systems, including safety valves, does not occur prior to reactor trip. This means that the best possible use is made of the depressurizing effect of the reactor trip. This approach minimizes the number of valve actuations and the potential for valves sticking in the open position.

Avoidance of Reactor Trip

Reactor trip is prevented by a fast reactor power cut back to part load for the following events:

- Loss of Main Feedwater Pumps (MFWPs), as long as at least one MFWP remains available and operable
- Turbine trip
- Full load rejection

Containment Building Volume

The internal volume of the Containment Building is larger relative to most existing U.S. PWR designs. The larger internal volume of the Containment Building provides the following advantages:

- There is more volume to absorb the mass and energy released from the RCS in accidents, such as a MSLB and Loss of Coolant Accident (LOCA). As a result, the peak containment pressure during these events is reduced.
- It is not necessary to credit active heat removal from the containment in the short-term phases of accidents, such as an MSLB and a LOCA. The SAHRS is available at approximately 12 hours after the beginning of a severe accident for long-term plant recovery.

Steam Generator Tube Rupture

The SGTR mitigation concept is based on having the Medium Head Safety Injection (MHSI) pump delivery shutoff head at a value less than the setpoints for the SG safety valves in order to minimize potential radioactive releases.

Partial secondary side cool down is started automatically on low-low pressurizer level (MHSI actuation signal). The main steam bypass or main steam relief valves open to depressurize the SGs at a rate of approximately 180°F/h to a pressure of 870 psia. This cool down is needed to bring the RCS pressure below the Main Steam Safety Valve (MSSV) response threshold and enable injection from the MHSI System.

Prevention against over-filling of the affected SG and consequential prevention of liquid release to the environment is a design requirement for the safety systems and the SG, including situations with MHSI actuation.

Isolation of the affected SG, that is, isolating all feedwater supply (including emergency feedwater); closing the Main Steam Isolation Valve (MSIV) and the Main Steam Relief Valve (MSRV), occurs automatically on a SG high level signal coincident with end of partial cooldown. The subsequent plant cool down to RHRS operation is accomplished using the remaining intact loops.

Leak-Before-Break

The RCS piping is designed for the Leak-Before-Break concept. However, LBB is not credited for the demonstration of adequate emergency core cooling (i.e., calculation of post-LOCA peak clad temperature and cladding oxidation) as required by 10 CFR 50.46.

3.3 Overpressure Protection

Overpressure Protection (OPP) protects the integrity of the RCPB in both hot and cold conditions. OPP is performed by the PZR safety valves in parallel with the reactor protection system and associated equipment.

The objective of OPP is to prevent the opening of non-isolable valves during all anticipated operational occurrences and accidents that have the potential for radioactive releases.

Reactor trip is taken into account as a pressure reducing measure on the EPR for the following reasons:

- Reactor trip in a nuclear power plant is highly reliable due to its redundancy and independence of design.
- Primary system OPP has always been designed considering reactor trip.
- Secondary side OPP was designed in the past according to conventional boiler rules. A nuclear power plant behaves differently after plant shutdown. In a nuclear power plant, the core power is reduced relatively quickly when the control rods drop into the core, whereas a conventional boiler maintains its temperature longer when shut down.
- Section NB-7120(b) of the ASME code recognizes reactor shutdown for OPP.
- All other safety systems in a nuclear power plant are designed considering reactor trip.
- Nuclear safety improves with the reduction of capacity and number of secondary side safety valves.

3.3.1 Primary Side Overpressure Protection

Three pilot-operated safety valve discharge trains are arranged at the top of the PZR for overpressure protection. Figure 3-3 shows the PZR with its discharge components.

Automatic opening of a safety valve upon detection of RCS overpressure is ensured by pilot actuators for each of the safety valves. During normal plant operation, a spring-loaded pilot valve is used to open the safety valve.

For overpressure protection at lower RCS temperatures (e.g., cold overpressure protection), two solenoid pilot valves in series are used to open each safety valve. The setpoint is adjustable. Additionally, the pilot valves may be manually operated.

Primary side OPP is classified as safety-related and the equipment used for this function is qualified for liquid, steam, and two-phase flow operation.

The PZR discharge performs the following safety functions:

- OPP of the RCS by automatic initiation of discharge of steam, water, or two-phase fluid, depending on the specific initiating event.

- Depressurization of the RCS by discharge of steam or water for those plant conditions for which depressurization by PZR spray via the Chemical and Volume Control System (CVCS) is not available or insufficient.
- Discharge of reactor coolant to enable continued RHR in the event of complete unavailability of the secondary side heat removal, in conjunction with injection of cooled borated water by the SIS (feed and bleed).

The PZR relief system discharges to the Pressurizer Relief Tank (PRT) which is located inside the containment. The PRT condenses the steam by mixing it with relatively cold water in the PRT. Thus, the PRT contributes to the protection of the RCS from overpressure in conjunction with the PZR relief system.

In addition to the PZR safety valves, a dedicated discharge line is provided to depressurize the RCS during a core melt condition. This guarantees depressurization to a pressure sufficiently below the level that would lead to a high pressure core melt accident. The Pressurizer Depressurization System (PDS) is manually actuated in the event of a severe accident and consists of one or more valves that discharge into the PRT.

PDS valves are independent of the electrical power supply units for operational considerations and for dependability during accident scenarios. These valves are supplied by the main emergency diesels, the SBO diesels, and the batteries dedicated to severe accidents.

3.3.2 Secondary Side Overpressure Protection

On each main steam line, three discharge trains are arranged outside the containment. The discharge trains on each line are arranged as follows:

- One discharge line is equipped with a relief valve and an isolation valve connected in series having approximately a 50% capacity of the full load flow of one SG.
- Two other discharge lines are equipped with a dedicated safety valve having approximately a 25% capacity of the full load flow of one SG.

Figure 3-4 shows a schematic arrangement of the secondary side OPP.

Both types of discharge trains (relief and safety valves) are safety grade. The relief train design meets ASME code requirements and will therefore be credited in OPP analyses.

In the overall concept of secondary side pressure limitation and heat removal, the following represents the hierarchy of the defense-in-depth principle.

- The first line of defense is the actuation of the turbine bypass.
- The second line of defense is the relief valve, which ensures safety grade controlled heat removal and pressure limitation.
- The third line of defense consists of the two safety valves.

Capacities of both the relief and safety valves are based on the principles discussed below.

- For anticipated operational occurrences, discharge is controlled in a way that prevents the opening of a non-isolable safety valve and applicable pressure limits are not exceeded. Protection system action (i.e., a reactor trip) is taken into account, and failure to open a discharge line is not considered for anticipated operational occurrences.

- For beyond design basis conditions, discharge capacity is sufficient to avoid exceeding applicable pressure limits, even if one of the discharge lines fails to open. For this scenario, the protection system action (reactor trip) is taken into account.

This approach provides diversity for both valve actuation and valve type.

With the selected configuration of discharge valves, the following safety functions are performed:

- OPP and controlled heat removal at normal or upset conditions is by means of the relief valve in the event of condenser unavailability.
- OPP for emergency conditions such that 110% of design pressure is not exceeded.
- With accidents (e.g., SBLOCA), the secondary side is cooled down to approximately 870 psia at a rate of approximately 180°F/h by means of the relief valves. This ensures adequate injection from the MHSI system. Adequate injection essentially means that the RCS inventory decrease is minimized so the respective design criteria are met.
- With a SGTR, as well as with any event involving the response of a steam dump, uncontrolled release of steam or water is prevented by closing the dedicated isolation valves if the water level increases beyond a certain limit or if a discharge valve sticks in the open position (the safety valve response is not challenged in the event of a SGTR).

3.4 Principal Mechanical Components

3.4.1 Reactor Pressure Vessel

The RPV is the main component of the RCS. The vessel is cylindrical, with a welded hemispherical bottom and a removable flanged hemispherical upper head with gasket. It is designed to provide the volume required to contain the reactor core, the control rods, the heavy reflector, and the supporting and flow-directing internals. The RPV nozzles are the fixed point of the RCS.

Figure 3-5 is an outline drawing of the RPV.

The RPV is made of low-alloy steel. The complete internal surface of the RPV is covered by stainless steel cladding for corrosion resistance.

The RPV has four inlet nozzles and four outlet nozzles located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant from the cold legs enters the vessel through the inlet nozzles and flows down through the annulus formed by the space between the core barrel and the reactor vessel inner wall. At the bottom of the vessel, the coolant is deflected to pass up through the core to the outlet nozzles. Heated reactor coolant leaves the RPV through four outlet nozzles, flowing into the hot legs and toward the SGs.

The cylindrical shell of the RPV consists of two sections, an upper and lower part. To minimize the number of large welds, which reduces the frequency of in-service inspections, the upper part of the RPV is machined from a single forging and fabricated with eight nozzles. Since the nozzles are fabricated into the massive plate used in the RPV shell, most of the reinforcement needed for the nozzle design is provided by the vessel material itself. Therefore, the nozzles used in this design are the “set-on” type requiring a less substantial weld bead than would otherwise be required.

The RPV closure head consists of the following single-piece forgings, welded together by a circumferential weld:

- The closure head dome
- The closure head flange

The closure head flange consists of a shaped forging with holes for the closure studs. The lower face of the flange is clad and locally machined to form two grooves in which two metallic gaskets are located.

The closure head is provided with penetrations for:

- Eighty-nine adapters for CRDMs, composed of tube and flanges. The flanges are connected to the CRDM latch-housings. The tubes are welded into the vessel head. Bolts to the CRDM latch-housing flanges connect the adapters. The seal arrangement at the interface between the two flanges consists of circular metallic seals.
- One adapter for a dome temperature measurement probe, and sixteen adapters for instrumentation (twelve lances for neutron and temperature instrumentation and four for reactor vessel water level). RPV aeroball and power density detectors, adapter flanges, RPV water level measurements, and thermocouple adapter flanges are welded into the RPV head.
- The vent pipe that is welded to the head dome penetrates the RPV closure head.

The lower part of the RPV body is made of two cylindrical shells at the reactor core level, one transition ring, and one bottom head dome.

The bottom head is a hemispherical shell connected to the RPV body through the transition ring. There are no penetrations in the bottom head.

The RPV is provided with thermal insulation to reduce heat losses into the Reactor Building. The insulation thickness ensures that the heat losses can be removed by the ventilation system under all operating conditions.

The thermal insulation for the RPV closure head consists of stainless steel frames filled with mineral wool insulation. The thermal insulation for the RPV cylindrical shell and bottom head consists of all-metal insulation sections (also called reflective type insulation) attached to a sealed liner. The thermal insulation of the RPV nozzle area consists of insulation cassettes fabricated from stainless steel sheets and filled with mineral wool.

Access provisions allow the welds to be inspected from both the external and internal sides of the RPV. The internal surface of the RPV walls is accessible from inside for visual and ultrasonic inspection.

The nozzle shell is equipped with an external seal ledge to keep any leakage or spills from flowing into the gap between the outer shell of the RPV and onto the insulation. This external seal ledge is connected to the reactor cavity seal ring to ensure leak tightness between the flange of the reactor vessel and the bottom of the reactor cavity.

3.4.2 Reactor Pressure Vessel Supports

The entire structure of the RPV is supported by pads located on the bottom of the eight nozzles for the reactor coolant loops. Each nozzle has its own support pad which rests on a support ring, also part of the RPV support structure. This arrangement is capable of withstanding the forces caused by design basis and severe accidents.

Figure 3-6 shows the RPV supports.

3.4.3 Reactor Pressure Vessel Internals

The RPV internals consist of the lower and upper sections. Figure 3-7 shows a cutaway view of the RPV internals. Most components of the internals are made of low carbon chromium-nickel stainless steel. The various connectors, such as bolts, pins, tie rods, etc. are made of cold-worked chromium-nickel-molybdenum stainless steel.

Lower Internals

The lower internals are made up of the core barrel, the lower core support structure, the heavy reflector, and the flow distribution device. These are vertically supported by a ledge machined into the flange of the RPV. Their movement is restricted vertically inside the RPV by an annular hold-down spring located between the flanges of the lower and upper internals. This design prevents them from lifting off the RPV ledge. The lower internals remain in place in the RPV during refueling but may be removed for in-service inspections of the RPV by means of a lifting rig.

Core Barrel

The core barrel is suspended from the RPV flange support edge and is centered at its upper flange by means of alignment pins. At the lower section, the lower radial support system restricts rotational and tangential movements, but allows for radial thermal growth and axial displacements.

The core barrel assembly consists of:

- An upper flange (the core barrel flange) that is located inside the RPV flange and serves to transmit the loads of both the fuel assemblies and lower assemblies to the vessel.
- A barrel cylinder (the core barrel) welded to the core barrel flange and made of cylindrical sections welded together. The upper section of the barrel has four outlet nozzles in front of the four vessel outlet nozzles. They provide the passageway for the reactor coolant to flow from the core to the RPV outlet nozzles. The maximum radial gap between the core barrel and the RPV nozzles is controlled to restrict the amount of bypass flow.
- Irradiation capsule baskets for holding irradiation specimens for brittle fracture surveillance of the RPV are bolted to the outside of the core barrel at locations where the irradiation neutron flux is higher than on the inside of the RPV core shells.

Lower Core Support Structure

The lower core support structure is the major supporting assembly of the complete RPV internals structure. The lower support plate is welded to the bottom shell of the core barrel. The thick forging supports all the fuel assemblies, the heavy reflector, and the flow distribution device. Holes are provided to direct and distribute the flow of reactor coolant to the inlet of the core. The lower support plate transmits the vertical loads to the RPV flanges and distributes the horizontal loads between the RPV flange and the lower radial support system.

The core barrel flange that sits on a ledge machined from the RPV flange is preloaded axially by a large Belleville type spring. The fuel assemblies sit directly on a perforated plate (the core support plate), which is approximately 17.7 in thick. This plate is machined from a forging of stainless steel and welded to the core barrel. Cooling water flows through the core support plate through four holes provided for each fuel assembly.

The lower internals are positioned in the bottom of the RPV by means of the lower radial support system.

The fuel assemblies are placed into the core cavity and rest on the lower support plate, which contains the lower fuel positioning pins that provide location and alignment for the bottom nozzles.

Heavy Reflector

The space between the multi-cornered radial periphery of the reactor core and the cylindrical core barrel is filled with an all-stainless steel structure, called the heavy reflector, the purpose of which is to reduce fast neutron leakage and flatten the power distribution. The reflector is inside the core barrel above the lower core support plate. To avoid any welded or bolted connections close to the core, the reflector consists of stacked forged slabs (rings) positioned one above the other with keys, and axially restrained by tie rods bolted to the lower core support plate.

The heavy reflector is dimensioned to accommodate expansion of the fuel assembly arrangement. Water cooling is provided by means of coolant channels inside the heavy reflector to prevent excessive stress and deflections of the rings due to the heat generated inside this steel structure by absorption of gamma radiation.

Figure 3-8 shows a cutaway view of the heavy reflector and RPV lower internals.

Flow Distribution Device

The flow distribution device is located below the lower support plate and bolted onto the lower support plate by means of vertically positioned columns. This device homogenizes the flow at the entrance of the lower support plate so that the reactor coolant flows upward in an even distribution through the reactor core.

Upper Internals

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

Upper Support Plate

The USP is a thick forged plate with a flange that is welded to the cylindrical skirt. This plate separates the upper plenum of the core barrel from the RPV upper head dome and is connected to the perforated UCP by the support columns for the RCCA guide, the normal support columns, and the level measurement position columns.

This structure compresses the fuel assemblies and is rigid enough to ensure flatness of the UCP.

A hold down spring (Belleville type) is located between the flange of the upper internals and the flange of the core barrel.

Perforated Upper Core Plate

The perforated UCP is made of austenitic stainless steel and is connected to the USP by RCCA guide support columns, normal support columns, and level measurement position columns. Collectively, these columns maintain the appropriate distance between the USP and the UCP. Additional parts of the UCP are the centering pins for the fuel assemblies and the RCCA guides.

Precise alignment between the UCP and heavy reflector is made possible by four sets of inserts that guide four alignment pins on the heavy reflector.

With the aid of centering pins, the UCP aligns the fuel assemblies (two centering pins each), and the RCCA guides (four centering pins each).

Support Columns

There are two types of support columns, one for the RCCA guides and another for the normal support columns.

Support columns for the RCCA guides are longitudinally-welded tubes. They are located above those fuel assembly positions that are equipped with RCCAs. The RCCA guides are located inside these support columns.

The normal support columns are also tubes and are located in the outer range of the core. They are connected to the UCP via an upper flange to the USP.

3.4.4 Pressurizer

The PZR consists of a vertical cylindrical shell, closed at both ends by hemispherical heads. It is constructed of ferritic steel, with austenitic stainless cladding on all internal surfaces in contact with the reactor coolant.

Figure 3-9 shows a cutaway view of the PZR.

The spray system inside the PZR consists of three separate nozzles welded laterally near the top of the upper cylindrical shell. Two nozzles are provided for the main spray lines (connected to cold legs) and one nozzle is provided for the auxiliary spray line connected to the CVCS. The spray heads inject the required spray flow in the steam space of the PZR.

The PZR is equipped with electric heater rods, installed vertically in the bottom head.

The upper head of the PZR has four large nozzles, one for each of the three safety valve connections on the upper head and one for the PDS line used for severe accident mitigation, and one small nozzle for venting. The three safety valves are actuated by pressure sensors installed laterally in the upper shell (safety valve 1) and by the sensing line nozzles in the steam volume (safety valves 2 and 3). A manway is also located on the upper head.

The PZR is connected to the RCS by a surge line that connects to one hot leg. The surge line PZR nozzle is located at the bottom of the PZR and is connected vertically.

The PZR is supported by three brackets welded to the lower cylindrical shell. The brackets rest on a supporting floor and allow free radial and vertical thermal expansion of the PZR. Lateral restraints prevent rocking in the event of an earthquake or a pipe break. Figure 3-10 shows the PZR supports.

The main functional requirements of the PZR are summarized below.

- The PZR forms part of the RCPB and provides RCS volume control (it is the coolant expansion vessel of the RCS) and RCS pressure control.

- The large water volume in the PZR prevents the heaters from being uncovered during out-surges and is also large enough to compensate for coolant expansion between 0% and 100% power under normal conditions.
- The large steam volume accommodates RCS OPP requirements.
- The large steam volume prevents frequent actuation of the pressure control equipment during normal operation.
- The PZR is designed not to empty on a reactor trip or turbine trip.

3.4.5 Steam Generator

The SGs are vertical shell, natural circulation, U-tube heat exchangers with integral moisture separating equipment. They are also fitted with an axial economizer to provide increased steam pressure.

Figure 3-11 shows a cutaway view of the SG.

The reactor coolant flows through the inverted U-tubes, entering and leaving nozzles located in the hemispherical bottom channel head of the SG. The bottom head is divided into inlet and outlet chambers by a vertical partition plate extending from the tube sheet.

The heat conveyed by the reactor coolant is transferred to the secondary fluid through the tube walls of the tube bundle. On the secondary side, the feedwater is directed to the cold side of the tube sheet by an annular skirt in which feedwater is injected by the feedwater distribution ring.

The axial economizer directs all the feedwater to the cold leg side of the tube bundle and about 90% of the recirculated water to the hot leg. This is made possible by the double wrapper in the cold leg of the downcomer. Feedwater is routed to the cold leg of the tube bundle and a secondary side partition plate (that extends up to the sixth tube support plate) separates the cold leg and the hot leg sides of the tube bundle. The internal feedwater distribution system (ring with oblong-shaped holes and deflecting sheet) of the SG covers only about 180 degrees of the wrapper on the cold side. Once the feedwater reaches the bottom of the cold side, the subcooled water makes a U-turn, flows up along the cold leg tube bundle between the wrapper and the partition plate, and is heated to a point where it boils. The steam-water mixture flows upward through the moisture separators and dryers and the dried steam exits the SG through the outlet nozzle located at the top of the SG elliptical head. This design enhances the heat exchange efficiency between the primary side and the secondary side and increases the outlet steam pressure by about 44 psi as compared with a boiler type SG with the same tube surface.

Figure 3-12 illustrates the principle of the axial economizer.

The tube material is alloy 690, which is widely used in SGs throughout the world, and is highly resistant to corrosion.

The cladding on the tube sheet is Ni Cr Fe alloy and the cladding on the channel head is stainless steel. The secondary shell is made with low alloy ferritic steel. Tube support plates are fabricated from an improved Type 410 stainless steel.

The SG is supported by four support legs or columns hinged to ball-jointed brackets or clevises. The ball joints allow the SG to move freely when the primary loop temperature is being raised or lowered.

Lateral supports guide normal SG thermal movement and restrain its movements during accident event loads.

Figure 3-13 shows the SG supports.

3.4.6 Reactor Coolant Pump

The RCPs are vertical, single-stage, shaft seal units, driven by air-cooled, three-phase induction motors. The complete unit is a vertical assembly consisting of (from top to bottom) a motor, a seal assembly, and a hydraulic unit.

Figure 3-14 shows a cutaway view of the RCP.

Reactor coolant is pumped by an impeller attached to the bottom of the rotor shaft. Coolant is drawn up through the bottom ring of the casing, up through the impeller, and discharged through the diffuser and an exit nozzle located in the side of the casing.

The shaft assembly consists of two parts, rigidly connected by a spool piece that is bolted to each half. The configuration allows for the shaft to be removed for performing maintenance on the shaft seals. The shaft is supported by three radial bearings: two oil bearings on the upper part; and one hydrostatic water bearing located on the impeller. The axial thrust is supported by a double acting thrust bearing located at the upper end of the shaft below the flywheel. The oil that lubricates the upper radial and thrust bearings is cooled in a low pressure oil-water cooler attached to the motor frame. The oil that lubricates the lower radial bearing is cooled by a low-pressure water coil integrated inside the oil pot.

The static part of the hydrostatic bearing is an integral part of the diffuser. The one-piece diffuser is bolted to the closure flange, thus permitting the entire assembly to be removed in one piece. Torque is transmitted from the shaft to the impeller by a Hirth assembly that consists of radial grooves machined on the flat end of the shaft and symmetrically on the impeller. A thermal barrier (a low-pressure water coil) cools the primary water in the event of a disruption of the seal injection water. This thermal barrier contains a secondary hydrodynamic radial graphite bearing. This bearing is not active in normal operation, but would provide support to the shaft should the hydrostatic bearing become ineffective (e.g., depressurization on primary side).

The shaft seals accommodate the pressure gradient from reactor coolant pressure to ambient conditions. The seals are located in a housing bolted to the closure flange. The closure flange and motor stand are jointly fitted to the casing with a set of studs.

The shaft sealing system consists of three seals staged into a cartridge. Seal number 1 is a hydrostatic seal that accepts the majority of the pressure gradient with a controlled leakage to the CVCS. The faceplates are comprised of a silicon nitrite ceramic. Seal number 2 is a hydrodynamic seal that accepts the remaining 25 psi in normal operation. In case of failure of the number one seal, the number two seal acts as a back-up during the limited period of time available to stop the pump and shutdown the plant. Seal number 3 is also a hydrodynamic seal with no significant differential pressure. Its purpose is to complete final leak tightness and prevent spillage of water.

The shaft seals are equipped with a Standstill Seal System (SSSS) actuated when the RCP is at rest after closure of all seal leak-off lines. A ring seal is moved upward by nitrogen pressure and closes against a landing on the rotor, thus creating a tight metal-to-metal seal. The SSSS ensures shaft tightness with the pump at standstill in the event of a simultaneous loss of CVCS seal injection and Component Cooling Water System (CCWS) water supply used to cool the shaft sealing system, a cascaded failure of all the stages of the shaft seal system, or a SBO.

Figure 3-15 shows the RCP supports.

3.4.7 Reactor Coolant Piping

The reactor coolant piping in each of the four coolant loops consists of a hot leg, a crossover leg, and a cold leg. The hot leg extends from the RPV to the SG; the crossover leg from the SG to the RCP; and the cold leg from the RCP to the RPV.

The nominal dimensions of the main coolant lines are:

Inside diameter	30.7 in
Thickness range	3 - 3.8 in

The piping material is austenitic stainless steel. The pipes are forged and the elbows are forged and bent by induction.

The RCS piping is designed using the Leak-Before-Break (LBB) concept. This eliminates the need to design RCS components and piping and supports to accommodate the dynamic effects of large or double-ended ruptures in these piping systems. Consequently, large pipe whip restraints and jet impingement shields are not required. To justify the LBB design, a monitoring system is required to detect leakage from this piping into the Reactor Building.

3.5 Reactor Coolant Chemistry

The main parameters for which limit values will be required for reactor coolant chemistry are listed in the table below. These parameters are required to:

- Limit the corrosion rate of system components in order to reduce the introduction of corrosion products (dissolved or suspended in the reactor coolant), since these products could foul the system and increase its activity
- Optimize the migration and limit the deposition of corrosion products to limit their accumulation on fuel cladding and the cold parts of the RCS, thus minimizing activated corrosion product build-up
- Prevent localized corrosion
- Suppress radiolytic decomposition of water

Expected pH at 572°F	from 7.1 to 7.3
Dissolved oxygen	< 5 ppb
Chlorides	< 150 ppb
Fluorides	< 150 ppb
Sulfates	< 150 ppb
Dissolved hydrogen during normal operation (T > 248°F)	25 to 100 ppm
Lithium hydroxide	0.4 to 6.0 ppm ^{7}Li

Table 3-1
Approximate Basic Thermal-Hydraulic Parameters for the RCS

Reactor thermal power	4,500 MWt
Hot leg temperature	624°F
Cold leg temperature	564°F
RCS operating pressure	2,250 psia
RCS design pressure	2,550 psia
Reactor coolant flow	125,000 gpm
SG tube bundle outlet pressure at full power	1,118 psia
Main steam pressure at hot standby	1,305 psia
Secondary side design pressure	1,450 psia
Steam flow per loop	5.1 Mlb/h
PZR size, total volume	2,650 ft^3
SG secondary side liquid mass at full power	182,000 lb

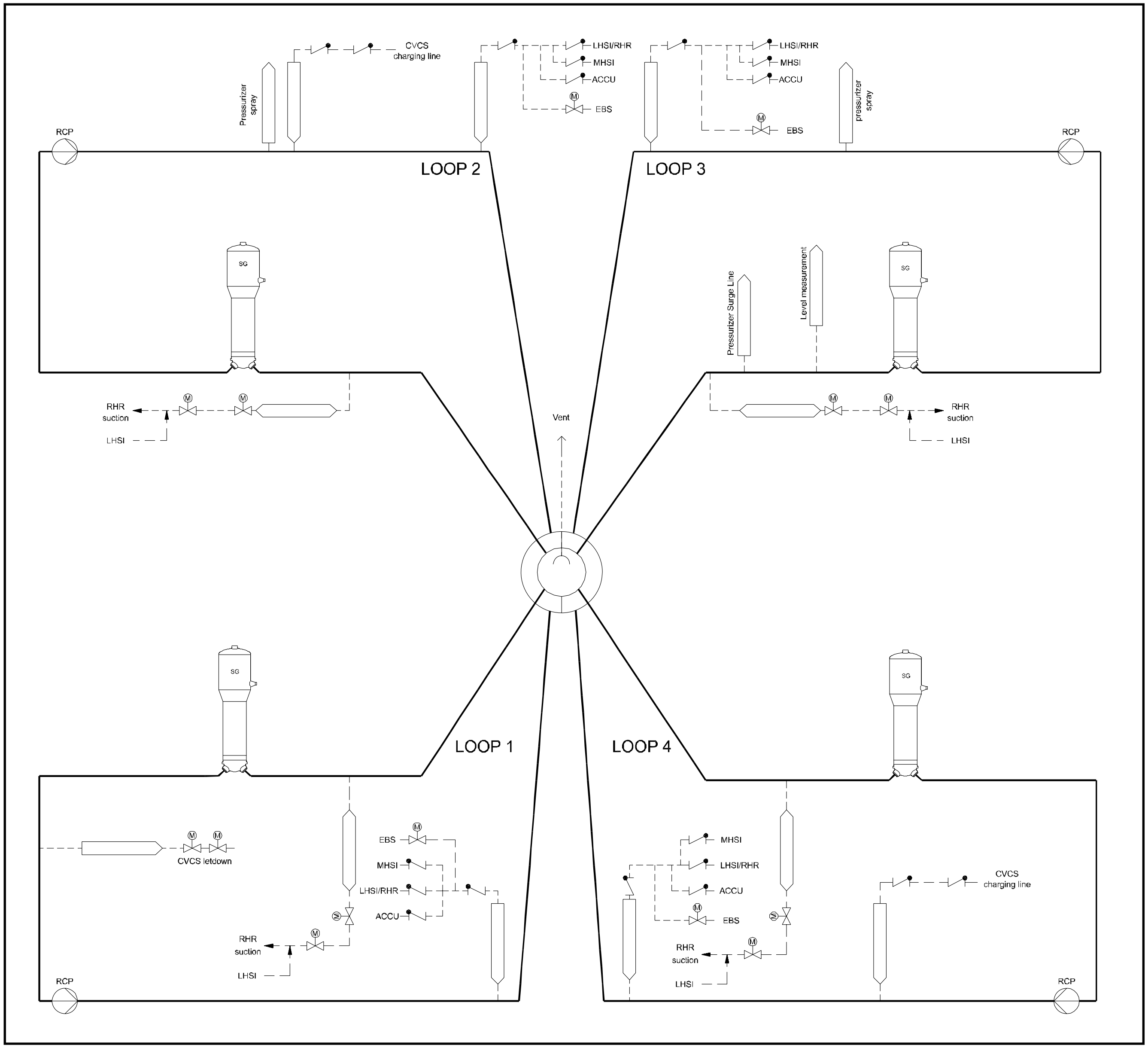


Figure 3-1
Reactor Coolant System

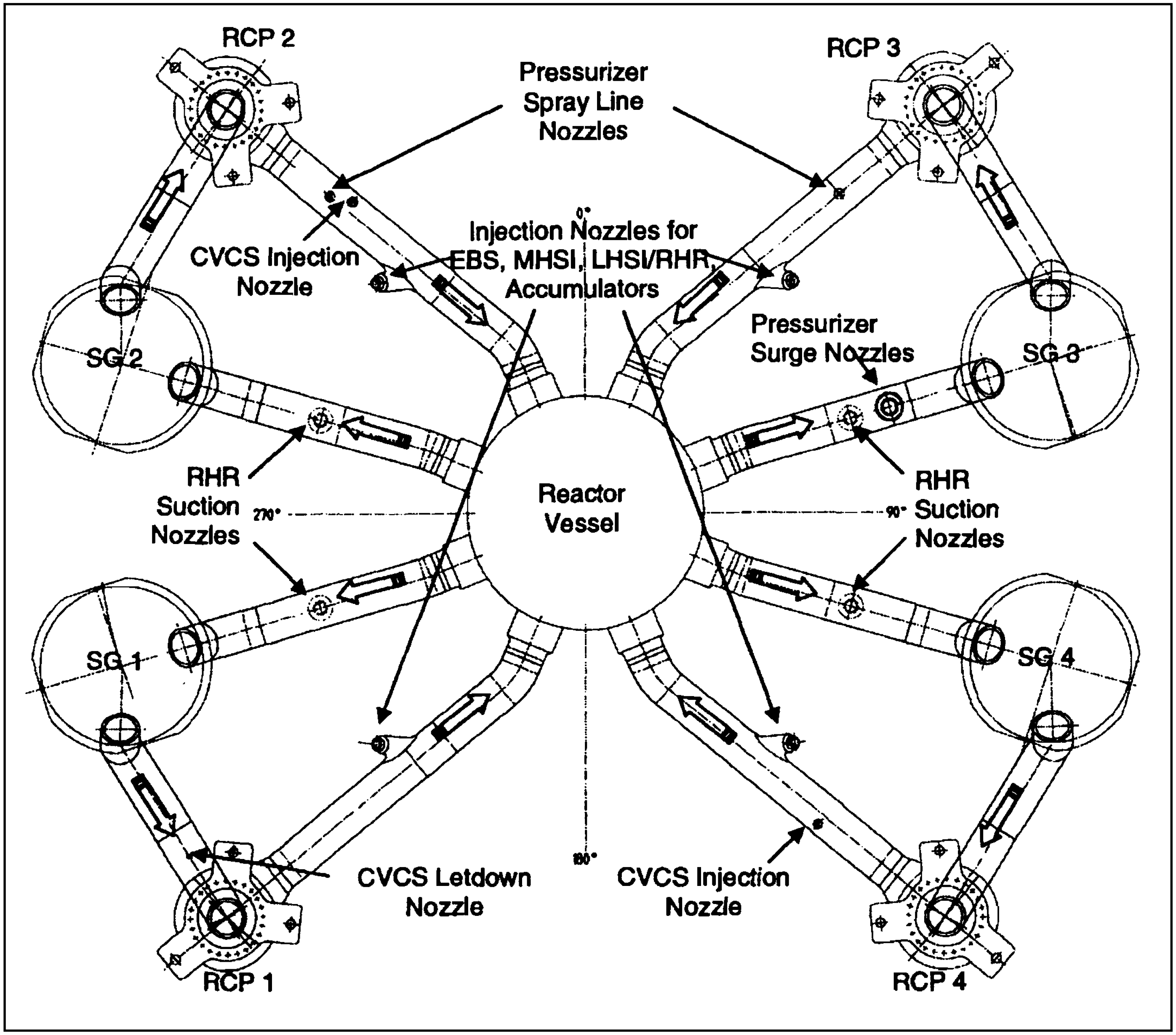


Figure 3-2

RCS Layout

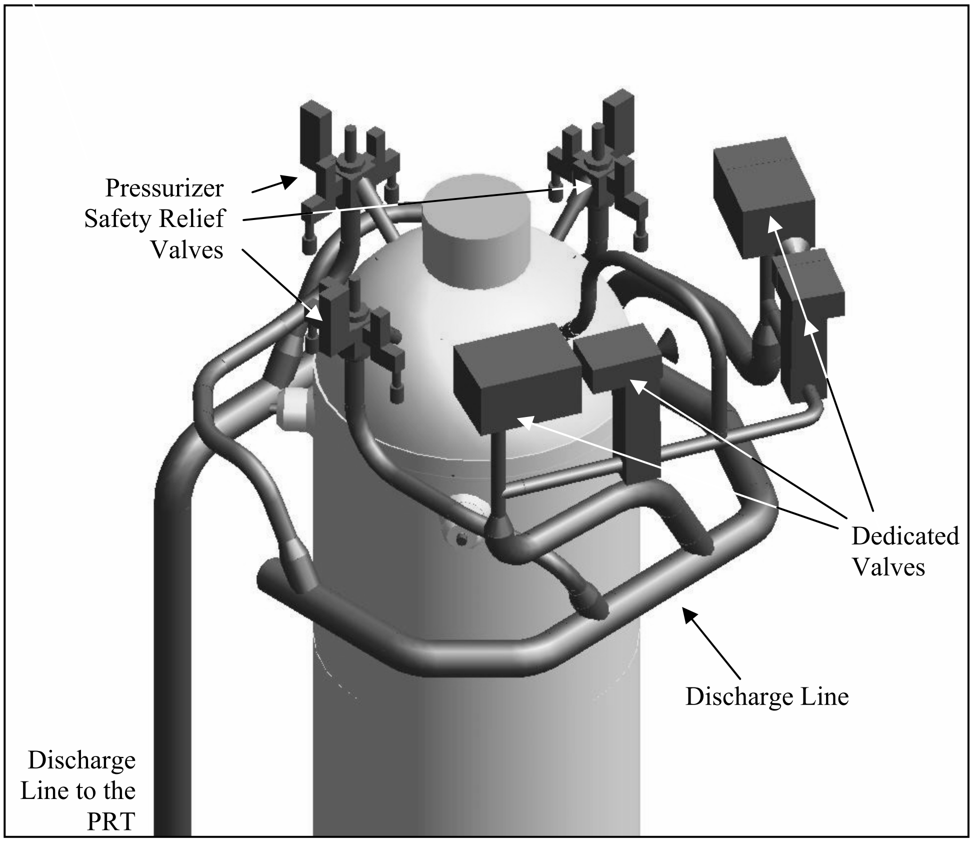


Figure 3-3
Pressurizer With Discharge Components

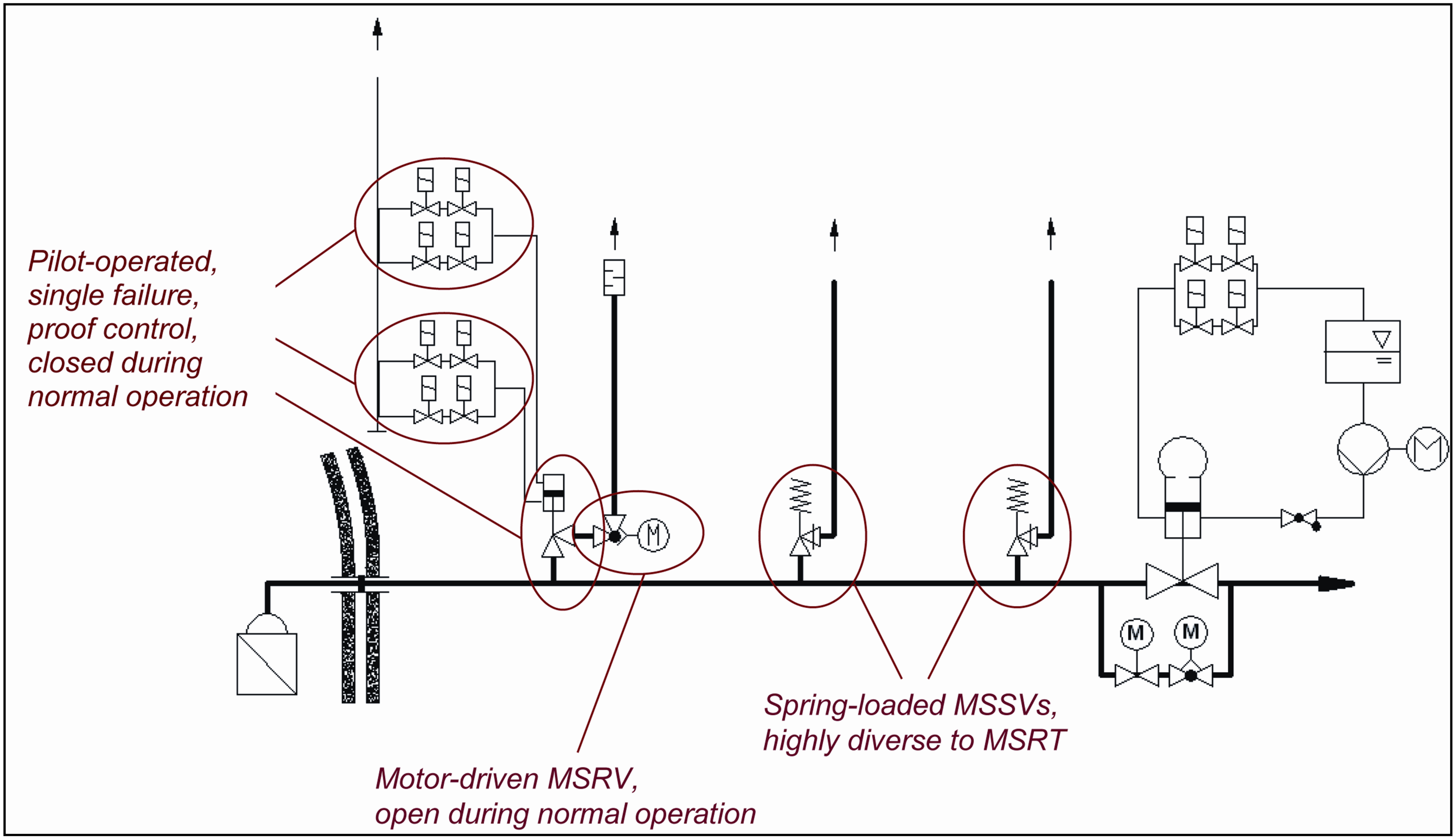


Figure 3-4
Arrangement of the Secondary Side Overpressure Protection

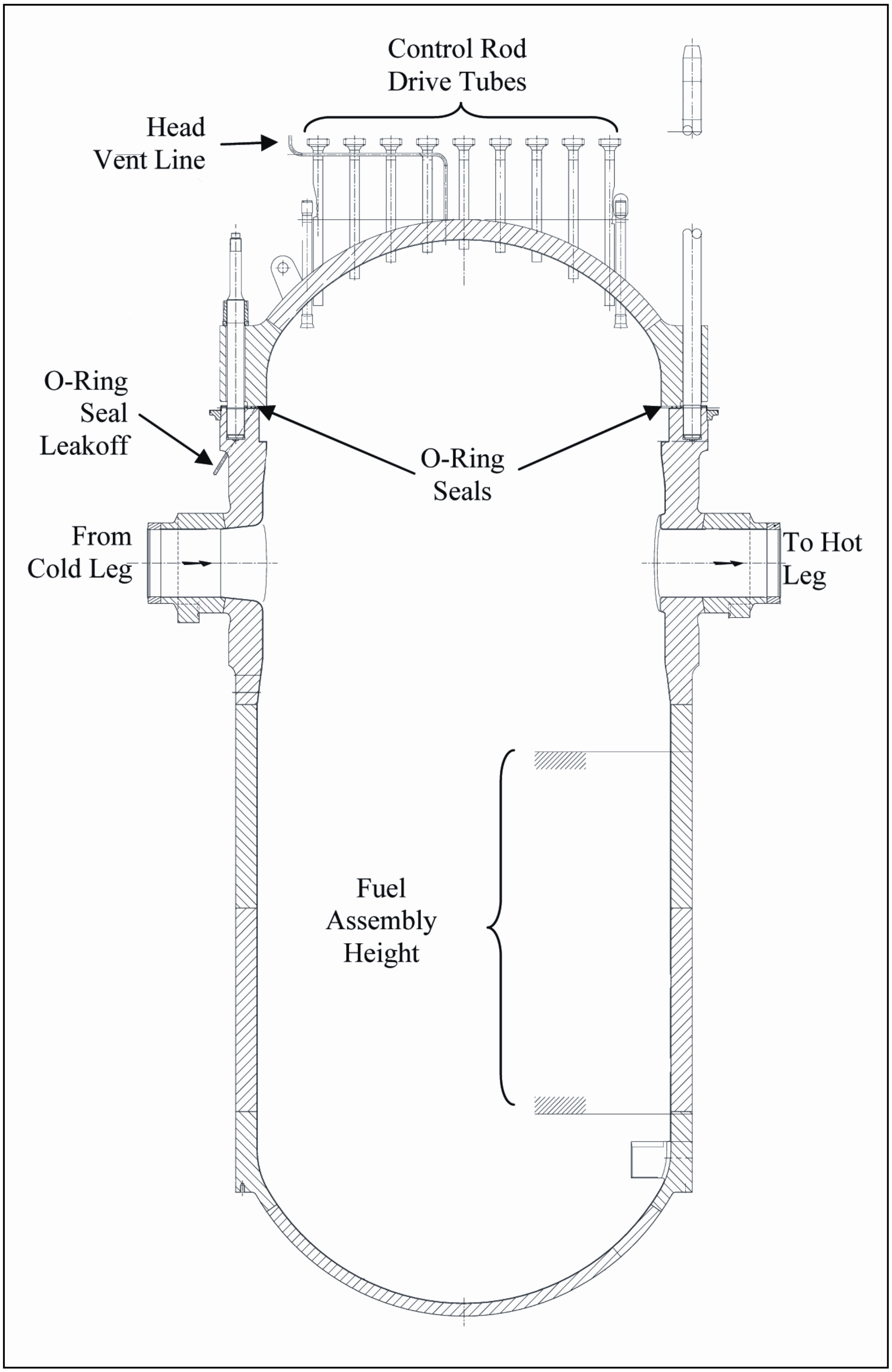


Figure 3-5

Reactor Pressure Vessel

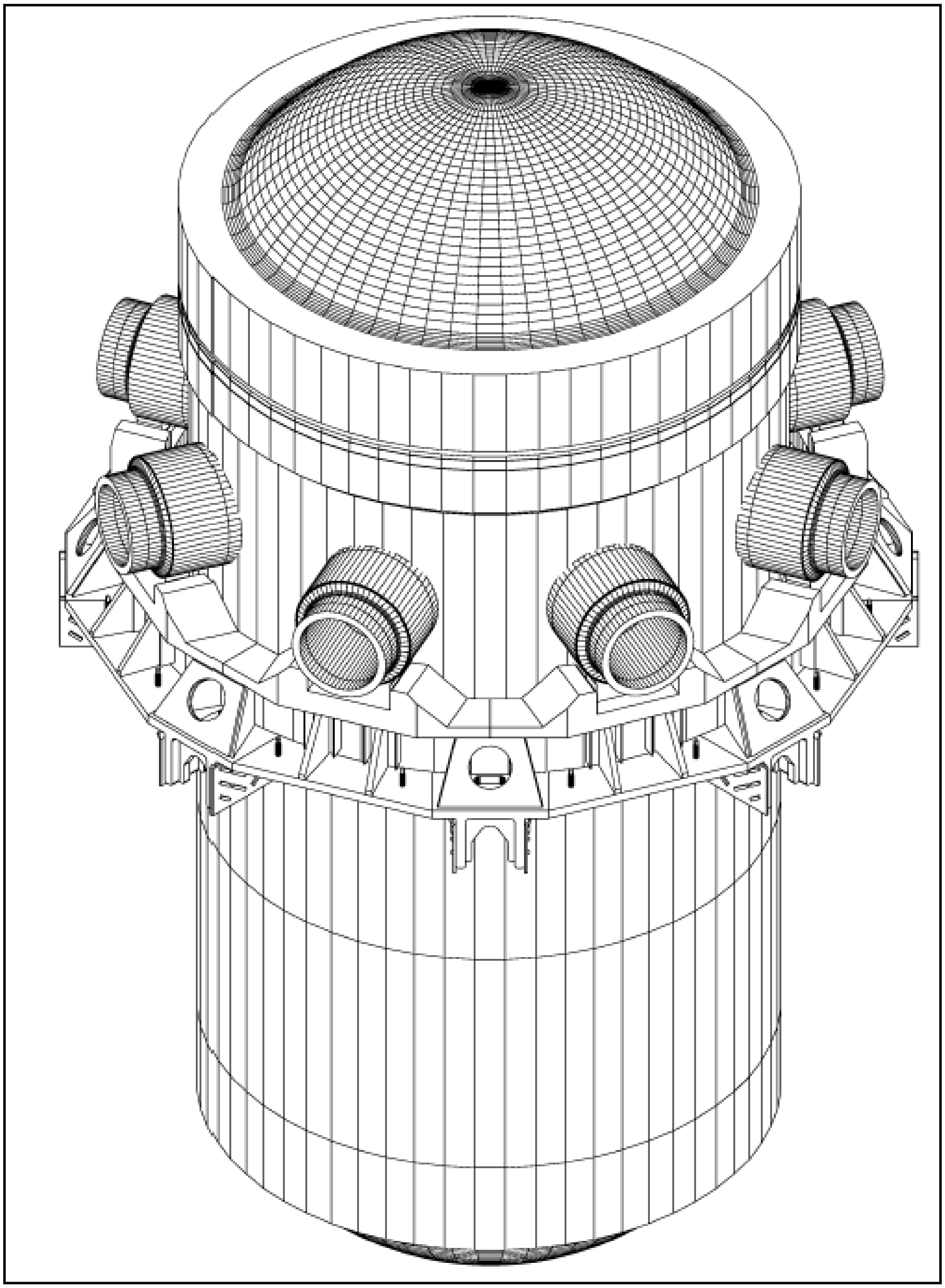


Figure 3-6
RPV Supports

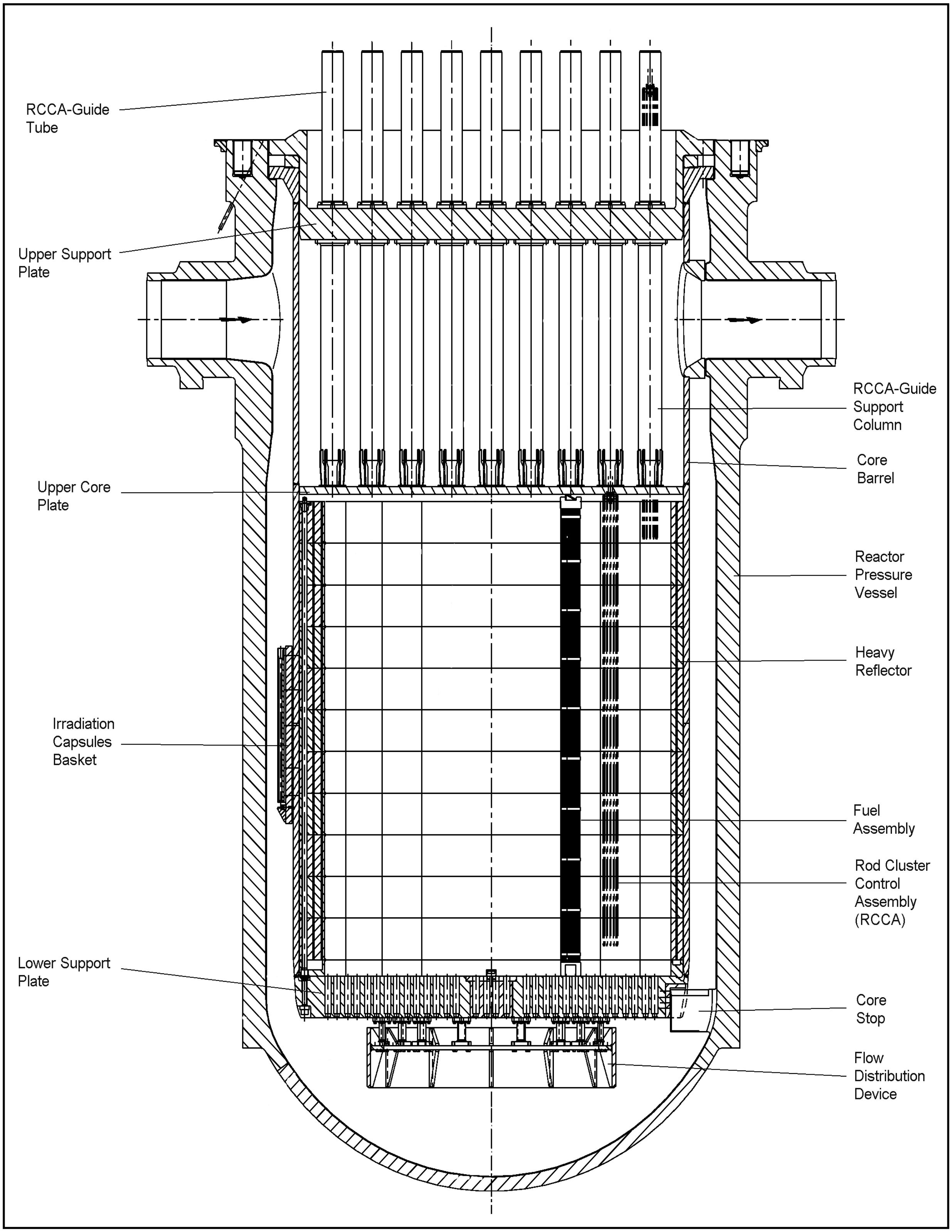


Figure 3-7
RPV Internals

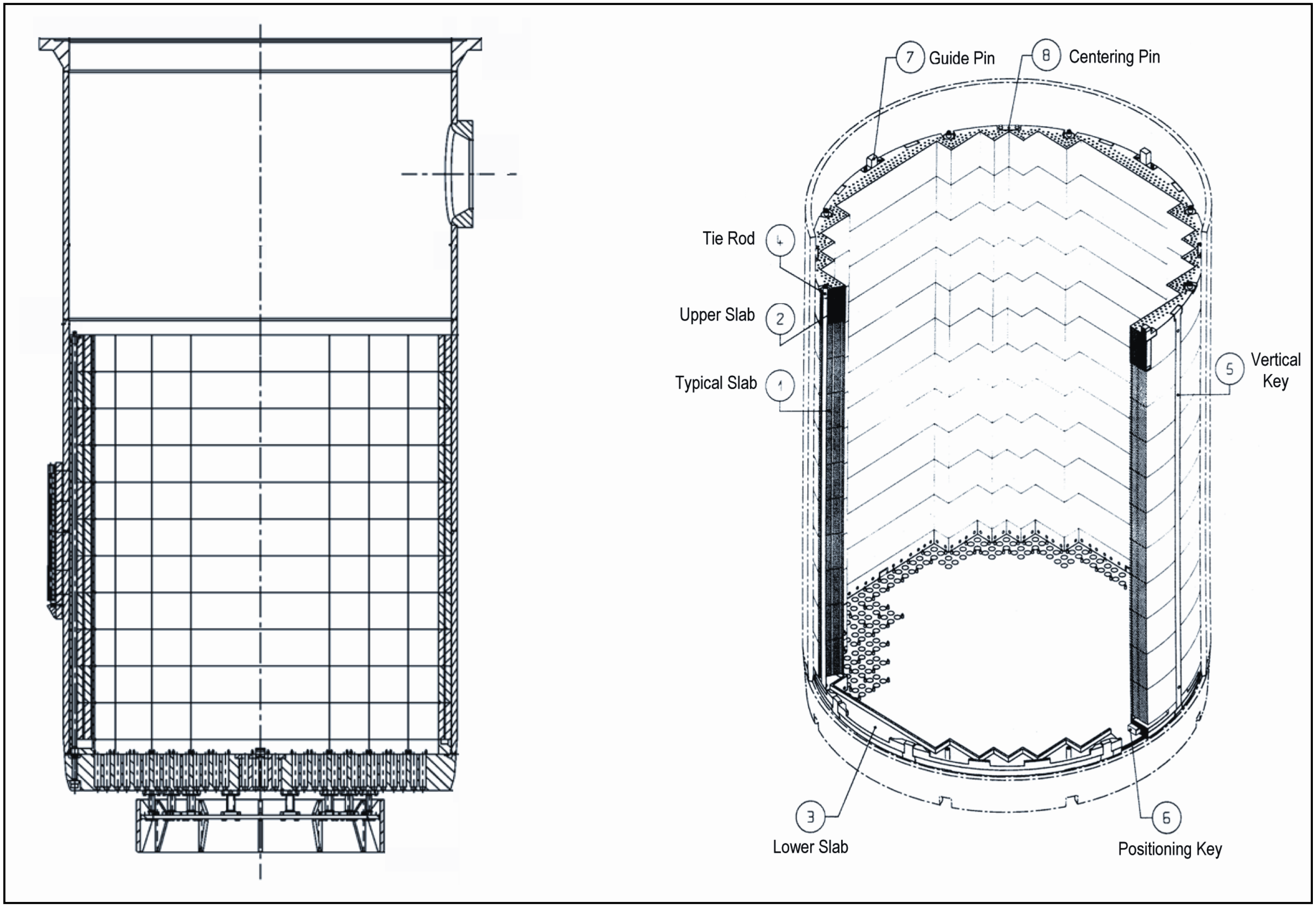


Figure 3-8
RPV Lower Internals

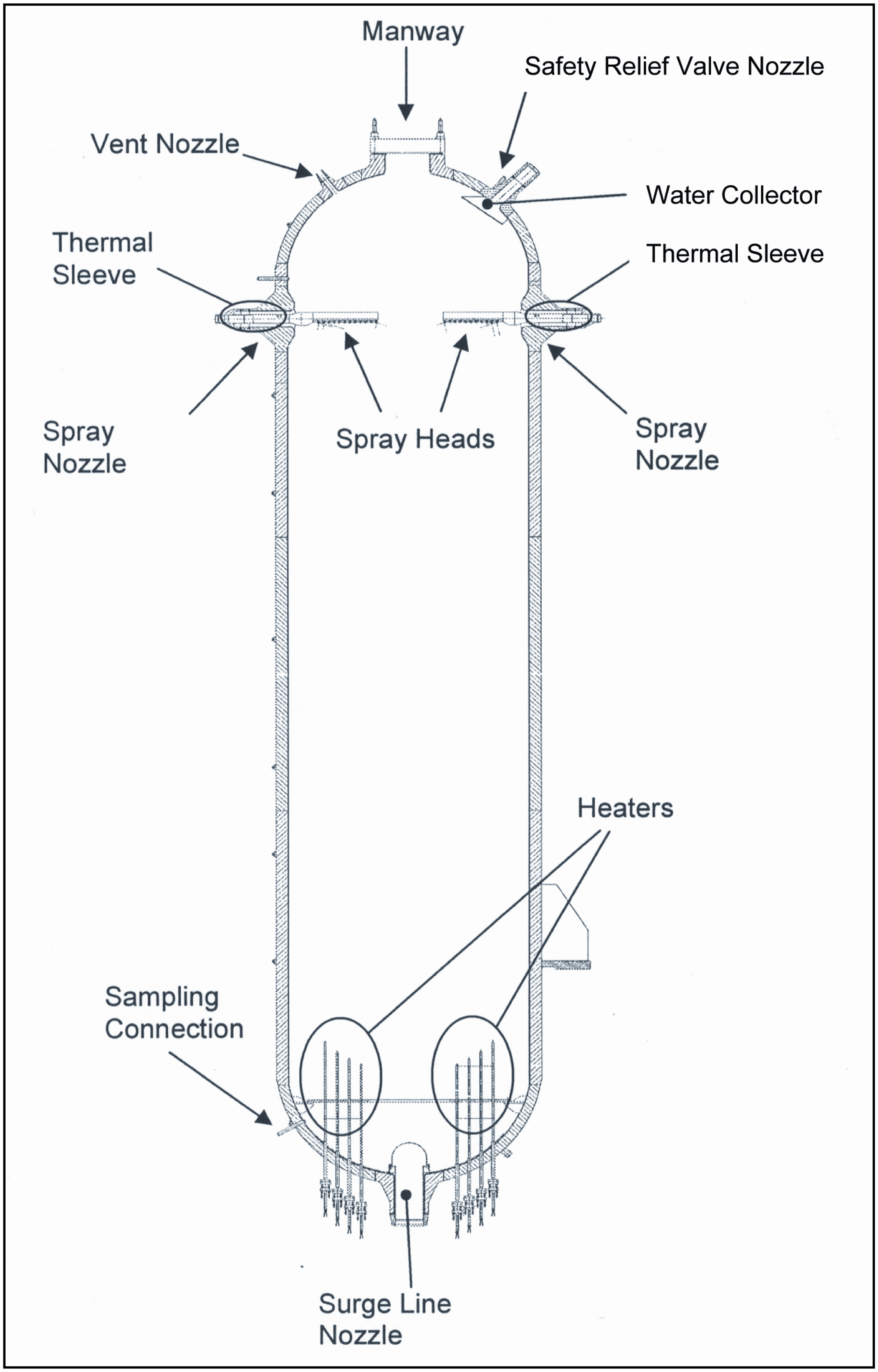


Figure 3-9
Pressurizer

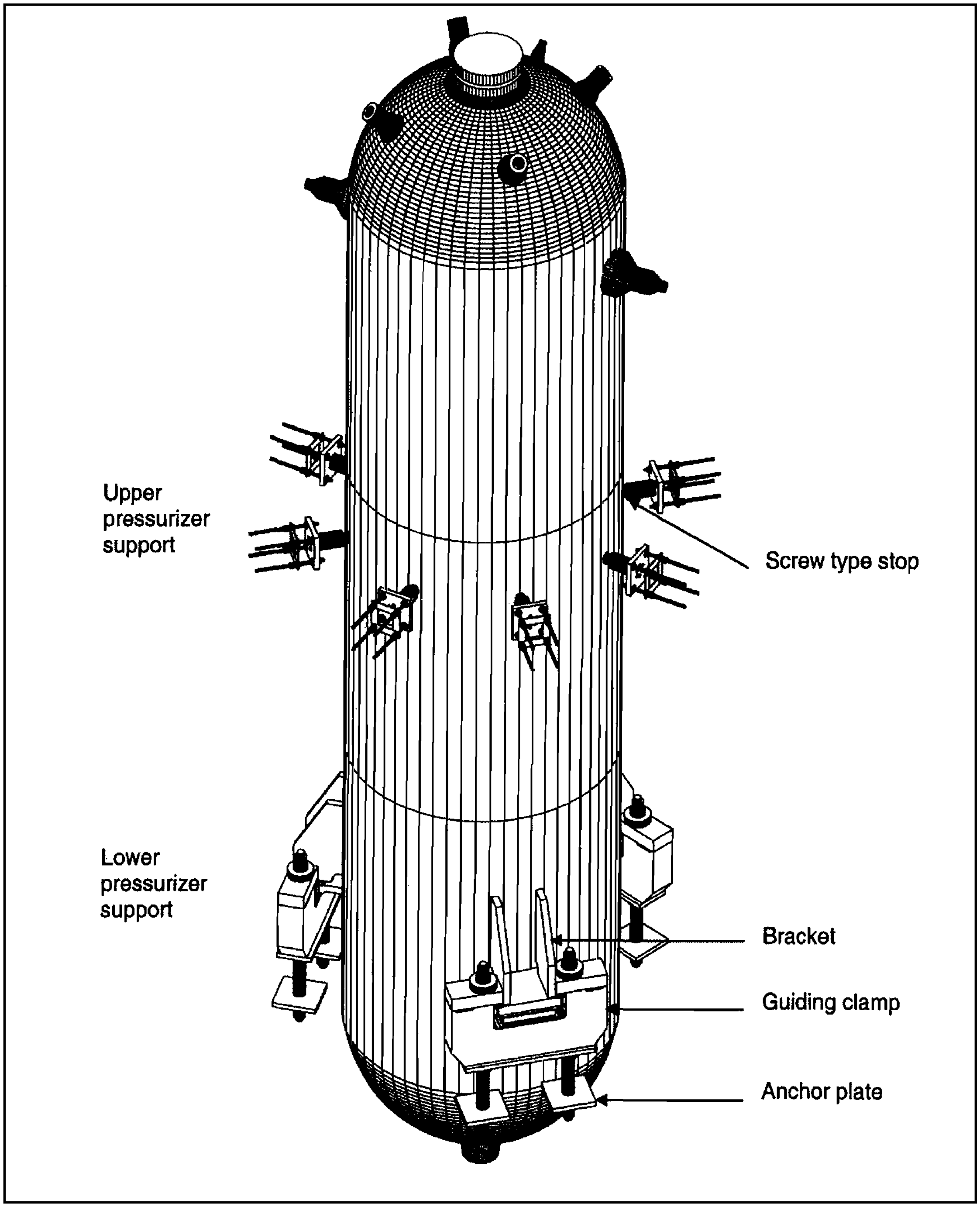


Figure 3-10
Pressurizer Supports

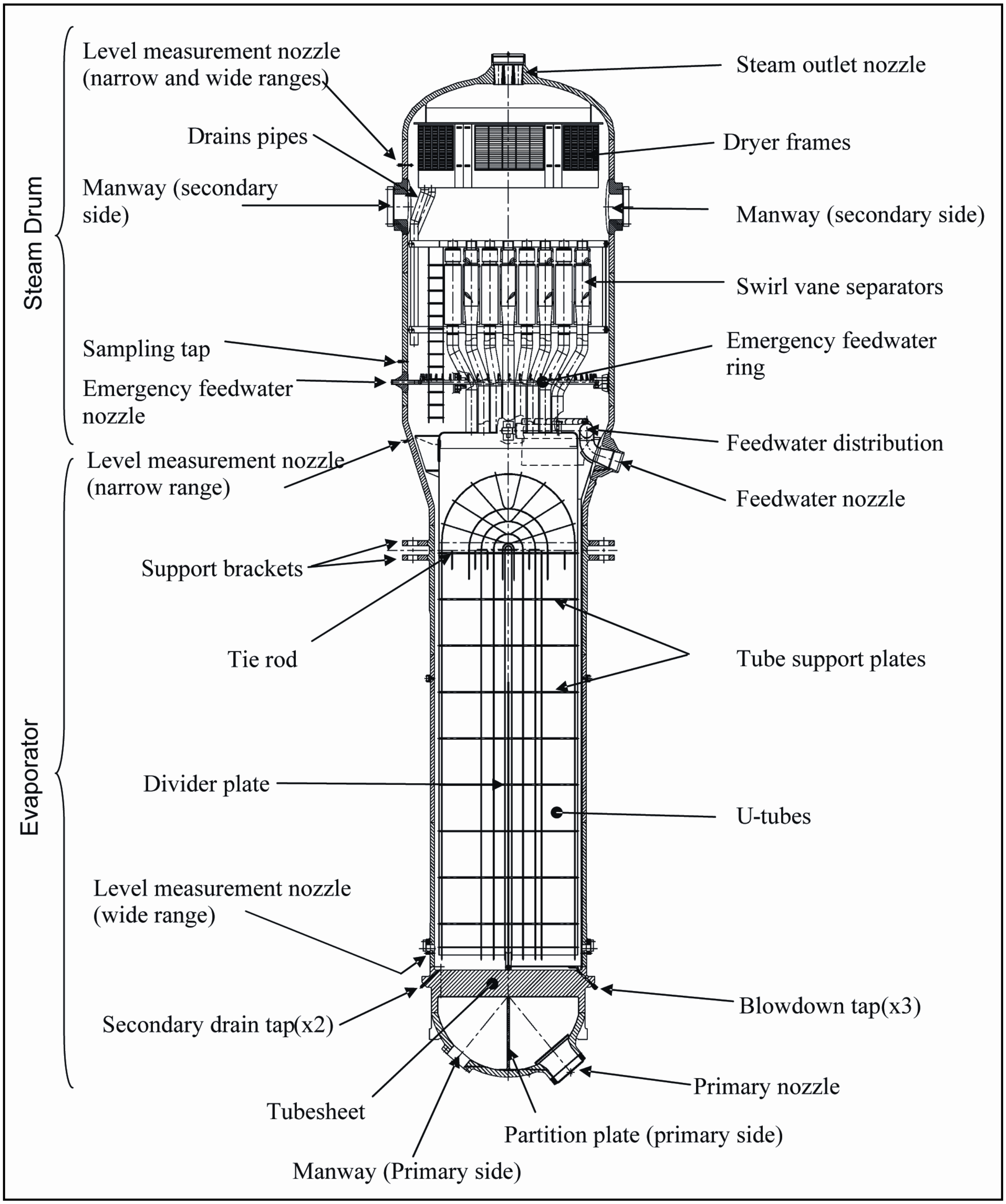


Figure 3-11

Steam Generator

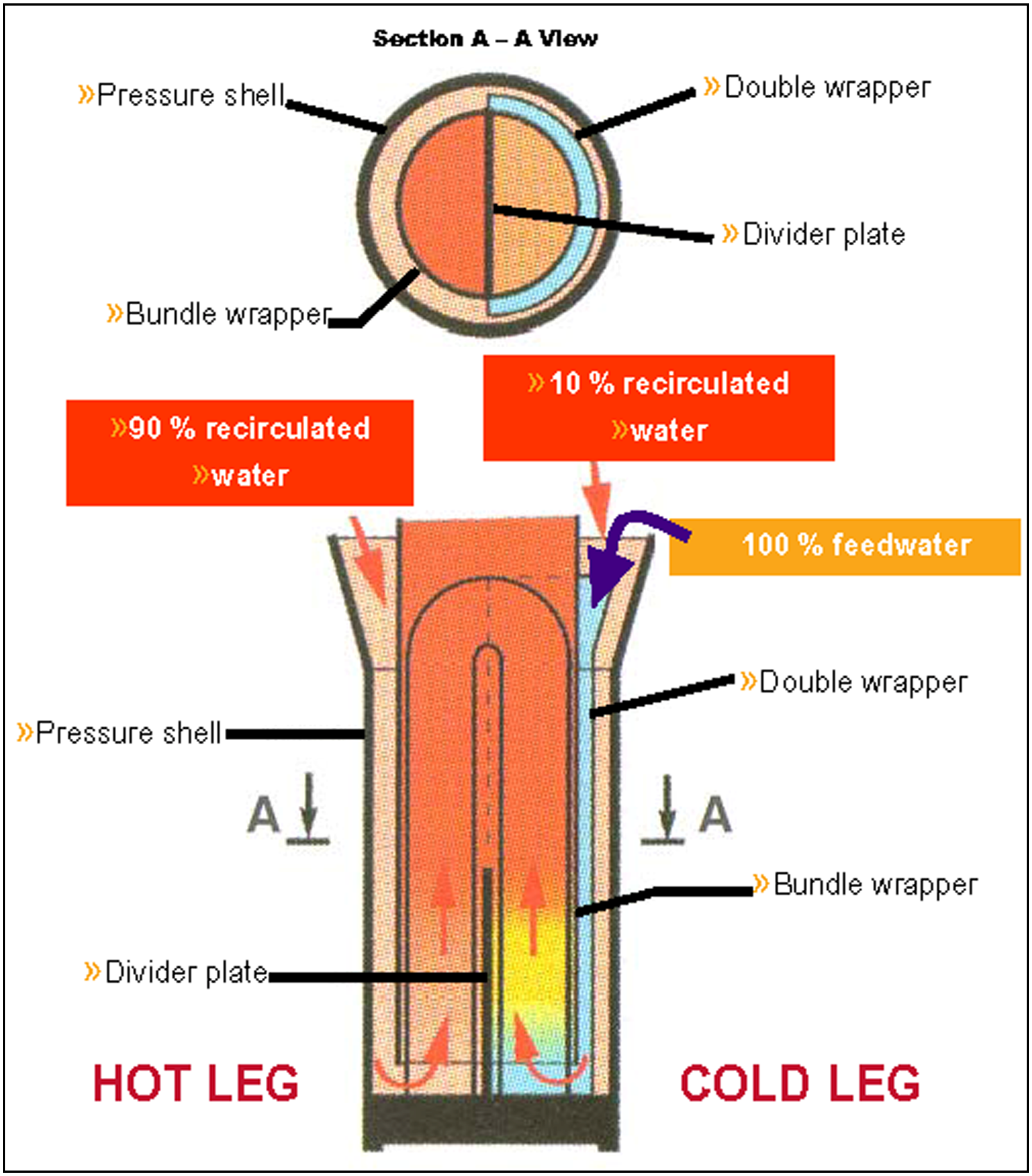


Figure 3-12
Steam Generator: Principle of the Axial Economizer

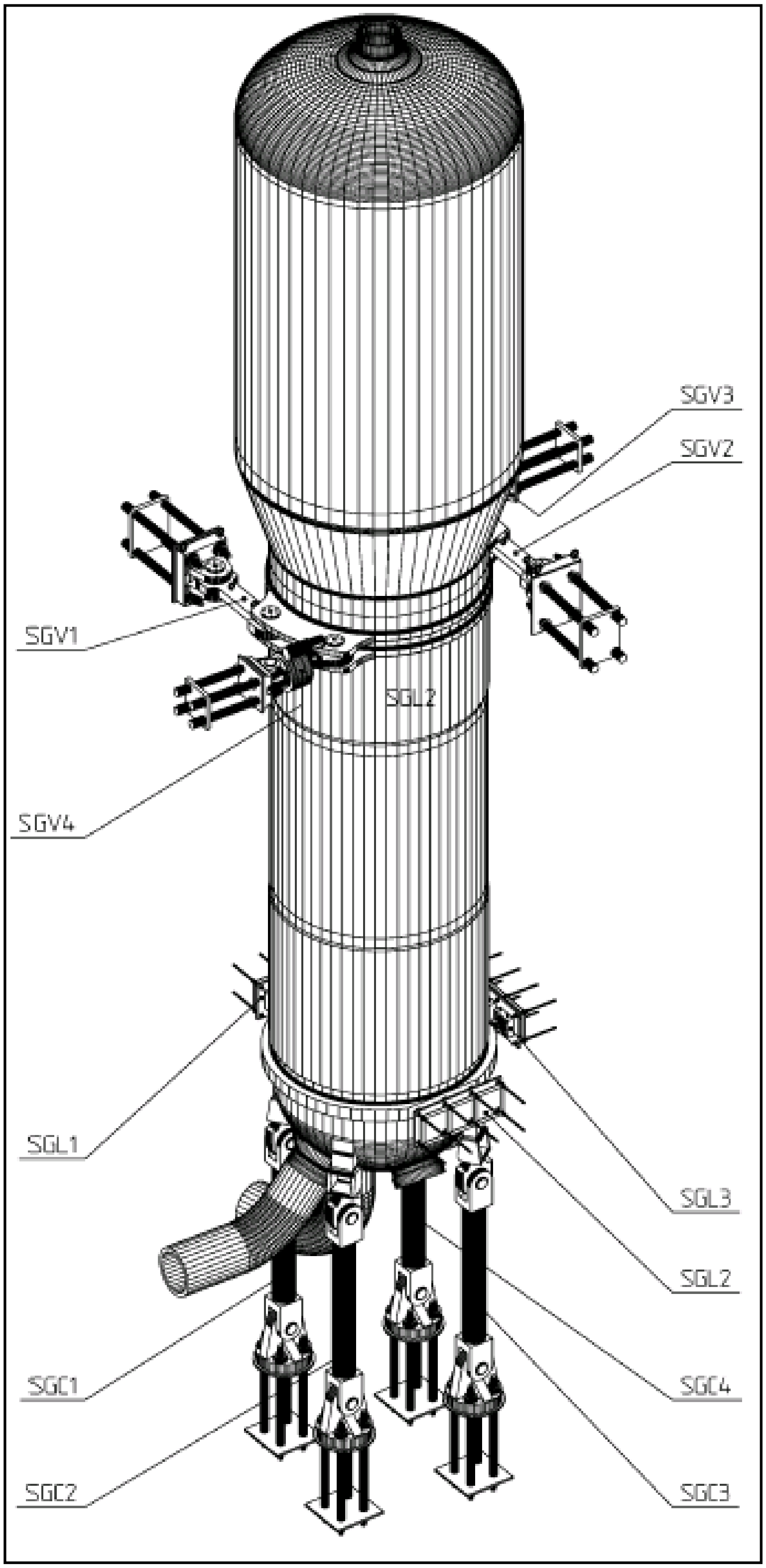


Figure 3-13
Steam Generator Supports

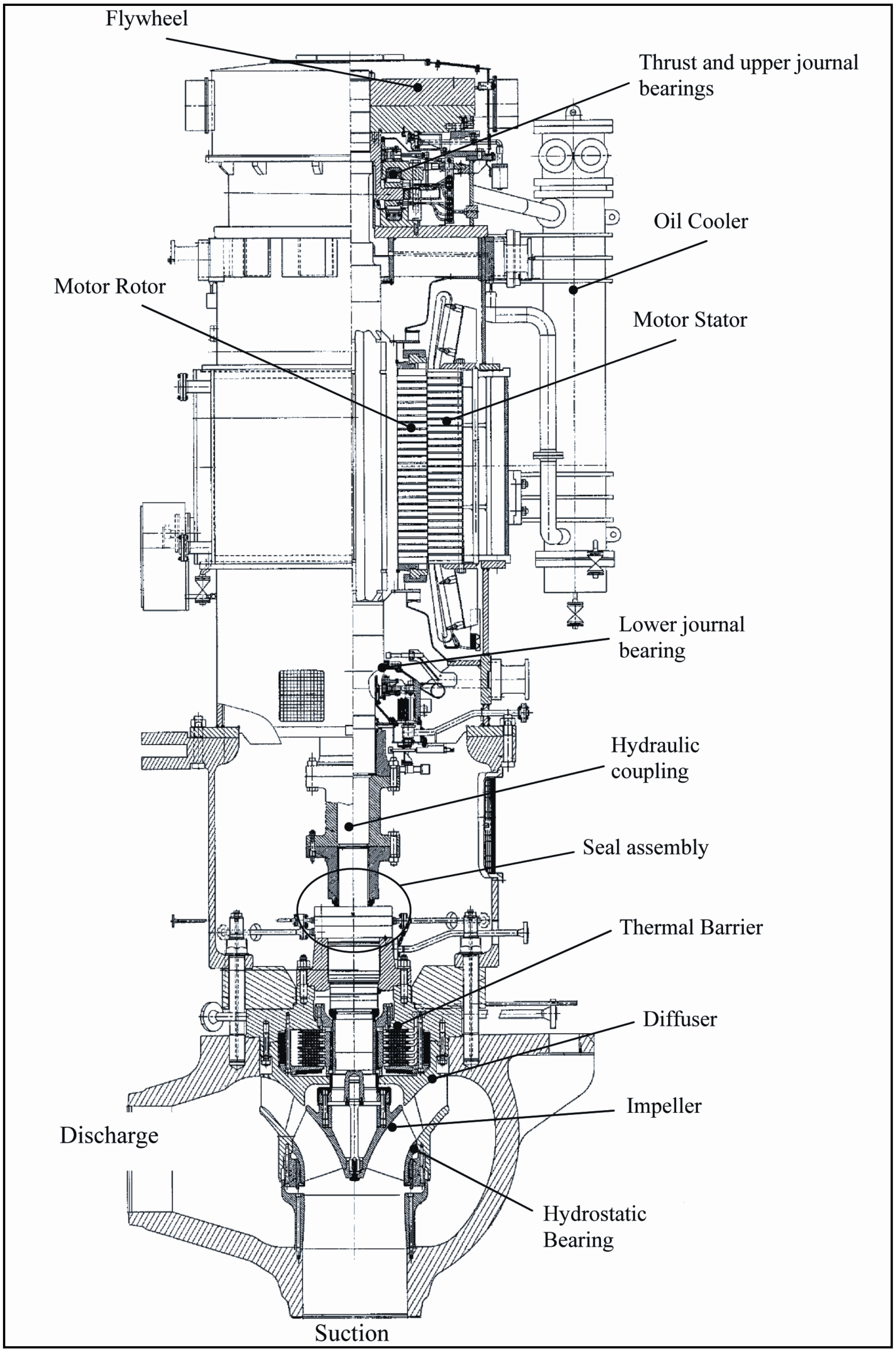


Figure 3-14
Reactor Coolant Pump Assembly

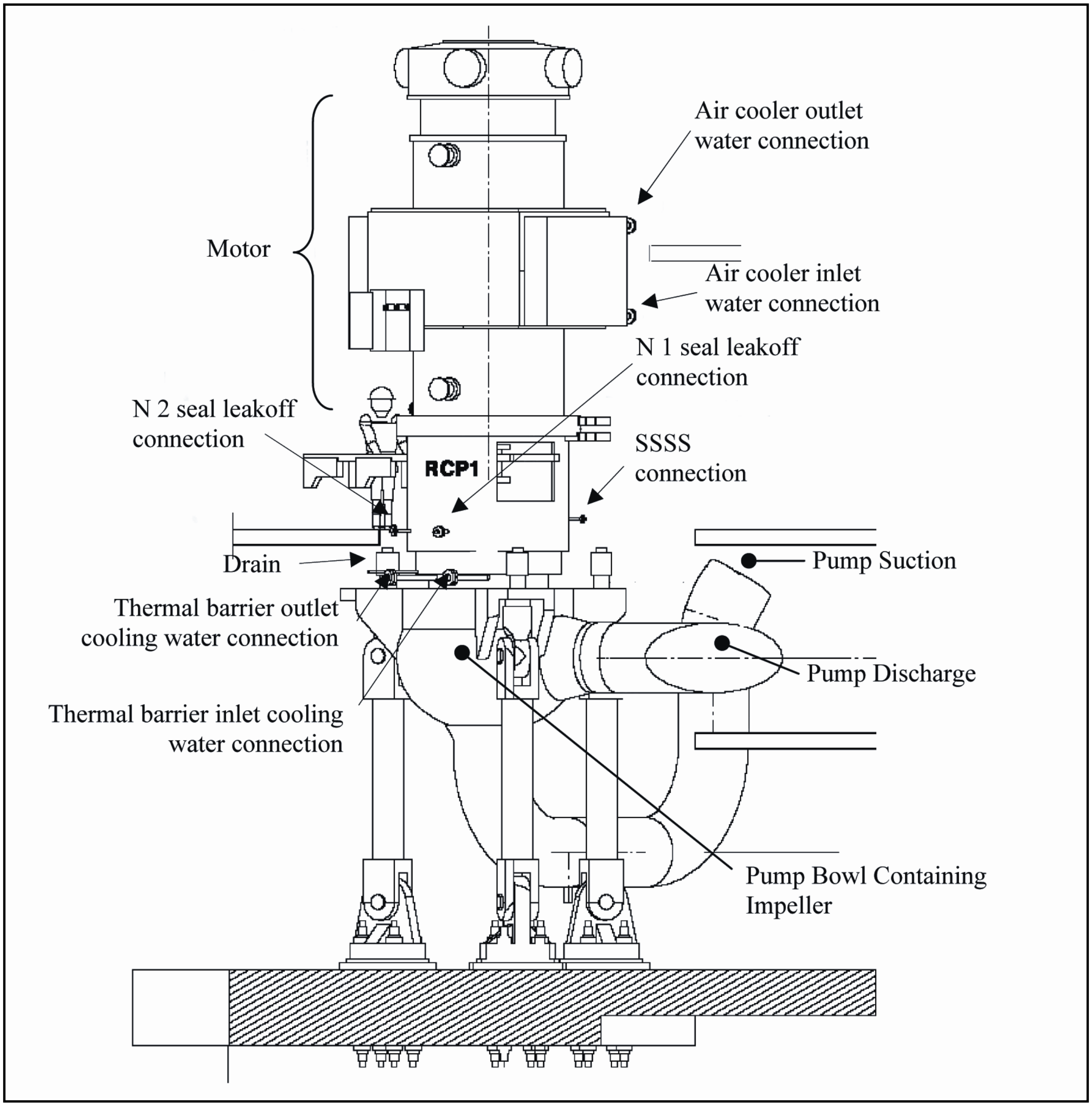


Figure 3-15
Reactor Coolant Pump Supports

4.0 PRINCIPAL FLUID SYSTEMS

4.1 Conceptual Features

This section describes the conceptual features of the principal fluid systems of the EPR.

4.1.1 Safety Functions

The safety functions provided by the principal fluid systems are:

- Control of reactivity
- RCS inventory and integrity
- Residual heat removal

Reactivity Control

The fluid systems that ensure or contribute to reactivity control are the Extra Borating System (EBS), the CVCS, and the SIS.

RCS Inventory and Integrity

Fluid systems that ensure control of the RCS inventory are the CVCS and the SIS. Fluid systems or equipment that contribute to ensuring RCS integrity are those required for OPP and those needed for cooling and supply of injection water to the RCP seals (since damage to an RCP seal could result in loss of RCS integrity). Cooling of the RCP seals is provided by the CCWS, with injection water ensured by the CVCS.

OPP of the RCS is ensured by the primary side OPP system.

Residual Heat Removal

RHR from the SG secondary side is ensured by the Emergency Feedwater System (EFWS) and the secondary side OPP system. The Start-Up and Shutdown System (SSS) also contributes to this function.

RHR from the primary side during shutdown conditions is ensured by the Residual Heat Removal System (RHRS) combined with the Low Head Safety Injection (LHSI) system.

4.1.2 Systems Required to Support Operation of Fluid Systems

The cooling water support systems for the principal fluid systems are described below.

The CCWS consists of both a safety-related portion and a non-safety-related portion. The safety-related portion has the same number of trains as the safety systems that require cooling by the CCWS. Common headers providing redundant and safety-grade isolation valves and train separation (as required during plant transients or accidents) connect both components of the CCWS.

The Essential Service Water System (ESWS) provides the heat sink for the CCWS and has the same number of trains as the CCWS.

4.1.3 Configuration of Systems

The following systems and their associated electrical power supply and I&C systems are arranged in a four-train configuration:

- SIS/RHRS
- EFWS
- CCWS
- ESWS

The four-train arrangement for the principal fluid systems, corresponding to the four-loop configuration of the RCS, leads to a simplified design concept for the fluid systems in that each system is connected to a single loop with no operator action required to balance flow between loops. This arrangement also allows flexibility and redundancy during plant shutdown conditions when capacity requirements for heat removal and other functions are reduced relative to the needs associated with normal power operations. The four-train configuration also offers the possibility of extending maintenance intervals on parts of the systems, which can be beneficial for preventive maintenance or general repair requirements. For instance, preventive maintenance of one complete safety train can be performed during power operation.

The following systems are arranged in a two-train configuration:

- EBS
- Fuel Pool Cooling System (FPCS), in which two FPCS pumps operate in parallel in each of the two trains

The organization of the systems that provide injection of water into the RCS is given below.

MHSI	
Medium Head Safety Injection (MHSI) System	4 trains; cold leg injection
Accumulator	4 accumulators; cold leg injection
LHSI/RHR	
Low Head Safety Injection/ Residual Heat Removal System	4 trains; cold leg injection for short term + cold and hot leg injection for long term
EBS	
Extra Borating System	2 trains; injection of borated water; cold leg injection
IRWST	
In-Containment Refueling Water Storage Tank	Storage of borated water inside containment

Figure 4-1 shows a schematic of the main fluid systems.

Each of the systems identified above is described in more detail below.

4.2 Safety Injection/Residual Heat Removal System

The SIS/RHRS performs normal shutdown cooling, as well as emergency coolant injection and recirculation functions to maintain reactor core coolant inventory and provide adequate decay heat removal following a LOCA. The SIS/RHRS also maintains reactor core inventory following a MSLB.

The safety functions of the SIS/RHRS are:

- Rapid reflood of the RPV and the reactor core following a Large Break LOCA (LBLOCA)
- Long-term injection of water to the core for small, intermediate and LBLOCA
- Injection of water for intermediate and LBLOCA to terminate the release of steam to the containment atmosphere as early as possible
- Injection of water to the RCS for small to intermediate size LOCA or SGTR, at any pressure less than the SG safety valve or relief valve discharge pressure. Partial cooldown of the SGs by the reactor protection system ensures adequate SIS flow (see Section 7.2.6.1).
- Cooling of the IRWST in the event of a LOCA
- Mixing of water recirculated in the long term after a LOCA to ensure homogeneous boron concentration and temperature
- Injection of water, in conjunction with reducing RCS pressure through use of the PZR safety valves, to ensure RHR from the RCS and cool down to a cold shutdown condition in the event of a loss of decay heat removal via the SGs (feed and bleed mode)
- Emergency makeup to the RCS in the event of a loss of water inventory during cold shutdown or refueling

The normal operational functions of the SIS/RHRS are:

- RHR to reach and maintain cold shutdown and refueling conditions
- Transfer water from the IRWST to the reactor cavity in preparation for refueling operations
- Cooling and mixing of the IRWST contents during normal plant operating conditions

The calculated cooling performance of the SIS/RHRS following postulated LOCAs will conform to the criteria of 10 CFR 50.46. The SIS/RHRS has sufficient capacity, diversity, and independence to perform its required safety functions following design basis transients or accidents assuming a single failure in one train while a second train is out-of-service for preventive maintenance.

The SIS/RHRS consists of four independent trains, each providing injection capability by an accumulator pressurized with nitrogen gas, a MHSI pump, and a LHSI pump. The LHSI/RHR pumps also perform the operational functions of the RHRS. Each of the four SIS trains is provided with a separate suction connection to the IRWST. Guard pipes are provided for sump suction piping between the sump connection and the suction isolation valve. The sumps are provided with a series of screens, ensuring protection of the SIS pumps against debris entrained with IRWST fluid.

Each pump is provided with a miniflow line routed to the IRWST. The LHSI/RHR pump miniflow also provides cooling and mixing of the IRWST.

In the injection mode, the MHSI and LHSI/RHR pumps take suction from the IRWST and inject into the RCS through nozzles located in the top of the piping. These pumps are located in the Safeguard Buildings, close to the containment. The LHSI/RHR pumps and the MHSI pumps normally inject into the cold legs. In the long term following a LOCA, the LHSI discharge can be switched over to the hot legs to limit the boron concentration in the core, thus reducing the risk of crystallization in the upper part of the core.

A LHSI/RHR heat exchanger is located downstream of each LHSI/RHR pump. These heat exchangers are installed in the Safeguard Buildings and cooled by the CCWS. The accumulators are located inside the containment and inject into the RCS cold legs when the RCS pressure falls below the accumulator pressure, using the same injection nozzles as the LHSI/RHR and MHSI pumps.

During RHR operation, the LHSI/RHR pumps take suction from the RCS hot leg and discharge through the LHSI/RHR heat exchangers back to the RCS cold leg. During shutdown, the LHSI/RHR pump is used in the RHR mode, but the MHSI pump remains available for water make-up in the event of a LOCA.

All four SIS/RHRS trains are powered from separate emergency buses, each backed by an EDG. The LHSI/RHR pumps in Trains 1 and 4 are also backed-up by the SBO diesels.

One SIS/RHRS train is located in the each of the Safeguard Buildings, thereby providing separation and/or physical protection from external and internal hazards.

Figure 4-2 shows the flow schematic of the SIS/RHRS.

IRWST

The function of the IRWST is to contain a large amount of borated water at a homogeneous concentration and temperature. The borated water is used to flood the refueling cavity for normal refueling. It is also the safety-related source of water for emergency core cooling in the event of a LOCA and is a source of water for containment cooling and core melt cooling in the event of a severe accident. During a LOCA, the IRWST collects the discharge from the RCS, allowing it to be recirculated by the SIS.

The IRWST is essentially an open pool within a partly immersed building structure. The wall of the IRWST is lined with an austenitic stainless steel liner to avoid interaction of the boric acid and concrete structure and to ensure water tightness. Each of the four SIS (safety-related) and two SAHRS (non safety-related) trains is provided with a separate sump suction connection to the IRWST. To prevent RCS thermal insulation and other debris from reducing the suction head of the SIS and SAHRS pumps during and following a LOCA, a series of barriers is used to minimize the amount of debris which can reach the sumps. The heavy floor beneath the RCS has strategically placed openings through which water drains to the IRWST. Each of these openings has a weir (curb) around it and a trash rack in the opening. Beneath the openings are retaining baskets which trap the larger sized debris while allowing water to flow into the tank. Each of the sumps is provided with a cage screen with reverse inclined sieves so caked debris can be backwashed to the floor of the tank. Each of the sump screens is sized such that, should all anticipated debris reach an individual screen, that screen will not be prevented from providing its function. Vortex suppressor grids under each sump screen prevent loss of suction if the IRWST water level is low. Screen backwashing functions are accomplished via the MHSI pump miniflow lines. The LHSI miniflow lines are operated continuously to permit cooling and mixing of the IRWST water.

Except for the sump suction isolation valves, all IRWST related components are passive. The isolation valves are powered from safety-related buses. Suction lines from the IRWST are equipped with guard pipes in addition to the sump isolation valves in order to satisfy single failure criteria to prevent loss of water inventory.

Figure 4-3 shows a flow diagram of the IRWST and related components.

4.3 Extra Borating System

The EBS provides high pressure boration to shut down the reactor following accidents when the CVCS is not available. The EBS is also designed to minimize fresh boron consumption and to avoid the use of active components during standby (i.e., complicated heat tracing devices to prevent crystallization). This design objective allows the RCS to be tested without first injecting fresh boron.

The EBS is a safety-related system that performs the following functions:

- Boration of the RCS in all anticipated operational transients and postulated accidents to reach a controlled state at all primary pressure levels
- Maintain the reactor in a shutdown state at any reactor temperature without control rods
- Ensure the RCS boration required to return the core to sub-critical conditions after reactor scram (trip)
- For a SBLOCA, less than ½-inch diameter break, used in combination with the SIS when the injected flow is not sufficient to reach the required boron concentration for RHRS connection (safe shutdown state)
- For Anticipated Transients Without Scram (ATWS), automatically starts to ensure the RCS boration required is provided to shut down the reactor (to a sub-critical condition)

The EBS consists of two identical primary trains. Each primary train is composed of its own boron tank, a high pressure 100% capacity pump, a test line, and injection lines to the RCS. The boron tanks and the primary train lines are filled with borated water and are located in a temperature controlled room that guarantees the non-crystallization of the boric acid.

The EBS does not perform any function supporting normal plant operation:

- The pumps are shut down except during periodic test or heating/mixing of the tanks.
- All hand-operated valves from the Extra Borating Tanks up to the RCS are normally open.
- The motorized and manual isolation valves of the test lines are normally closed.
- The motorized isolation valves of the injection lines connected to the RCS are normally closed.
- The containment isolation valves are normally open.

No active components are in service during EBS standby except the boron room heaters.

A header connecting the bottom of the two boron tanks is normally isolated by a remotely operated closed manual valve and can allow injection (or draining) from both tanks using only one EBS pump.

The two EBS trains are assigned to Safety Divisions 1 and 4. These are remotely operated and powered by emergency buses, each backed by an emergency diesel and further backed-up by the SBO diesels. The two trains are installed in two separate layout divisions within the Fuel Building. Each of the two trains can inject into two RCS loops via the cold legs.

4.4 Emergency Feedwater System

The EFWS supplies water to the SGs to maintain water level and remove decay heat following the loss of normal feedwater supplies due to anticipated operational transients and design basis accident conditions. This ensures the removal of heat from the RCS, which is first transferred to the secondary side via the SGs and then discharged as steam to the condenser, or via the SG MSRVs.

During normal power operation, the feedwater supply to the SGs is provided by the Main Feedwater System (MFWS). For start-up and shutdown operation of the plant, a dedicated system, the SSS, is provided. The SSS is actuated automatically in the event of a low level in the SGs following a reactor trip with the loss of the MFWS. The SSS actuation reduces the frequency of the EFWS actuation and increases the reliability of the entire feedwater system.

The EFWS is a safety-related system that performs the following functions:

- Provide sufficient flow to the SGs to recover and maintain SG water inventory and remove residual heat from the RCS via the SGs and MSRVs to assist in the cooldown and depressurization of the RCS to RHR/SIS conditions under design basis transient and accident conditions
- Isolation of EFWS flow to the affected SG following a MSLB to prevent overcooling the RCS and associated positive reactivity
- Isolation of EFW pump flow to the SG with a tube rupture upon SG high level to mitigate the potential radiological consequences of a SGTR event
- Provide sufficient volume in the storage pools to maintain hot shutdown conditions for 24 hours following beyond design basis events (SBO and loss of ultimate heat sink)

The EFWS has sufficient capacity and independence to perform its required safety functions following design basis transients or accidents assuming a single failure in one EFW pump train and a second train being out-of-service for preventive maintenance.

The EFWS does not perform any functions supporting normal plant operation.

The EFWS has four separate and independent trains, each consisting of a water storage pool, pump, control valves, isolation valves, piping, and instrumentation. A supply header is provided that allows cross-connecting the storage pools to the pump suctions, and another header that allows cross-connecting the discharge of the pumps to the SGs. These headers are normally isolated and require local operator action to change storage pool or pump discharge alignment. The non-safety Demineralized Water Distribution System can be used to provide make-up to the EFWS storage pools.

Each of four emergency feedwater pumps is powered by a separate electrical division supplied by its own EDG. In case of common mode failure of all EDGs, two of the motor-driven EFW pumps are powered by two diverse SBO diesels.

One EFWS train is located in each of the Safeguard Buildings, providing separation and/or physical protection from external and internal hazards. The storage pools are internally lined concrete and are structurally part of each Safeguard Building.

Figure 4-4 shows the flow schematic of the EFWS.

4.5 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) is the interface system between the high pressure RCS and the low pressure systems in the Nuclear Auxiliary Building and Fuel Building. The CVCS provides a flow path for the continuous letdown and charging of RCS water. The CVCS maintains the RCS water inventory at the desired level via the PZR level control system and provides RCP seal water injection and auxiliary spray for PZR cooldown.

The CVCS is an operational system and is not required for the mitigation of design basis accidents. However, the CVCS may be utilized to preclude the use of safety systems during minor transients (e.g., boron dilution events). The system is normally in continuous operation during all modes of plant operation from normal power operation to cold shutdown.

The system performs the following operational functions:

- Continuous control of the RCS water inventory during all normal plant operating conditions utilizing the charging and letdown flow path

- Provide make-up to the RCS in the event of a loss of inventory due to limited leakage
- Adjust the RCS boron concentration as required for power variation control, plant start-up or shutdown, or core burnup compensation through the addition of boron and/or demineralized water
- Reduce PZR pressure by diverting charging flow to the PZR auxiliary spray nozzle and condensing the steam bubble in the PZR to reach SIS/RHRS conditions
- Inject cooled and purified water into the RCP seals to ensure cooling and leak tightness and return any seal leakage to the CVCS
- Provide primary coolant chemical control by interfacing with the coolant purification, treatment, degasification, and storage systems
- Control the concentration and the nature of dissolved gases in the RCS by maintaining the required hydrogen concentration in the charging flow and degasifying the reactor coolant, when required
- Filling and draining the RCS during shutdown conditions

The major components of the CVCS are two redundant centrifugal Charging Pumps, a Volume Control Tank (VCT), a Regenerative Heat Exchanger, two high pressure coolers in parallel (cooled by Component Cooling Water), two parallel high pressure Reducing Stations, a low pressure Reducing Station, and associated valves and piping.

The letdown portion of the system receives water from the RCS Loop 1 crossover leg and exits the RCS through two motor-operated isolation valves connected in series. The flow then passes through the tube side of a regenerative heat exchanger transferring heat to the charging flow returning to the RCS on the shell side. The letdown flow is further cooled in the high pressure cooler and depressurized by a pressure reduction valve. Downstream of the pressure reduction valves, a safety valve provides overpressure protection of the letdown piping inside the Reactor Building. A bypass connection is provided to allow discharging letdown flow to the Reactor Coolant Drain Tank. This connection permits letdown from the RCS if a portion of the CVCS system or equipment outside containment is not available. Also, a connection to the discharge of the RHRS is arranged to allow letdown flow when the RCS is depressurized. The letdown flow then passes through the Reactor Building into the Fuel Building. In the Fuel Building, the letdown flow is sampled and purified. During power operation, the purification flow rate is sufficient to treat at least one RCS volume in one-half day under normal conditions.

Letdown flow is degasified in the Coolant Degasification System if required. Letdown flow is then directed to the VCT and to the Hydrogenation Station where hydrogen gas is mixed into the flow stream to provide oxygen scavenging which results from the radiolysis of reactor coolant. The VCT acts as a surge tank to permit smooth control of variations in charging and letdown flow rates. Provisions also allow for the diversion of any excess letdown flow to the Coolant Storage and Supply System due to the volume expansion of RCS resulting from system heatup or any boration or dilution. Connections are provided to the CVCS to allow for chemical additions and for boric acid and demineralized water makeup.

The Charging Pumps take suction from the VCT/letdown line and increase the pressure to allow the purified coolant to be returned to the RCS. The charging pumps can also take suction from the IRWST in the event of low level in the VCT or if a dilution accident is detected. If either condition is detected, the motor-operated valves from the IRWST automatically open and the motor-operated valves from the VCT/letdown line automatically close.

The main charging pump discharge flow passes through the shell side of the regenerative heat exchanger where the temperature is increased prior to injection into the cold legs of RCS loops 2 and 4. The charging flow rate is adjusted by a motor-operated control valve in the charging pump discharge flow path, which is controlled by the PZR level control system to maintain a constant PZR level during normal power operation.

A portion of the charging flow is delivered to the RCPs for shaft seal water. The seal water is automatically controlled by motor-operated control valves to each RCP during plant conditions when seal injection is required for RCP operation. Leakage through the RCP seals is returned to the VCT to maintain CVCS inventory.

A three-way motor-operated valve downstream of the regenerative heat exchanger is provided for aligning CVCS injection flow to the pressurizer auxiliary spray nozzle to allow for reducing RCS pressure in order to reach SIS/RHR conditions.

A low-pressure reducing station is provided to allow the RHRS to utilize the letdown flow path and the coolant purification system during shutdown conditions when the RHRS is in operation.

Even though the CVCS is not required to perform any design basis accident mitigation functions and is only an operational system, the CVCS charging pumps and motor-operated valves are powered from emergency buses which are further backed-up by EDGs.

Major components of the CVCS are located in the Reactor Building and the Fuel Building. These components are protected from external hazards by the building design and are physically separated or provided with protection from internal hazards.

Figure 4-5 shows a flow schematic of the CVCS.

4.6 Component Cooling Water System

The CCWS ensures the capability to transfer heat from safety-related systems and operational cooling loads to the heat sink via the ESWS under all normal operating conditions.

Figure 4-6 shows a flow schematic of the CCWS.

The CCWS performs the following safety functions:

- Heat removal from the Safety Injection/Residual Heat Removal System (SIS/RHRS) to the ESWS
- Heat removal from the FPCS to the ESWS as long as any fuel assemblies are located in the spent fuel storage pool outside containment
- Cooling of the thermal barriers of the RCP seals
- Heat removal from the Heating, Ventilation, Air Conditioning (HVAC) chillers of Divisions 2 and 3
- Cooling of the SAHRS by two separated trains that are part of a dedicated cooling chain (this function is used for prevention of core melt and severe accident mitigation)

The CCWS consists of four separate safety classified trains (1, 2, 3 and 4) corresponding to the four layout divisions (1, 2, 3 and 4) and two separate common loop sets. One of the common loops (common 1) is connected normally either to safety train 1 or to safety train 2. The other common loop (common 2) is connected either to safety train 3 or to safety train 4.

Each safety classified train consists of the following equipment:

- One pump, equipped with the necessary minimum flow line and cooling line. Each train is capable of providing 100% of the train needs.
- One heat exchanger located downstream of the CCWS pump. This heat exchanger is cooled by the ESWS and its bypass line which is connected to the CCWS side and equipped with a control valve to control the CCWS temperature when the ESWS temperature is low.
- One surge tank (concrete tank with a steel liner) which is connected to the pump suction line and located above the highest CCW load. The surge tank is connected to a demineralized water make-up to compensate for CCWS normal leaks or component draining water.
- One sampling line, which is connected permanently to a radiation monitor.
- One chemical additive supply line.
- A set of isolation valves that separate the train from the common load set.

Two separate dedicated trains cool the SAHRS when it is used during a severe accident scenario involving core melt. Each train consists of the following equipment:

- One pump with power supplied by the SBO diesel
- One heat exchanger located downstream of the pump
- One surge tank
- One demineralized water supply line with pressurizing pump

4.7 Essential Service Water System

The ESWS consists of four separated safety-classified trains that provide cooling of the CCWS heat exchangers with water from the heat sink during all normal plant operating conditions, transients, and accidents. The ESWS also includes two additional trains of dedicated cooling for severe accident mitigation.

Figure 4-7 shows a flow schematic of the ESWS.

The safety function of the ESWS is to cool the CCWS. It also provides cooling of the SAHRS (via the dedicated SAHRS cooling chain).

The ESWS provides cooling water to the four CCWS/ESWS heat exchangers which, in turn, cool components of the safety systems. Each train consists primarily of the suction pipe from the heat sink, the pump, the discharge pipe from the pump to the ESWS/CCWS heat exchanger, and the outlet pipe from the heat exchanger to the heat sink.

The divisions of the ESWS trains are grouped two-by-two into separate rooms belonging to the same civil structures in such a way that an internal hazard affecting one train does not affect the other train.

Electrical power is supplied by independent power trains, which are backed-up by the main EDGs.

The ESWS pumps are installed in the Service Water Buildings.

4.8 Start-up and Shutdown System

During shutdown from power operation to hot standby (after the reactor and turbine are manually tripped), the MFWPs are switched off consecutively and the SSS pump takes over SG feed, with main steam discharging to the condenser via the main steam bypass.

During shutdown from hot standby to LHSI/RHR entry conditions (248°F), automatic cooldown of the RCS is provided by the secondary side at a prescribed temperature reduction rate with SG feed by the SSS and main steam bypass operation.

The SSS is described in more detail in Section 9.3

4.9 Fuel Pool Cooling and Purification System

The Fuel Pool Cooling and Purification System is made up of the FPCS and Fuel Pool Purification System (FPPS). The FPCS cools the Spent Fuel Pool (SFP). The FPPS provides purification of the Fuel Building pool and Reactor Building pool compartments and provides the capability to provide make-up water or transfer water between the various pool compartments or the IRWST.

The FPCS is safety-related and removes decay heat from the SFP during normal plant operation, outages, and accidents.

The FPPS, which is not safety related (except for the containment isolation boundary) performs the following functions:

- Purification of the water in the Fuel Building pool compartments (Fuel Building transfer compartment, cask loading pit, and SFP); Reactor Building pool compartments (reactor cavity, Reactor Building transfer compartment, instrumentation lances compartment, and the internals compartment); and the IRWST
- Maintain boron concentration in the Fuel Building pool, Reactor Building pool, and IRWST at the refueling concentration
- Transfer water between the compartments in both the Fuel Building pool and Reactor Building pool and also to and from the IRWST

The FPPS also has the capability to perform the following:

- Spray down the cask loading pit, Fuel Building transfer compartment, and the reactor pool compartments
- Fill and purify the water in a spent fuel cask while in the pool
- Skim the surface of the SFP and reactor cavity
- Provide make-up to the SFP or the instrumentation lances compartment from the Demineralized Water Distribution System or the Fire Protection System
- Provide borated water from the Reactor Boron and Water Make-Up System (RBWMS)
- Sample the Fuel Building pool, Reactor Building pool, and IRWST

The FPCS has the capability to adequately cool the spent fuel and the FPPS to isolate its containment penetrations following design basis accident conditions, assuming a single failure.

Figure 4-8 shows the flow schematic of the Fuel Pool Cooling and Purification System.

The FPCS has two separate and independent trains, each consisting of two pumps installed in parallel, a heat exchanger cooled by the CCWS, and associated piping and valves. The pipe penetrations to the SFP are above the required level of water that must be maintained over the spent fuel, while providing the required pump suction head. The pipes that penetrate the pool are equipped with siphon breakers to limit water loss resulting from a leak in the piping system.

The FPPS includes two purification pumps that operate in parallel. One pump is generally used for Fuel Building pool purification and the other pump for Reactor Building pool purification. Headers are provided upstream and downstream of the purification pumps that allow for the alignment of each pump to either building. There are two purification paths, one is part of the FPPS and the other path utilizes the Coolant Purification System. The purification paths each consist of a pre-cartridge filter, a mixed bed demineralizer, and a post-cartridge filter installed in series. The purification pipes enter and exit the pool from above the water level of the pools and are equipped with siphon breakers. Drain lines that penetrate the bottom of the pool in some compartments are normally locked closed and are designed to withstand loadings associated with internal and external hazards. The SFP, which is a single pool with two regions, does not have drain lines penetrating the bottom of the pool.

Both FPCS trains are powered from separate emergency buses, each backed-up by an EDG. The FPPS containment isolation valves are powered from emergency buses, while the rest of the system is supplied by a normal power supply.

Both trains of the FPCS are located in the Fuel Building, which is physically protected from external hazards. The two FPCS trains are installed on either side of the SFP, which provides adequate separation to minimize the effect of internal hazards.

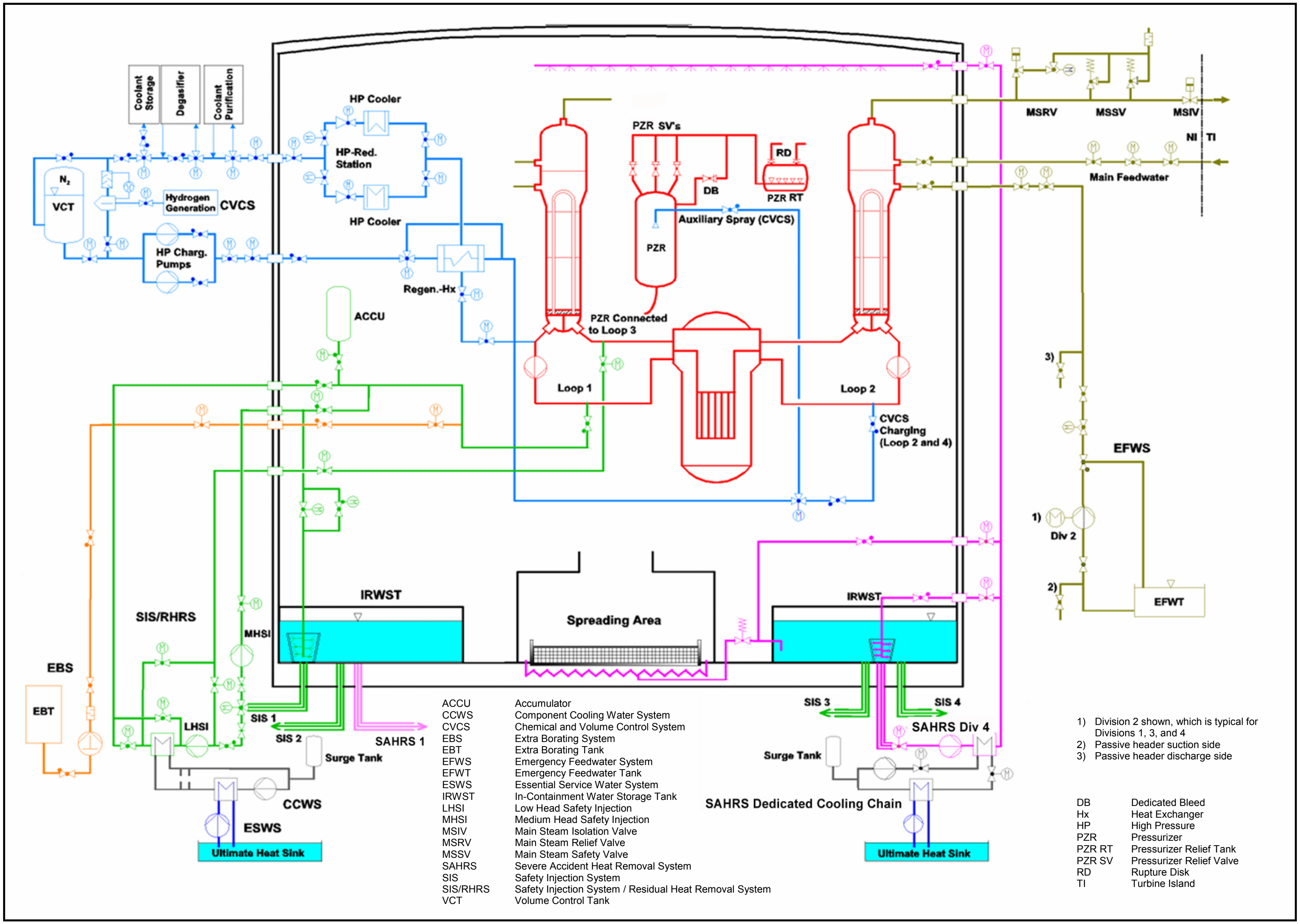


Figure 4-1

Main Fluid Systems

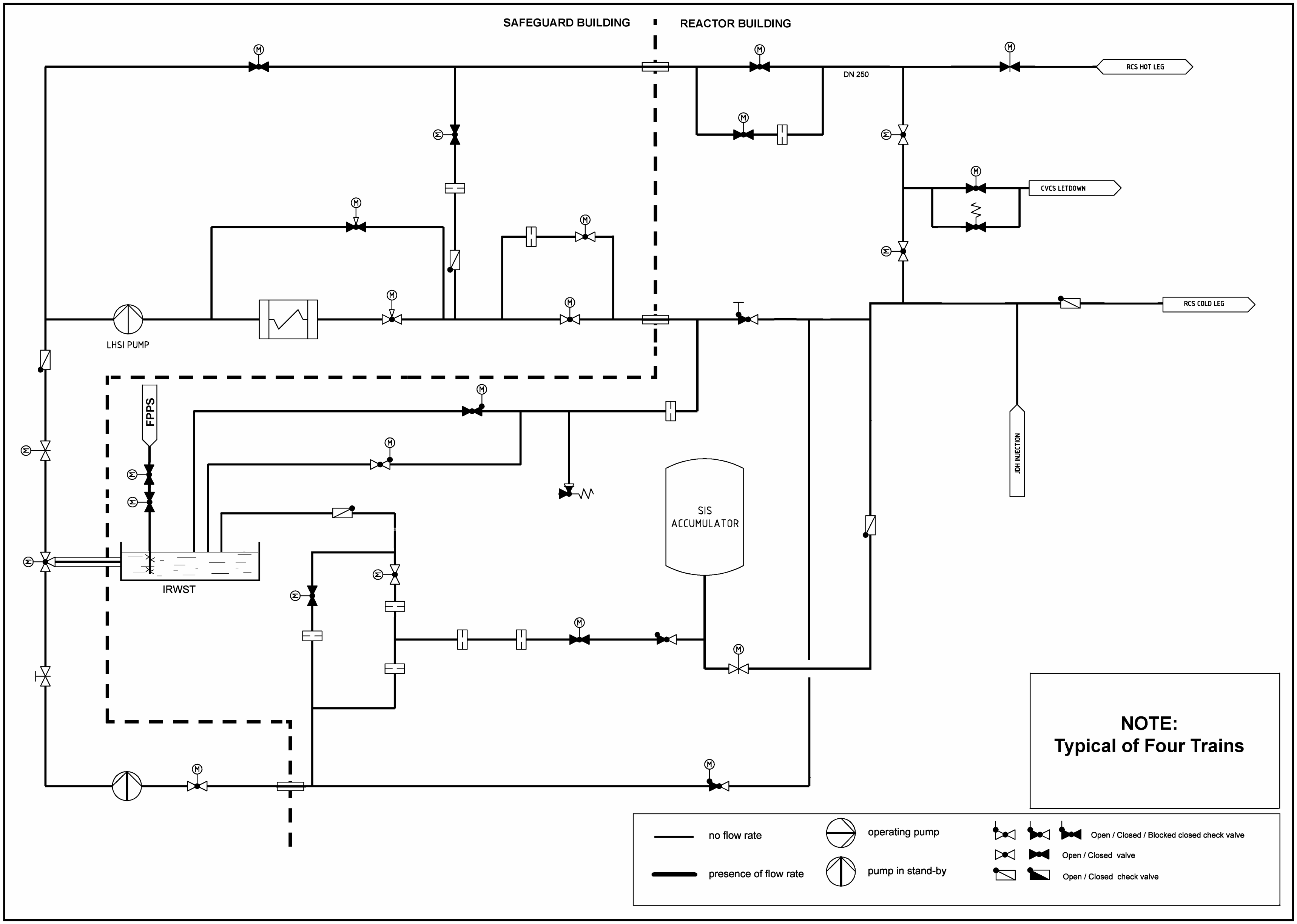


Figure 4-2
Safety Injection System

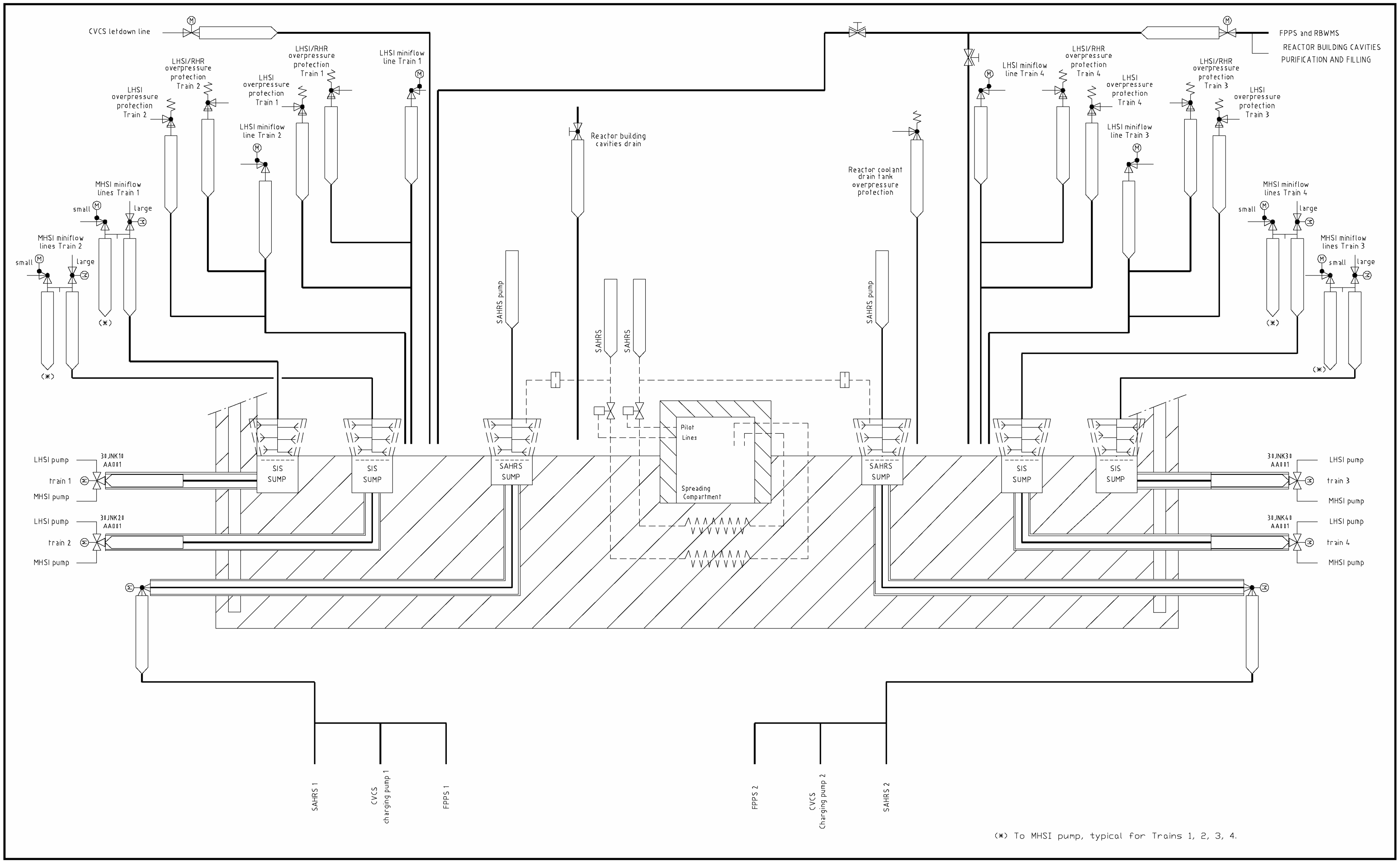
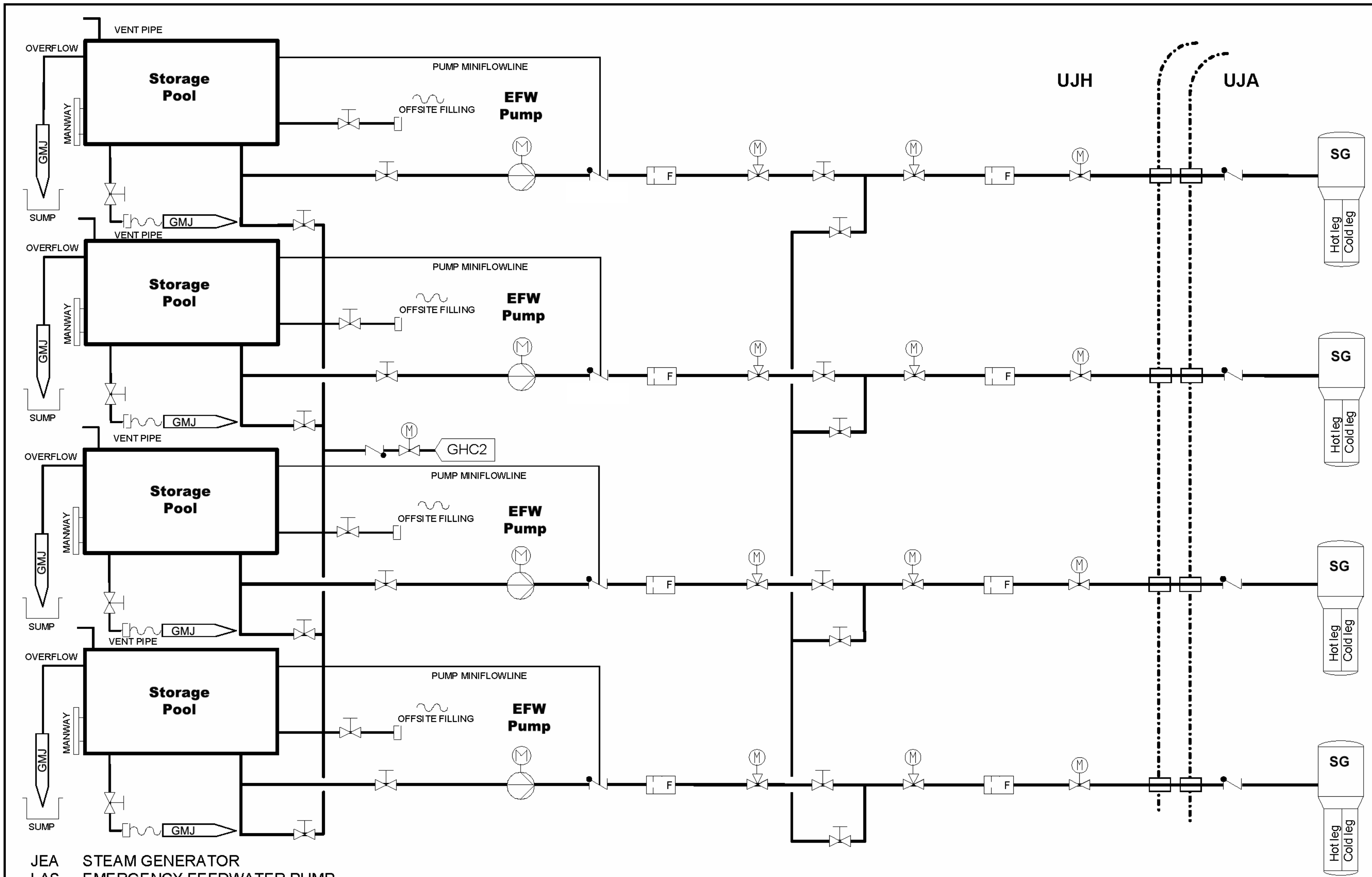


Figure 4-3

IRWST and Related Components



JEA STEAM GENERATOR
LAS EMERGENCY FEEDWATER PUMP
KTB COMPONENT VENT AND DRAINING SYSTEM
GHC2 DEMINERALIZED WATER DISTRIBUTION SYSTEM
GMJ PROCESS DRAINS COLLECTION AND DISPOSAL SYSTEM
UJA REACTOR BUILDING
UJH SAFEGUARD BUILDING

Figure 4-4
Emergency Feedwater System

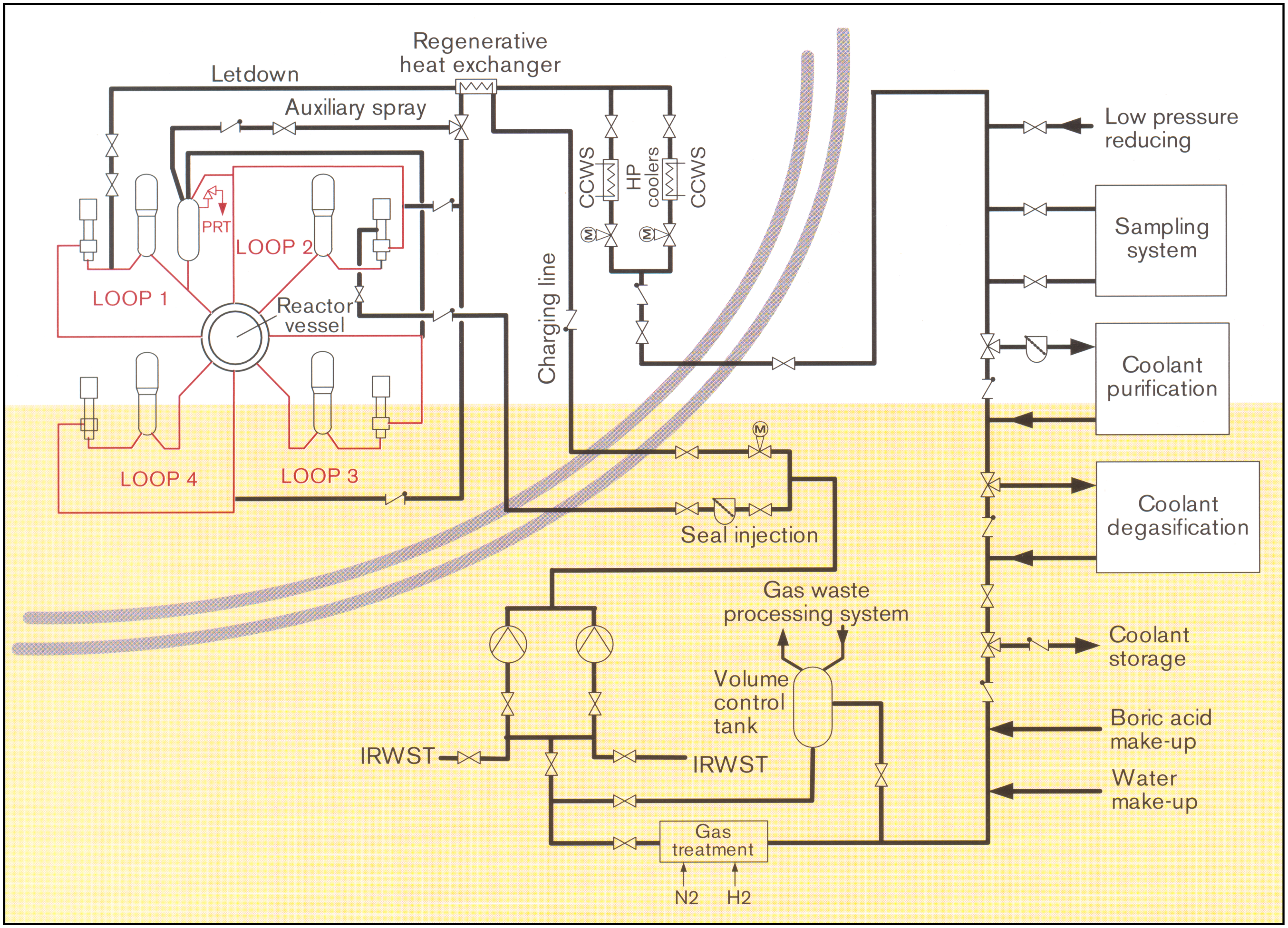


Figure 4-5
Chemical and Volume Control System

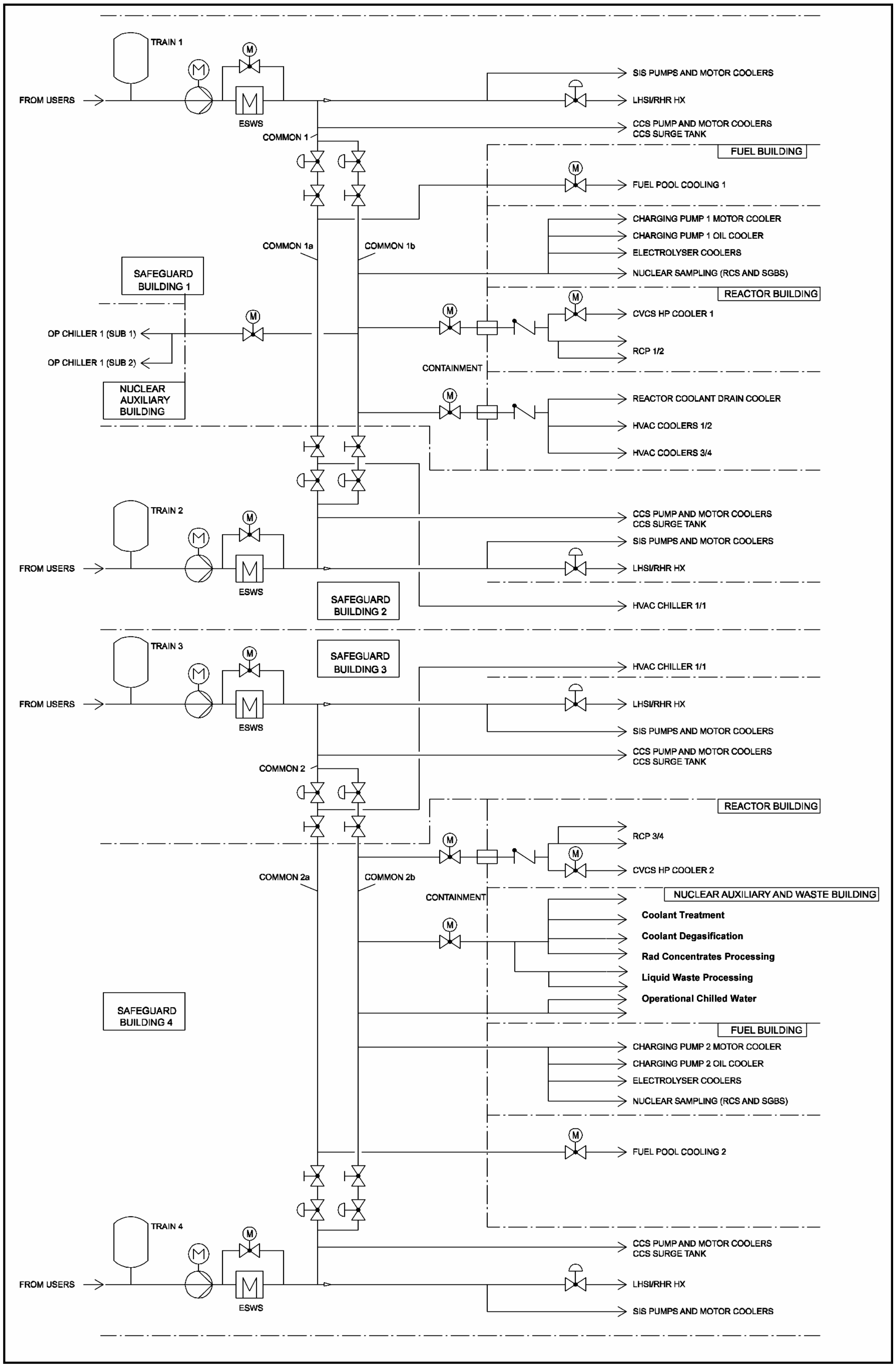


Figure 4-6
Component Cooling Water System

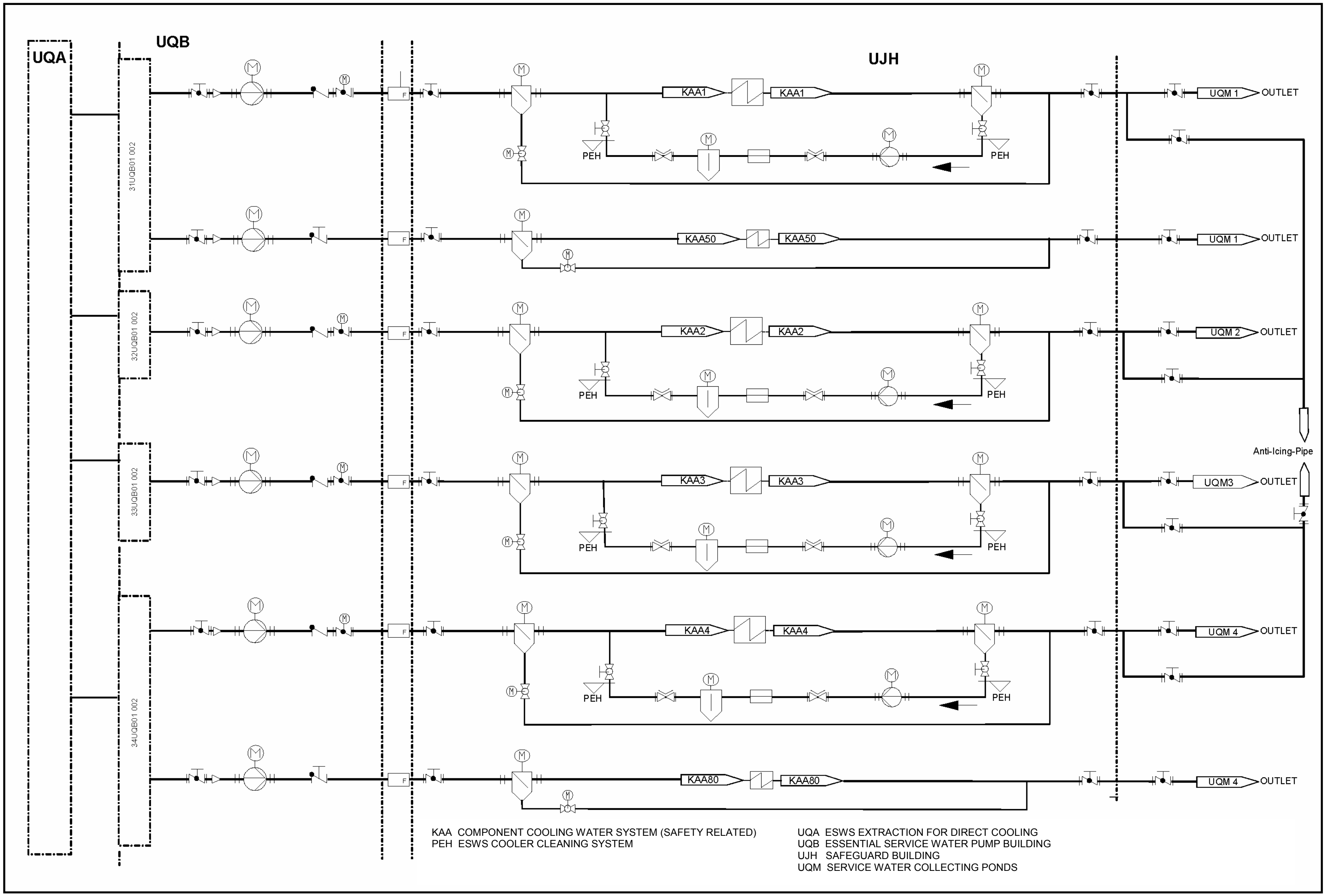


Figure 4-7
Essential Service Water System

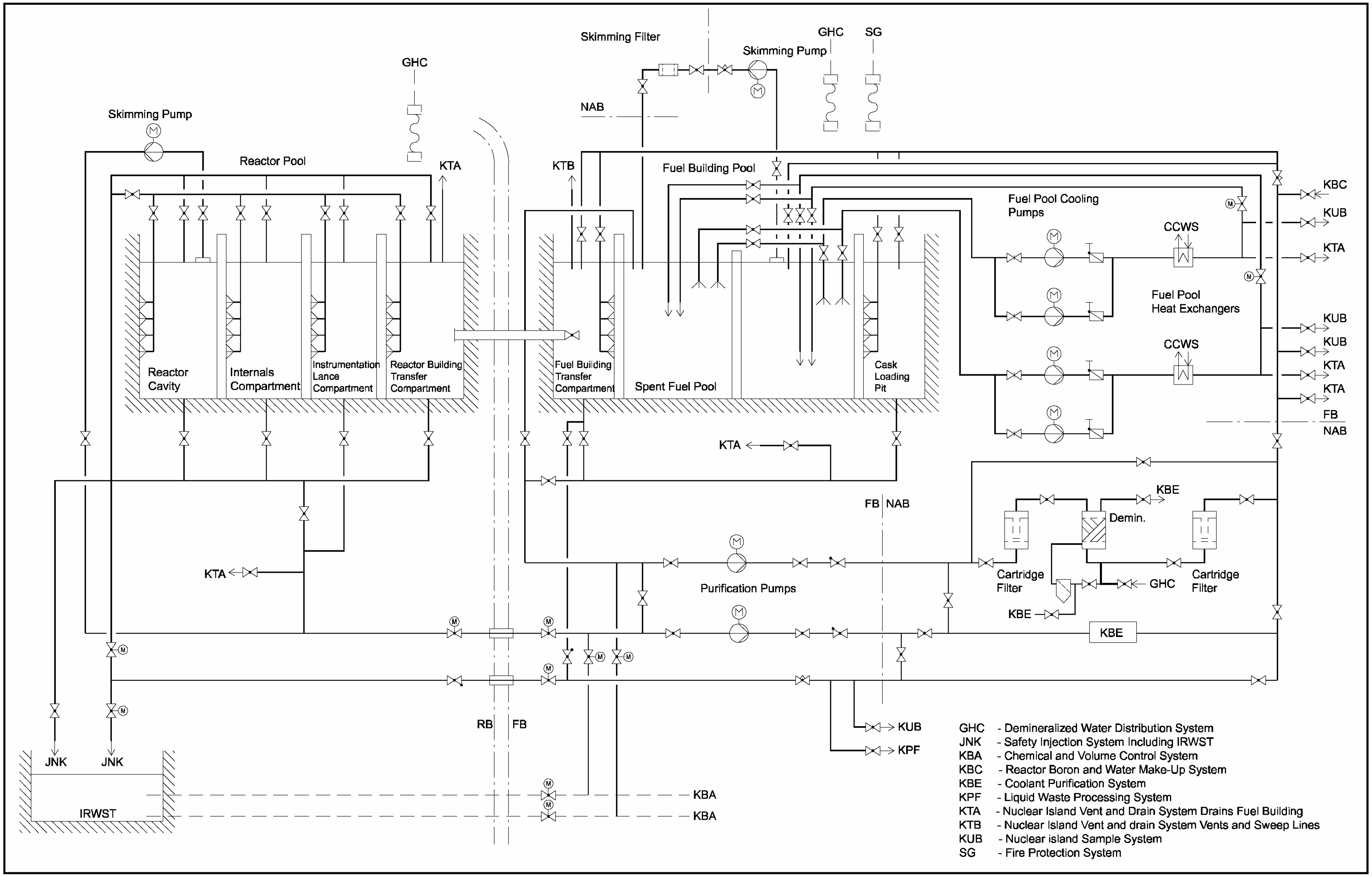


Figure 4-8
Fuel Pool Cooling and Purification System