1.0 INTRODUCTION

The purpose of this Chapter of the Volume 1 Head Document is to provide a summary of the standard design features of the nuclear island of the EPR. Additionally, it provides functional descriptions of the major systems, components and structures of the EPR. This document is for information only as an overview and is not intended for formal review and approval. For more definitive information on the proposed UK EPR design, the reader is referred to Volume 2 of this Fundamental safety Overview.

The EPR is an evolutionary Pressurized Water Reactor (PWR) designed by AREVA NP, a jointly-owned subsidiary of AREVA and Siemens. It is a four-loop plant with a rated thermal power of 4 500 MW and an electrical power output of about 1600-1660 MW depending of the conventional island technology and heat sink characteristics. 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.

1.1 Design Philosophy

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 standard EPR shares the basic set of design features such as four redundant trains of emergency core cooling, containment and Shield Building, and a core melt retention system for severe accident mitigation, and it is adapted to meet applicable regulatory and commercial requirements.

The EPR design is based on the combined design and operating experiences of AREVA NP, formerly Framatome, and Siemens. The EPR meets the French Nuclear Safety Authority's safety requirements adopted during the Standing Committee on Reactor Safety including German experts on October 19th and 26th, 2000 known as "Technical Guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors.

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 "defence-in-depth," has been improved on all levels, resulting in:

- reductions in radiological consequences and accident initiator frequencies,

- favourable transient plant behaviour,

- 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 of the safety systems 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),

- increased robustness of the containment with regard to loads from hypothetical core melt and subsequent RPV 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 Pressuriser (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 “defence-in-depth” concept which comprises five 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
5) Off-site emergency response is not per se part of the safety design.

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 conditions. 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 probabilistic safety objectives:

- the cumulative Core Damage frequency considering the contribution from internal and external events (sabotage excluded) is less than 10-5/reactor-year,
- the cumulative Large Release frequency of radioactive materials to the environment from a core damage event is 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 Containment Heat Removal System (CHRS) 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 pressuriser (PZR) including control systems to maintain system pressure, one Reactor Coolant Pump (RCP) per loop, one steam generator (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 made of 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; 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) arranged in four trains 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 - i.e. loss of off-site AC power with coincident failure of all four EDGs.

Water storage for safety injection is provided by the In-containment Refuelling 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.

1.3 EPR Technical Design Codes

The design, construction and commissioning of the EPR reactor is based on the application of internationally accepted codes and standards. The codes and standards used are of three types:

- technical codes referred to as RCCs (Rules for Design and Construction) which describe industry practice for PWR reactors currently in operation in France. The RCC codes are applicable to mechanical, electrical and fuel design.
- technical codes referred to as ETCs (EPR Technical Codes) which have been produced specifically for the EPR reactor and which supersede existing RCCs. The ETCs are applicable to civil design and the design against fire.
- other codes and standards applicable within the European context of the EPR project (at both the regulatory and industrial level), which can in certain cases supersede the RCC or ETC standards. These include ASME III-NC, KTA class 2 and other European standards for pressurised equipment, as well as international standards for equipment qualification.

Further details are given in Volume 2 Chapter B.6.

1.4 Comparison with Currently Operating PWRs

EPR design parameters are compared with those of a French N4 plant, and a German KONVOI plant in Table 1-1.

TABLE 1- 1: COMPARISON OF EPR DESIGN PARAMETERS

	EPR	KONVOI	N4 PLANTS
Overall			
Net electrical output	≈ 1 660 MW	1 365 MW	1 475 MW
Reactor thermal power	4 500 MW	3 850 MW	4 250 MW
Efficiency	≈ 36%	35.40%	34.50%
Plant design life	60 years	40 years	40 years
Reactor Coolant System			
Number of loops	4	4	4
RCS operating pressure	15.5 MPa	15.8 MPa	15.5 MPa
RCS design pressure	17.6 MPa	17.6 MPa	17.2 MPa
Secondary Side			
SG tube bundle outlet pressure at 100%	7.7 MPa	6.45 MPa	7.3 MPa
Main steam pressure at hot standby	9.0 MPa	8.0 MPa	8.1 MPa
Secondary side design pressure	10.0 MPa	8.83 MPa	9.1 MPa
Water inventory on SG secondary side at full load	≈ 77 200 kg	46 000 kg	62 000 kg
Main steam flow rate	2553 kg/s	2050 kg/s	2400 kg/s
Feed water temperature at 100 % power	230°C	218°C	229.5°C
Core Design			
Number of fuel assemblies (FA)	241	193	205
Type of FA	17 x 17	18 x 18	17 x 17
Active length	420 cm	390 cm	427 cm
Total FA length	480 cm	483 cm	480 cm
Linear heat rate	166.7 W/cm	166.6 W/cm	179 W/cm
Number of control rods	89	61	73
Total flow rate (T.H. design cond.)	22 235 kg/s hot leg	18 800 kg/s hot leg	19 420 kg/s hot leg

	EPR	KONVOI	N4 PLANTS
Total flow rate (B.E. design value)	23 135 kg/s	19 875 kg/s	20 193 kg/s
Vessel inlet/outlet temp. (T.H. cond.)	295.6 / 329.8 °C	292 / 326 °C	292 / 330 °C
Enrichment (max)	5 % U 235	4 % U 235	4 % U 235
Batch discharge burn up	55 to 65 MWd/kg	50 MWd/kg	50 MWd/kg
Core Control			
Number and kind of control rods	89 "black” rods	61 "black” rods	65 "black” rods 8 "grey” rods
Control principle at rated power	No control rods deeply inserted.	No control rods deeply inserted.	4 “grey” rods partly inserted.
Core Instrumentation			
Ex core instrumentation	Flux channels	Flux channels	Flux channels
In core instrumentation	“Top mounted”	“Top mounted”	“Bottom mounted”
	40 aero ball fingers. 12 fixed in core detector fingers. 36 (12x3) core outlet thermocouples	40 aero ball fingers. 8 fixed in core detector fingers. 24 (8x3) core outlet thermocouples (4 for post accident, 20 for additional info)	6 fission movable detectors instrumenting 60 fuel assemblies. 52 core outlet thermocouples
Primary Side Overpressure Protection			
	3 discharge trains on top of the pressurizer, plus dedicated severe accident depressurization trains (2 trains with 2 valves per train)	Connection of 3 discharge trains to top of PZR (valves mounted to relief tank)	3 discharge trains on top of the pressurizer

	EPR	KONVOI	N4 PLANTS
		First train provided with relief valve and upstream isolation valve, Other trains provided with safety valve without isolation valve	Each train provided with two safety valves (“tandem”) in series to ensure the possibility of isolation of a stuck open valve
	Opening of safety valves ensured by solenoid or by spring loaded. Discharge flow to relief tank and via rupture disc into containment	Automatic opening by pilot valves (solenoid for relief valve, spring loaded for safety valves Discharge flow to relief tank and via rupture disc into containment	Automatic opening ensured by pilot actuators Discharge flow to relief tank and via rupture disc into containment.
Secondary Side Overpressure Protection			
	1 discharge line equipped with a relief valve and an isolation valve 2 other discharge lines equipped with safety valves Valve sizing taking credit for reactor trip	1 discharge line equipped with a relief valve and an isolation valve 1 discharge line equipped with 2 safety valves in series (tandem design) Valve sizing not taking credit for reactor trip	2 redundant discharge lines each equipped with 1 relief valve and 1 isolation valve in series 7 discharge lines each equipped with 1 spring-loaded safety valve Safety valve sizing: no credit taken for first reactor trip, N+2 design
PRIMARY COMPONENTS			
Reactor Pressure Vessel			
Fluence (design target)	60 years 1.2 E19 nvt.End of life RT $_{NDT}$ = 30°C	40 years 1.10 E19 nvt	40 years 3.6 E19 nvt
Material	16 MN D5	20 MnMoNi 5 5	16 MN D5 / 20
Nozzles	Set on	Set on	Set in

	EPR	KONVOI	N4 PLANTS
Support	Underneath nozzles	Ring girder design: with extra support lugs and horizontal stops.	Underneath nozzles.
Steam Generator			
Heat transfer surface area	7960 m² (with economizer)	5400 m² (without economizer)	7308 m² (with economizer)
Tube material	Inconel 690	Inconel 800	Inconel 690
Pressure boundary material	18 MN D5	20 MnMoNi 5 5	18 MN D5
Pressurizer			
Pressure boundary material	18 MN D5	20 MnMoNi 5 5	18 MN D5
Surge line connection	Axial	Radial	Axial
Internal volume	75 m^3	65 m^3	60 m^3
Total heating power	≈ 2600 kW	2142 kW	2160 kW
Reactor Coolant Pump			
Supplier	AREVA NP	KSB / Andritz	Framatome
Casing material	Stainless	Ferritic with cladding	Stainless
Shaft seals	3 seals, standstill seal.	3 seals, standstill seal.	3 seals, diverse power supply for seal injection
Flow rate (best estimate)	28 320 m^3/h	22 700 m^3/h	24 850 m^3/h
Head (best estimate)	100.2 m	90 m	106 m
Power of Motor (hot/cold)	8 000 / 10 850 kW	5 500 / 7 400 kW	6 500 / 8 750 kW
Reactor Coolant Piping			
Material	Stainless steel	Forged ferritic steel with single-layer austenitic cladding).	Stainless steel
Break preclusion	Yes	Yes	No

	EPR	KONVOI	N4 PLANTS
EMERGENCY CORE COOLING			
Medium Head Safety Injection (MHSI)			
Location	Cold leg	Hot leg (with auto. switchover to cold leg in case of hot leg injection pipe break)	Cold leg (cold + hot legs in long term)
Number of pumps	4	4	2 via header
Shutoff head	85 / 97 bar	110 bar	145 bar
Accumulators			
Location	Cold leg	Cold and hot leg	Cold leg
Number of accumulators	4	8	4
Injection pressure	45 bar	25 bar	45 bar
Total volume	47 m³	45 m³	47 m³
Low Head Safety Injection (LHSI)			
Location	Cold leg (cold + hot legs only in long term for LOCA)	Cold and hot legs	Cold leg (cold + hot legs only in long term for LOCA)
Number of pumps	4	4	2 via header
Shutoff head	Min. 20 bar	12.5 bar	25 bar
Water Storage Tank (IRWST)			
Arrangement	Inside containment	Annulus	Outside reactor building
Number	1	4	1
Total volume	1 940 m³	1 880 m³	min. 2600 m³ (available for injection) total volume: 3037
RESIDUAL HEAT REMOVAL SYSTEM (RHR)			
Arrangement	Combined with LHSI outside containment	Outside containment	Inside containment
Number of pumps	Combined with LHSI	4 combined with LHSI	2
Shutoff head	23 bar	12.5 bar	9.6 bar

	EPR	KONVOI	N4 PLANTS
CONTAINMENT HEAT REMOVAL SYSTEM (CHRS)			
Containment spray system for design basis accidents	None	None	Spray system, 2 trains (100%) outside containment
Containment pressure control for severe accidents	Containment heat removal system (spray system + sump cooling) Equipment located outside containment	Filtered venting	Filtered venting
BORATION SYSTEMS			
Operational functions	Chemical and volume control system (CVCS)	Chemical and volume control system (CVCS)	Chemical and volume control system (CVCS)
Safety functions	Extra borating system (2 trains, each 100%)	Extra borating system (4 trains)	Chemical and volume control system (CVCS) Use of CVCS in the long term phase (manual phase) MHSI + Bleed (PZR safety valves when CVCS unavailable or inefficient)
FEED WATER SUPPLY			
Normal operation	Main feed water system	Main feed water system	Main feed water system
Start-up and shutdown conditions	Dedicated system for start-up and shutdown operation (1 pump)	Dedicated system for start-up and shutdown operation with 2 pumps, both emergency power supplied	Use of emergency feed water system

	EPR	KONVOI	N4 PLANTS
Emergency conditions	Emergency feed water system 4 separate and independent trains with passive headers Pumps driven by motors (emergency power supplied) and by 2 SBO diesels	Emergency feed water system 4 separate and independent trains with passive headers each pump is driven by diesel (directly) and motor (not emergency power supplied)	Emergency feed water system 4 pumps via headers (2 by 2) 2 electrical motor driven pumps, + 2 turbine driven pumps
FUEL POOL COOLING SYSTEM			
Spent fuel volume	1 486 m^3	1 330 m^3	1 150 m^3
	2 trains (two pump per train) 1 backup train (1 pump)	3 trains (one pump per train)	2 trains (one pump per train)
Normal flow rate	Main train cooling pumps : 222 kg/s, Backup train: 153 kg/s	Pump delivery: 170 kg/s	Cooling pumps : 105.6 kg/s
COMPONENT COOLING WATER SYSTEM (CCWS)			
	4 trains (1 pump per train, 1 heat exchanger per train)	4 trains (2 pumps and 1 heat exchanger per train, 2 trains with emergency pump)	2 trains (2 pumps 100% per train, 2 half exchangers per train)
Nominal flow rate	950 kg/s	500 kg/s per pump 400 kg/s per emergency pump	944 kg/s
Total head at nominal flow	57 m	45 m 30 m (emergency pump)	50 m

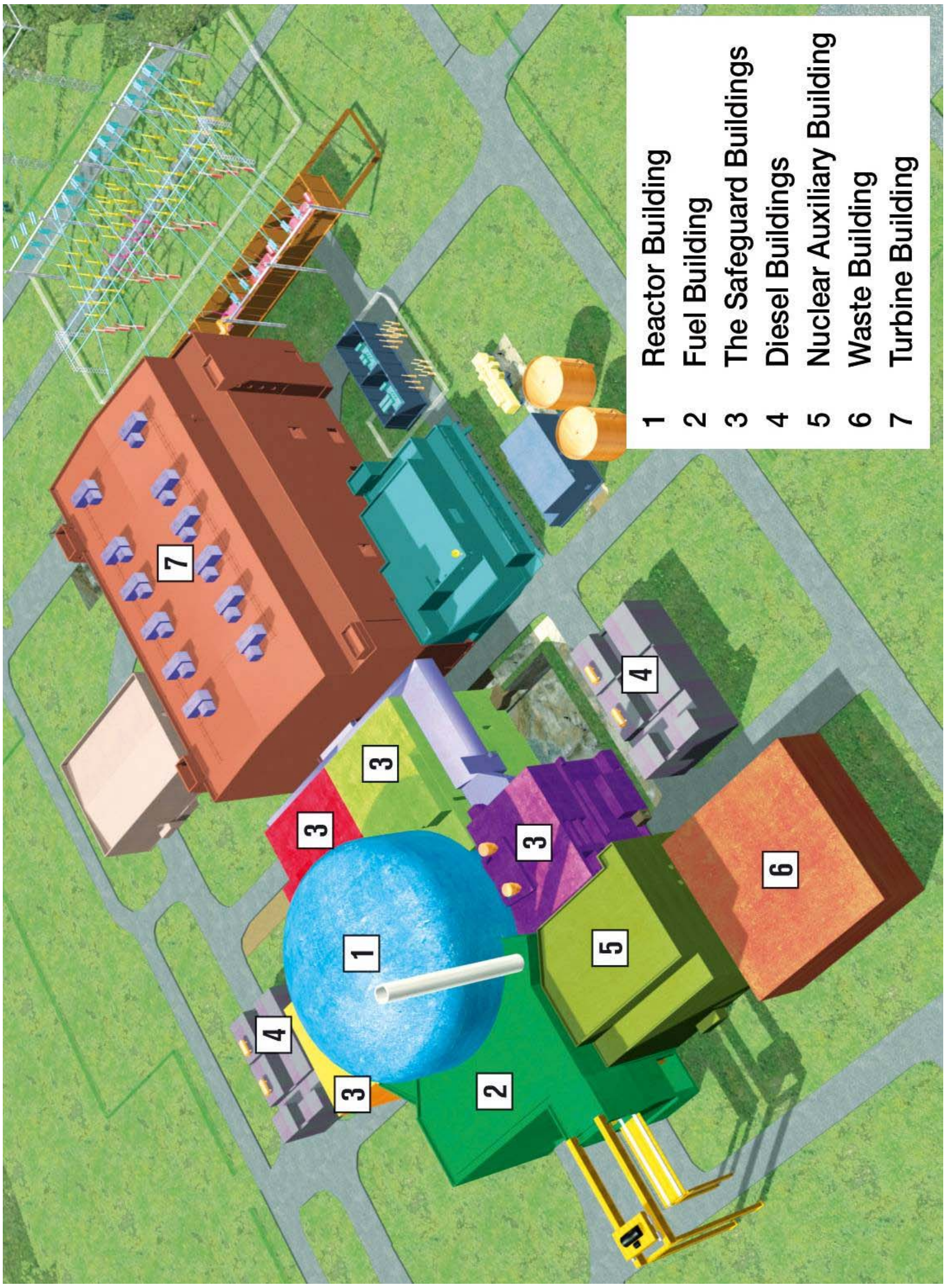
	EPR	KONVOI	N4 PLANTS
ESSENTIAL SERVICE WATER SYSTEM (ESWS)			
	4 trains (1 pump per train)	4 trains (1 pump and 1 heat exchanger per train, 2 trains with emergency pump)	2 trains (2 pumps 100% per train)
Nominal flow rate	3420 m^3/h/pump	3600 m^3/h/pump 2000 m^3/h/emergency pump	2600 m^3/h/pump
ELECTRICAL SYSTEMS			
Operational power supply	4 train arrangement (within at least 2 divisions)	4 train arrangement 4 divisions	2 train arrangement (in 2 divisions)
Emergency power supply	4 train arrangement 4 divisions concept 4 diesel generator sets of ≈ 7 MW each in dedicated separated buildings (10 kV)	4 train arrangement 4 divisions concept 4 diesel generators (5 MW each) in a dedicated building	2 train arrangement 2 division concept for emergency power supply 2 diesel generator sets of 8 MW each in two separated buildings
	2 small diesel generator sets for Station Blackout mitigation (≈ 1 MW each). Diversity by different diesel size and different voltages (10KV, 690V)	4 emergency feed diesel generators (0.85 MW each) in a separate, fully bunkered building. Diversity by different diesel sizes	Diversity to the 2 diesels by: 1 turbine generator in the short term, 1 gas turbine in the long term

	EPR	KONVOI	N4 PLANTS
I&C SYSTEMS			
Technology	Digital technology (conventional instruments and drive control interface).	Hardwired technology, special applications of digital technology, e.g. processing for reactor controls.	Full digital technology (except safety control area and remote shutdown panel).
Main Control Room	Screen-based (except for safety control area). Protected against all external hazards.	Conventional with a supplementary digital process information system. Protected against earthquake.	Screen-based (except for safety control area). Protected against all external hazards.
Remote Shutdown Station	Station for maintaining the plant in safe shutdown conditions in case of unavailability of the Main Control Room.	Emergency control station (in a separate building) for maintaining the plant in safe shutdown conditions in case of unavailability of the Main Control Room.	Station for maintaining the plant in safe shutdown conditions in case of unavailability of the Main Control Room.
	Protected against all external hazards.	Protected against all external hazards.	Protected against all external hazards.
CONTAINMENT			
Containment concept	Double wall containment concept with: a primary wall in prestressed concrete (1.3 m thick) with metallic liner, about 80 000 m³ free volume,	Double wall containment concept with: a spherical steel primary wall, 70 000 m³ free volume,	Double wall containment concept with: a primary wall in prestressed concrete without liner, 73 000 m³ free volume,

	EPR	KONVOI	N4 PLANTS
	a secondary wall in reinforced concrete (1.8 m thick),	a secondary wall in reinforced concrete,	a secondary wall in reinforced concrete,
	a common basemat in reinforced concrete for RB, SAB 1 - 4 and FB	a common basemat in reinforced concrete.	a common basemat in reinforced concrete.
Design pressure for DBAs (abs).	LOCA or steam line break, 0.55 MPa	LOCA 0.63 MPa	LOCA or steam line break 0.53 MPa
Test pressure (abs)	0.65 MPa	0.77 MPa	0.53 MPa
Integral leakage rate for primary wall	< 0.3% vol/day at the design pressure of 0.55 MPa abs.	< 0.25% vol/day	< 1% vol/day
Containment Functions fulfilled by:	Primary and secondary walls with an annulus between them, collection of possible leakage through the primary wall in the annulus and filtration before release to the environment via stack,	Primary wall (steel sphere), collection of possible leakage through the primary wall and filtered release via stack (annulus air extraction systems),	Primary and secondary walls with an annulus between them, collection of possible leakage through the primary wall in the annulus and filtration before release to the environment via stack,
	systems for the retention and control of leakages and leak-off system for some penetrations systems for containment isolation monitoring,	systems for the retention and control of leakages and leak-off system for some penetrations, systems for containment isolation monitoring,	systems for the retention and control of the leakages through the peripheral buildings, systems for containment isolation monitoring,

	EPR	KONVOI	N4 PLANTS
	systems to control the pressure and temperature conditions inside containment (HVAC, heat removal from IRWST, CHRS).	systems to control the pressure and temperature conditions inside containment (HVAC, heat removal from the sump, filtered venting system).	systems to control the pressure and temperature conditions inside containment (HVAC, containment spray system, filtered venting).
PROTECTION AGAINST EXTERNAL HAZARDS			
Design Basis Earthquake (DBE)	0.25 g	Modified USNRC / 0.3g (Floor Response Spectra enveloping all KONVOI sites).	0.15 g
Airplane Crash (APC)	Light aircraft Military aircraft Large commercial aircraft	Military aircraft	(Cessna, Lear Jet)
Explosion Pressure Wave (EPW)	10.0 kPa incoming wave	30.0 / 45.0 kPa load function	5.0 kPa incoming wave
Reactor Building			
SSE	Yes	Yes	Yes
APC	Yes	Yes	Yes
EPW	Yes	Yes	Yes
Safeguard Building			
SSE	Yes	Yes	Yes
APC	2 protected 2 by segregation	Yes	Yes
EPW	Yes	Yes	Yes
Fuel Building			
SSE	Yes	-	Yes
APC	Yes	-	Yes
EPW	Yes	-	Yes

FIGURE 1- 1: TYPICAL PLANT CONFIGURATION
Location of some buildings depends on site layout



2.0 CORE DESIGN

2.1 Overall Features

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. Assemblies described below are for design and performance assessment as described in this document. The EPR core can accommodate different fuel assembly design.

- 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% 235U.
- Gd_2O_3 integral burnable poison with Gd concentration of 2 wt% to 8 wt%.
- Highly corrosion-resistant and low-growth M5^{TM} cladding and tubing.
- Monobloc^{TM} guide thimbles to increase structural strength.
- Low growth M5^{TM} intermediate spacers.
- Alloy 718 end spacers providing improved fuel rod support and flow-induced fretting resistance.
- Debris-resistant robust FUELGUARD^{TM} 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^{TM} 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^{TM} 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 hold-down springs of the fuel assembly, which are used to prevent hydraulic lift-off of the fuel assembly during operation. The hold-down 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 Technical performance

The levels of performance – in terms of capacity to reach very high burn-up fractions, reliability of cladding and assembly –, required for the EPR fuel assembly are equivalent to or better than the performance levels required from the burnable products used in existing reactors.

The capacity of M5™ to reach very high burn up fractions, with margins, led to the natural decision to adopt this alloy. The experience feedback from 33 reactors shows that the HTP fuel assembly design meets the required performance levels, due to excellent mechanical behaviour.

Corrosion-resistant tubular cladding

The M5™ alloy is AREVA NP's reference material for fuel rod tubular cladding. This alloy is the outcome of a vast optimisation and industrial development programme, which started in the 1990s and reached completion five years ago. The M5™ alloy is entirely re-crystallised and made up of zirconium, niobium and oxygen. Its performance is reproducible. It is stable when irradiated and compatible with an increase of the burn up fraction to more than 70 GWd/t, including under increasingly severe operating conditions. Its in-reactor performance has been demonstrated by a monitoring programme covering the most varied and severe operating conditions. More than 740 000 fuel rods in M5™ tubular cladding have been irradiated up to 78 GWd/t in 41 pressurised water reactors, including more than 49 fuel assembly reloads in the whole of the world, of which 4 have been irradiated since 1998 in a 1300 MW EDF reactor.

The main features of this material, assessed in terms of its performance after irradiation programmes rising to burn up fractions of at least 78 GWd/t, include very high dimensional stability, considerable corrosion withstand, low hydrogen take-up, very good behaviour when ramping up power and in LOCA (loss of coolant accident) and RIA (reactivity insertion accident) situations.

Absence of deformation

The measurements taken on the HTP fuel assemblies as a whole have highlighted the absence of significant deformation. This satisfactory behaviour is attributable to the geometrical characteristics of the grid, which allow for satisfactory embedding of the fuel rods. The M5 alloy used for guide thimbles and the grids provides for significant margins even at very high burn up fractions, due to great dimensional stability.

Excellent handling behaviour

The handling of the HTP fuel prevents little difficulty on the reactors where it is in service. This very satisfactory behaviour is attributable to the HTP grid, which generates little interaction between fuel assemblies in the in-core handling phase and are fitted with guide vanes between each rod in the lower and upper parts of the outer plates, which give good protection against snagging.

Fretting wear withstand

A major generic wear test programme has demonstrated the original HTP/HMP support concept (fuel rod support along the outer tube surface at 8 points) offers major benefits in regard to the risk of wear. 1000 h of endurance trials were also undertaken using HTP/HMP type grids, presenting a fuel rod/support set-up at the end grid level. These trials confirmed the earlier tests over shorter durations, and showed that wear at the HTP/HMP grid level remained negligible. It was regarded as being equivalent to surface level adjustment, with no indication of incipient wear or evidence of wear developing in the long term.

Grid buckling withstand

The HTP grid design gives it both low outer rigidity (which under equivalent accidental stresses absorbs the force of impact), and considerable withstand to buckling effect when at the end of service life.

2.4 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 absorbers is a hybrid Ag-In-Cd (AIC) alloy and B4C design, with AIC in the lower part and B4C 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 15.5 MPa. The coolant contains soluble 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.

2.5 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:

- average discharge burn-up up to 60 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 (transients – events that might be expected to occur at least once during life of the unit), 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, 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 which might potentially occur at least once in the life of a group of units). 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 behaviour 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 (accidents – events which are not expected to occur during the life of any unit in a group of similar units) 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 burn up with a safety analysis accounting for irradiation-induced changes.

- For fast reactivity transients, the fuel enthalpy shall be limited to 921 kJ/kg and 837 kJ/kg for unirradiated and irradiated fuel, respectively.

2.5.1 Nuclear Design

The main characteristics of the core are derived from the objectives set, namely minimisation of cycle cost, a plutonium recycling capability of up to 50% of the MOX in-core fuel assemblies), the desired flexibility in respect of campaign durations (from 12 to 24 months) and reloading strategies (out/in and in/out).

The number of fuel assemblies is fixed at 241, so allowing for high reload fractions, thus meeting the objective of high discharge burn up (average burn up of fuel assemblies on unloading), namely more than 60 GWd/t. With an active height of 420 cm and power of 4500 MW, linear power is 163.4 W/cm as against 179 W/cm for the N4 plants. This reduction allows for an equivalent increase in the admissible in-core hot spot factor ($F_{\Delta H}$), other things being equal. In practice, after allowance for all the factors contributing to $F_{\Delta H}$ definition, limit $F_{\Delta H}$ for the EPR at 4500 MW is the order of 1.61, as against 1.40 for N4 plants. This high value accommodates low leakage in/out management modes. In regard to MOX recycling capability, the EPR is able to accommodate fuel management with a higher MOX fuel assembly fraction.

The design of the voluminous core contributes to reducing radial leaks by reducing to a minimum the ratio of Surface/Volume, as shown in the table below:

Type of power plant	EPR	KONVOI	N4 plants
Number of FA	241	193	205
Surface/Volume ratio	1.79	1.86	2.01

Another factor contributing to lower radial leaks is the presence of a heavy stainless steel reflector located in the space between the core and the core containment (thickness between 8 and 30 cm). In addition to the gains in vessel fluence, the heavy reflector improves neutron economy, which translates either into an extension of cycle length or a reduction in the required level of enrichment:

	Extension of cycle length	Saving of enrichment
Cycle 1	33 efpd (on 484)	0.15%
Equilibrium cycle 18 months in/out	8 efpd (on 488)	0.08%

The choice of a voluminous core combined with low linear power has resulted in gains in terms of fuel cycle costs.

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 2-3 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 = 2.8 (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 4500 MW.

Boron Concentration, Reactivity Coefficients, Shutdown Efficiency

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

Burnable absorbers concentration is adjusted to limit the critical boron concentration at beginning-of-cycle (BOC), 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 260°C 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.

The RCCA pattern is shown in Figure 2-6. There are 36 rod cluster control assemblies in 5 sub-groups shown as P in the figure below. The remaining 53 clusters are reserved for reactor shutdown (N clusters). As all clusters are identical, there is great flexibility in their distribution among the control sub-groups. This limits the effects of sub-burnout of the fuel assemblies, and the effects of cluster wear, by permutation or alternation of cluster location.

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. 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 < 1400 ppm at BOC, HZP without Xe.

The theoretical boron concentration of the IRWST is in the range of 2600 ppm natural boron and ~1600 ppm based upon the use of enriched boron. The actual IRWST boron concentration will be somewhat higher to account for uncertainties and partial dilutions.

2.5.2 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 DNBRLCO -- 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 is 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 ~2810°C for unirradiated UO_2, decreasing by ~ 7.6°C per 10000 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 hydraulic” flow rate, minimizing the flow entering the core, is used for the core’s thermal hydraulic 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 hydraulic characteristics of the EPR are shown in Table 2-4.

Core Control Principles/Control Rods/Control Rod Manoeuvring

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.

2.6 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.

Ex-core 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 ex-core 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 ex-core 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 (0.2 mm); their bend radii are small (only a few cm); 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 three or 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 ~ 61 mm.

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 burn up 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.

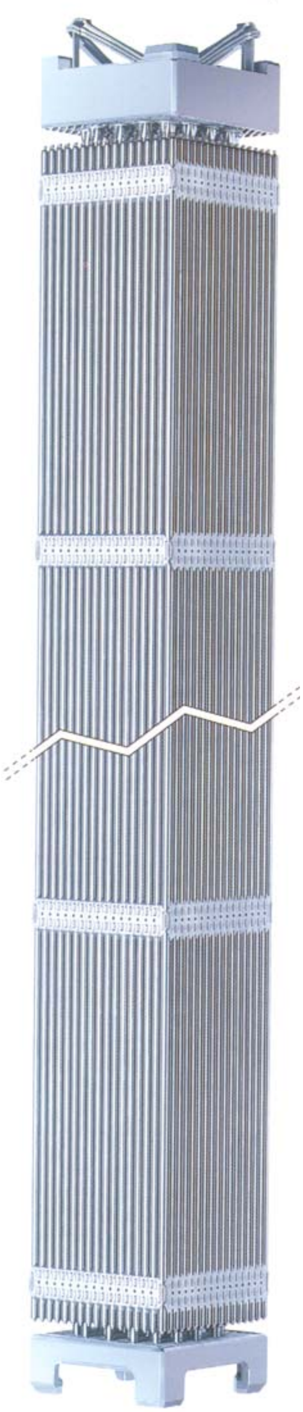
The COTCs are located in the top nozzle of the instrumented fuel assemblies and are radially distributed over the core in the same way as the SPND fingers. At every location, there is space for three thermocouples. These sensors are primarily used for post-accident measurement purposes, but they may 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	4500 MW
Number of fuel assemblies	241
Fuel assembly pitch in core (cold)	215.04 mm
Number of fuel rods/fuel assembly	265
Fuel assembly length (cold, without springs)	4805 mm
Active length (cold)	4200 mm
Average linear power (at rated power)	166.7 W/cm
Outside diameter of fuel rods	9.50 mm
Number of guide tubes per assembly	24
Guide tube outer diameter	12.45 mm
Number of spacer grids per assembly	10
Fuel rod pitch	12.6 mm
Total fuel rod length (cold)	4550 mm
Total fuel assembly mass	784 kg
Assembly UO_2 mass (oxide)	598 kg
Total cross section area	11.14 m^2

Length given at cold conditions

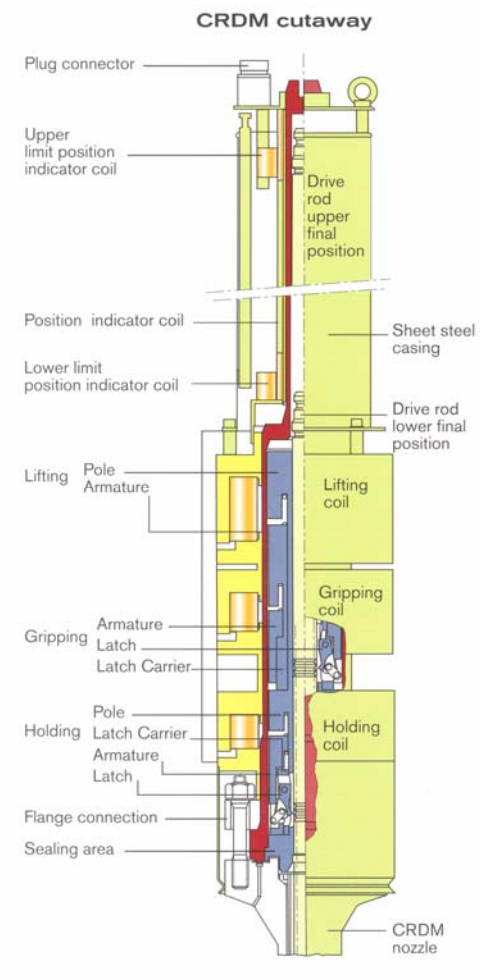
TABLE 2- 2: KEY FUEL ASSEMBLY CHARACTERISTICS



Characteristics	Data
Fuel Assemblies	
• Fuel rod array	17 x 17
• Lattice pitch	12.6 mm
• 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
• Hold-down springs	Inconel 718
Fuel Rods	
• Outside diameter	9.50 mm
• Active length	4200 mm
• Cladding thickness	0.57 mm
• Cladding material	M5™
• **Co-Mixed Burnable Poison (Typical)**	
• Material	Gd_2O_3
• Gadolinium enrichment (wt%)	2 – 10
• UO_2 carrier enrichment (wt% ^{235}U)	Up to 3*

* Host enrichment is the non-gadolinium fuel pin enrichment

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	10.17 g/cm^3
- Absorber outer diameter	7.65 mm
• B4C part (upper part)	
- Natural boron	19.9% ^{10}B
- Density	1.79 g/cm^3
- Absorber diameter	7.47 mm
Cladding	
• Material	AISI 316 stainless steel
• Surface treatment (externally)	Ion-nitriding
• Outer diameter	9.68 mm
• Inner diameter	7.72 mm
Fill Gas	
Control Rod Drive Mechanisms (CRDMs)	
• Quantity	89
• Displacement speed	375 mm/min or 750 mm/min
• Maximum scram time allowed	3.5 s
• Materials	• Forged 304 stainless steel • Magnetic 410 stainless steel • Non-magnetic stainless steel

TABLE 2- 4: THERMAL HYDRAULIC DESIGN DATA

Characteristics	Data
Total core heat output Number of loops Nominal system pressure	4 500 MW 4 15.5 MPa
Coolant flow (*): • Core flow area • Core average coolant velocity • Core average mass velocity • Thermal flow rate (per loop) • Core mass flow rate • Core bypass flow rate (for TH conditions)	$5.9\ m^2$ 5 m/s $356\ g/cm^2.s$ $27\ 180\ m^3/h$ 22225 kg/s 5,5 %
Coolant temperature (for TH conditions): • Nominal inlet temperature • Average temperature rise in vessel • Average temperature rise in core • Average temperature in core • Average temperature in vessel	295.7 °C 34.2 °C 36 °C 313.7 °C 312.8 °C
Core heat transfer: • Heat transfer surface area • Average core heat flux • Maximal core heat flux (nominal conditions) • Average linear power density • Peak linear power for normal conditions w/ uncertainty • Peak linear power protection threshold	$8005\ m^2$ $54.7\ W/cm^2$ $157.3\ W/cm^2$ 163.4 W/cm 470 W/cm 590 W/cm
DNB ration • Minimum DNBR under nominal operating conditions with $F_{\Delta H} = 1.61$	~ 2.6

- ** with heat exchange coefficient without plugging and with clean tubes (closed warm dome configuration)*

FIGURE 2- 1: RADIAL CROSS SECTION OF A FUEL ASSEMBLY

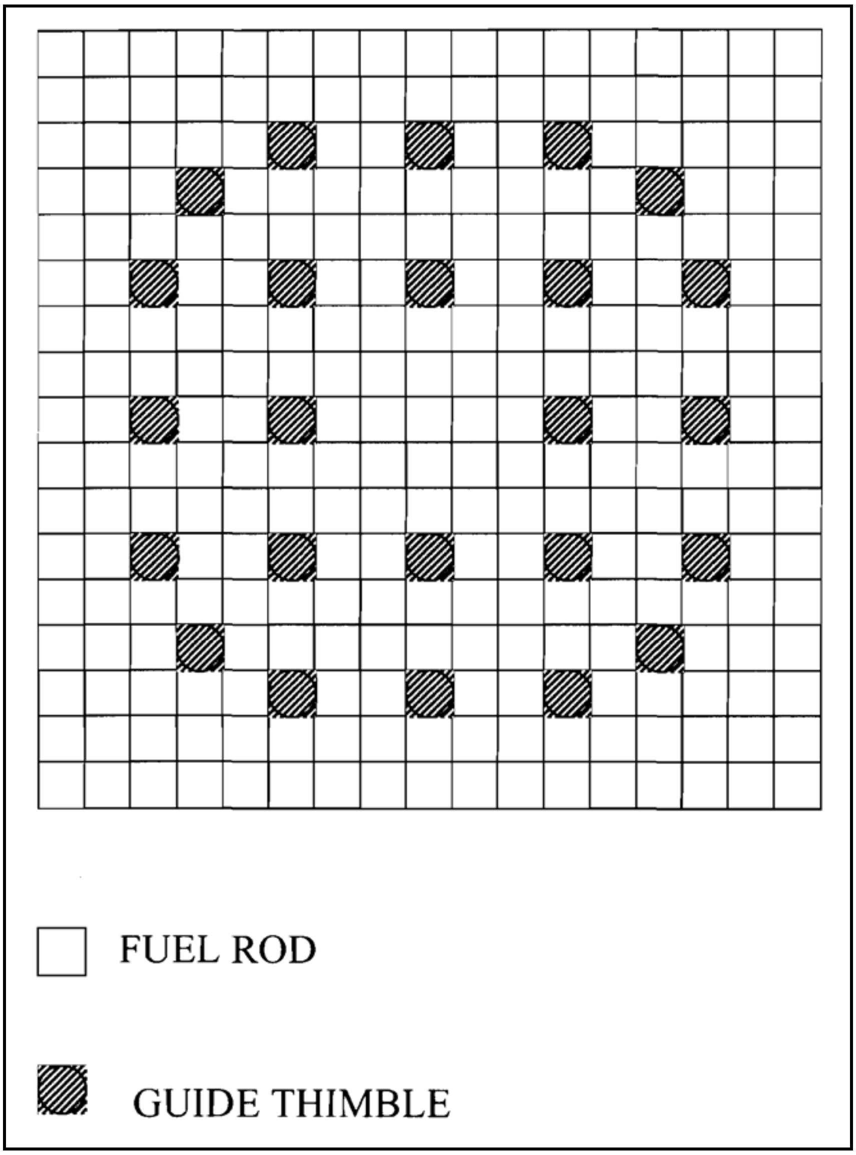


FIGURE 2- 2: FUEL ASSEMBLY

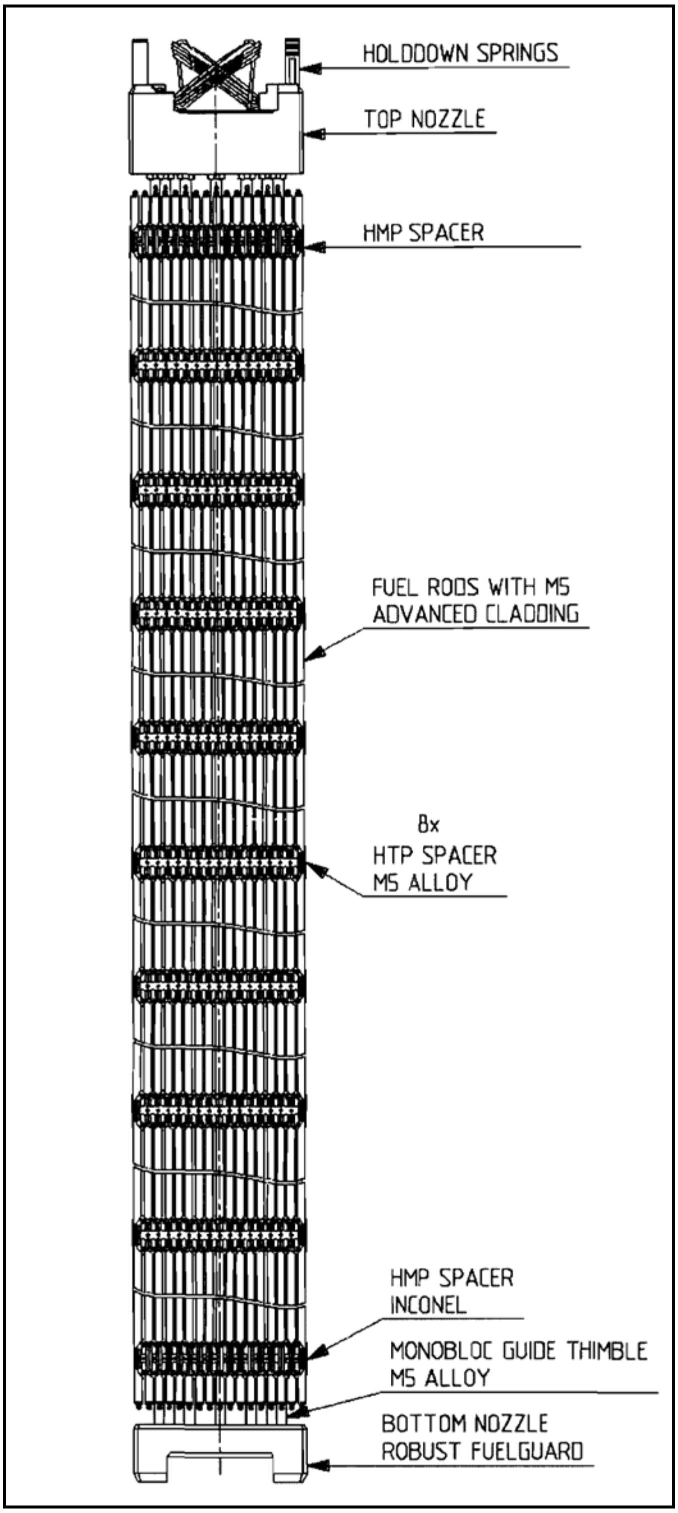
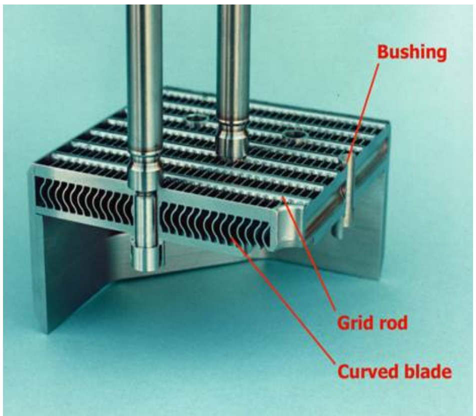


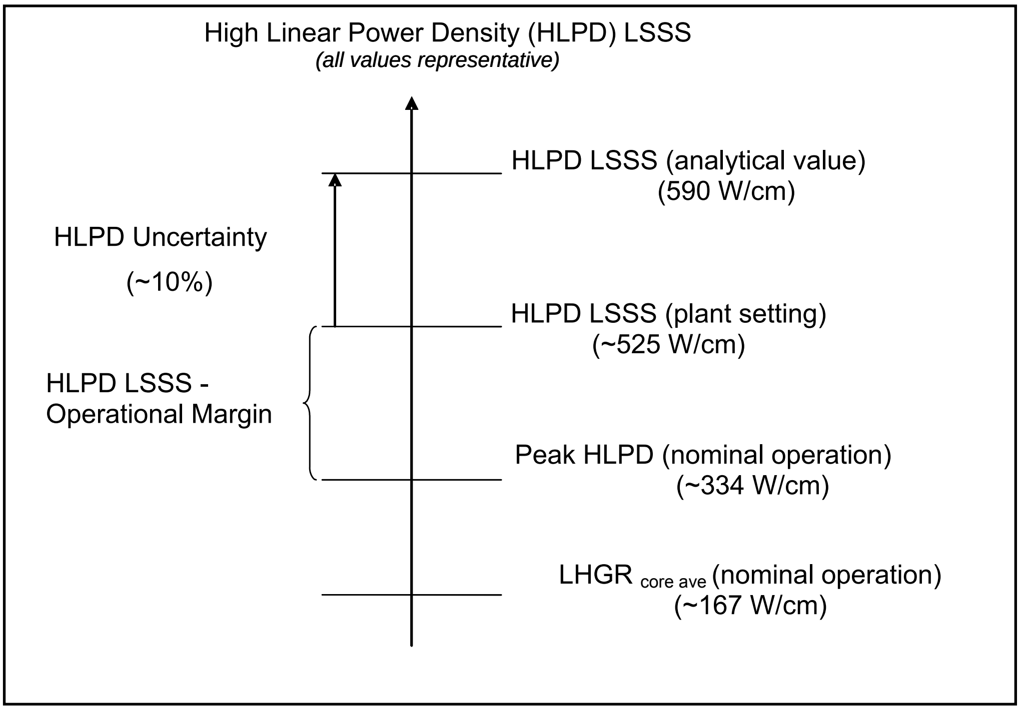
FIGURE 2- 3: FUEL ASSEMBLY TOP AND BOTTOM NOZZLES

Top Nozzle



Robust FUELGUARD™ Bottom Nozzle

FIGURE 2- 4: HIGH LINEAR POWER DENSITY



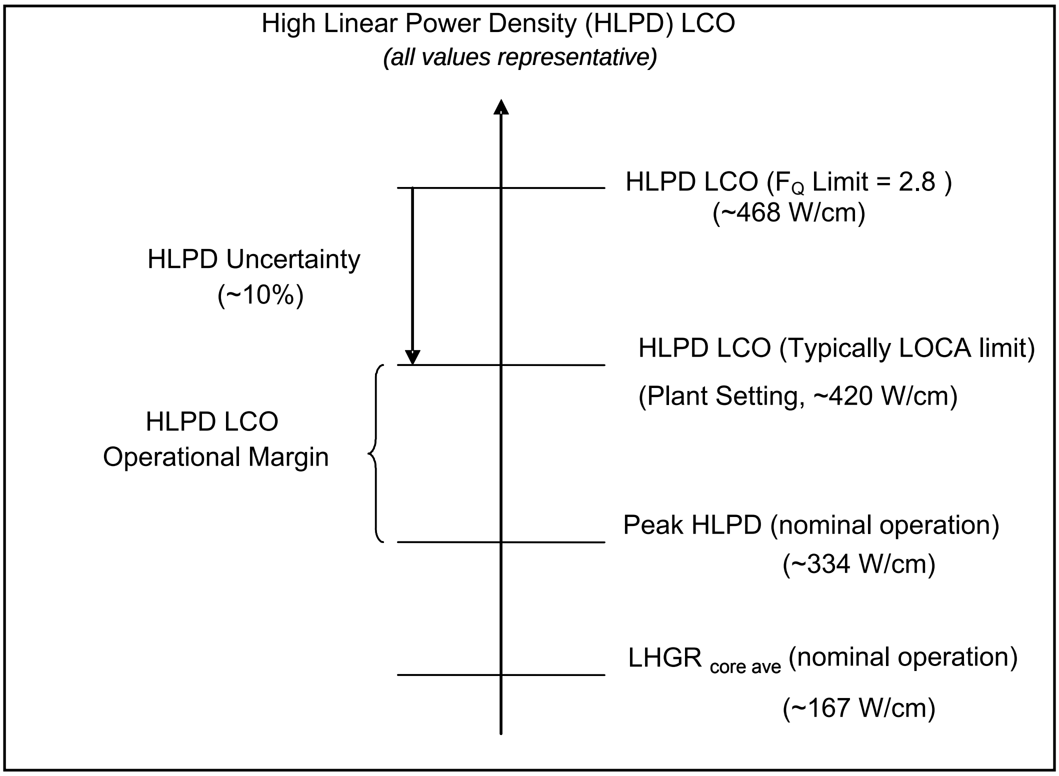


FIGURE 2- 5: TRANSIENT ANALYSIS METHOD – DNBR CRITERION

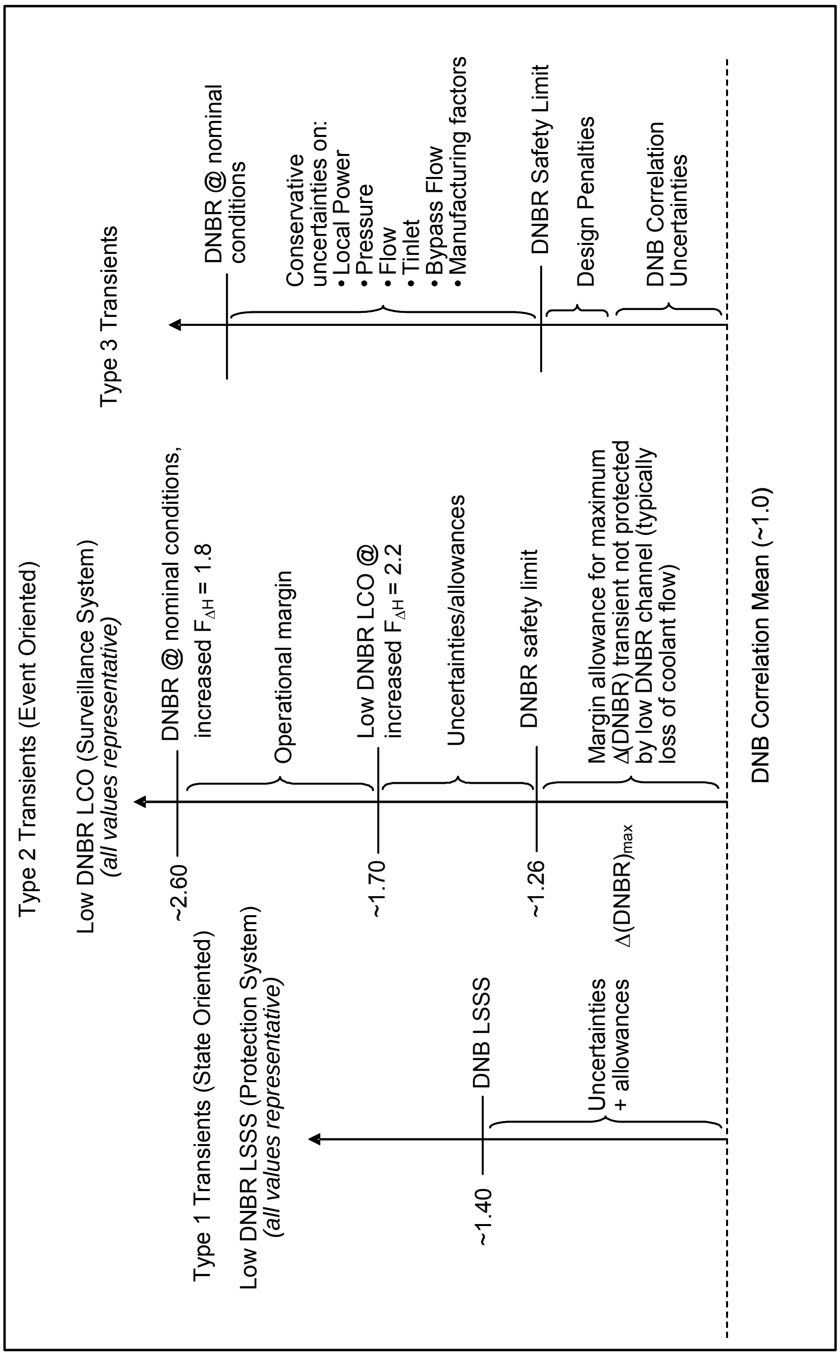


FIGURE 2- 6: RCCA PATTERN

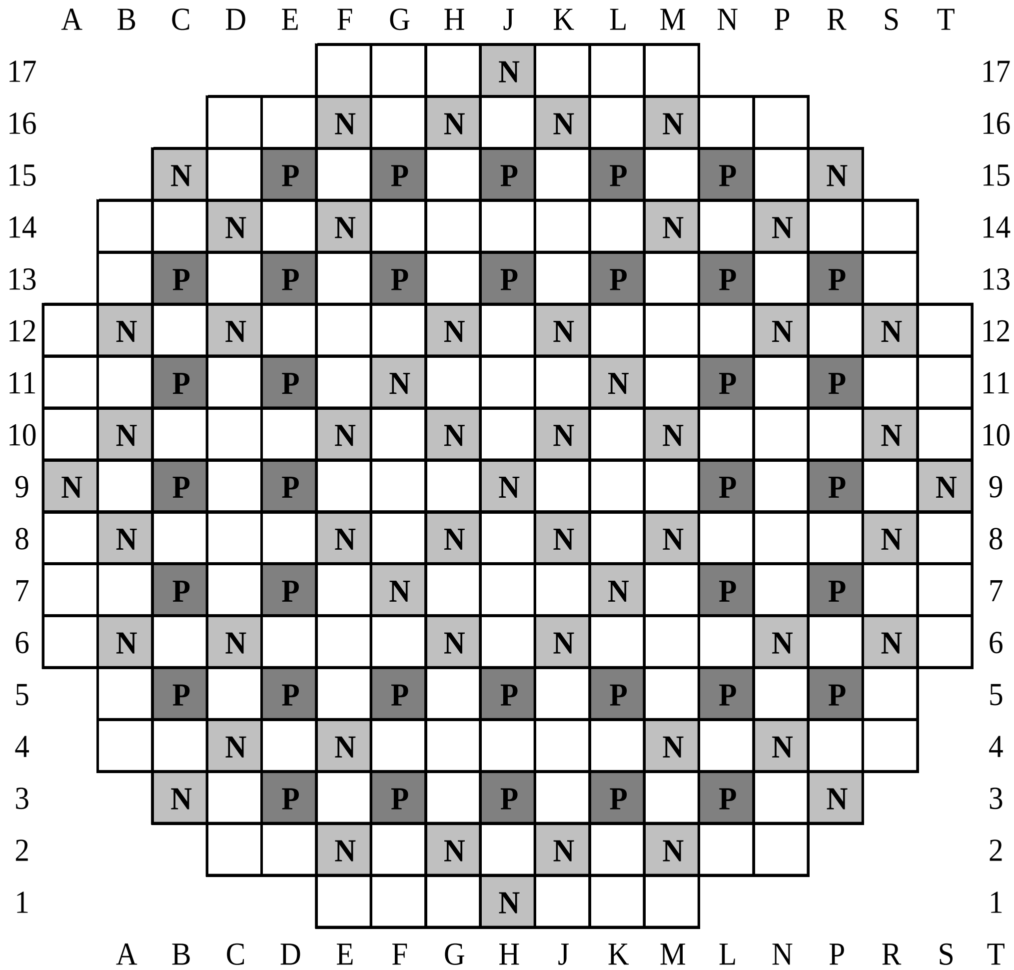


FIGURE 2- 7: AEROBALL SYSTEM

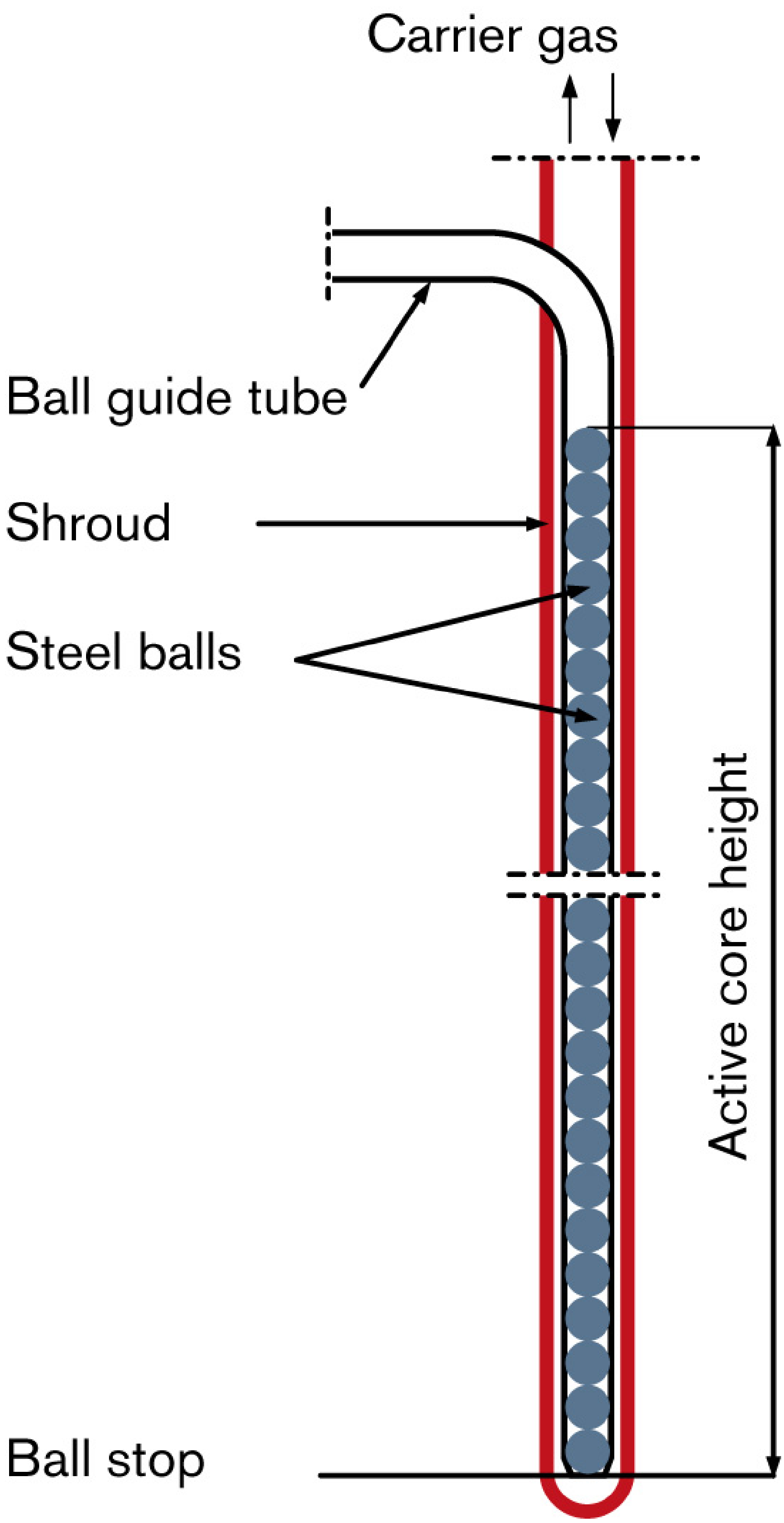


FIGURE 2- 8: ARRANGEMENT OF IN-CORE INSTRUMENTATION FINGERS

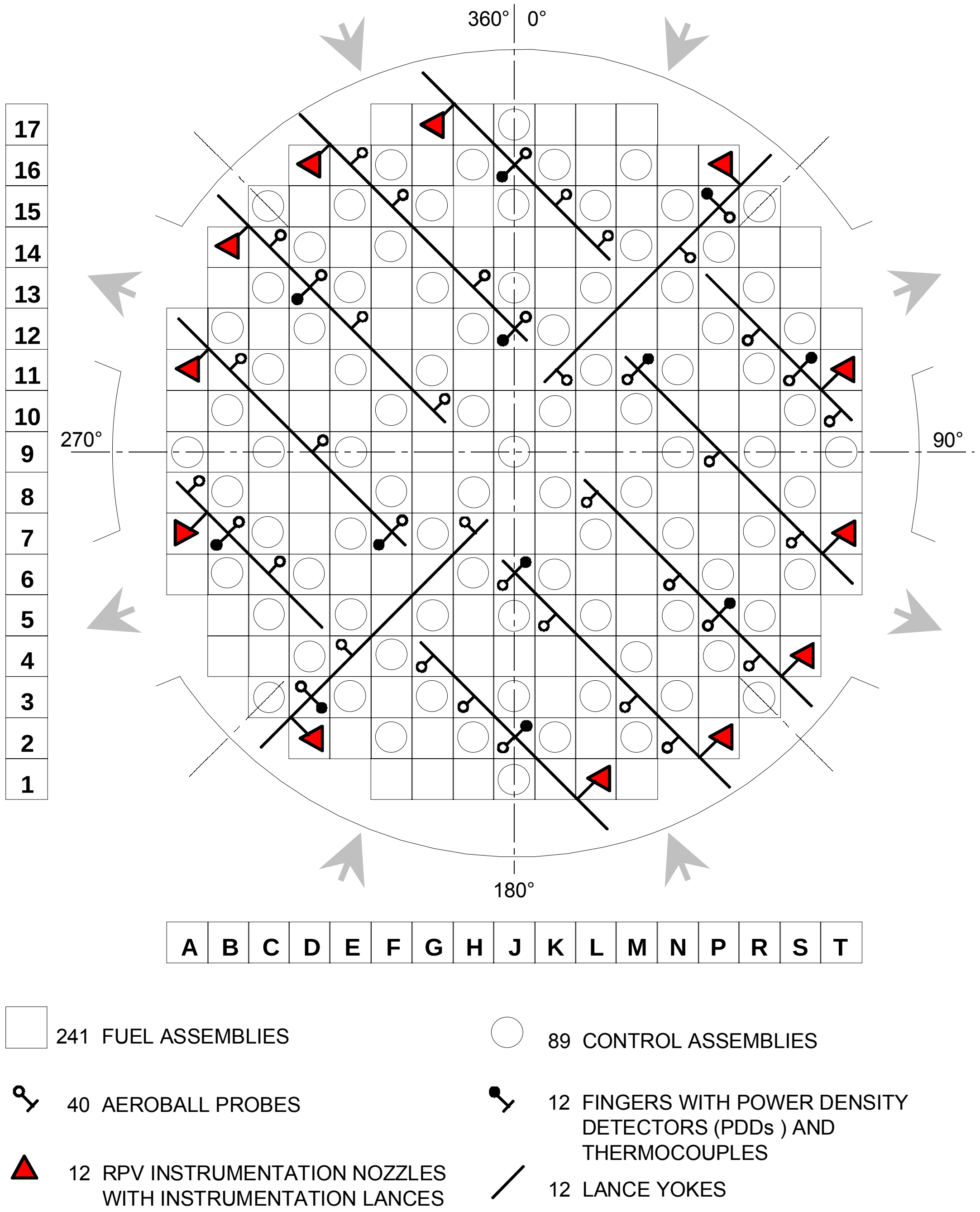
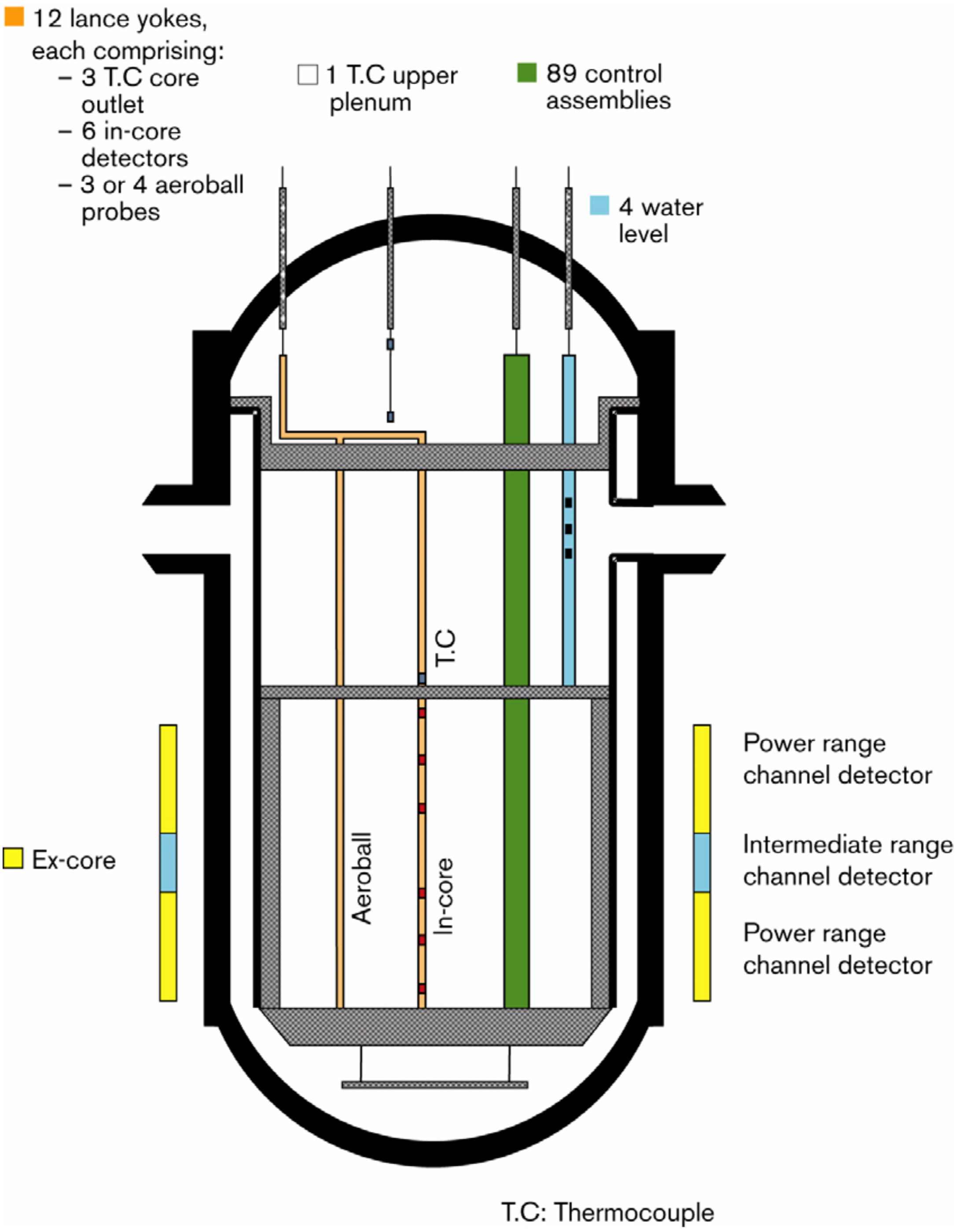


FIGURE 2- 9: IN-CORE INSTRUMENTATION



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 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 favourable 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 1.1 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,
- loss of one RCP (Even though an RCP trip does not lead to an automatic reactor trip, continued operation with one pump unavailable is not part of the licensing basis).

Containment Building Volume

The internal volume of the Containment Building is larger relative to most existing 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 CHRS need not be used before 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 of pressure 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 pressure (MHSI actuation signal). The main steam bypass or main steam relief valves open to depressurize the SGs at a rate of approximately 100°C/h to a pressure of 6 MPa. 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.

Break Preclusion

The RCS piping is designed to meet the criterion for Break Preclusion. However, the rupture of the major RCS loop pipework (LBLOCA) is assumed to be credible for the design of emergency core cooling systems (i.e., calculation of post-LOCA peak clad temperature and cladding oxidation) as required in the Technical Guidelines issued by the Standing Committee on Nuclear Safety of the French Safety Authority.

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,
- 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 3-2 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 two trains each with two valves of a different technology than the safety 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 3-3 shows a schematic arrangement of the secondary side OPP.

Both types of discharge trains (relief and safety valves) are safety-related and are credited in OPP analyses.

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

1) The first line of defence is the actuation of the turbine bypass.
2) The second line of defence is the relief valve, which ensures safety grade controlled heat removal and pressure limitation.
3) The third line of defence 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 6 MPa at a rate of approximately 100°C/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 General Characteristics

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.

Figure 3-4 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 flows out 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 extent 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 forging used in the upper part of the RPV, 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 shrinked on and 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 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 insulation material.

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, as well as seismic loads.

Figure 3-5 shows the RPV supports.

3.4.3 Reactor Pressure Vessel Internals

The RPV internals consist of the lower and upper sections. Figure 3-6 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 refuelling 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 centred 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 the annular hold-down spring that acts as a large Belleville type spring. The fuel assemblies sit directly on a perforated plate (the core support plate), which is approximately 45 cm 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 core 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 sized 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 3-7 shows a cutaway view of the heavy reflector and RPV lower internals. The picture in attachment shows one of the forged slabs under fabrication.

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 centring 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 centring pins, the UCP aligns the fuel assemblies (two centring pins each), and the RCCA guides (four centring 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 forged 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 3-8 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 man-way 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 3-9 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 is large enough to compensate for coolant expansion between 0% and 100% power under normal conditions and prevents the heaters from being uncovered during out-surges,
- 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 transients.

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 3-10 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 0.3 MPa as compared with a boiler type SG with the same tube surface.

Figure 3-11 illustrates the principle of the axial economizer.

The tube material is Inconel 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 3-12 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 3-13 shows a cutaway view of the RCP.

Reactor coolant is pumped by a 4-blades impeller attached to the bottom of the rotor shaft. Coolant is drawn up through the bottom ring of the casing, up through the 12-blades 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 pressure ($\approx$ 0.17 MPa) 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 station blackout (SBO).

Figure 3-14 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 : 780 mm,
- thickness range : 76.0 – 96.5 mm.

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 Break Preclusion (BP) concept that encompasses (a) preventive measures based on design and on manufacturing quality; (b) the Leak-Before-Break (LBB) concept and; (c) redundant surveillance measures during operation.. 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 BP 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 Table 3-1.

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.

In order to limit the corrosion rate and to avoid the production, transport and deposition of non active and activated corrosion products, the lithium concentration is adjusted to obtain an alkaline and constant pH at nominal temperature. A minimum value of pH (300°C) of 7.2 is defined.

TABLE 3- 1: REACTOR COOLANT CHEMISTRY

CHARACTERISTICS	DATA
Expected pH at 300°C	from 7.1 to 7.4
Dissolved oxygen	< 5 ppb
Chlorides	< 150 ppb
Fluorides	< 150 ppb
Sulfates	< 150 ppb
Dissolved hydrogen during normal operation (T > 120°C)	1.5 to 4.5 ppm
Lithium hydroxide	0.4 to 4.0 ppm ^{7}Li

FIGURE 3- 1: RCS LAYOUT

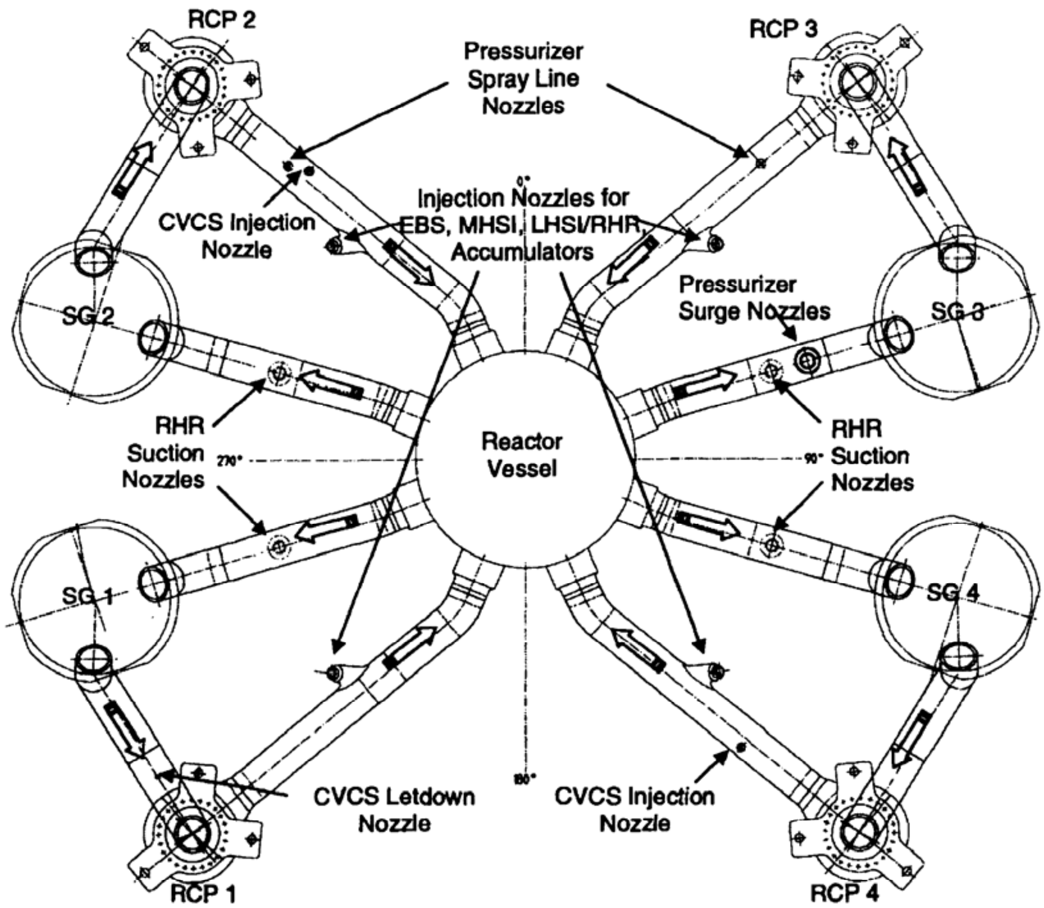


FIGURE 3- 2: PRESSURIZER WITH DISCHARGE COMPONENTS

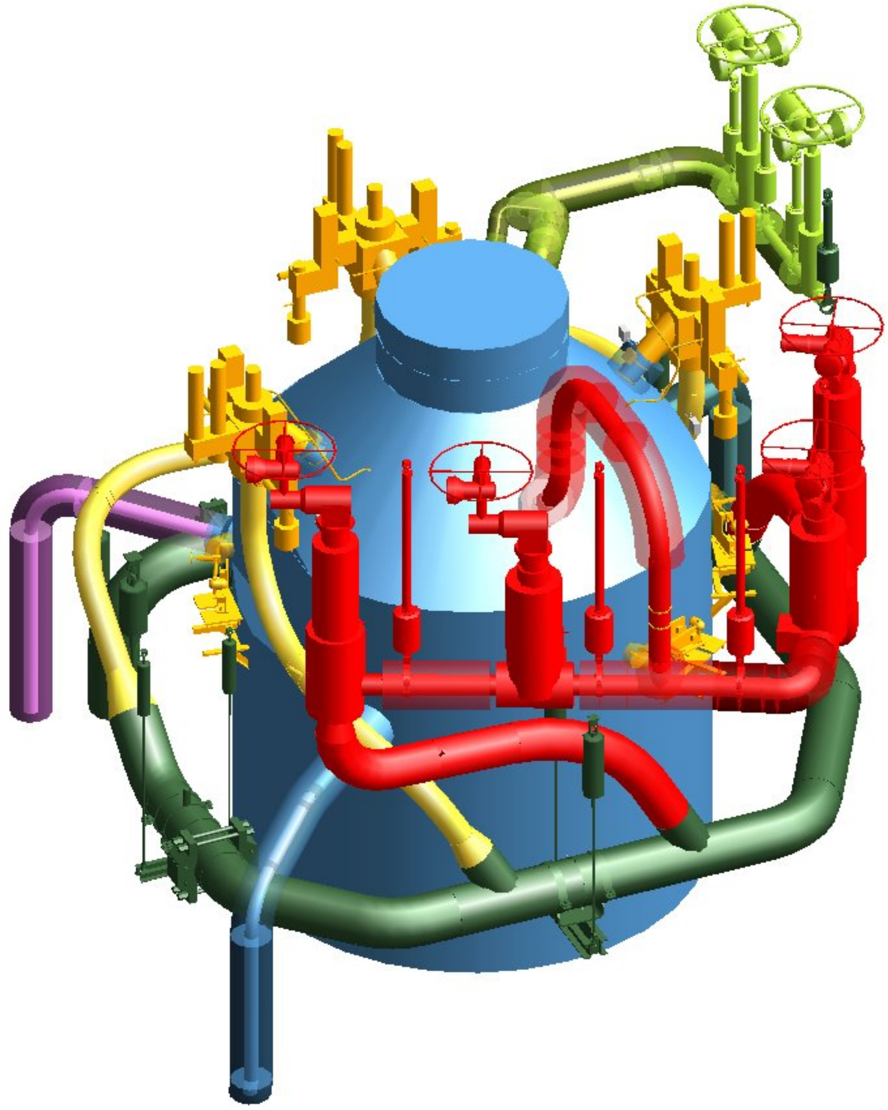


FIGURE 3- 3: ARRANGEMENT OF THE SECONDARY SIDE OVERPRESSURE PROTECTION

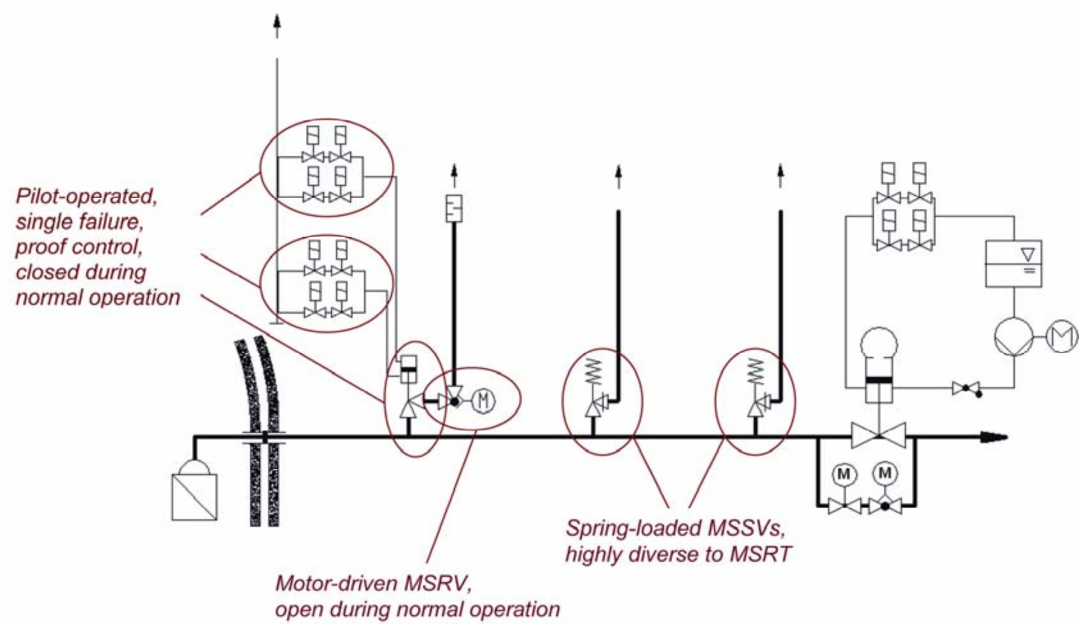


FIGURE 3- 4: REACTOR PRESSURE VESSEL

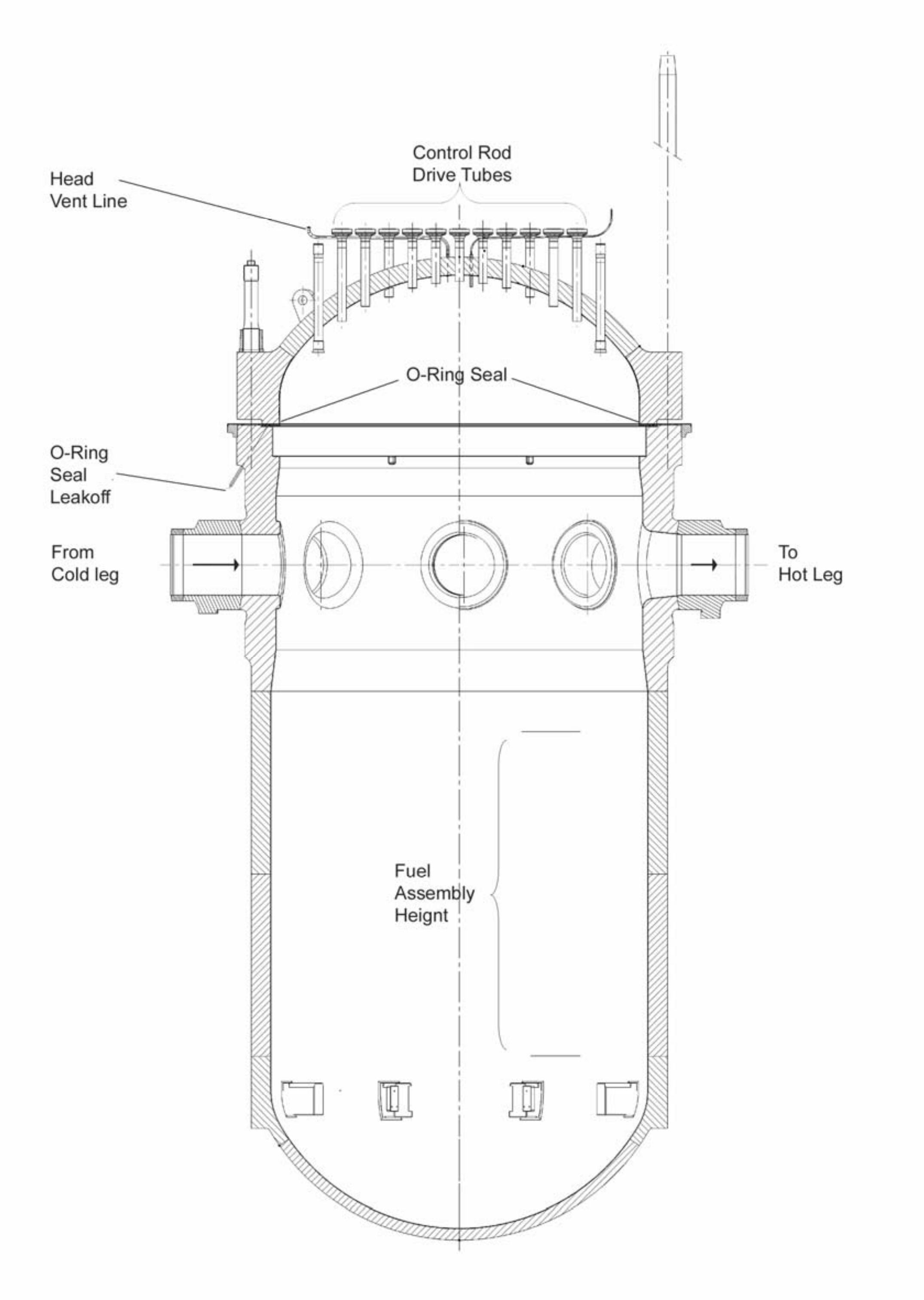


FIGURE 3- 5: RPV SUPPORTS

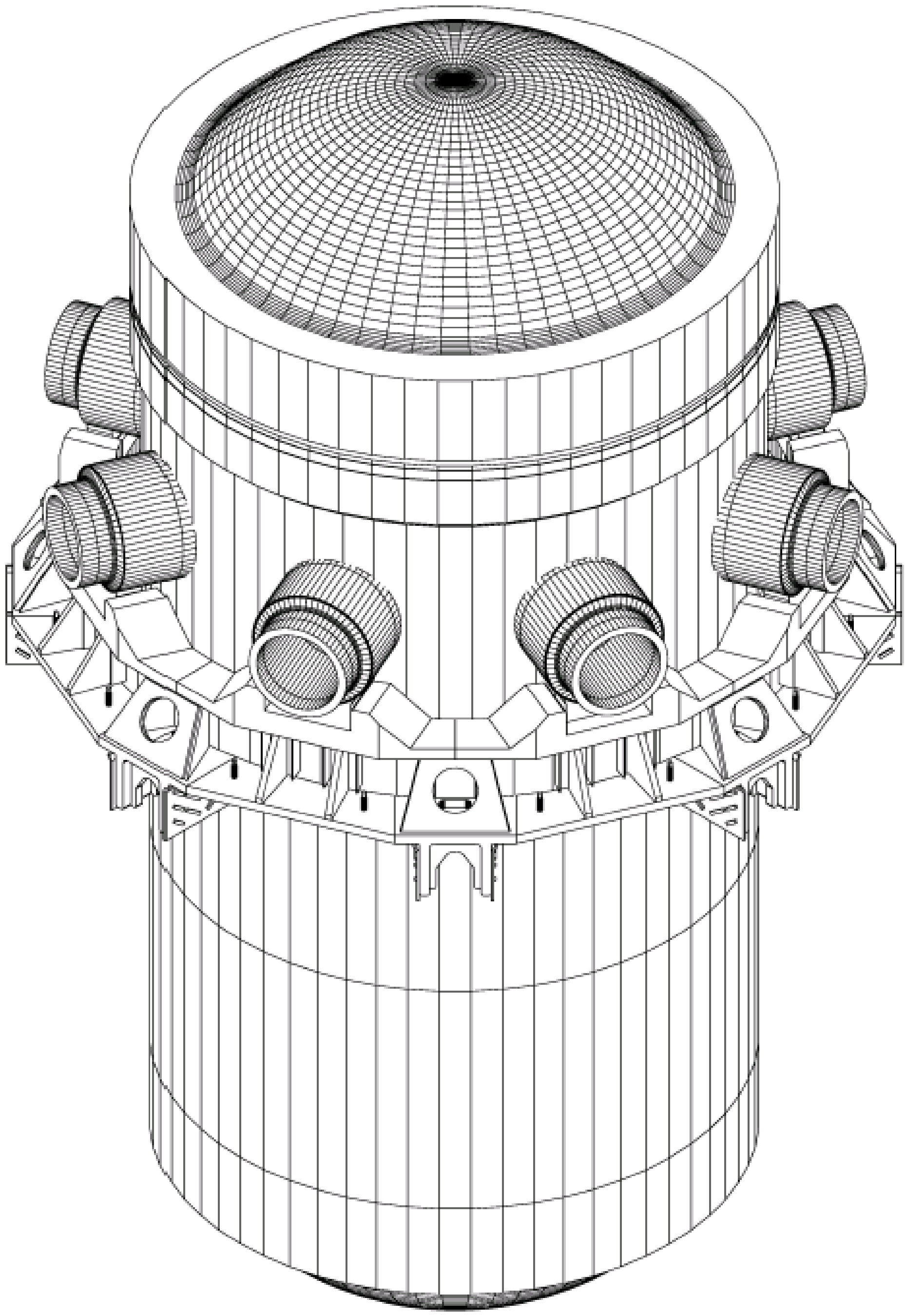


FIGURE 3- 6: RPV INTERNALS

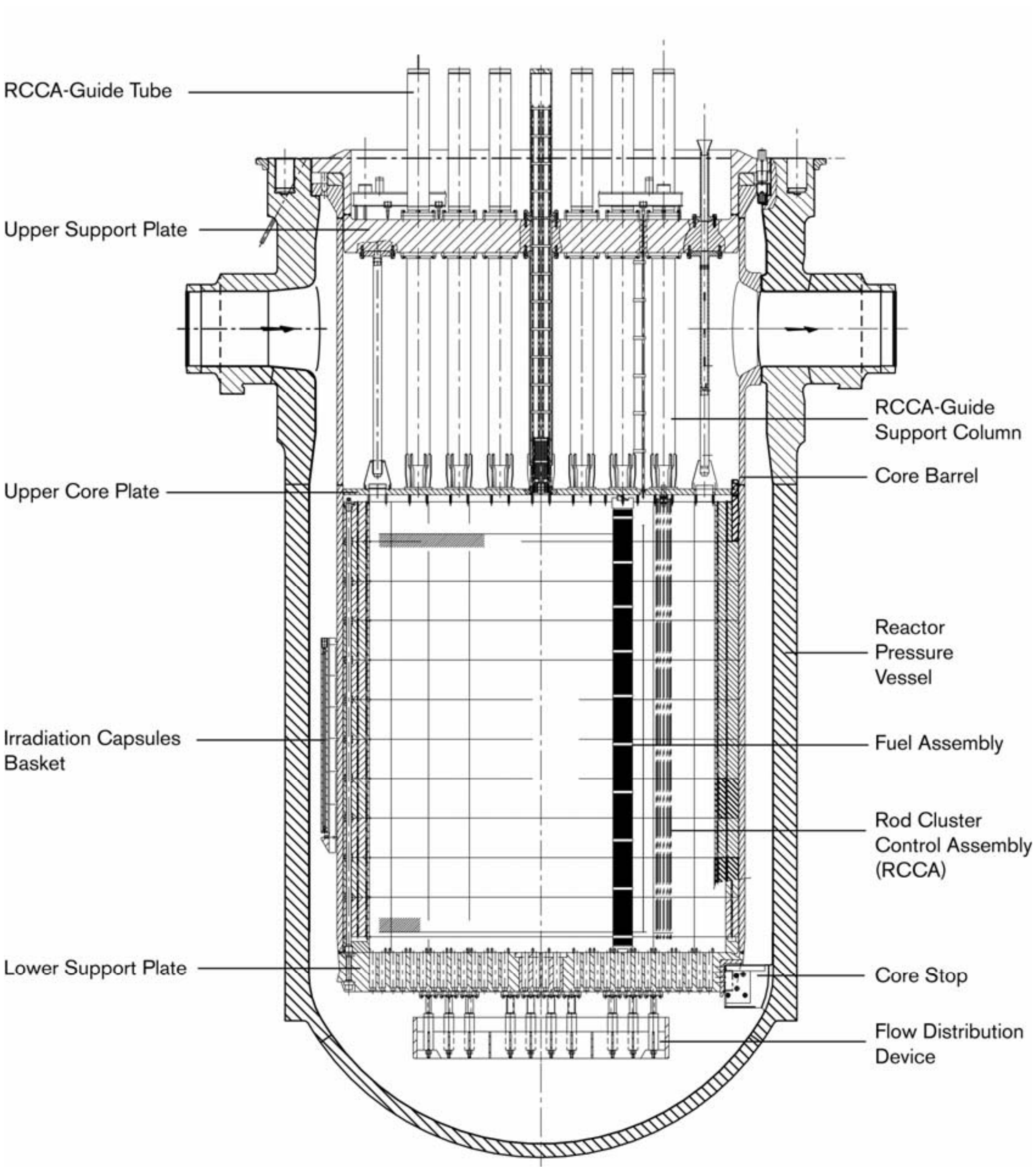
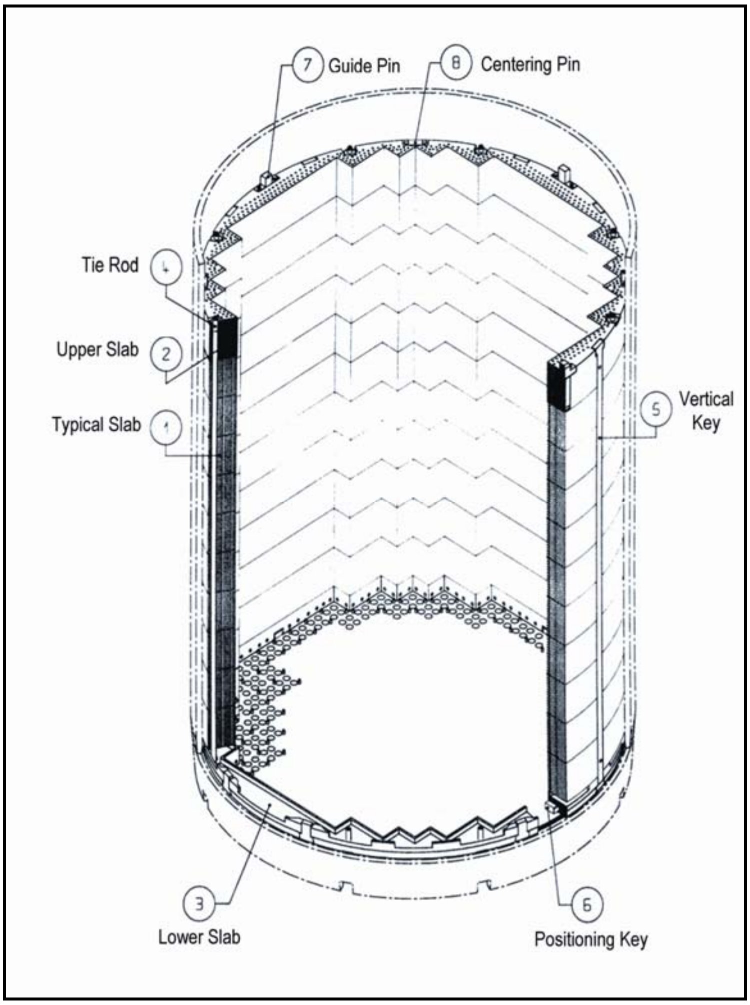


FIGURE 3- 7: HEAVY REFLECTOR



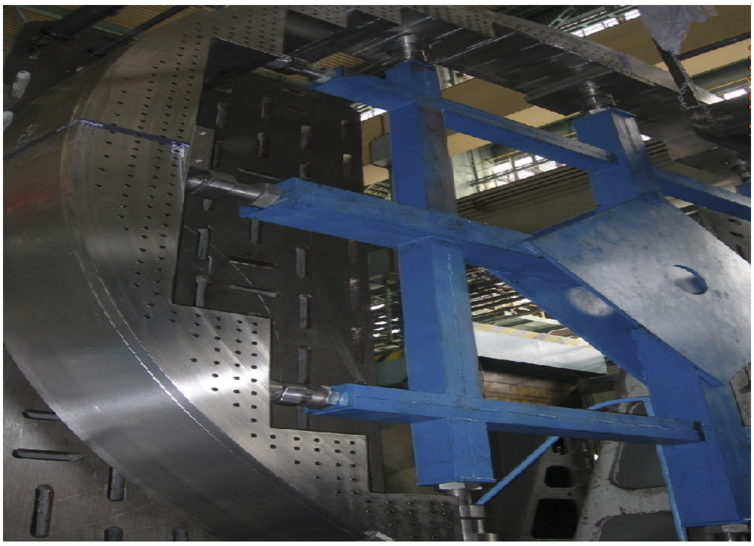


FIGURE 3- 8: PRESSURIZER

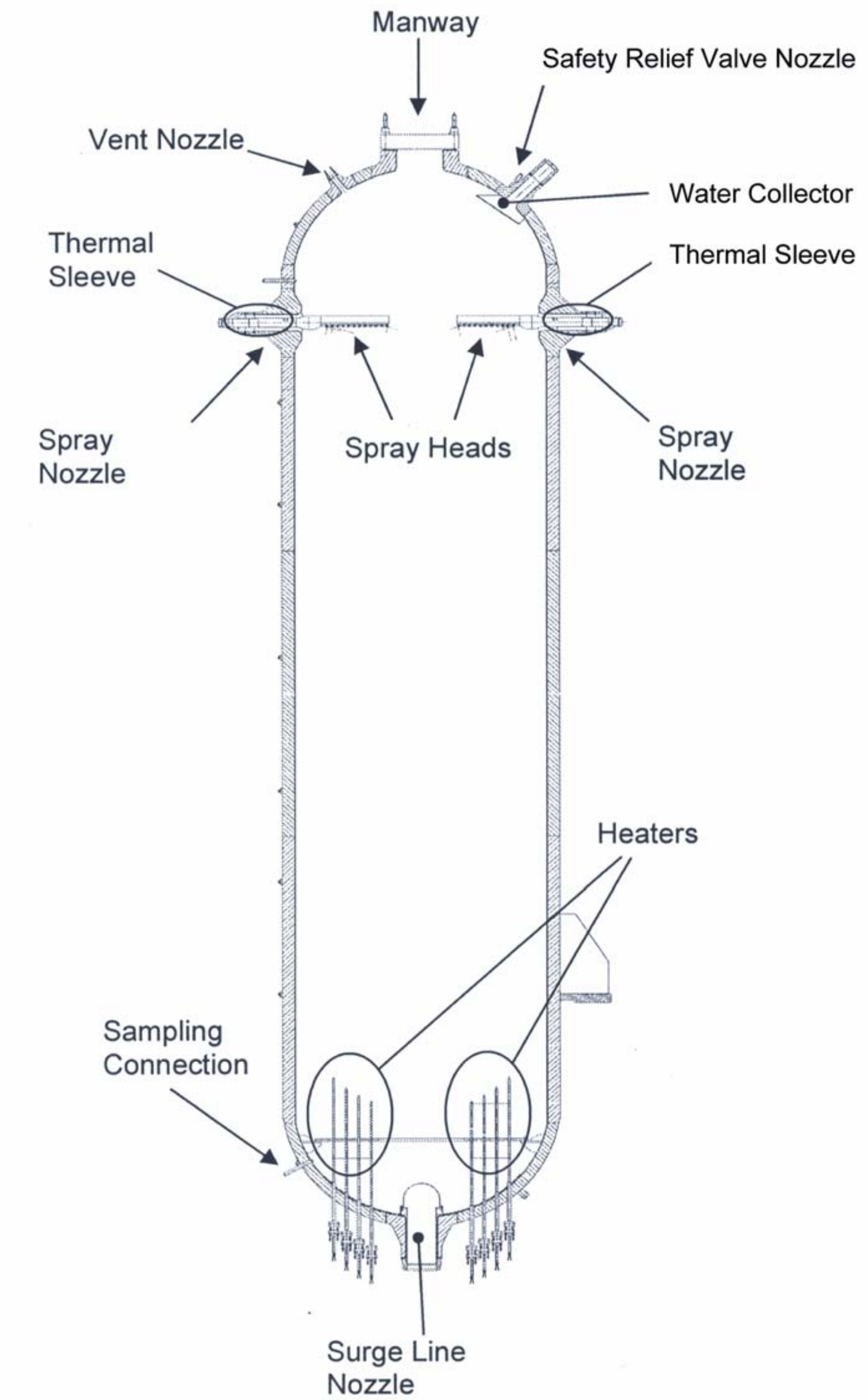


FIGURE 3- 9: PRESSURIZER SUPPORTS

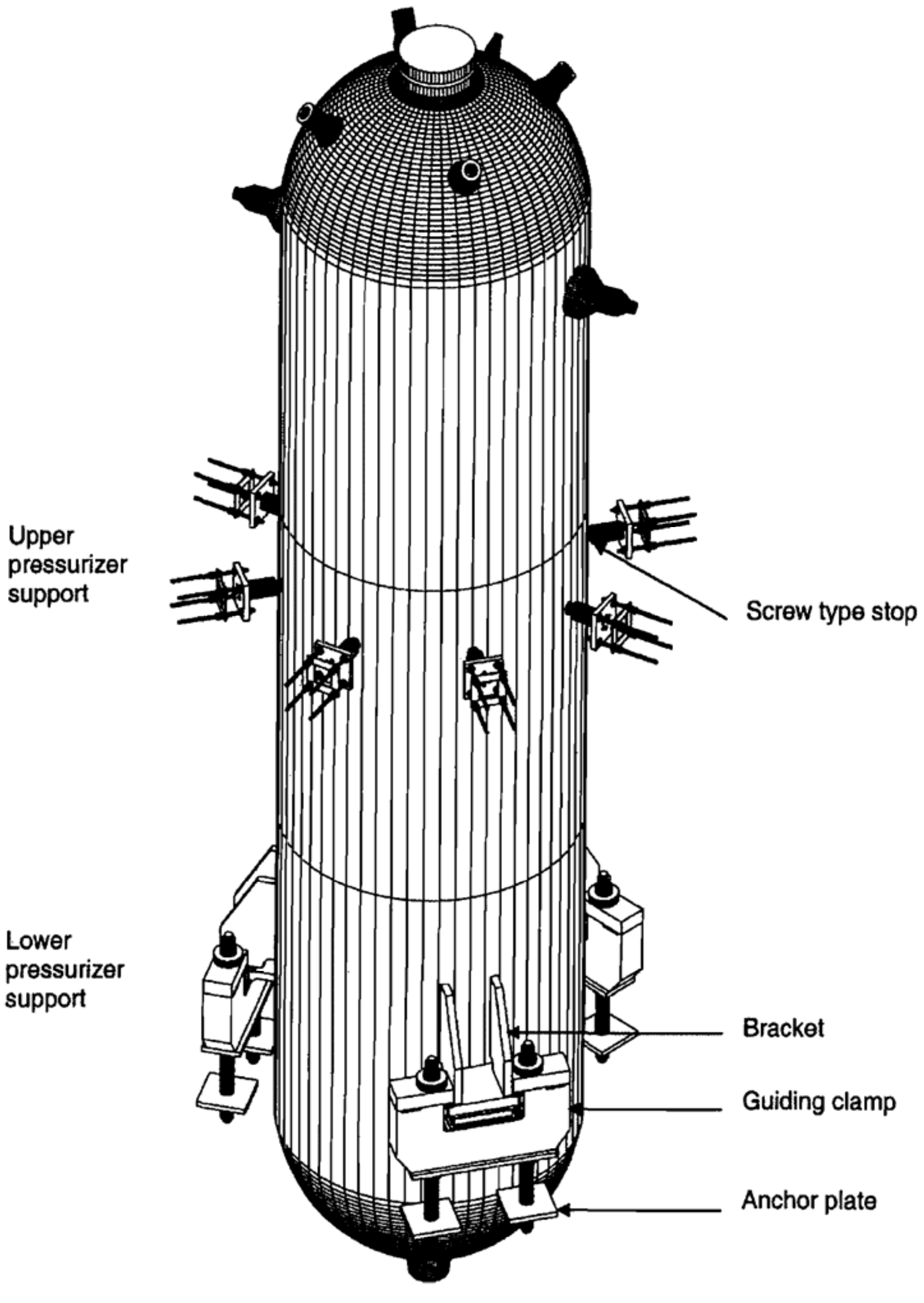


FIGURE 3- 10: STEAM GENERATOR

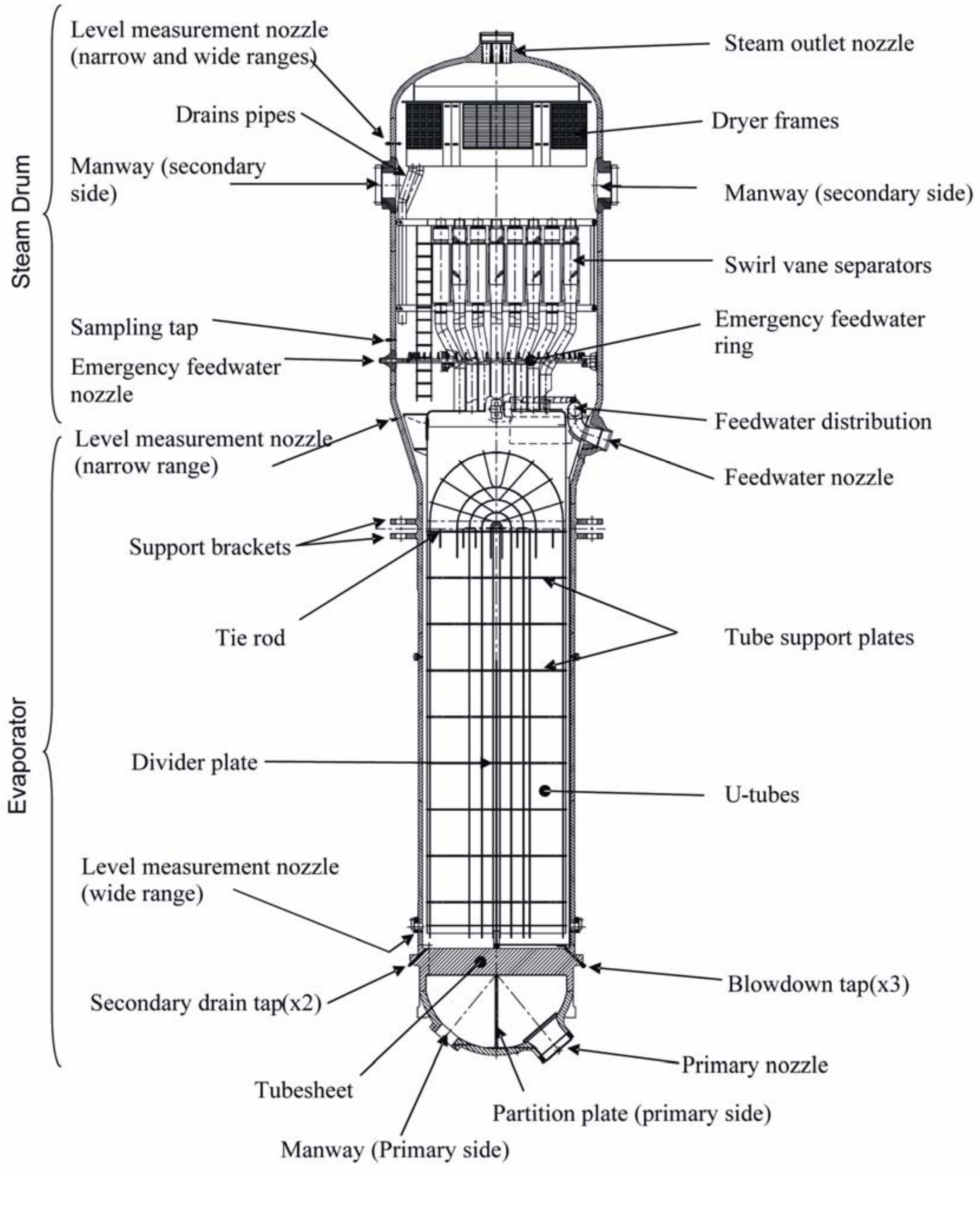


FIGURE 3- 11: STEAM GENERATOR: PRINCIPLE OF THE AXIAL ECONOMIZER

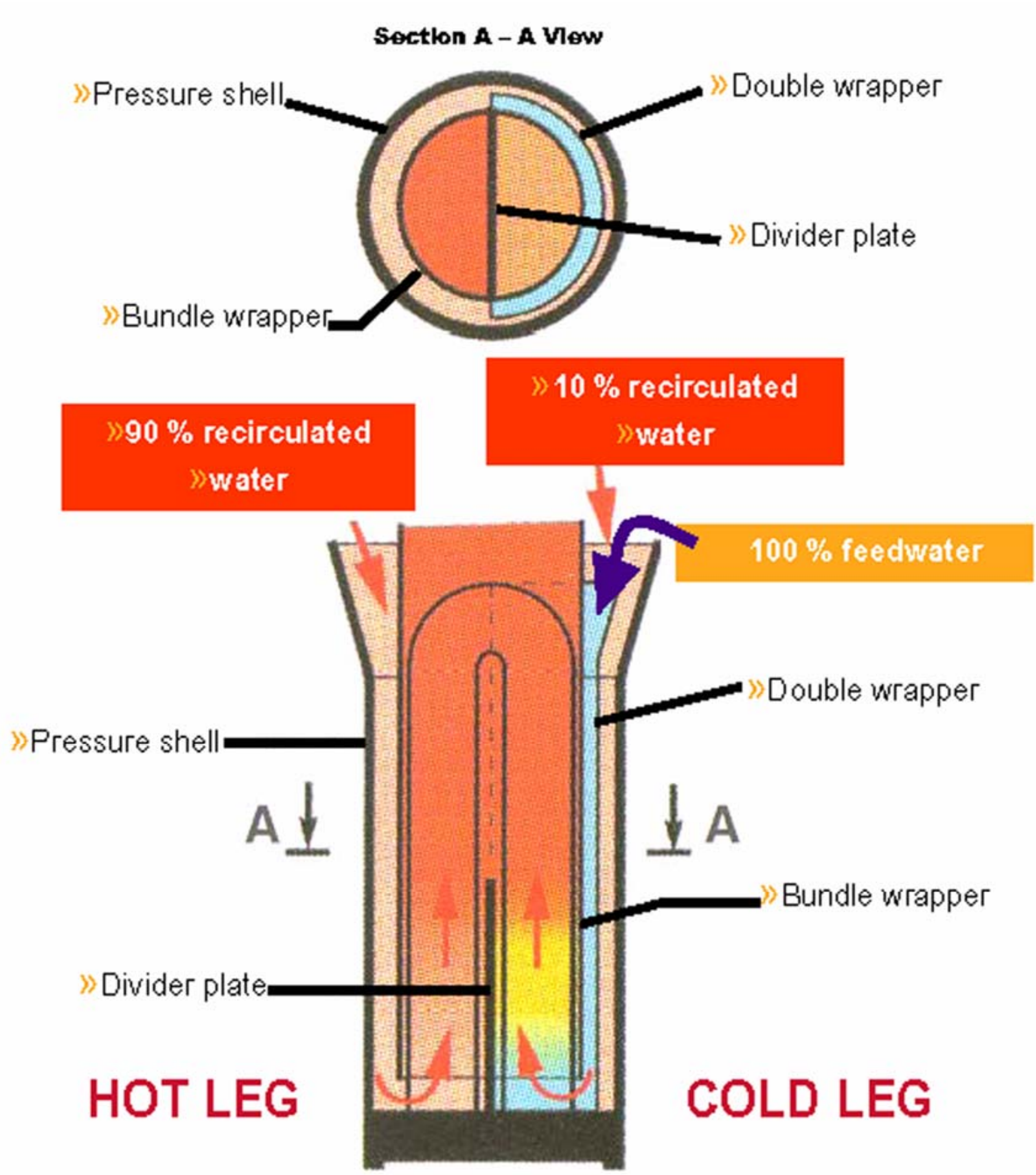


FIGURE 3- 12: STEAM GENERATOR SUPPORTS

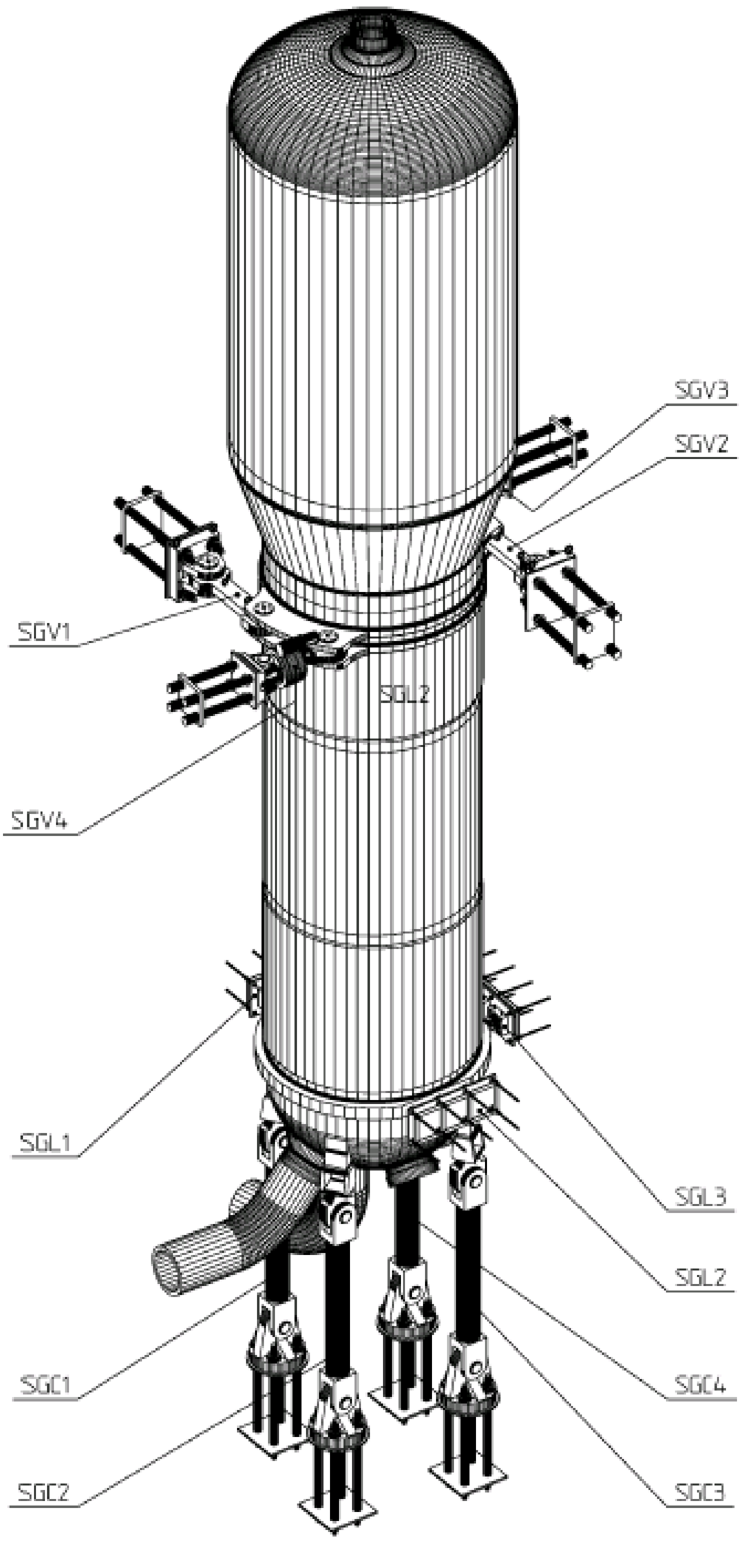


FIGURE 3- 13: REACTOR COOLING PUMP ASSEMBLY

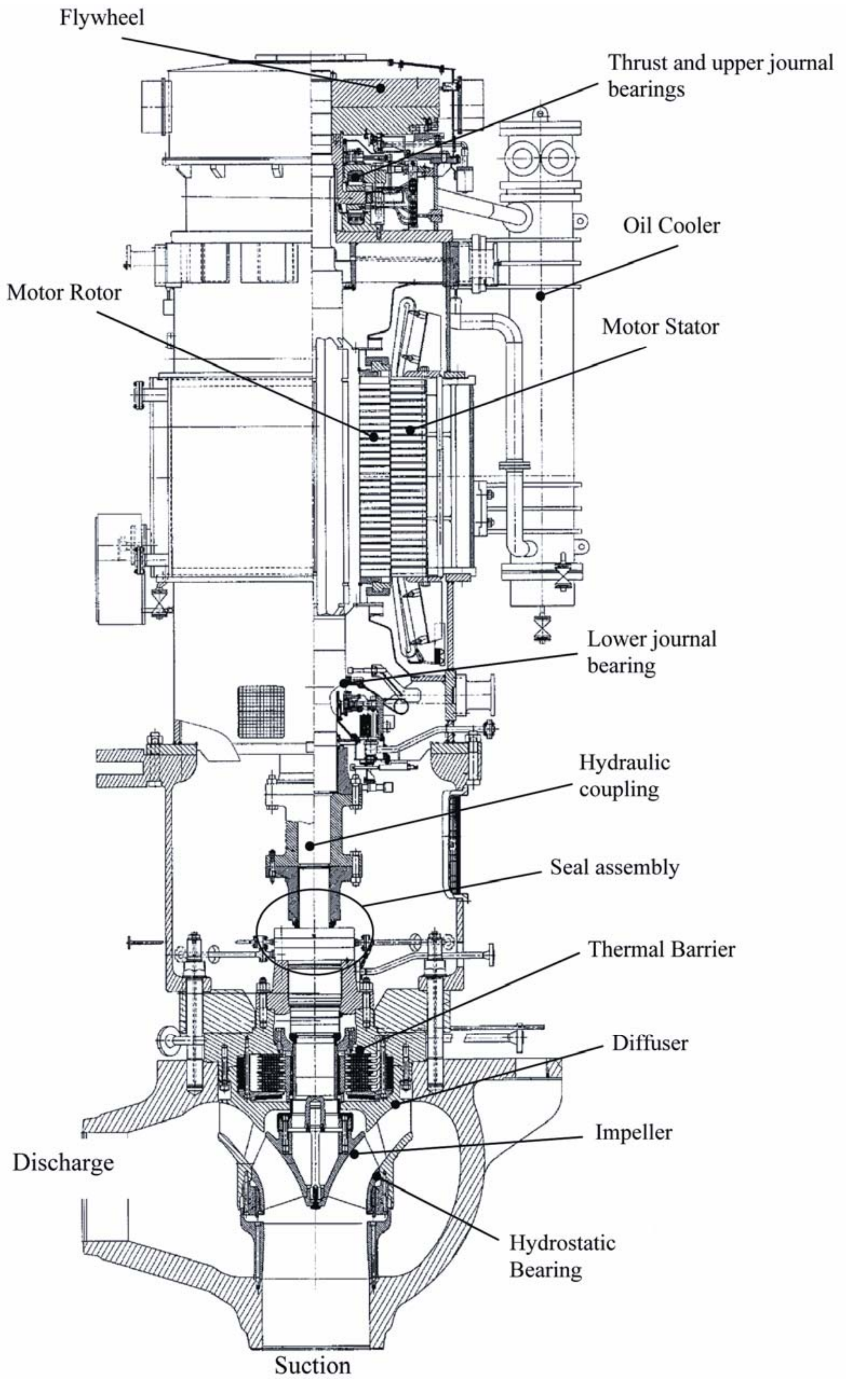
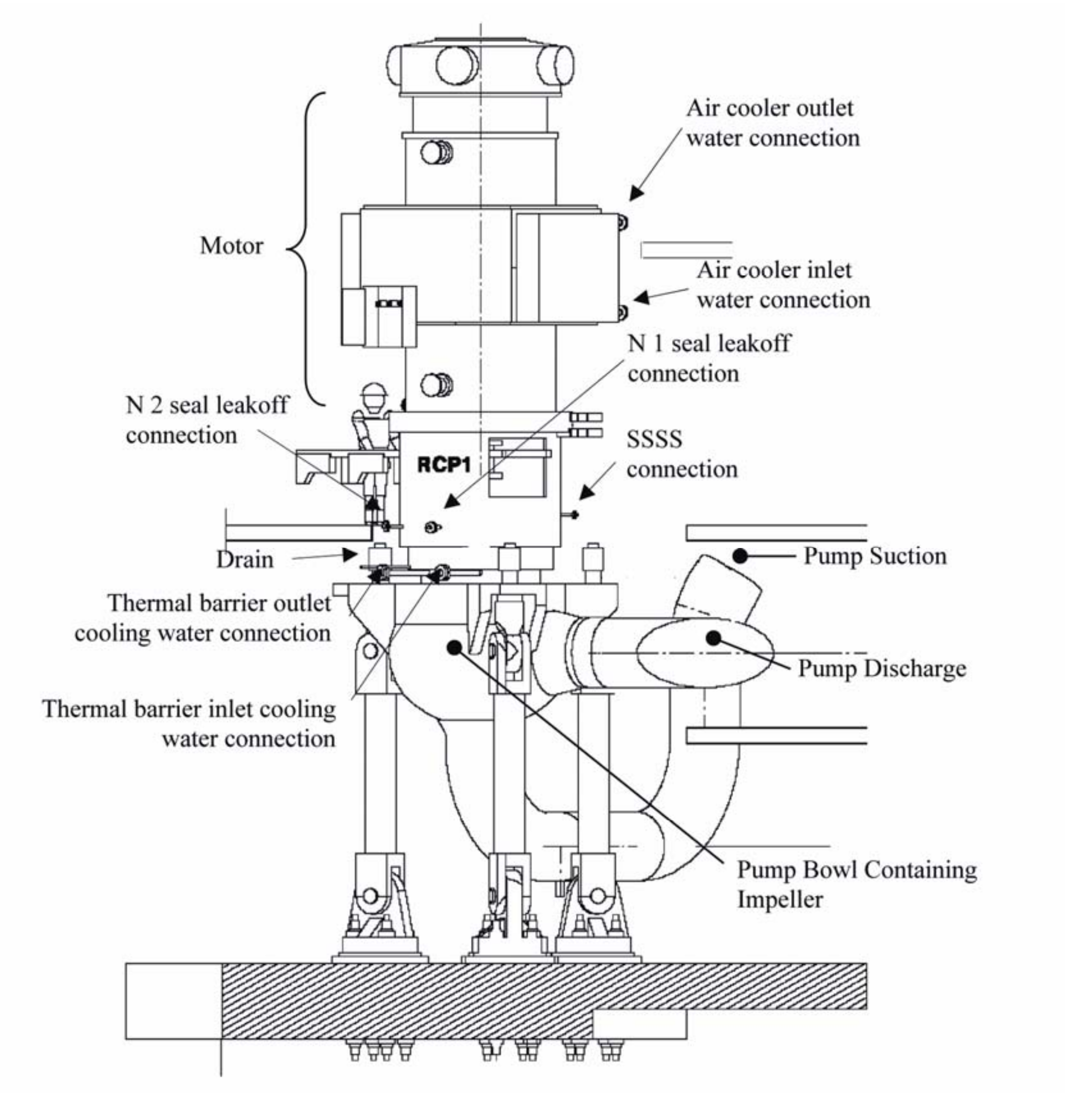


FIGURE 3- 14: REACTOR COOLING PUMP SUPPORTS



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 Component Cooling Water System (CCWS), with injection water ensured by the Chemical and Volume Control System (CVCS).

Overpressure Protection (OPP) of the RCS is ensured by the primary side OPP system.

Residual Heat Removal

The Start-Up and Shutdown System (SSS) remove the residual heat during normal operation. When SSS is not available, the RHR from the SG secondary side is ensured by the Emergency Feedwater System (EFWS) and the secondary side OPP system.

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. 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 to perform preventive maintenance of one complete safety train 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. A backup train is also provided.

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 Refuelling 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.

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.

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 relief valve discharge pressure,
- partial cooldown of the SGs by the reactor protection system ensures adequate SIS flow,
- 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 refuelling.

The normal operational functions of the SIS/RHRS are:

- RHR to reach and maintain safe shutdown state and refuelling conditions,
- transfer water from the IRWST to the reactor cavity in preparation for refuelling operations,
- cooling and mixing of the IRWST contents during normal plant operating conditions.

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.

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 refuelling cavity for normal refuelling. 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 CHRS (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 CHRS 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.

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), automatic start to ensure the RCS boration required is provided to shut down the reactor (to a sub-critical condition).

During normal plant operation, the EBS pump is used to perform the hydrostatic pressure test of the Reactor Coolant Pressure Boundary (RCPB) in plant shutdown conditions.

The EBS does not perform any other functions 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 tank room heaters.

The EBS consists of two identical trains. Each 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 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 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. 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.

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.

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 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 Demineralised 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 when the normal pressurizer spray is unavailable.

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 burn-up compensation through the addition of boron and/or demineralised 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 heat-up or any boration or dilution. Connections are provided to the CVCS to allow for chemical additions and for boric acid and demineralised 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 CHRS 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 liner) which is connected to the pump suction line and located above the highest CCW load. The surge tank is connected to a demineralised 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 both the CHRS when it is used during a severe accident scenario involving core melt and the backup train of the FPCS.

Each dedicated 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 demineralised water supply line with pressurizing pump.

4.7 Essential Service Water System/Ultimate Cooling Water System

The Essential Service Water System (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 Ultimate Cooling Water System (UCWS) includes two 50% trains of dedicated cooling for severe accident mitigation.

The safety function of the ESWS is to cool the CCWS. The UCWS provides cooling of the CHRS (via the dedicated CHRS 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 (120°C), 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.

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 refuelling 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 Demineralised 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 4-7 shows the flow schematic of the Fuel Pool Cooling and Purification System.

The FPCS has two separate and independent main trains, each consisting of two pumps installed in parallel, a heat exchanger cooled by the CCWS, and associated piping and valves. A third train equipped with one pump and one heat exchanger cooled by an intermediate cooling chain shared with the CHRS is provided. 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.

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.

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

4.10 Steam and Power Conversion System

The steam and power conversion system includes the Main Steam System (MSS), the turbine generator[1], the main condenser, the feedwater system, the feedwater storage tank, and other auxiliary systems. Most of these systems are part of the conventional island which is described in section 9.

The main condenser condenses the turbine exhaust and transfers the heat rejected in the cycle to the circulating water system. Regenerative feedwater heaters heat the condensate and the feedwater and return it to the SGs.

[1] The NI is technology-neutral with regard to turbine technology.

A feedwater storage tank is integrated into this cycle to deaerate and heat the condensate. This tank also provides a buffer volume to accommodate minor system transients.

The following parts of the steam and power conversion system have safety-related functions with respect to RHR:

- EFWS (see section 4.4),
- MSS inside the Nuclear Island,
- MFWS inside the Nuclear Island.

4.10.1 Main Steam System

The MSS routes the steam produced in the four SGs to the HP turbine inlet valves.

Each main steam line has a MSIV located just outside the containment. A bypass line with shut-off and control valves are provided around each MSIV for warming the piping system downstream from the cold condition. After pressure balance has been achieved between the secondary side of the SGs and the main steam lines in the Turbine Building, the MSIVs are opened and the warm-up valves are closed.

Overpressure protection on each main steam line is provided by a Main Steam Relief Train (MSRT) and two MSSVs. Each MSRT consists of a Main Steam Relief Isolation Valve (MSRIV) and a downstream MSRV. The MSRIVs are fast opening valves that are normally closed. The MSRVs are normally open control valves. The MSRIVs open quickly in the event of an overpressure transient. The MSRVs allow termination of flow through a stuck open MSRIV.

In addition to steam supply to the main turbine, the MSS supplies backup auxiliary steam for miscellaneous uses such as deaerator pegging. Additionally, the MSS features a non-safety grade turbine bypass to the condenser for operational flexibility.

The safety functions of the MSS are to provide reactivity control, RHR, and containment of radioactive substances.

The Break Preclusion concept applies to the main steam lines from the steam generators to the fixed points downstream of the MSIVs.

Figure 4-8 shows a simplified flow diagram of the MSS.

Reactivity Control

The MSS does not directly affect reactivity control. However, the safety-related portion of the MSS indirectly supports reactivity control by isolation of the steam lines in the event of excessive steam flow. An excessive increase in steam flow causes overcooling of the reactor coolant and thus introduces positive reactivity feedback to the core.

In the event of a steam line break, quick closure of the steam line isolation devices enables the broken line to be isolated to limit the cooldown of the RCS and the energy release, so that the allowable limits specified for the fuel and the design conditions for the RPV and the containment are not exceeded. The isolation devices stop the steam flow (or two-phase mixture) that may flow through them in the normal flow or reverse flow direction.

Residual Heat Removal

The MSS removes residual heat by steam dump to the condenser via the turbine by-pass (if available) or to the atmosphere via the MSRT from the hot shutdown condition until RHRS entry conditions are reached.

In case of SIS signal following a small or intermediate break LOCA or SGTR, the MSS cools the primary side down to the MHSI pressure by means of the MSRTs (i.e., partial cooldown) or turbine bypass (if available).

Main steam release to the atmosphere is designed so that the fuel temperature remains within specified limits and the RCPB remains within the design conditions.

Heat removal is performed even in the event of a loss of external power combined with a single failure (failure to open one MSRT).

Containment of Radioactive Substances

The MSS retains radioactivity in the event of a SGTR by isolating the SG on the steam side.

To ensure the containment of radioactive substances in the event of a SGTR, the affected SG is detected and isolated; thus, the release of reactor coolant to the atmosphere is minimized.

Operational Functions

The MSS supplies main steam to the turbine and all other main steam consumers in the turbine building during normal operation, and removes the residual heat by steam transfer to the condenser during non-power operation. The main steam is transported through four lines from the SGs via the main steam valve stations in compartments on top of the Safeguard Buildings to the main stop and control valves of the HP turbine.

From each of the SGs, the main steam flows in a main steam line out of the Reactor Building via the valve compartment, into the Turbine Building and up to the turbine valves.

A main steam valve station consists of:

- one MSIV,
- two MSSVs,
- one MSRIV,
- one MSRV.

The MSIV is welded into a straight piping section between the containment penetration and a fixed point downstream of the penetration. The MSIV is an oil-pneumatic operated gate valve.

The MSRV is a motor-driven control valve that is welded into the discharge piping downstream of the MSRIV.

The MSRIV is a fast opening, open–shut valve that is welded to the main steam line section between containment penetration and MSIV.

The two MSSVs are spring-loaded safety valves. Each one is welded onto the main steam line section between the containment penetration and MSIV.

The warm-up line incorporates one motor-driven isolation valve and one motor-driven control valve.

Downstream of the MSIVs, pipes branch off the main steam lines to the turbine bypass station. The heating steam for the two steam reheaters is extracted between the main stop and control valves of the HP turbine.

Following expansion in the HP turbine, the steam is dried, reheated and fed to the three double-flow LP turbine cylinders. Each LP cylinder is assigned a condenser in which the steam condenses following expansion in the LP turbine. The heat of condensation is removed by the condenser circulating water system.

The HP and LP turbines comprise a permanently coupled unit with the generator. During start-up or on turbine shutdown, main steam is dumped directly into the condensers via the turbine bypass.

Condensate which collects in the condenser hotwell is pumped through four stages of LP feedwater heating and delivered to the deaerator by the condensate pumps.

Feedwater is pumped from the deaerator through two stages of HP feedwater heating and delivered to the SGs by the feedwater pumps. The drains accumulating in the feedwater heaters, reheaters, moisture separators, and in drains traps downstream of the third stage of feedwater heating are cascaded back and subsequently pumped forward by a drain pump. Drains upstream of the third stage of feedwater heating cascade back to the condenser. The feedwater control valves are located in the valve compartments and are accessible at all times. A swing check valve is located inside the containment upstream of each SG.

During start-up and shutdown, the SGs are supplied with feedwater by means of the SSS.

The required degree of purity of the water in the steam/water cycle is maintained by means of the SGBS. The blow down water is cooled, cleaned, and returned to the steam/water cycle. The required makeup water for the cycle is conditioned in a demineralising system, stored in the demineralised water storage tank, and fed as required to the cycle.

To ensure heat removal from the RCS via the SGs, three systems for supply of feedwater to the SGs are provided, namely the MFWS, the SSS, and the EFWS with the latter being a safety system.

Two additional possibilities for steam dumping are provided with either the condenser or the atmosphere acting as the heat sink. The latter path is designed on the basis of safety considerations.

Under normal operating conditions, there are no detectable radioactive contaminants present in the steam and power conversion system. The system is monitored for increases in radioactivity by means of the main steam line monitors (N16, noble gases), the SGBS, and the activity monitoring system for the condenser evacuation system (non-condensing gases extracted from the condenser).

4.10.2 Main Feedwater System

The MFWS extends from the feedwater tank through the feedwater pumping system, the HP feedwater heaters, feedwater isolation valves, and up to the SG main feedwater inlet nozzles. During normal power operation, the feedwater supply to the SGs is provided by the 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 feedwater reliability.

Figure 4-9 shows a flow schematic of the MFWS and SSS.

The MFWS discharges feedwater from the feedwater tank by the feedwater pumping system via the feedwater piping system to the SGs. Feedwater is heated in two HP feedwater heater stages by the turbine extraction steam system. The condensed steam is cascaded back by the heater drains system to the feedwater tank. These systems are not required to operate during or after an accident. The system layout ensures that no malfunction of any component or piping of these systems will affect the safe operation of the plant or any system which is important to safety. Only the function of MFWS containment isolation is important to safety. Thus, the portion of the MFWS from the main feedwater containment isolation valves and feedwater piping system (from the isolation valve inlets to the SG main feedwater inlet nozzles) is safety class. The safety requirements of the MFWS are described below.

For accident scenarios, the MFWS participates indirectly in the reactivity control function by closure of the main feedwater isolation valve and the full-load and low-load isolation valves so as to prevent an overcooling transient due to SG overfeed.

During normal operation, the MFWS controls the SG supply of feedwater at the required flow rate as long as the start-up and shutdown system is available.

To provide containment of radioactive substances in the event of a SGTR, the affected SG is detected and isolated. The MFWS provides isolation of the affected SG during a SGTR by means of the main feedwater isolation valve and full-load and low-load isolation valves. Thus, the potential for release of reactor coolant to the environment is minimized.

The single failure criterion is applied to the isolation valves of the MFWS to provide safe isolation of the feedwater supply by the MFWS and the start-up and shutdown system.

The isolation valves of the MFWS are provided with emergency power backup so their functions can be performed in the event of a loss of off-site power.

The main feedwater piping system from the inlet of the MFWS containment isolation valves up to the SGs fulfils the following safety functions:

- prevention of excessive mass flow by isolation via the main feedwater isolation valve and the high and low load isolation valves to prevent SG overfeed,
- retention of radioactivity in the affected SG in case of a SGTR by main feedwater isolation,
- control of the feedwater supply from the start-up and shutdown system to the SGs.

The feedwater piping system supplies feedwater from the feedwater tank to the SGs during power operation and start-up/shutdown operations. The supply of feedwater is provided by the feedwater pumps or by the start-up and shutdown pump.

The feedwater piping system outside the turbine building up to the SGs conveys the feedwater leaving the feedwater heating system to the SGs, and controls the SG water level by means of full-load and low-load control valves. The feedwater piping system outside the turbine building up to the SGs shuts off the feedwater supply in the event of a feedwater control malfunction, thus preventing overfeeding of the SGs. The feedwater piping system from the inlet of the MFWS containment isolation valves up to the SGs performs the following functions during accidents:

- isolates the SG in the event of feedwater line breaks,
- shuts off the feedwater supply in case of main steam or feedwater line break to prevent containment over pressurization,
- retains the radioactivity in the affected SG in the event of SGTR,
- isolates the SG in case of LOCA to prevent containment bypass,
- prevents depressurization of the unaffected SGs in the event of a non-isolable feedwater line break inside the containment,
- prevents depressurization of the SGs in the event of an isolable feedwater line break,
- reduces overcooling in the event of a main steam line break.

FIGURE 4- 1: MAIN FLUID SYSTEMS

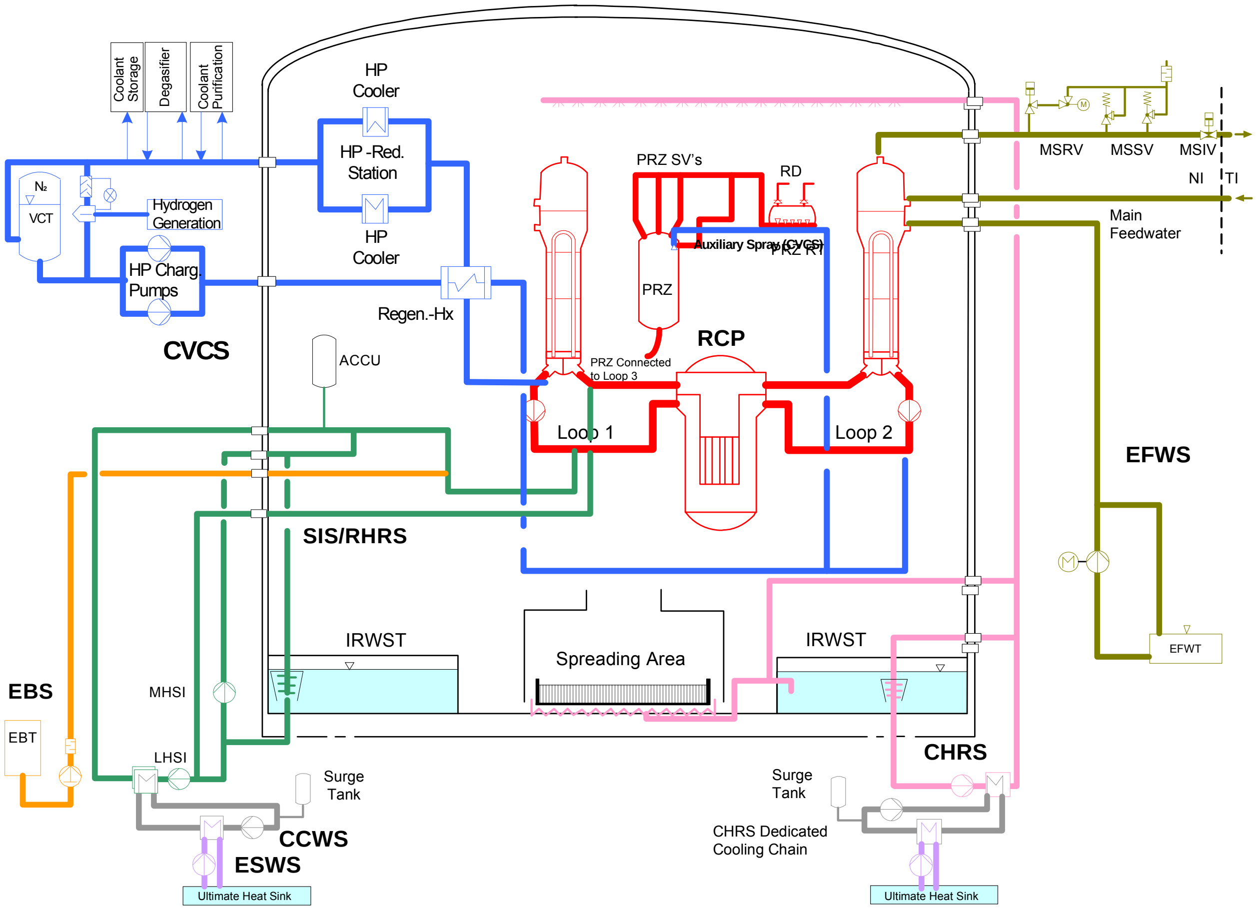


FIGURE 4- 2: SAFETY INJECTION SYSTEM

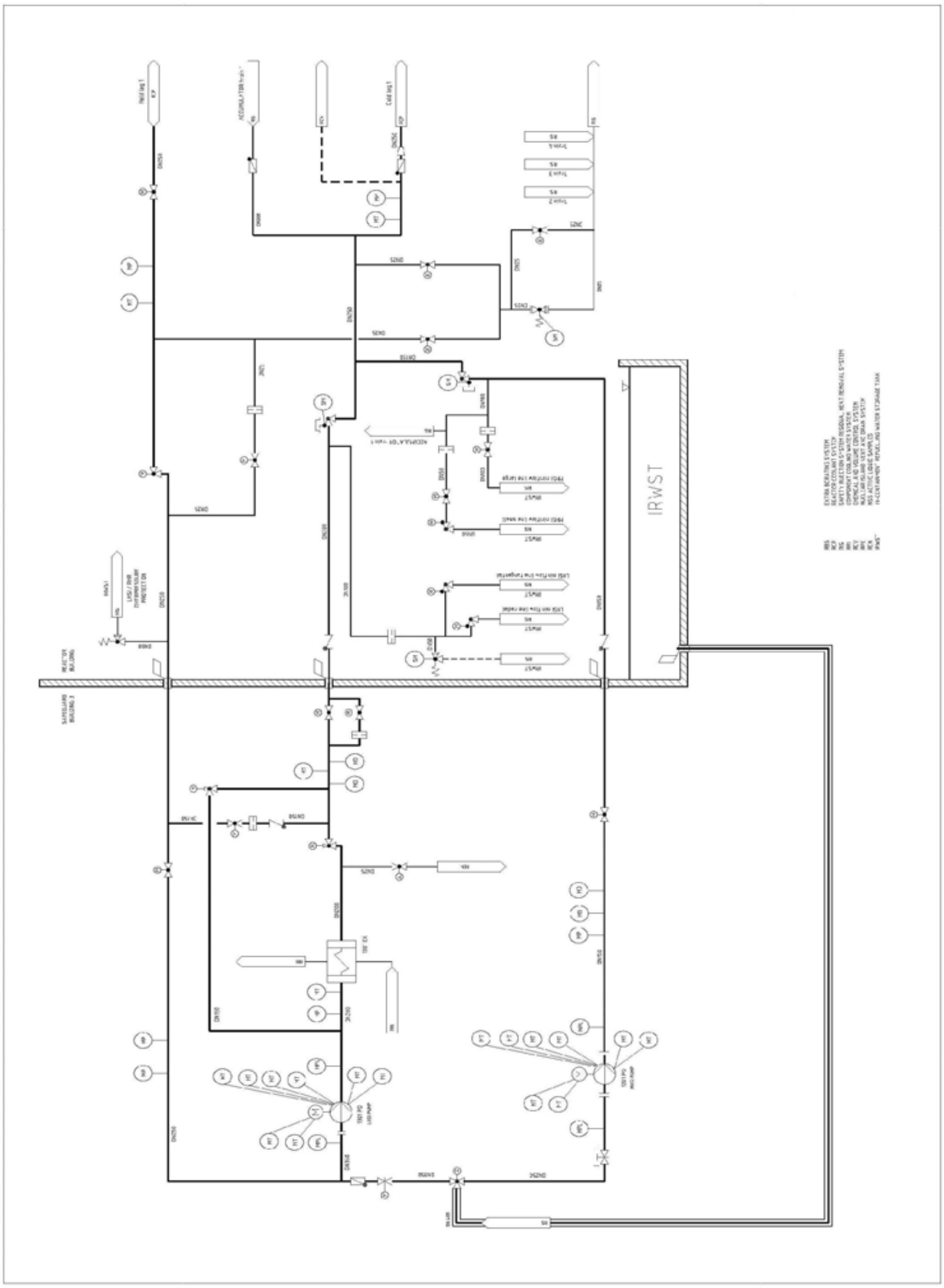


FIGURE 4- 3: IRWST AND RELATED COMPONENTS

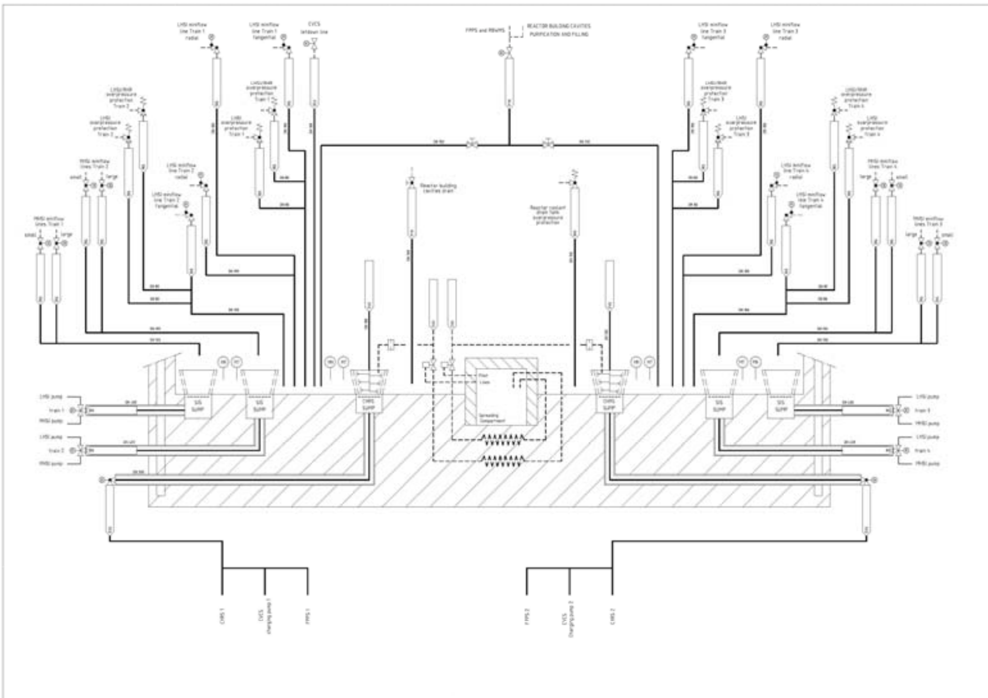


FIGURE 4- 4: EMERGENCY FEEDWATER SYSTEM

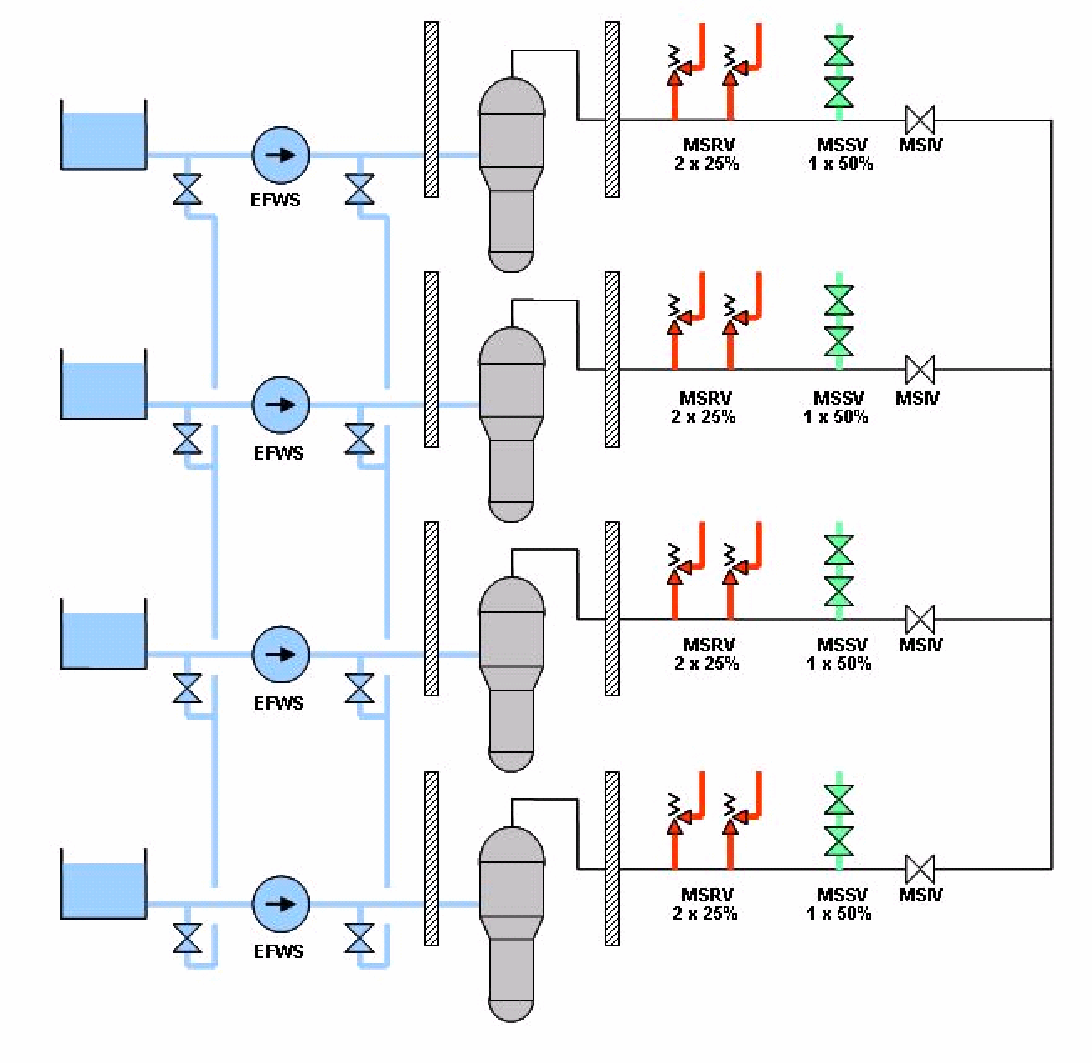


FIGURE 4- 5: CHEMICAL AND VOLUME CONTROL SYSTEM

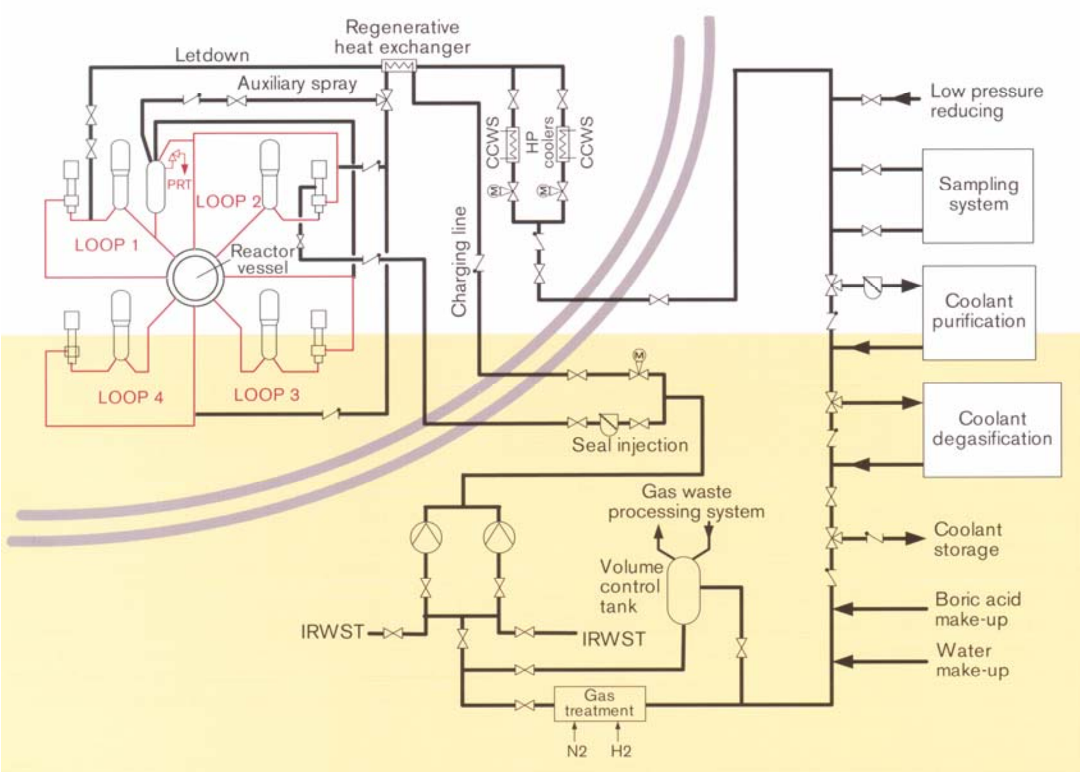


FIGURE 4- 6: COMPONENT COOLING WATER SYSTEM

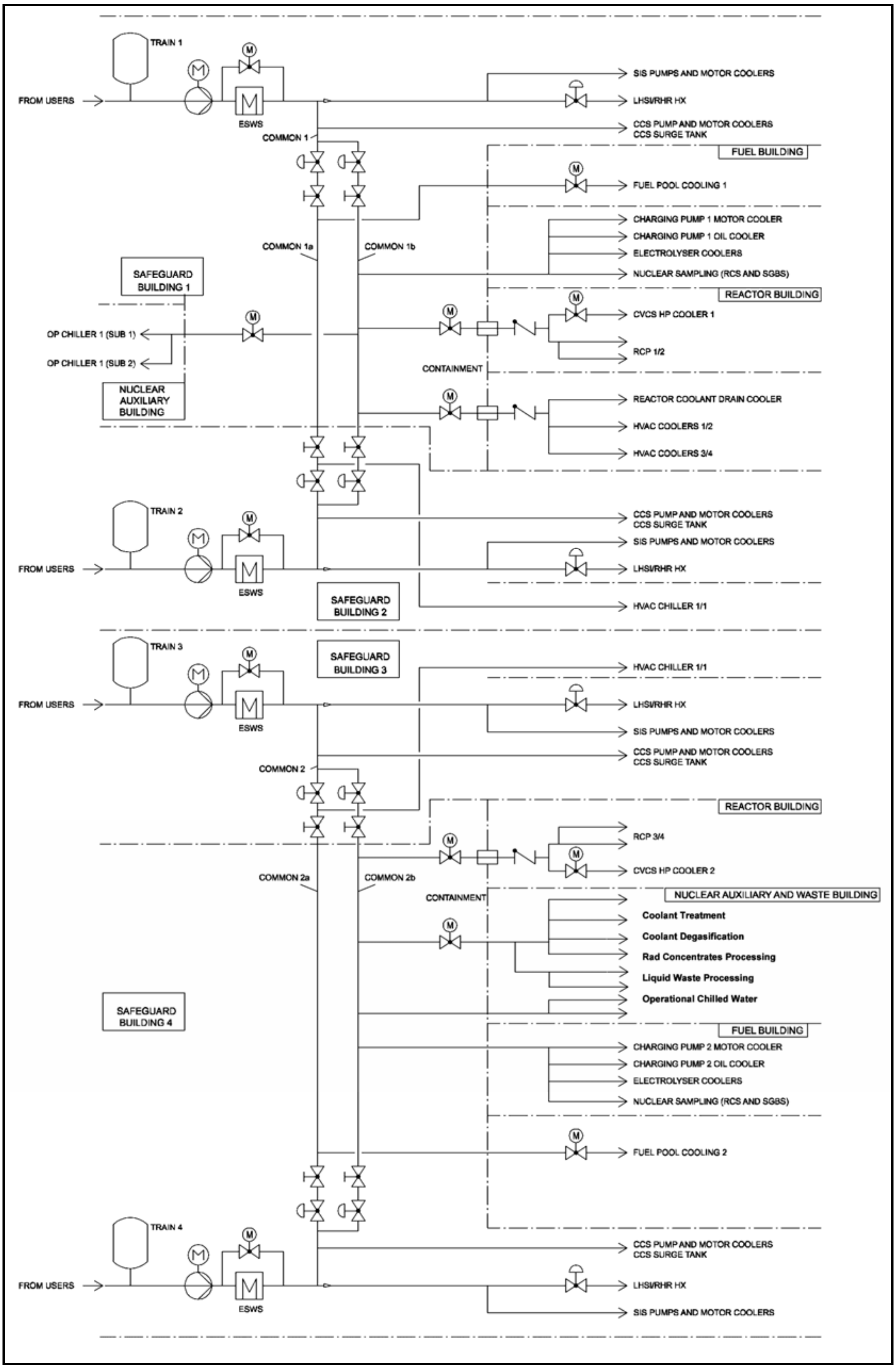


FIGURE 4- 7: FUEL POOL COOLING AND PURIFICATION SYSTEM

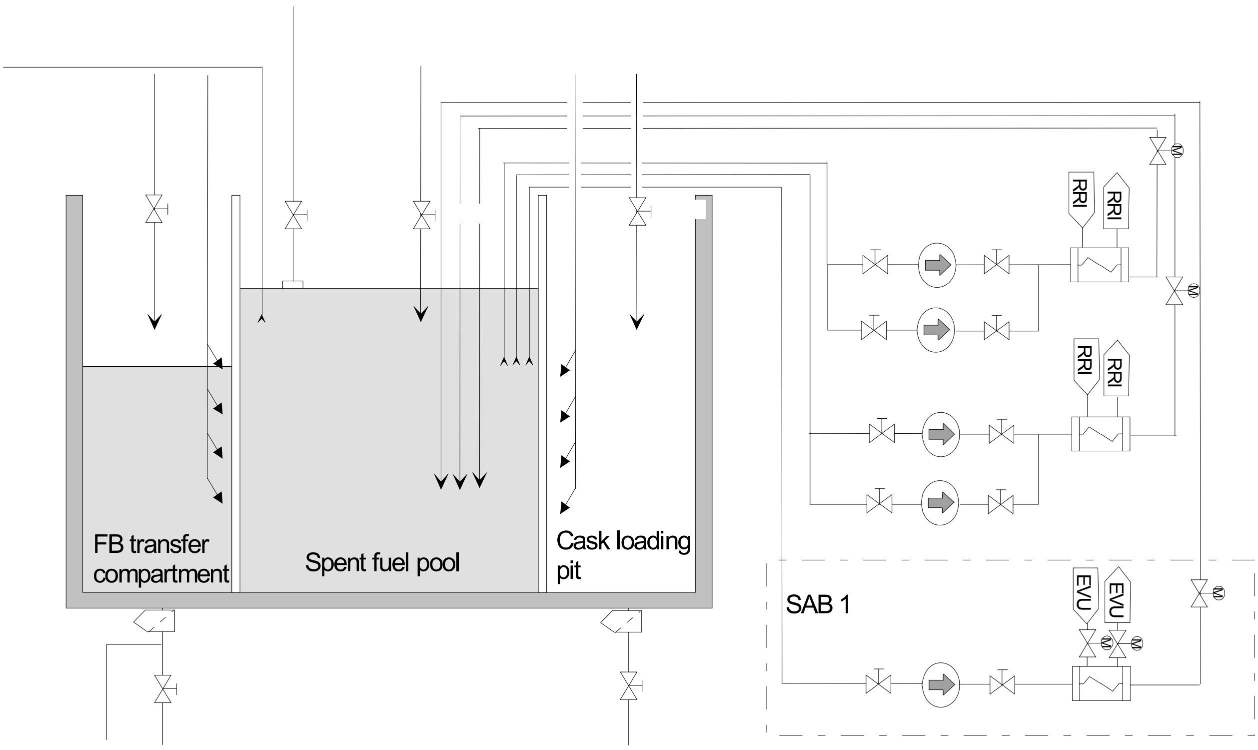


FIGURE 4- 8: MAIN STEAM SYSTEM (SAFETY CLASSIFIED PORTION)

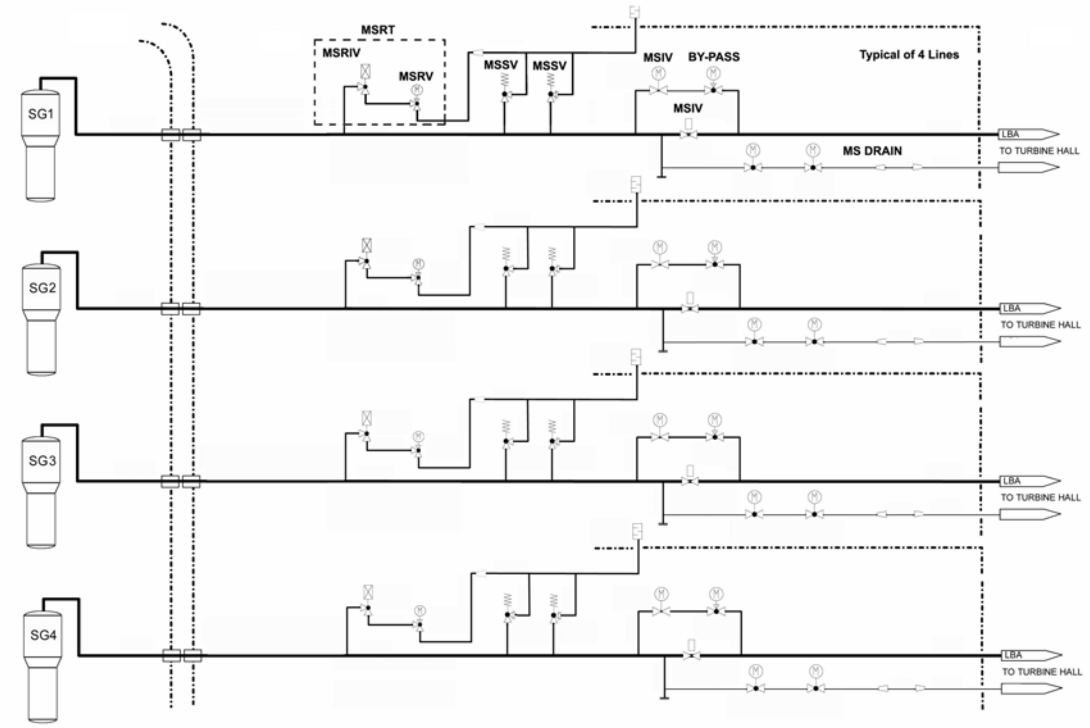
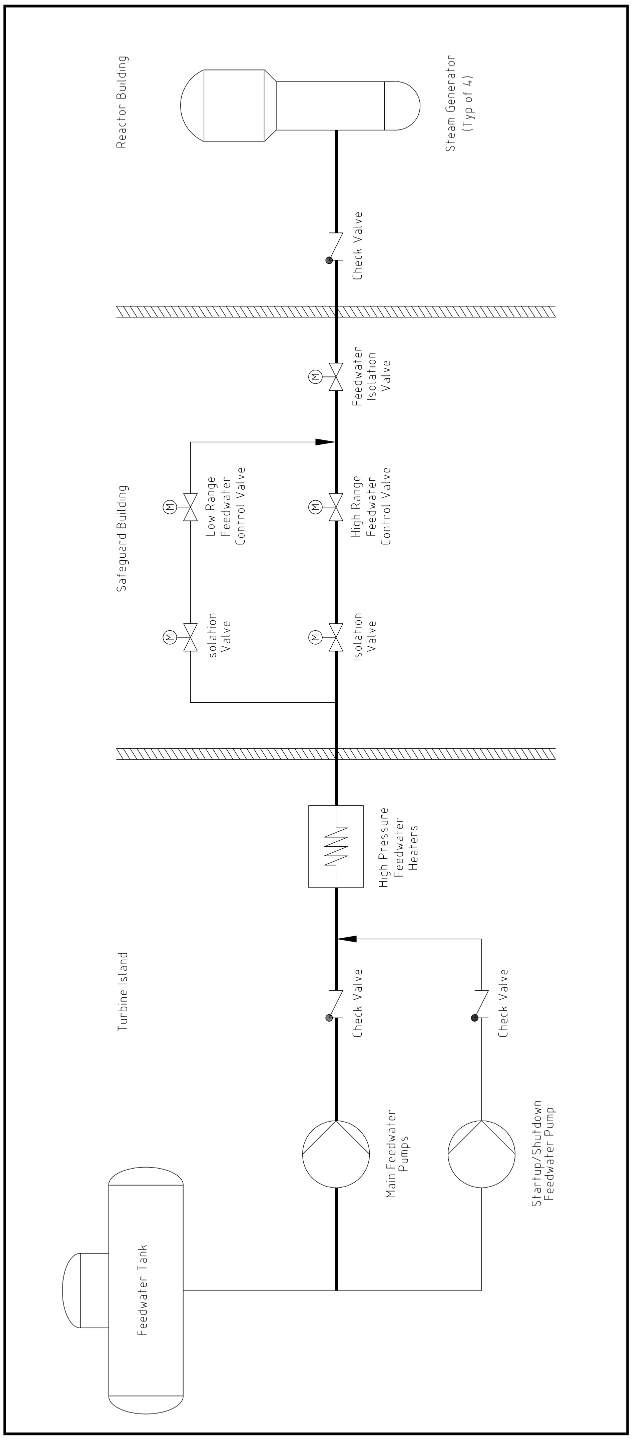


FIGURE 4- 9: MAIN FEEDWATER SYSTEM AND START UP/SHUTDOWN SYSTEM



5.0 REACTOR BUILDING AND SUPPORTING SYSTEMS

5.1 Reactor building structure

The Reactor Building consists of a cylindrical reinforced concrete outer Shield Building, a cylindrical post-tensioned concrete inner Containment Building with a 6 mm thick steel liner, and an annular space between the two buildings. The Shield Building functions to protect the Containment Building from external hazards. The Containment Building contains the RCS and portions of associated structures, systems, and components. In the event of a LOCA or severe accident, the Containment Building serves to retain all radioactive material and to withstand the maximum pressure and temperature resulting from the release of stored energy.

Figure 5-1 through Figure 5-3 show 3D and cutaway views of the Shield and Containment Buildings.

Figure 5-4 through Figure 5-9 show elevation and plan views of the Nuclear Island. The Reactor Building is located at the center of the Nuclear Island and is situated on a common basemat with the other Nuclear Island structures.

The Reactor Building is designed to withstand internal accidents as well as external hazards including the following: aircraft hazard, EPW, seismic events, missiles, tornado, and fire. The EPR design bases are expected to envelope most of the potential sites.

The Design Basis Earthquake (DBE) design includes consideration of various soil conditions, including rock sites. Figure 5-10 shows the free-field ground motions anchored to a 0.25g peak ground acceleration.

The Shield Building provides a hardened structure designed to withstand an aircraft hazard. The Containment Building interior structures and equipment are decoupled from the impact forces of an aircraft hazard. Figure 5-12 shows the decoupling of the containment interior structures from the outer walls.

The common basemat of the Nuclear Island (the Reactor Building, Safeguard Buildings, and Fuel Building) ensures that overturning due to a seismic event or aircraft hazard will not occur.

The Shield Building is designed to withstand the potential effects of an external explosion.

The design pressure and temperature of the Containment Building are defined by the following events:

- double-ended rupture of a reactor coolant pipe (LBLOCA),
- main steam line break,
- severe accidents.

Sprays or fan coolers are not required to mitigate short-term containment pressure or temperature responses to design basis accidents or for long-term containment pressure response. The Residual Heat Removal System (RHRS) is sufficient to reduce the pressure to half the peak in less than eight hours after a LOCA.
The design pressure and temperature of the containment are 0.55 MPa and 170°C, respectively. The maximum pressure and temperature for a LBLOCA or a steam line break are below the design values of the containment.

The bottom part of the containment has an allowable flooding volume of approximately 1300 m^3, which is sufficient to protect safety-related components against flooding after a piping failure.

The reactor pool (reactor cavity with storage compartment above the RPV, used during refuelling) and the IRWST are major internal structures.

The Containment Building also contains the reactor vessel, PZR, SGs, steam generator blow down flash tank, and a portion of the main steam and feedwater lines.

Containment Building

The key dimensions of the Containment Building are:

- free volume (approx.) : 80 000 $m^{3,}$
- internal diameter : 46.80 m,
- thickness of Inner Wall : 1.30 m,
- thickness of Dome : 0.90 m,
- IRWST Water Volume (Normal Operation) : 1940 m^3.

Shield Building

The approximate dimensions of the Shield Building wall are:

Cylindrical portion up to Safeguard Buildings height:

- inner diameter : 53 m,
- thickness : 1.30 m.

Cylindrical portion above Safeguard Buildings height:

- inner diameter : 53 m,
- thickness : 1.80 m,
- dome Diameter / Thickness : 66 m / 1.80 m.

Basemat

The Reactor Building basemat is reinforced concrete. The basemat thickness is typically 4.5 m. A circumferential pre-stressing gallery of vertical tendons of the inner wall is situated underneath the basemat. The Containment Building steel liner continues into the basemat to prevent release of radioactivity to the ground.

5.2 Containment Isolation

Containment isolation valves minimize the release of radioactive fluids to the environment in the event of an accident with fission product releases.

The containment is designed for an integrated leak rate equal to or less than 0.3% per day of the containment free volume at the containment design pressure and temperature. The containment integrated leak rate will be verified through tests.

The isolation function is assured through penetration and isolation valve arrangements. Mechanical and electrical penetrations are designed to withstand the consequences of external hazards and an accident in the containment.

The type, number, and arrangement of isolation valves are in accordance with French and German requirements for existing PWRs. These requirements consider isolation valves on piping for systems in normal operation and in accidents. This includes systems that are part of the RCPB, connected directly to the containment atmosphere, or form a closed loop inside of containment.

The single failure criterion is satisfied for piping directly connected to the RCS or to the inner containment atmosphere along with the installation of a minimum of two valves for each line. The two valves operate independently of each other, with one installed inside the containment and one outside.

The exception to this principle involves the lines from the IRWST sumps to the SIS and CHRS pumps, which have only one isolation valve. In this case, the piping between the sump and the valve is contained in a sealed envelope (guard pipe), thus providing a double leak-tight penetration barrier. This double barrier is designed to be leak-tight and withstand the design basis environmental conditions inside the containment (from the containment penetration to the containment isolation valve) and takes into account a single failure (functional failure or passive failure).

5.3 Provisions for Severe Accident Mitigation

The goal of the severe accident mitigation concept of the EPR is to ensure the function of the containment in the event of an accident resulting in a significant structural degradation of the reactor core. To meet this design goal, specific design features have been incorporated for the retention and stabilization of the molten core inside the containment as well as for the mitigation of environmental effects that can compromise its fission product retention capability.

The dedicated features to address severe accident challenges incorporated in the EPR design include:

- dedicated valves for rapid depressurization of the RCS,
- multiple Passive Autocatalytic Hydrogen Recombiners (PARs) to minimize the risk of hydrogen detonation,
- a containment designed to promote atmospheric mixing with the ability to withstand the loads produced by hydrogen deflagration,
- a dedicated compartment to spread and cool molten core debris for long-term stabilization,
- a CHRS with 2 trains allowing one train to be serviced or repaired in the long term if necessary,

- electrical and I&C systems dedicated and qualified to support severe accident mitigation features,
- the Reactor Building consisting of an inner Containment Building and an outer Shield Building with a sub-atmospheric annulus.

These features ensure that the EPR has the ability to mitigate a broad spectrum of severe accident challenges and is consistent with advanced light water reactor expectations regarding severe accidents.

Core Melt Retention

The EPR is equipped with a dedicated Core Melt Retention System for molten core debris up to and including the total inventory of the core, internals, and lower RPV head. The functional principle of the Core Melt Retention System is to spread the molten core debris over a large area and stabilize it by quenching it with water. Spreading increases the surface-to-volume ratio of the melt to promote fast and effective cooling and limit further release of radionuclides into the containment atmosphere. The main components of the Core Melt Retention System are shown in Figure 5-13. These features ensure a passive transformation of the molten core into a cooled, solid configuration without operator action.

Picture of Figure 5-19 shows a view of the Core Melt Retention System under erection on Olkiluoto 3.

After release from the RPV, a period of melt retention in the reactor cavity will occur. The need for this temporary retention addresses the prediction that the release of molten material from the RPV will, most likely, not take place in a single release, but over a period of time. Without a retention period, this release could create undefined and potentially unfavourable conditions for subsequent melt spreading.

Accumulation and temporary retention within the reactor cavity is ensured by a layer of sacrificial material that must be penetrated to escape into the transfer channel. This delay ensures that, in case of an incomplete first release of melt from the RPV, practically the entire core inventory will be collected in the cavity. The admixture of the sacrificial material equalizes the spectrum of possible melt states prior to spreading and makes the melt properties (and, therefore, subsequent stabilization measures) independent of the uncertainties related to the initial release of melt from the RPV.

The sacrificial concrete in the reactor cavity is backed by a refractory layer of sintered Zirconium blocks (Figure 5-13). Consequently, the molten core-concrete interaction will proceed in a quasi one-dimensional manner. The progression front is subsequently limited by the fixed position of a cavity retention gate (Figure 5-13), thus, the total mass of concrete that has to be incorporated into the melt during temporary retention is well-defined. This temporary retention also ensures that the final temperature of the oxidic melt prior to spreading is reduced while the admixture of concrete ensures that the viscosity of the melt is maintained in a favourably low range.

The relocation of the melt from the reactor cavity into the spreading area is initiated by the failure of a retention gate centred in the lower portion of the cavity. This gate, which isolates the reactor cavity from the spreading compartment, consists of a steel framework enclosed by an aluminium outer layer and covered with a layer of sacrificial concrete.

The gate's concrete cover is an integral part of the sacrificial layer in the cavity and has approximately the same thickness. The retention time in the pit is primarily driven by the thickness of this concrete cover and not by the delay-to-failure time of the gate after melt contact. The gate is the only location in the reactor cavity where the sacrificial concrete is not backed by a protective layer. Therefore, the melt plug represents the predefined failure location for melt retention in the cavity. Following the failure of the cavity retention gate, the melt will progress through the transfer channel in a single pour and into the spreading area. Figure 5-14 shows a schematic of the reactor cavity retention gate.

The spreading compartment is a dead-end room (Figure 5-16) which is virtually isolated from the rest of the containment. It is also protected from sprays, leaks, or other kinds of spillage. Since there is no direct water inflow into this compartment the spreading area will be dry at the time of the melt arrival.

The area within which the molten core debris will ultimately be retained is a shallow crucible. Its bottom and sides are assembled from individual elements of a cast iron cooling structure. The cooling structure is covered with a layer of sacrificial concrete and provides protection against thermal loads resulting from melt spreading as well as a sufficient delay to ensure that the cooling elements will be flooded prior to the initial contact between the molten core debris and the metallic cooling structure.

The melt stabilization process in the spreading area is passively actuated. When the melt enters into the spreading area, spring-loaded valves will be opened by a thermal actuator, initiating a controlled gravity-driven flow of water from the IRWST.

The configuration of the IRWST relative to the spreading area is represented in Figure 5-15. The incoming water will fill a central supply duct underneath the spreading area where it will enter the system of parallel channels formed by finned cooling structure elements (Figure 5-17).

The water will continue to rise along the sidewall of the cooling structure and pour onto the surface of the melt from the circumference (Figure 5-18). Water overflow will continue until the spreading room and IRWST are balanced, resulting in the submersion of the spreading area and transfer channel as well as a portion of the reactor cavity, thereby stabilizing any residual core debris in those areas.

The stabilization of the melt in the spreading area is based on cooling and crust formation. Consequently, there are no limiting thermal-chemical constraints and no need to ensure a certain range of melt compositions or a predefined melt layering or distribution. Due to the high surface-to-volume ratio created by the spreading process and the fact that the melt is completely surrounded by cooled surfaces, a safe enclosure of the molten core debris within a crust envelope will be achieved soon after the end of the molten core-concrete interaction in the spreading area. The denser metallic melt fraction at the bottom is predicted to solidify within the first few hours. Solidification of the decay-heated oxidic melt will take longer, due to internal heat generation from radioactive decay.

Combustible Gas Control

The Combustible Gas Control System (CGCS) is designed to reduce the concentration of hydrogen produced during severe accidents and can also reduce the concentration of hydrogen within the containment following a LOCA. The CGCS (in standby mode during normal operation) becomes operational when subjected to a hydrogen/steam environment.

The CGCS is a completely passive system with three components as described below:

- passive Autocatalytic Recombiners (PARs) are arranged throughout the containment to reduce the concentration of hydrogen and promote convection. The arrangement of the PARs supports global convection, homogenizes the atmosphere, and reduces the global hydrogen concentration as well as peak hydrogen concentrations,

- rupture foils are installed in the structural steelwork forming the ceiling above the SGs and open passively on pressure differential to promote global convection within the containment,

- hydrogen mixing dampers are arranged in the lower annular rooms and above each of the four SGs. The mixing dampers are stainless steel louvers that open passively to promote a global convection loop within the containment.

The CGCS contributes to the containment integrity by ensuring that hydrogen concentrations remain low enough to prevent excessive loads on the containment and internal structures. In order to meet this objective, the CGCS was designed to satisfy the following requirements:

- the local hydrogen concentration is maintained below 10% based on the free volume of the containment. Any region having a concentration of hydrogen above 10% by volume shall be small enough to prevent flame acceleration,

- the maximum amount of hydrogen, resulting from the oxidation of all Zirconium in the core, is reduced to below 4% within 12 hours,

- the adiabatic isochoric complete combustion pressure is kept below the containment design pressure for all scenarios that involve hydrogen combustion.

The PARs and mixing dampers are designed to remain operational following an earthquake and under adverse environmental conditions such as the irradiation, temperature, pressure, and humidity conditions resulting from a severe accident. The PARs are also able to withstand the various products released during a severe accident including aerosols, iodine, and spray, as well as local heat-up.

The CGCS interfaces with the atmosphere of the containment, but has no direct interface with other systems. Operational interfaces exist between the CGCS and the RCS depressurization system, the containment spray of the CHRS, and the hydrogen monitoring system.

Severe Accident Heat Removal

The CHRS is used in the event of a severe accident, to control the containment pressure and achieve long-term cooling of the IRWST and the molten corium in the spreading compartment. The CHRS may also be employed to transfer residual heat to the ultimate heat sink during a beyond design basis event involving failure of all other RHR capability without core melt.

The CHRS provides the following functions:

- provides containment isolation in the event of an accident that does not require CHRS actuation,

- provides a containment spray function to rapidly control containment pressure and temperature following passive melt stabilization,
- provides long-term containment pressure and temperature control through operation in the recirculation mode,
- transfers residual heat from the containment atmosphere to the IRWST during a severe accident in order to control the containment pressure and temperature,
- removes fission products from the containment atmosphere during a severe accident,
- transfers residual heat from the spread melt to the IRWST during a severe accident,
- transfers residual heat from the IRWST to the ultimate heat sink via an intermediate, dedicated cooling system, during a severe accident or during a beyond design basis event without core melt in which all other RHR capability has failed,
- backflushes sump screens in the IRWST to remove potential accumulated debris from the sump screens of the CHRS pump suction nozzle.

The CHRS consists of two separate and identical trains. Each train consists of a dedicated suction line from the IRWST and a pump and heat exchanger located in a dedicated room in Safeguard Buildings 1 and 4. The secondary side of the CHRS heat exchanger is cooled by the CCWS.

Figure 5-20 shows a schematic of the CHRS.

There are three possible flow paths downstream of the pump and heat exchanger:

- to a dome spraying system consisting of a ring header with spray nozzles located in the dome of the containment to reduce containment pressure, temperature, and airborne fission products,
- to a basemat cooling device with an overflow to the spreading compartment to remove the decay heat from the spread melt. The decay heat is transferred to the containment atmosphere by steam generation. The steam is then condensed into the IRWST, completing the heat transfer from the spread melt to the IRWST. The flow limiter on the passive flooding path limits direct backflow into the IRWST to assure flooding of the transfer channel and reactor cavity during the CHRS operation,
- to a sump screen flushing device which is used to detach accumulated debris from the sump screens of the CHRS pump suction nozzle.

The CHRS is designed for both short-term and long-term operation phases. The volume, design and thermal capacity of the containment structures and the IRWST provide approximately 12 hours, after the beginning of the severe accident, for operator action. During this time, no active system is needed to maintain the containment pressure below its design value.

During the short-term operation phase, the containment pressure and temperature are controlled by spraying via the CHRS spray line. Fission products are scrubbed from the containment atmosphere during this timeframe.

When the containment pressure and temperature are sufficiently reduced, one or both trains of the CHRS may be operated to directly flood the spread melt to more efficiently remove decay heat and control containment temperature and pressure. The CHRS operation mode may be changed as necessary to provide further containment atmospheric heat and fission product removal in the long-term phase.

To minimize the potential of a radioactivity release caused by leakage of recirculating highly contaminated water outside the containment, appropriate design provisions are provided. These include a specific leak-tight compartment for the system components and adequate shielding or purge connections and filters to allow repairs to be made during the long-term operation phase.

FIGURE 5- 1: SHIELD BUILDING AND CONTAINMENT BUILDING

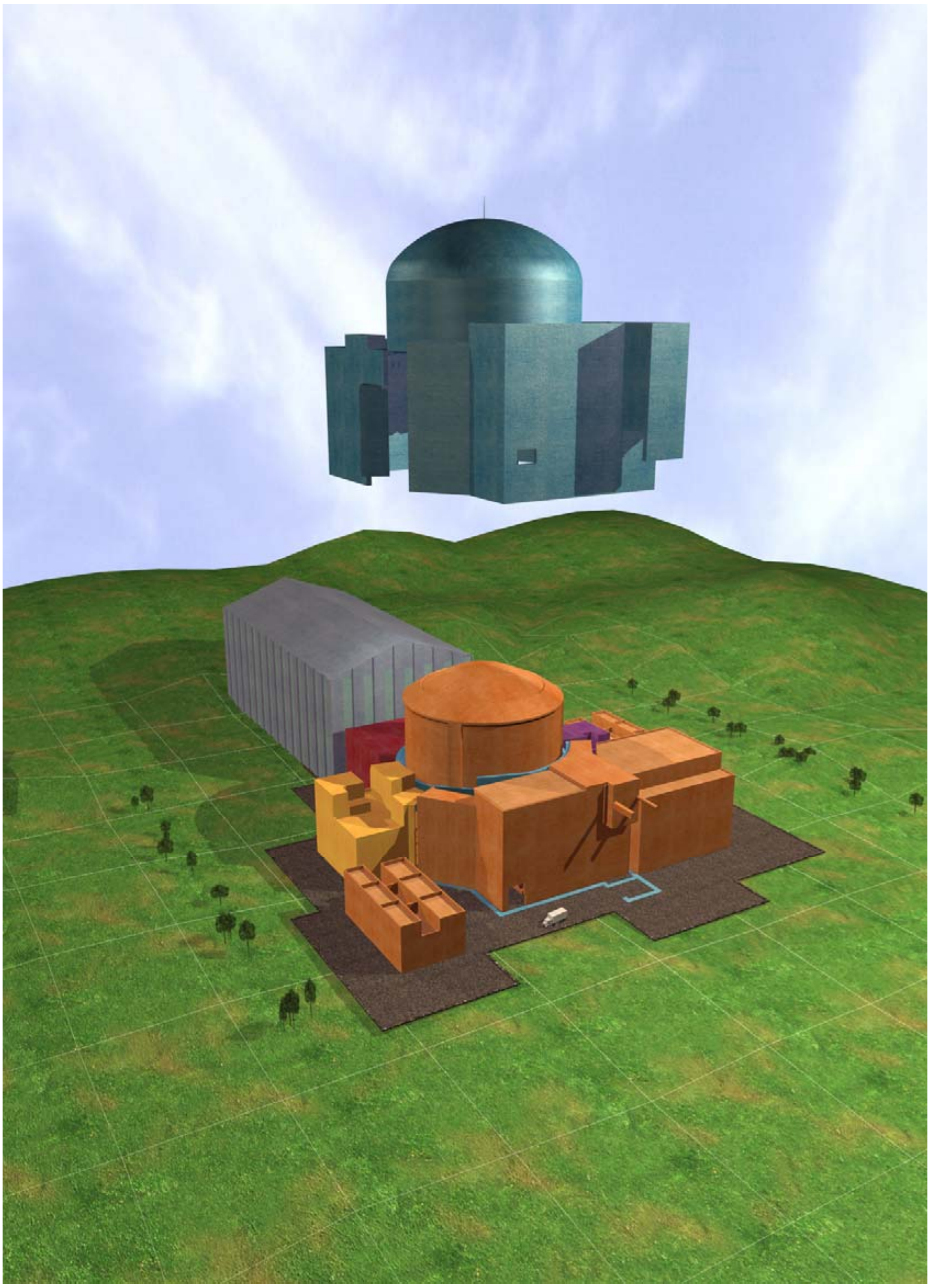


FIGURE 5- 2: SHIELD BUILDING AND CONTAINMENT BUILDING INTERIOR STRUCTURES AND EQUIPMENTS

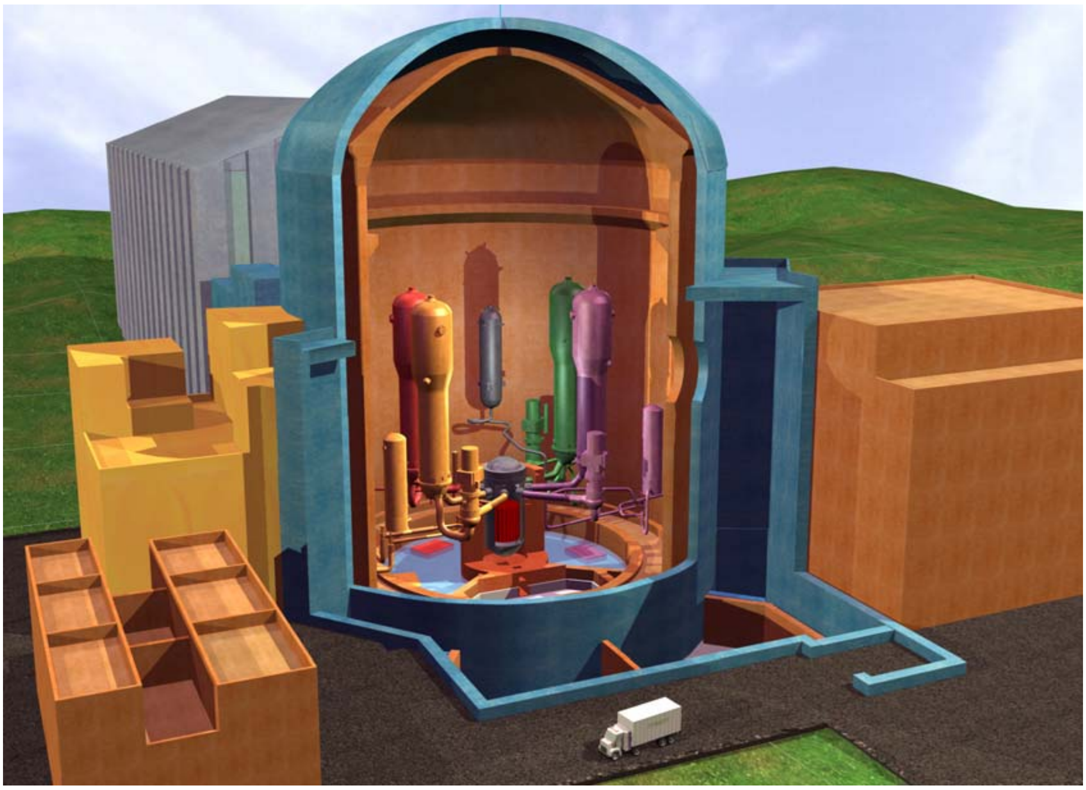


FIGURE 5- 3: CONTAINMENT BUILDING INTERIOR STRUCTURES AND EQUIPMENTS

FIGURE 5- 4: REACTOR BUILDING PLAN VIEW -6 M

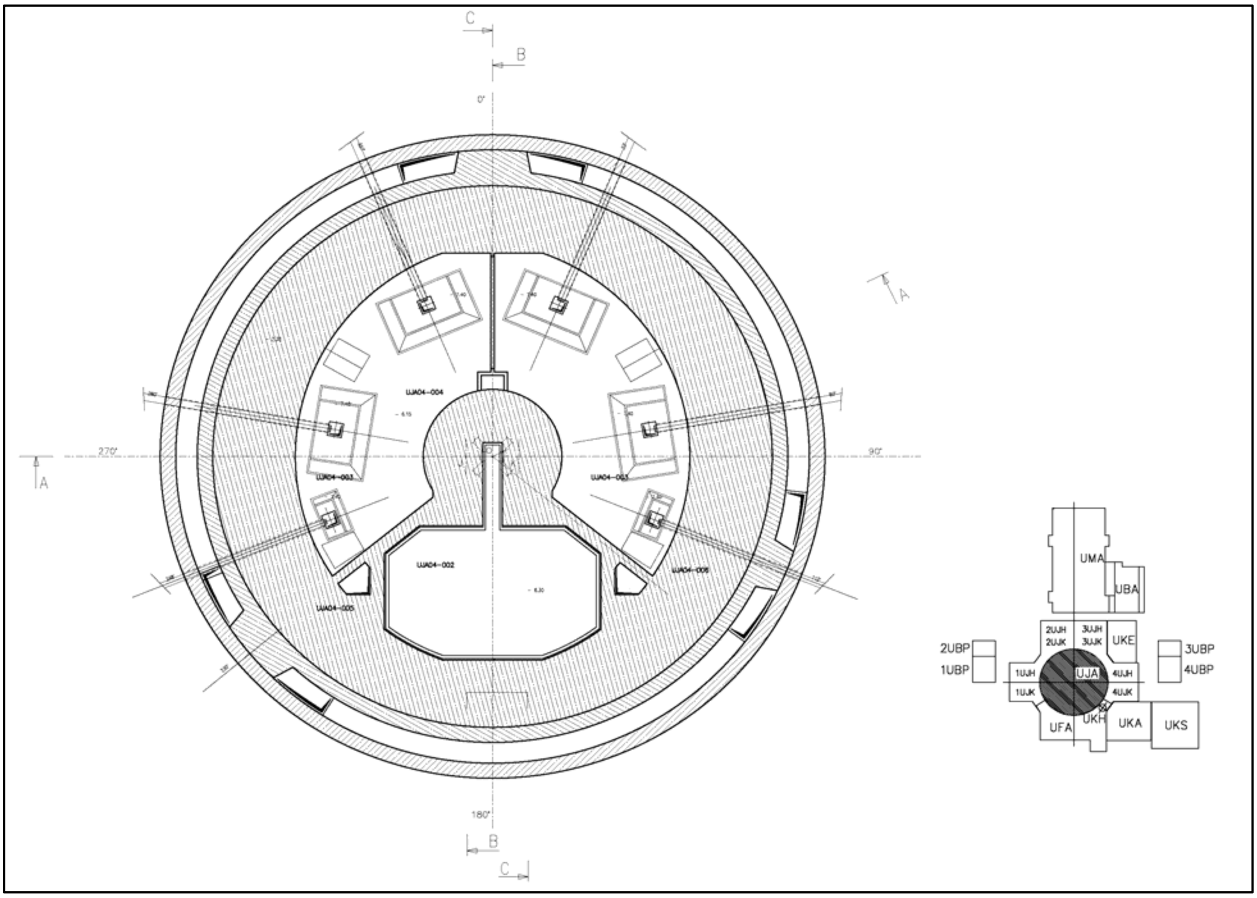


FIGURE 5- 5: REACTOR BUILDING PLAN VIEW +1.5 M

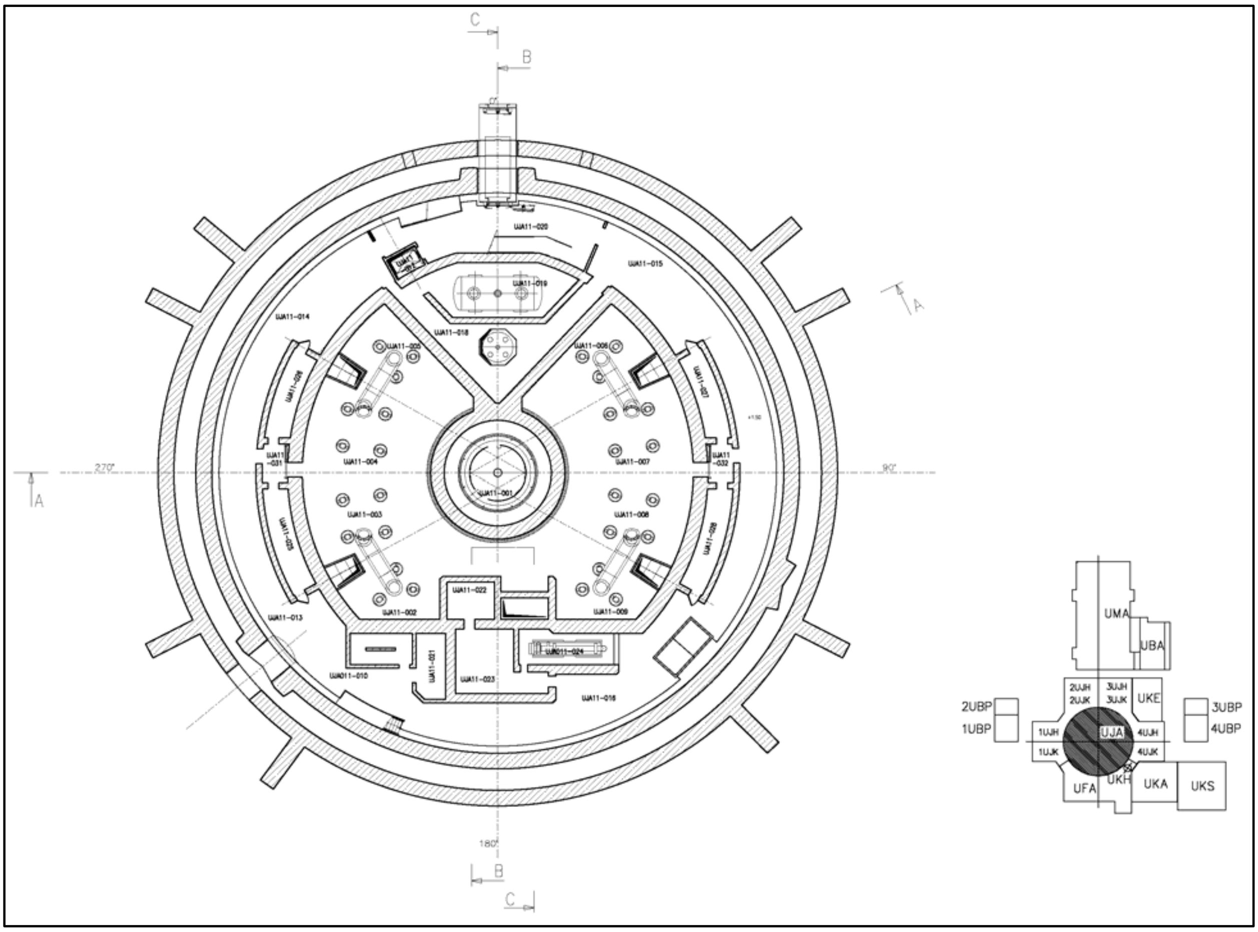


FIGURE 5- 6: REACTOR BUILDING PLAN VIEW +5 M

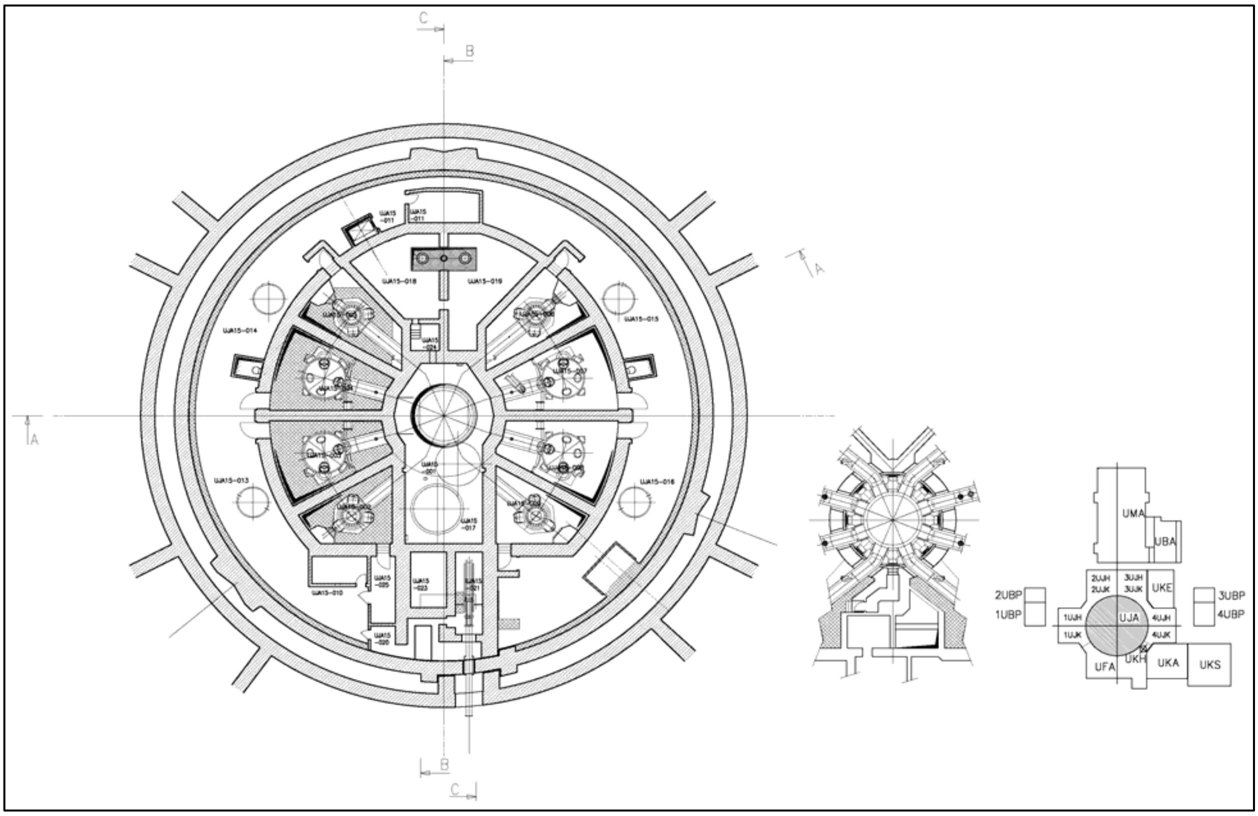


FIGURE 5- 7: REACTOR BUILDING SECTION A-A

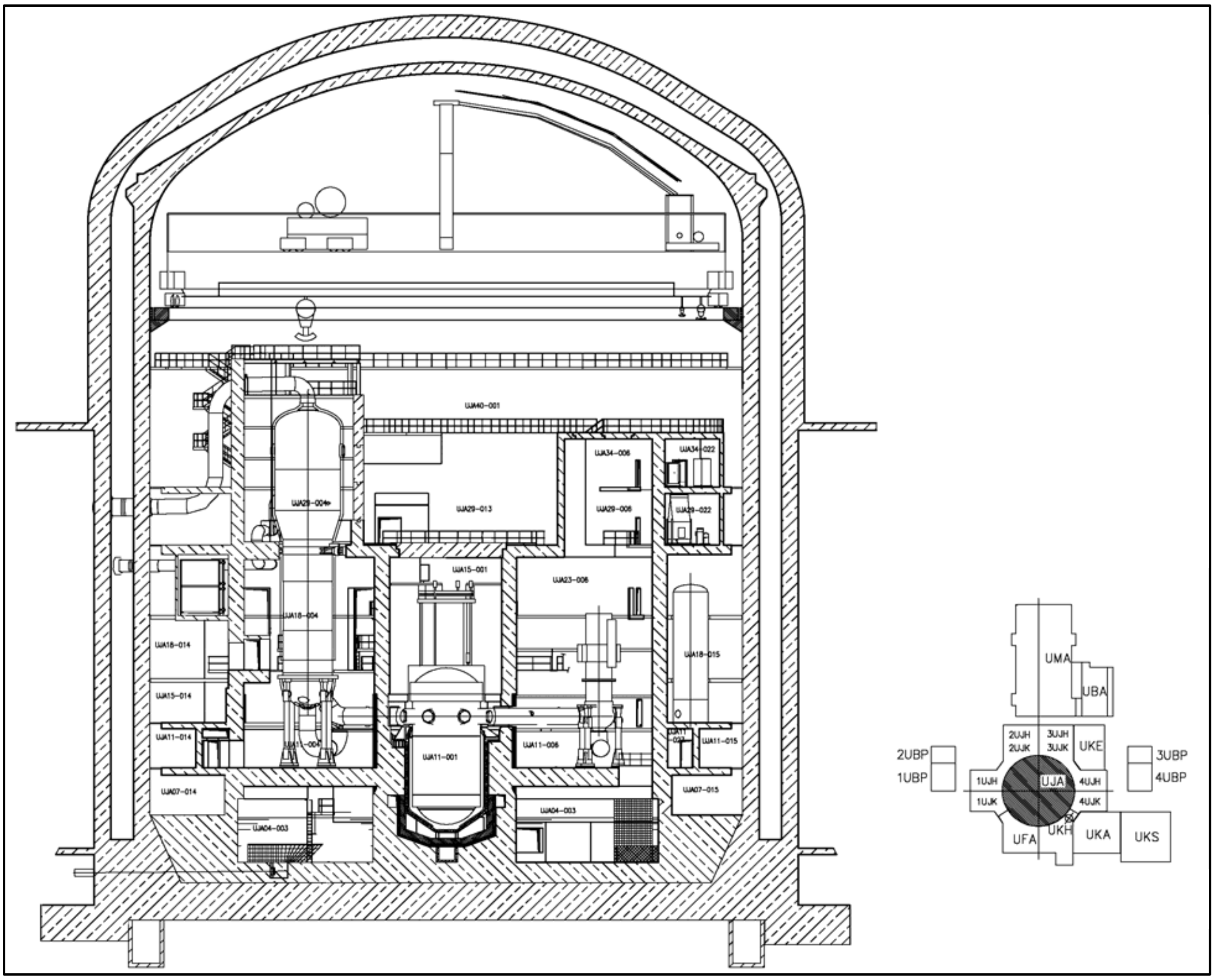


FIGURE 5- 8: REACTOR BUILDING SECTION B-B

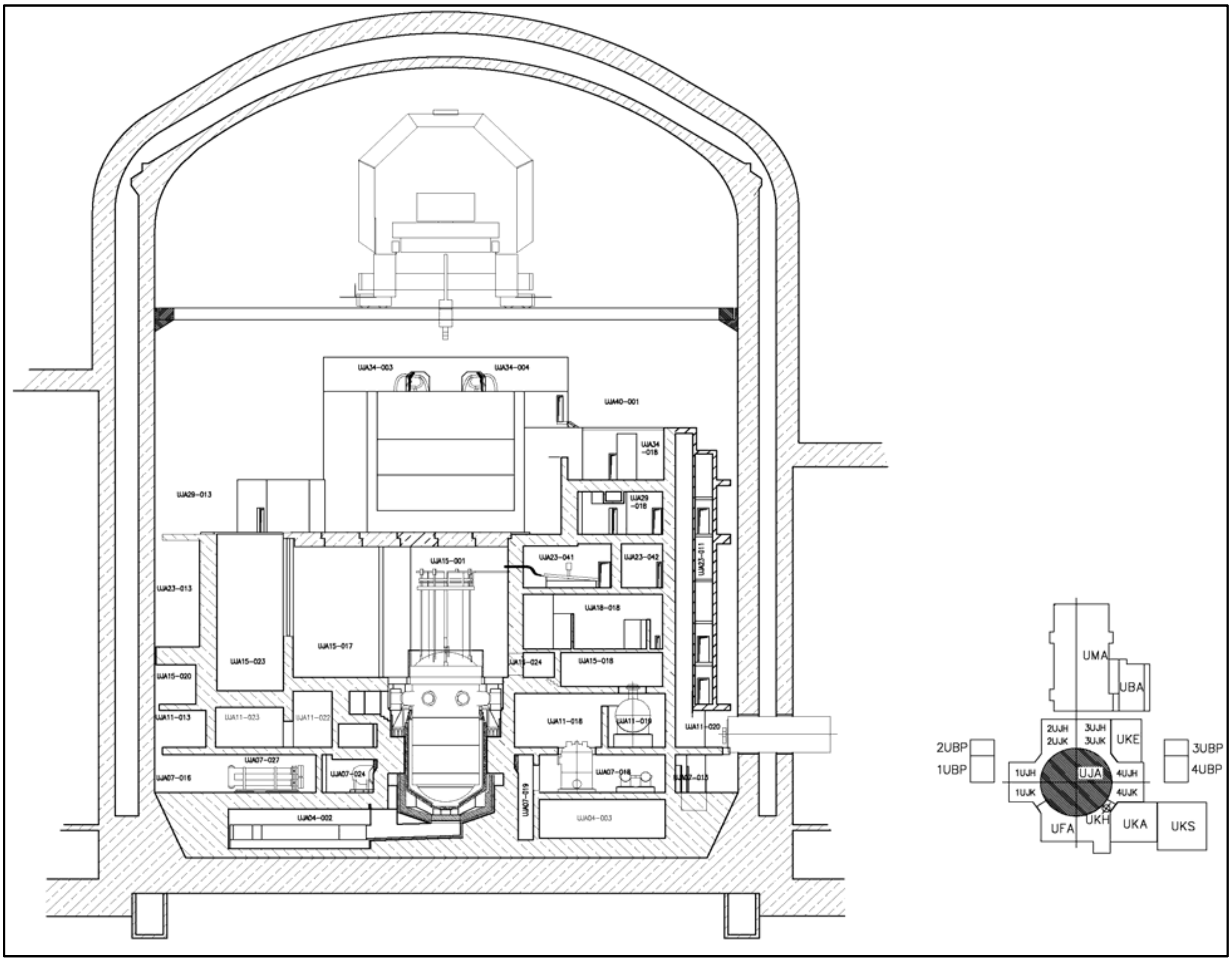


FIGURE 5- 9: REACTOR BUILDING SECTION C-C

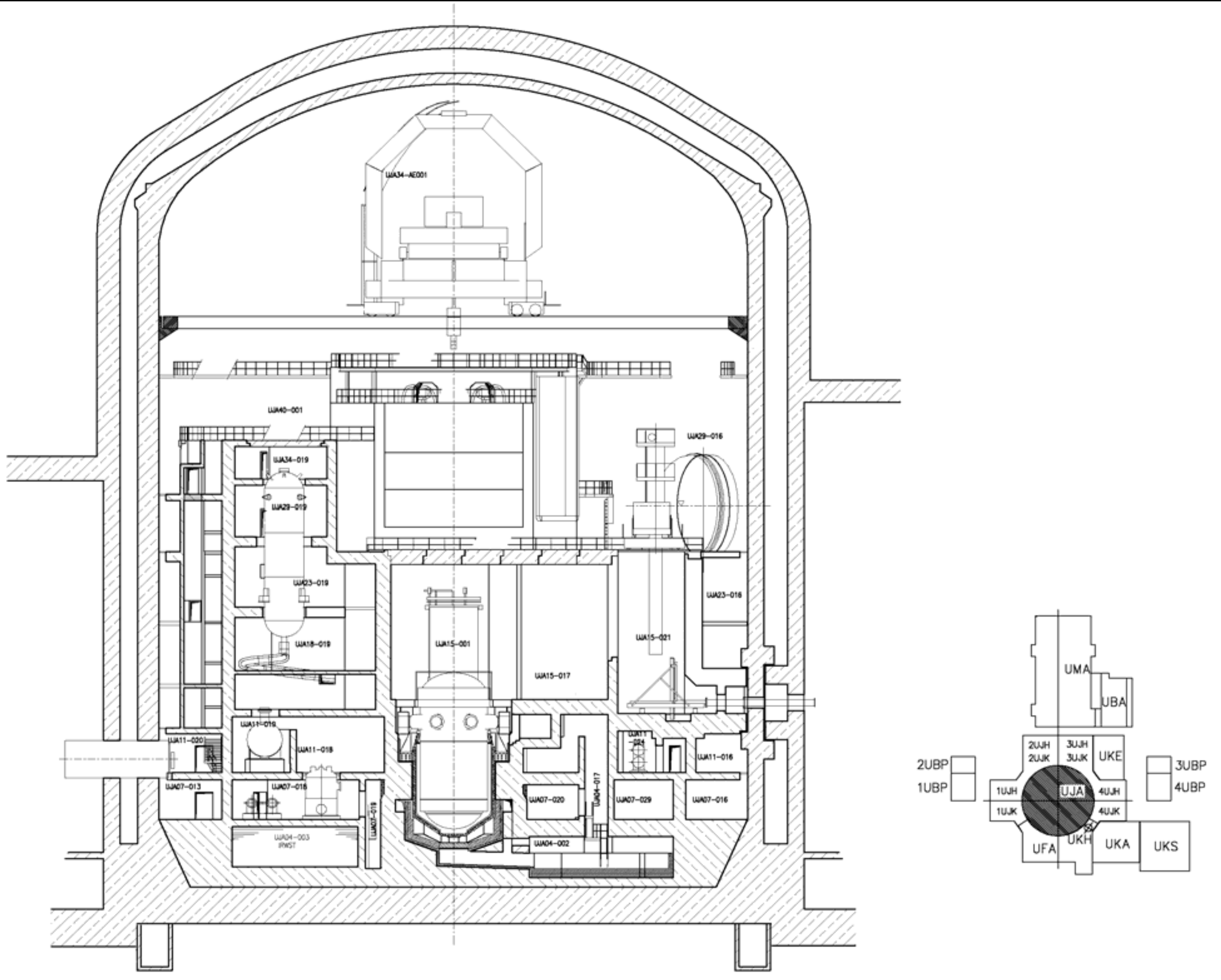


FIGURE 5- 10: FREE-FIELD GROUND MOTIONS

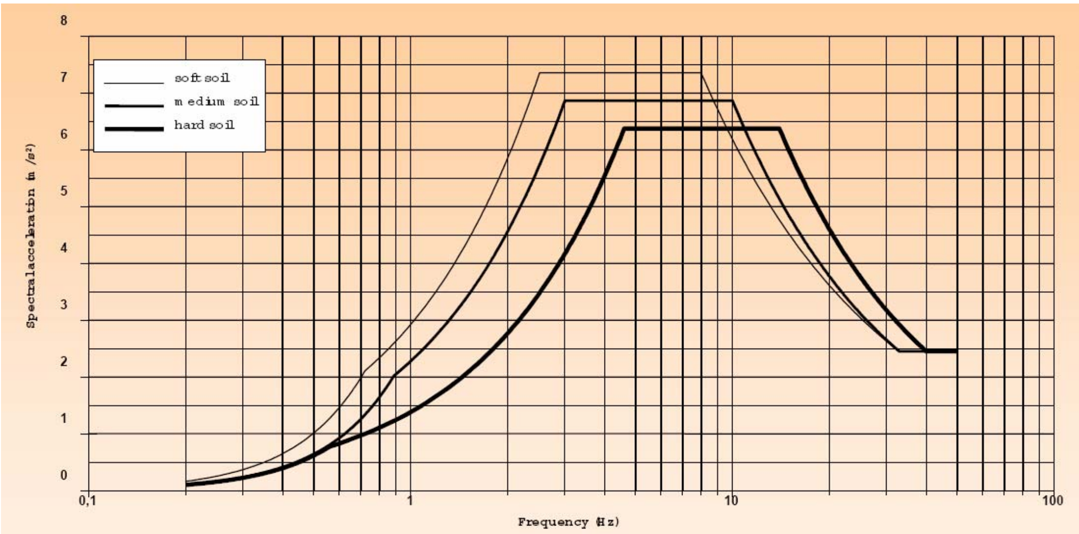


FIGURE 5- 11:

Not used.

FIGURE 5- 12: CONTAINMENT DECOUPLING

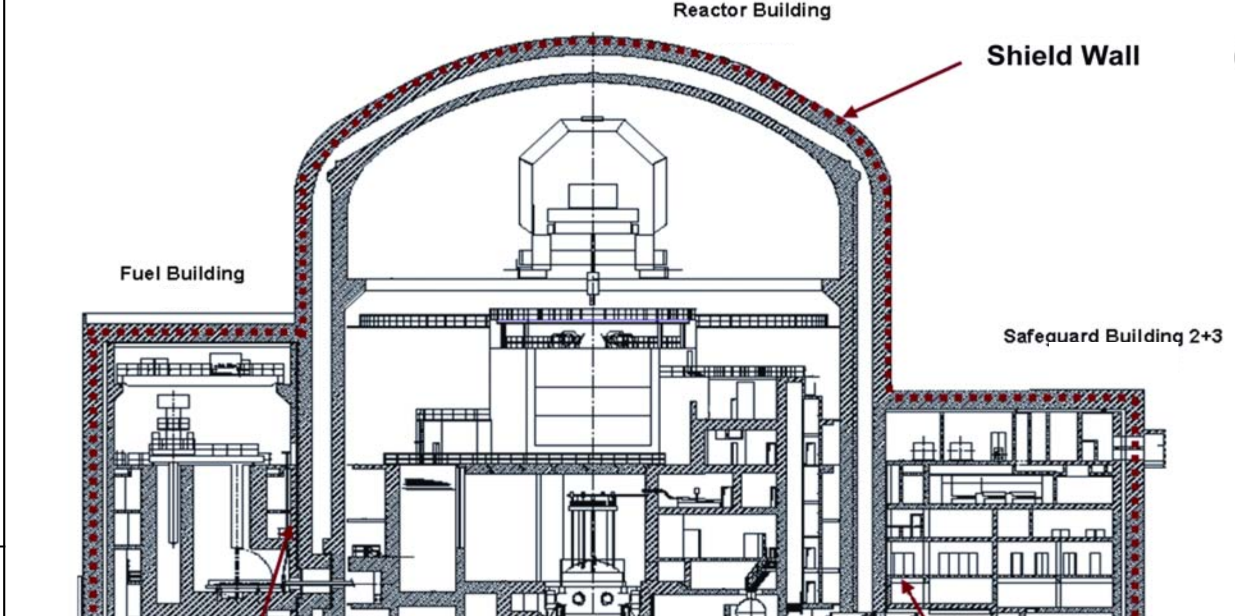


FIGURE 5- 13: CORE MELT RETENTION SYSTEM

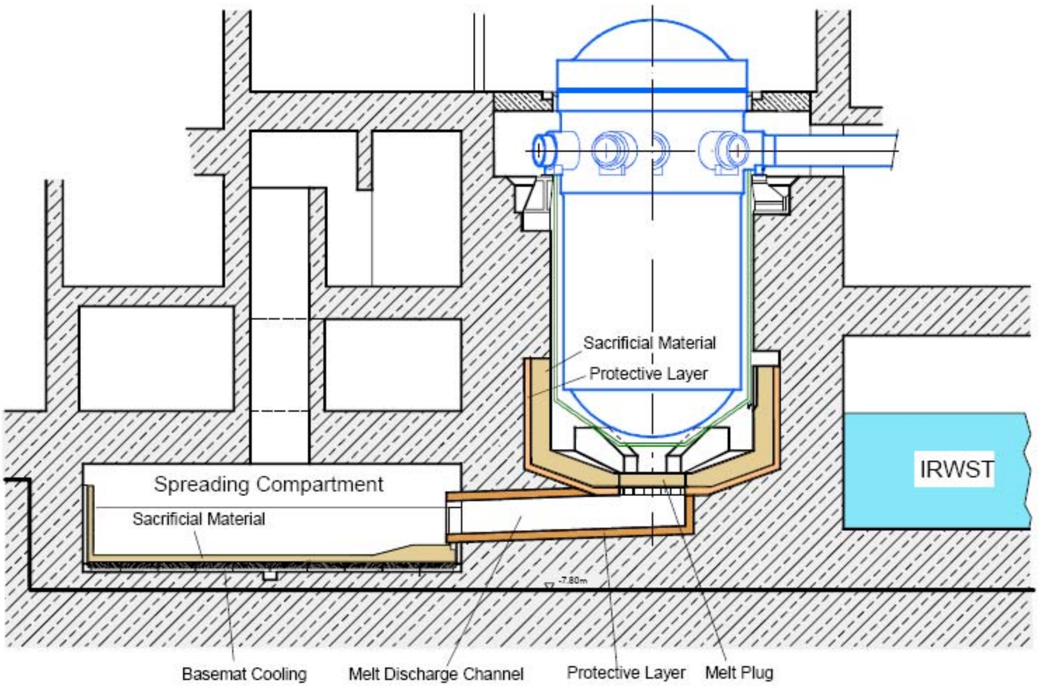


FIGURE 5- 14: REACTOR CAVITY RETENTION GATE

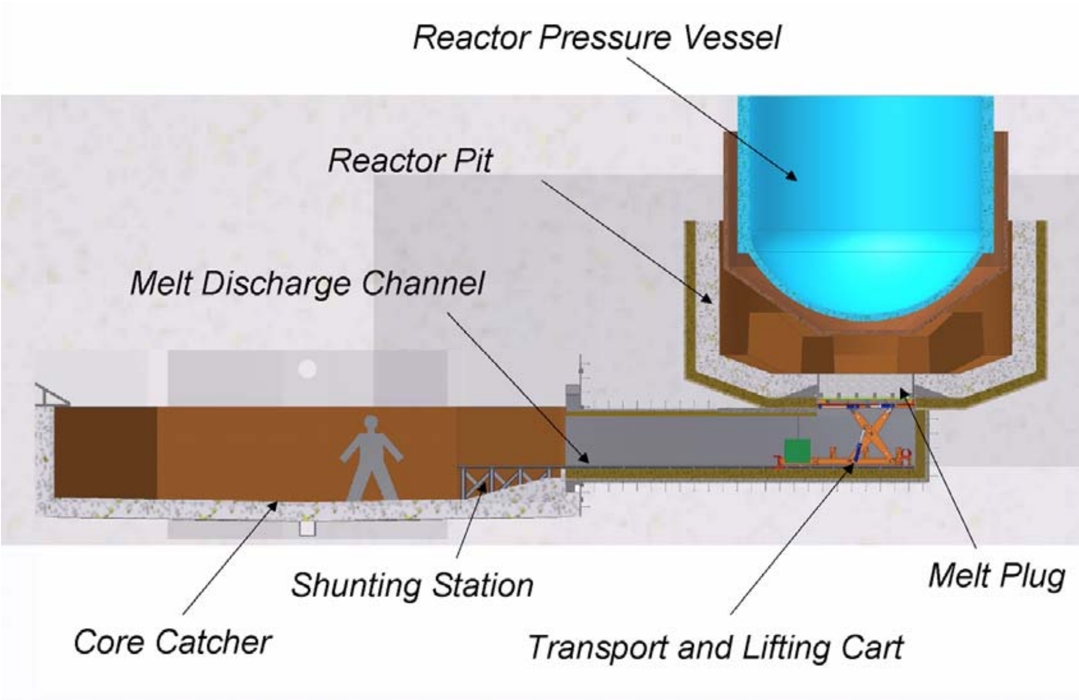


FIGURE 5- 15: IRWST AND CORE MELT SPEADING AREA

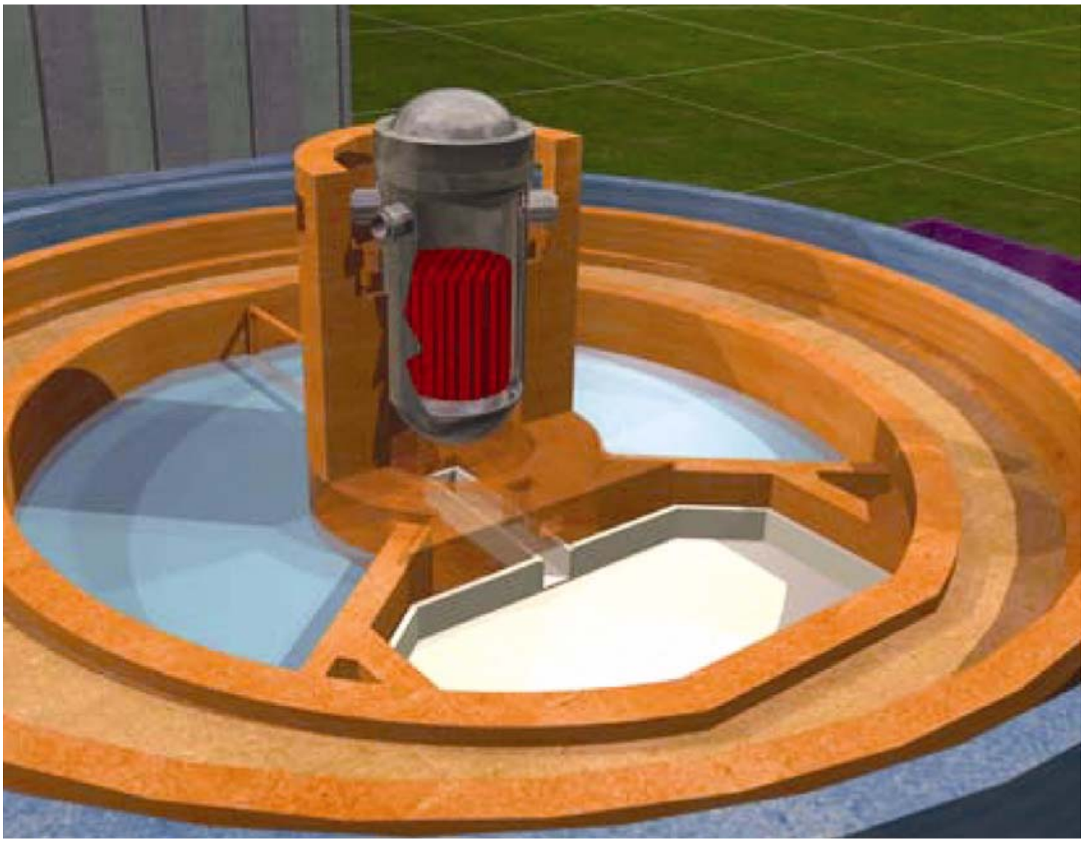


FIGURE 5- 16: CORE MELT SPREADING COMPARTMENT

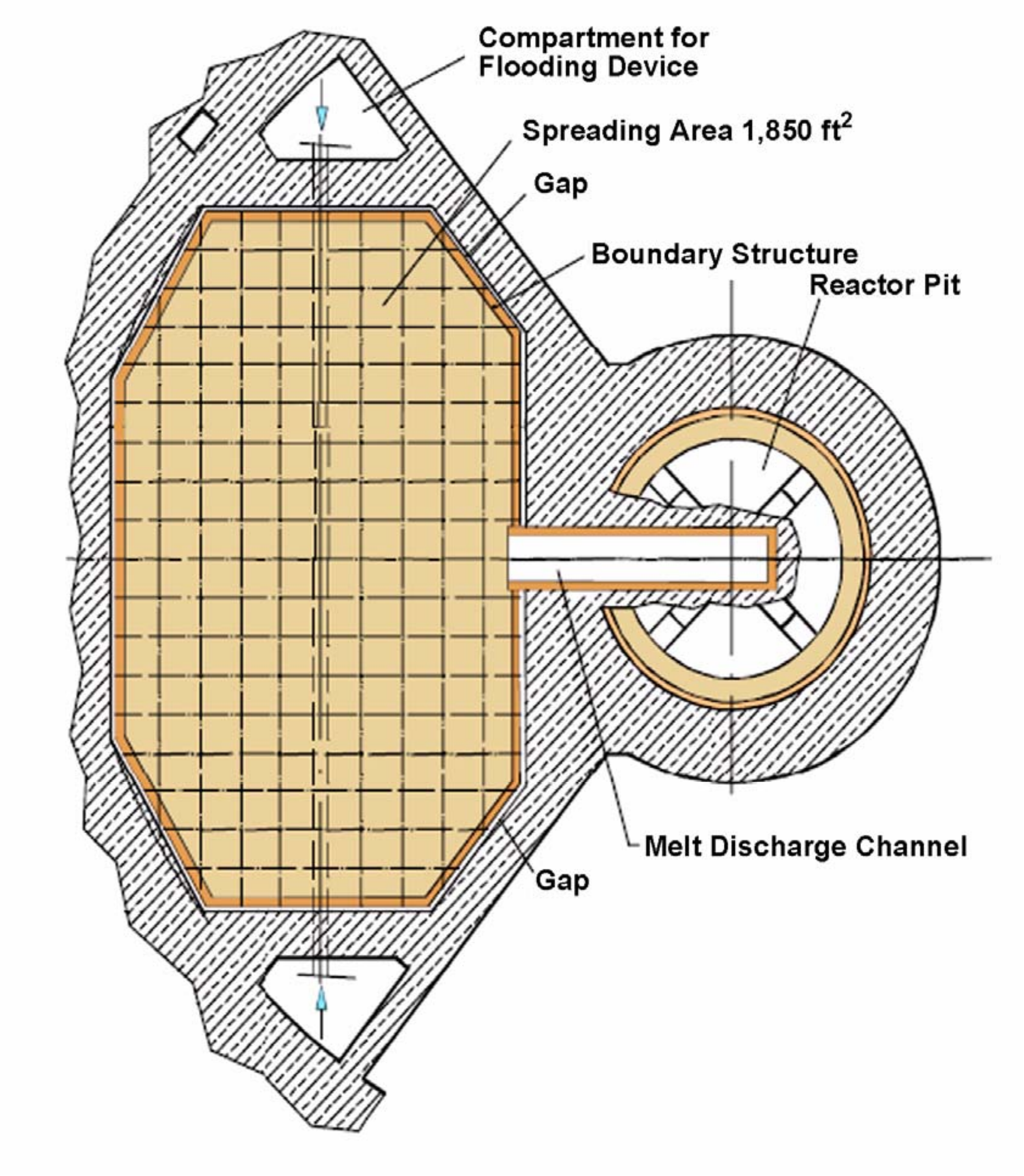


FIGURE 5- 17: SPREADING AREA COOLING SYSTEM

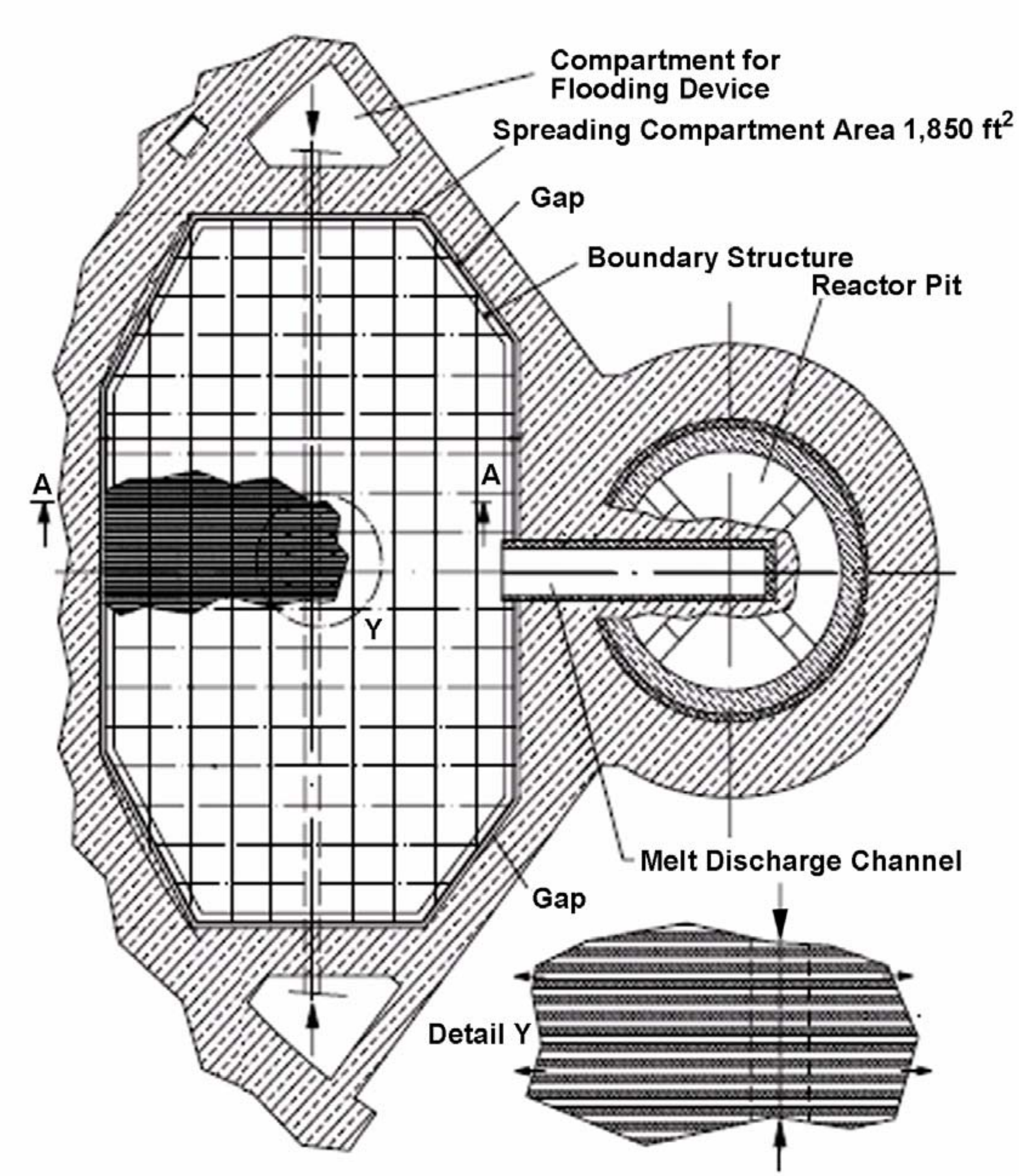


FIGURE 5- 18: DETAIL OF CORE MELT SPREADING COMPARTMENT

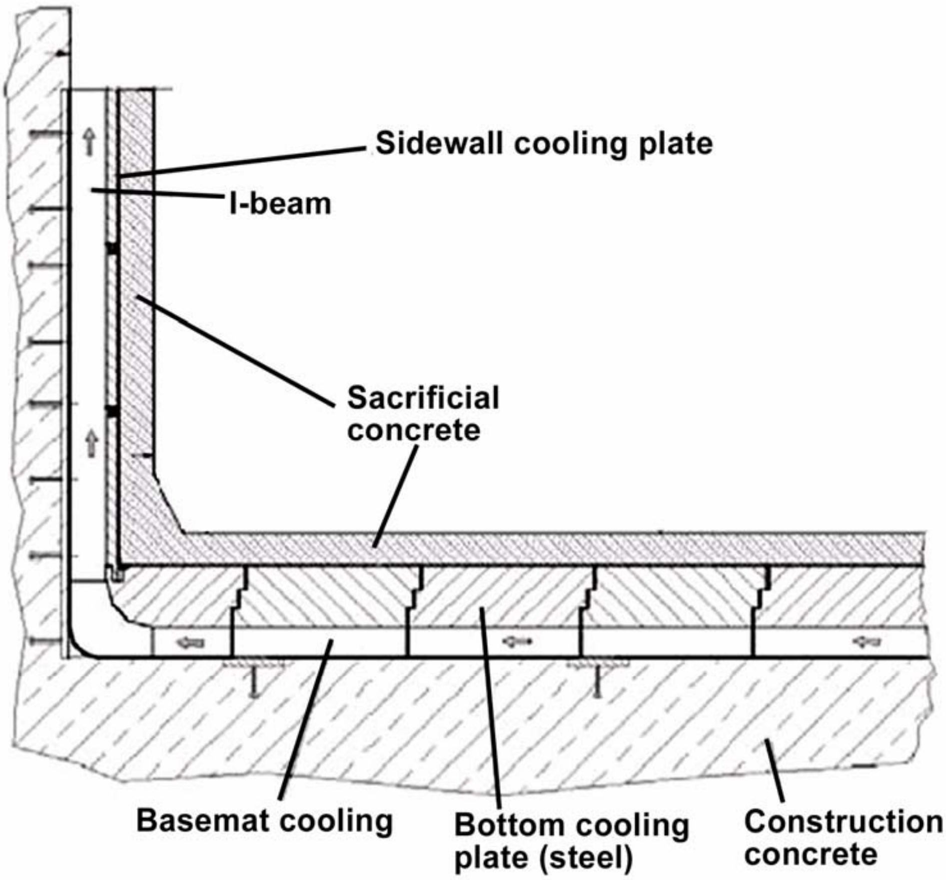
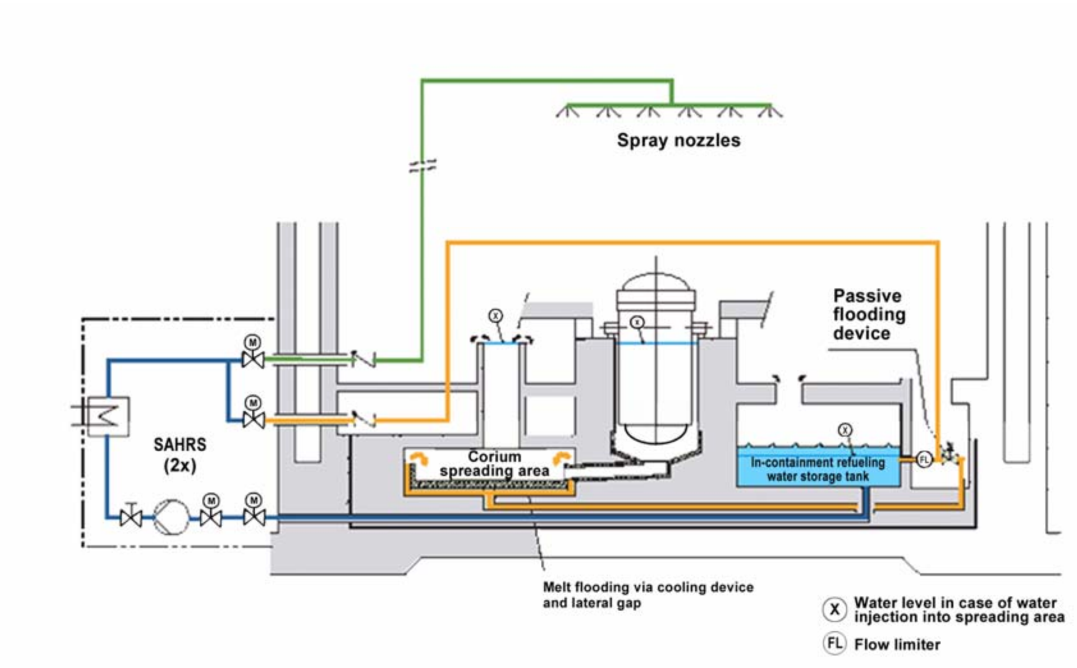


FIGURE 5- 19: CORE MELT SPREADING COMPARTMENT UNDER ERECTION (OLKILUOTO 3)

FIGURE 5- 20: DETAIL OF CORE MELT SPREADING COMPARTMENT



6.0 OTHER SAFETY-RELATED BUILDINGS

6.1 General description

The Nuclear Island consists of the Reactor Building, Safeguard Buildings, and the Fuel Building, all of which are located on a common basemat (foundation raft). The Nuclear Auxiliary Building, two emergency Diesel Buildings, the Radioactive Waste Processing Building, and ESW intake structures are located on individual basemats.

Figure 6-1 and Figure 6-2 show the general layout of the major EPR Buildings.

Of these buildings and structures, those that are safety-related include the Reactor Building, the four Safeguard Buildings, the Fuel Building, the two Diesel Buildings that house the four EDGs and the two SBO generators, the vent stack, the ESW intake structure, the Nuclear Auxiliary Building, and the Radioactive Waste Processing Building. These safety-related buildings provide a physical arrangement that supports a four-train divisional separation for the integral systems and components of the EPR.

Each train of the safety systems is protected against propagation of internal hazards (e.g., fire, high-energy line break, flooding) from one train to any other. This requirement leads to an allocation of each train into a specific area or division that is separated from the other trains. According to the number of trains, the four Safeguard Buildings correspond to the four safety divisions.

Both the structural design and physical arrangement of the buildings provide protection from both external and internal hazards. Additionally, all safety-related buildings are designed to withstand the effects of the SSE.

The EPR is designed to withstand an aircraft hazard and an EPW. The specific design basis for the EPR against the external hazards of an aircraft hazard and an EPW will be determined by the specific characteristics of the site on which the EPR will be located and in accordance with local regulatory requirements.

The safety concept of aircraft hazard for the EPR is a combination of hardened structures and spatial separation.

The Turbine Building is independent from the Nuclear Island. The turbine itself is located in a radial position with respect to the Reactor Building to avoid the impact of a turbine missile on buildings containing safety classified equipment.

The Switchgear Building, which contains the power supply and the I&C for the balance of plant, is located next to the Turbine Building. Both buildings are designed on the basis of technical requirements and safety regulations and do not impact the Nuclear Island.

Radiation Protection

Radiation protection requirements reduce personnel exposure during operation and while performing in-service inspection and maintenance. The design target for the maximum collective dose exposure is less than 0.5 man.Sv/yr.

The mechanical part of the Safeguard Buildings is separated into radiologically controlled (“hot”) and non-controlled (“cold”) areas. Those systems that are radiologically “cold” under normal conditions, such as CCWS and EFWS, are separated (including the dedicated access locations) from the systems that are radiologically “hot,” such as the RHRS. This arrangement minimizes the necessity for personnel to enter contaminated areas.

Primary layout design features providing radiation protection include the following:

- equipment is located in separate compartments (tanks/heat exchangers, pumps and valves) according to access requirements and anticipated radiation levels,
- access to radioactive components is provided via shielded service routes and transition from areas with lower radiation levels to areas with higher radiation levels.

6.2 Safeguard Buildings

There are four Safeguard Buildings. Safeguard Buildings 1 and 4 are spatially separated on opposite sides of the Reactor Building while Safeguard Buildings 2 and 3 are housed together in a hardened enclosure.

Each of the four Safeguard Buildings is separated into two functional areas:

- mechanical Area,
- electrical, I&C and HVAC Area.

Building isolation and filtering in the event of a release of radioactivity is ensured. In the event of an RHRS high energy line break, depressurization devices open to avoid inadmissible pressure build-up inside Safeguard Buildings 1 and 4. These depressurization devices reclose after RHRS isolation, thereby preventing release of radioactivity to the environment. Release of radioactivity to the ground water is also prevented by ensuring that all flood volumes are contained within the building.

The Safeguard Buildings are located close to the Reactor Building and contain the following:

- SIS,
- CCWS,
- EFWS,
- CHRS (only in Safeguard Building 1 and Safeguard Building 4),
- main Control Room and Technical Support Center (TSC),
- equipment for I&C and electrical systems of the Nuclear Island,
- safeguard Building ventilation and safety chilled water systems.

The SIS design basis is a four-train independent system. In order to minimize the connection lengths to the RCS, the individual trains are radially assigned to the RCS loops. Net Positive Suction Head (NPSH) requirements for the safety injection pumps are satisfied by locating the pumps on the lowest level of the Safeguard Buildings.

The CCWS supplies the SIS/RHRS heat exchangers with cooling water. The CCWS is installed close to the connecting SIS/RHRS, but in a different radiation zone, as the activity level of both systems is different. The CCWS is located in a second outer row around the Reactor Building, in the radiologically non-controlled area of the Safeguard Buildings.

The EFWS is also located in the mechanical, radiologically non-controlled area of the Safeguard Buildings.

Mechanical Area

Each of the divisions in the mechanical area includes a LHSI and a MHSI SIS. The LHSI combines the LHSI functions and the RHRS functions. These systems are arranged at the inner areas in the radiological controlled area, whereas the corresponding CCWS and the EFWS are installed at the outer areas in the radiological non-controlled areas on several levels.

In addition to the SIS in Safeguard Building 1 and Safeguard Building 4, the CHRS is installed in the radiological controlled area to protect operating and maintenance personnel from high radiation from the CHRS components following a severe accident.

Internal flood protection is provided on a divisional basis by ensuring that there are no connections between divisions below the flood level.

Electrical, I&C, and HVAC

The I&C and electrical equipment related to safety tasks, as well as those related to the operational functions of the Nuclear Island, are located within the Safeguard Buildings. The safety-related electrical systems, the I&C, the Main Control Room, and the divisional HVAC systems are arranged in the upper building levels. These areas are all classified as radiologically non-controlled. The Control Room complex is placed above them. At this level, the Main Control Room is installed in Safeguard Building 2 and the TSC in Safeguard Building 3.

All I&C and electrical equipment related to the functional requirements for the balance of plant are located in the Switchgear Buildings.

Heating, ventilation, and air conditioning for each electrical division (Safeguard Buildings 1, 2, 3, and 4) are provided by its own HVAC system. Normally, the functions are ensured by a 1 x 100% safety train without cross connection to neighbouring divisions.
During maintenance, these functions are ensured by a 1 x 100% non-safety-related train that is common for two divisions (1/2 and 3/4).

The air supply for the mechanical area of each Safeguard Building is provided by the air supply system of its own electrical division. For Safeguard Buildings 1 and 4, the Safety Chilled Water Systems are equipped with air-cooled chillers installed on upper levels within the Safeguard Buildings. For Safeguard Buildings 2 and 3, the safety-related Chilled Water Systems are equipped with CCWS-cooled chillers. Each train of the Chilled Water System is designed to cool one of the trains of the air supply and the corresponding Main Control Room ventilation systems.

The HVAC for the Main Control Room and associated rooms is located inside the fully-hardened Safeguard Building 2 and Safeguard Building 3.

Main Steam and Feed water Valve Stations

The Main Steam and Feed water Valve Stations are arranged in a 2-by-2 configuration, with the lines of Divisions 1 and 2 located in a main steam valve house enclosure at the top of Safeguard Building 1 and the lines of Divisions 3 and 4 at the top of Safeguard Building 4. This physical separation and the main steam valve house enclosure provide protection against external hazards. The valve stations are physically protected by walls against internal hazards.

6.3 Switchgear Building

The Switchgear Building is divided into two divisions. Each division is separated into an electrical equipment and a cable distribution area, each having its own shafts for cables, supply/exhaust air, and smoke control. The Switchgear Building is situated on a foundation slab with two basement levels plus three stories above. Ventilation equipment for these buildings is installed on the roof. The building structure consists of reinforced concrete columns and walls as well as masonry walls, where applicable.

6.4 Fuel Building

There is a complete separation within the Fuel Building between operating compartments and passageways, equipment compartments, valve compartments, and the connecting pipe ducts. Areas of high activity are separated by means of shielding from areas of low or no activity.

External hazard protection is achieved by full hardening of the structure. The inner building structures are decoupled from the outer protection wall to ensure the integrity of the systems and components within the Fuel Building.

The pool for spent fuel assemblies is located outside the containment in the Fuel Building to:

- provide the possibility of cask loading outside the containment during plant operation,
- provide a sufficient spent fuel capacity without influencing the containment diameter.

Inadvertent release of radioactivity to the environment and ground water is prevented. Building isolation and filtering is ensured in the event of a release of radioactivity inside the building.

Fuel storage and handling equipment include the following features:

- storage of spent and new fuel assemblies outside the containment,
- sufficient storage capacity including full core unloading during outage,
- tube to transfer fuel assemblies from inside the containment to the outside and vice versa,
- the fuel transfer tube is closed from both sides during normal operation,
- for fuel cask handling, the layout of the building is based on loading via the bottom using a dedicated cask loading device.

The SFP consists of a single pool with two regions.

The mechanical floor levels below grade house the Fuel Pool Cooling System (FPCS), the Extra Borating System (EBS), and the CVCS. Redundant trains of these systems are physically separated by a wall.

Above grade level and up to the operating floor, one side of the Fuel Building is dedicated to the SFP, cask loading pit, transfer station, and the storage and inspection compartments for new fuel assemblies. The accident exhaust air filtration units, AVS, and parts of the containment sweep ventilation system are located on the other side of the Fuel Building, as well as the VCT of the CVCS and the two boric acid storage tanks.

The operating floor is divided into two areas: the SFP area and the setdown handling/transport area in front of the equipment hatch. The setdown handling/transport area is connected to the outside by two doors where all of the main equipment, components, and tools are handled. The setdown/handling area in front of the hatch is also connected via a large door to the maintenance and setdown area of the Nuclear Auxiliary Building.

If the large door for the spent fuel cask docking station is opened, the cover for the transportation opening inside this room can be closed. The cover is air-tight, fulfilling the airlock function between the loading area and the operating floor of the SFP area.

6.5 Nuclear Auxiliary Building

The Nuclear Auxiliary Building is located on a separate basemat from the Nuclear Island and contains additional systems including:

- boron Recycle System (coolant and demineralised water storage, coolant treatment and coolant purification),
- fuel Pool Treatment System,
- gaseous Waste Processing System (GWPS),
- portions of the Steam Generator Blow down System,
- nuclear Auxiliary Building ventilation and Operational Chilled Water System.

There is complete separation within the building between operating compartments and passageways on one hand, and equipment compartments, valve compartments and the connecting pipe ducts on the other. Areas of high activity are separated by means of shielding facilities from areas of low or no activity. It is not necessary to pass through compartments with a high dose rate to enter ones with a lower dose rate.

The isolation of the building is ensured in the event of a release of radioactivity inside the building to avoid inadvertent release of radioactivity to the environment. The basemat of the building is designed to be leak tight to avoid release of radioactivity to the ground water.

The Nuclear Auxiliary Building is safety-related and is designed against a SSE, winds, external flooding, and an EPW.

A portion of the building is designed as a radiological non-controlled area in which parts of the operational chilled water system are located. Special laboratories for sampling systems are located on the lowest level. Air exhausts from the radiological controlled areas of the Nuclear Island Buildings are routed, collected, and controlled within the Nuclear Auxiliary Building prior to release through the stack.

6.6 Diesel Buildings

There are two Diesel Buildings. The two Diesel Buildings house all four EDGs and the two SBO diesel generator sets along with their diesel fuel storage tanks. Related electrical, I&C and HVAC equipment are also contained in the Diesel Buildings.

The two Diesel Buildings are located on opposite sides of the plant, providing physical separation for protection against external hazards.

Each Diesel Building contains two redundant trains comprised of the main diesel generators for emergency power supply, and one SBO diesel generator. The two redundant main diesels and the SBO diesel generator, including related equipment, are protected against internal hazards by divisional separation.

The general layout of each Diesel Building from bottom to top is listed below:

- fuel storage tanks (within a dedicated fire compartment),
- diesel generators including local control panels,
- HVAC for electrical and HVAC exhaust air,
- battery room with electrical and service tank (zone SBO-Diesel only),
- service tank, handling room and HVAC room for underground piping and cable tray duct runs (main diesels only),
- engine air cooling equipment (zone SBO-Diesel only),
- air cooling equipment (main diesel only),
- roof level with silencer for diesel engines.

The doors at grade level leading into the open air are sound-absorbing and include special fittings to prevent entrance by unauthorized persons.

6.7 Radioactive Waste Processing Building

The Radioactive Waste Processing Building is used for the collection, storage, treatment and disposal of liquid and solid radioactive waste and is adjacent to the Nuclear Auxiliary Building. A basement extends underneath the entire building. The building has a total of seven full stories that contain all the components required for liquid and solid radioactive waste processing.

As with the Nuclear Auxiliary Building, there is a complete separation within the building between operating compartments and passageways on the one hand and equipment compartments, valve compartments and the connecting pipe ducts on the other. Areas of high activity are separated by means of shielding facilities from areas of low or no activity. It is not necessary to pass through compartments with a high dose rate to enter ones with a lower dose rate.

The Radioactive Waste Building is designed against the SSE.

Isolation of the building is ensured if radioactivity is released inside the building and the basemat of the building is designed to be leak tight.

The Radioactive Waste Building is supported on a shallow reinforced-concrete slab foundation which is separated from the adjacent foundation of the Nuclear Auxiliary Building by a settlement joint permitting relative movement. The design of the outer and internal walls is based on the results of structural analyses and on requirements for radiation protection.

6.8 Vent Stack

The vent stack is located between the Fuel Building and the Safeguard Building.

The vent stack design basis is Seismic Category 1, constructed of reinforced concrete with smoothed surfaces inside and outside.

The vent stack is designed for:

- exhaust air flow parameters defined by the HVAC equipment,
- wind loads,
- SSE,
- EPW.

The vent stack system ensures discharge of filtered air from the buildings within the controlled access area to the atmosphere at an elevation that meets dispersion requirements.

The Nuclear Auxiliary Building HVAC system exhaust air ducts lead to the roof of the staircase between the Fuel Building and Safeguard Building 4 and from there directly to the vent stack.

6.9 Essential Service Water Pump Buildings

The ESW Pump Buildings are located near the circulating water pump house and are connected to it by means of bypass ducts. The ESWS is arranged in two trains, each containing two ESW pumps at each side of the circulating water structures, separated by a sufficient distance to ensure protection against an aircraft hazard.

The circulating water, which is mechanically cleaned in the Circulating Water Pump Building, is transferred to the essential service water pumps in the Essential Service Water Pump Building. From there the circulating water is pumped into the service water lines.

The Essential Service Water Pump Buildings are designed for external hazards, including SSE and EPW.

FIGURE 6- 1: LAYOUT OF MAJOR EPR BUILDINGS

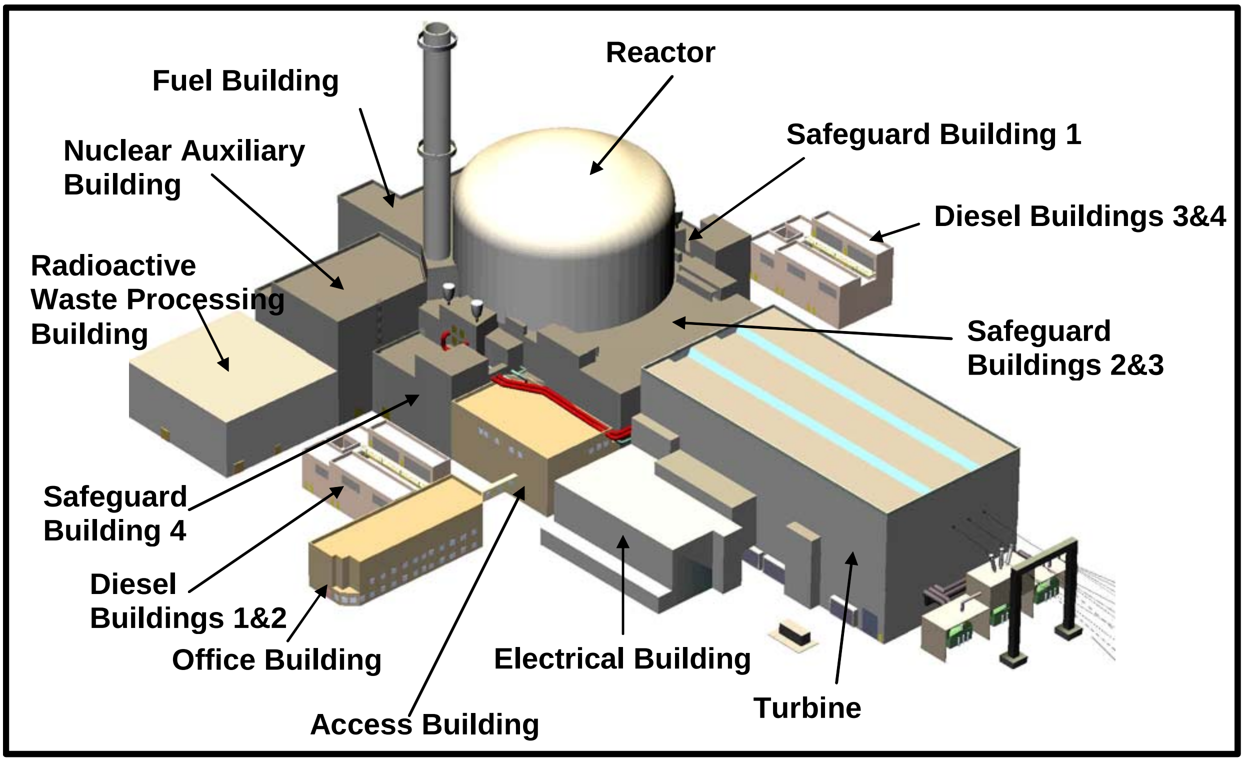
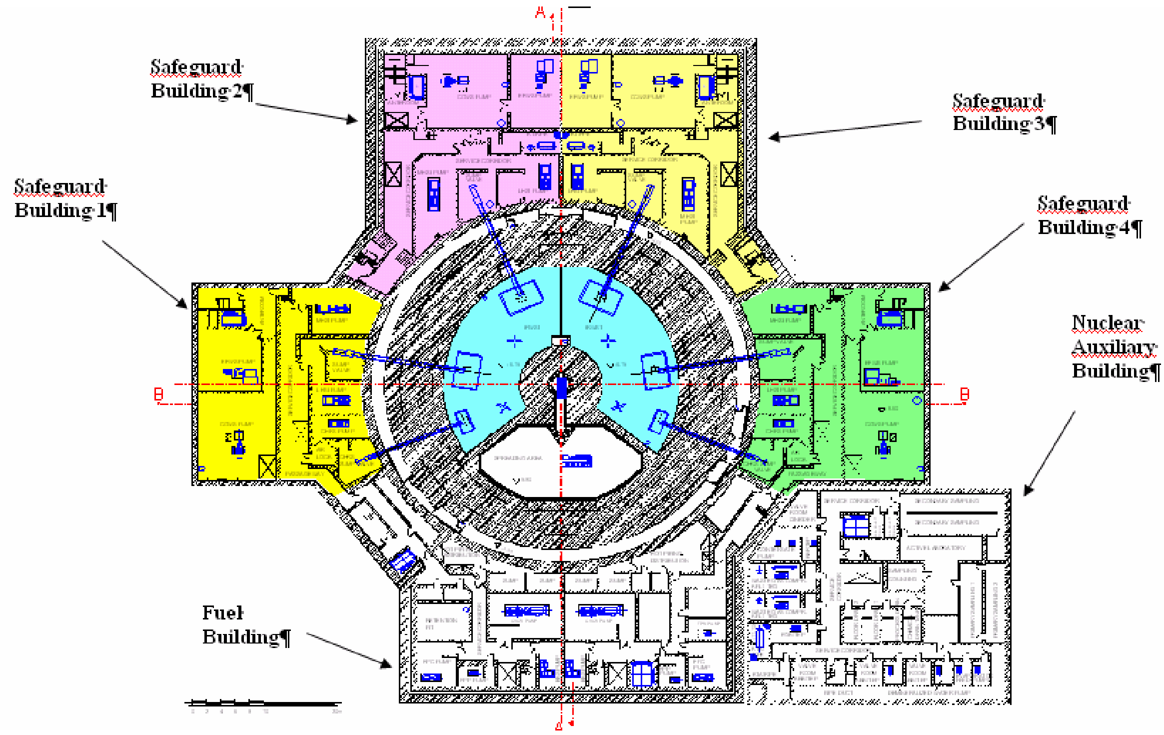


FIGURE 6- 2: ORGANIZATION OF SAFEGUARD BUILDINGS ENSURES STRICT PHYSICAL SEPARATION



7.0 ELECTRICAL DISTRIBUTION AND INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Electrical distribution system

Some of the more important electrical system features of the EPR are described in this section.

7.1.1 General

The basic power supply for the EPR operates at 50Hz, with voltage regulated through a site/utility specific transmission grid. Figure 7-1 is a single-line diagram illustrating the distribution network.

The Electrical Distribution System (EDS) is designed as a 4-train, 4-division system. Most non-safety related plant loads are powered from the Turbine Island 4-train Normal Power Supply System (NPSS) of the EDS while engineered safeguard loads as well as a few non-safety related loads are powered from the Nuclear Island 4 division, Emergency Power Supply System (EPSS) of the EDS.

The RCPs and a few non-safety related loads are powered from the Nuclear Island NPSS.

During plant on-line/power operation, electrical power is supplied from the main generator via the main step-up transformer to the main switchyard and the plant EDS via two auxiliary normal transformers. Each transformer powers two trains and two divisions and about half of the total plant auxiliary load.

During plant off-line/shutdown periods, power to the EDS is supplied from either the main switchyard via the two auxiliary normal transformers, or the standby switchyard/transmission grid via a single auxiliary standby transformer.

The plant can accept a generator load rejection from 100 percent power or less without a reactor or turbine trip while stable operation continues. During such an event, the generator breakers (i.e., those that connect the main step-up transformer and auxiliary normal transformers to the main switchyard) will open, but the connection from the generator to the auxiliary normal transformers via the main step-up transformer remains online. Consequently, the plant can continue to autonomously operate (within administrative/license limits associated with available off-site sources), disconnected from the grid while powering all house loads.

The EPSS is normally powered directly from the Turbine Island NPSS. However, in the event of a loss of off-site power or a degraded grid event (with or without a concurrent design basis accident), the EPSS is automatically disconnected from the NPSS while four Emergency Diesel Generators (EDGs) (one for each division) re-power the EPSS. As described in section 6, the EDGs are housed in buildings separated from the rest of the plant.

The EPSS has four Uninterruptible Power Supply (UPS) channels (one per division) consisting of batteries, battery chargers, inverters, and associated distribution panels. Emergency power is also available from two additional diesel generators provided as alternate AC sources for coping with postulated SBO events.

7.1.2 Off-Site Power

Design criterion requires that each plant's onsite EDS be supplied by two physically independent circuits designed and located to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. For the EPR, these two circuits consist of:

- the auxiliary normal transformers and their high-side connections to the main switchyard and their low-side connections to the NPSS buses,

- the auxiliary stand-by transformer and its high-side connections to the standby switchyard and its low-side connections to the NPSS buses.

7.1.3 Emergency Power Supply System

The EPSS is arranged in four divisions, each separate and independent from the other three and each with its own EDG consistent with 4-division process safety systems. Safety loads and some non-safety loads are connected to the EPSS, as previously mentioned. These loads are those necessary to shut down the reactor safely; maintain it in a shutdown condition; remove residual and stored heat; mitigate accidents; and prevent excessive release of radioactive substances under accident conditions.

7.1.4 Station Blackout Power Supply System

The SBO power supply system is part of the EPSS. Two separate and independent SBO diesel generator units are provided as alternate AC sources for coping with postulated SBO events (i.e., loss of both the off-site power supply and all four of the onsite EDGs). Loads necessary for non-accident safe plant shutdown are powered from the SBO diesels. Until the start of the SBO event, the two-hour rated EPSS batteries supply DC power for required loads including inverters and their critical loads. Early in the two-hour period, the SBO diesels are started manually from the control room.

7.1.5 Uninterruptible Power Supply System

The UPS system is part of the EPSS and provides four UPS channels. Batteries and inverters provide an assured source of power for essential low-voltage DC and AC loads.
Each battery charger is sized for the capability to recharge its fully discharged battery while concurrently supplying its largest combined demand of various essential steady-state loads. With a full charge and the charger not operating, each battery is capable of supplying power under the worst-case design basis event loads for two hours.
In addition to the four two-hour rated batteries, two supplemental 12-hour rated batteries are provided, one each for divisions 1 and 4. These supplemental batteries are provided for severe accident mitigation and increase the coping time for restoration of AC power.

7.2 Instrumentation and Control System

The functions of plant I&C systems are:

- control and monitoring of the plant systems functions during normal conditions,

- control of plant functions that are implemented to initiate corrective measures in the case of deviation from LCO,

- control of limitation functions that are implemented to initiate corrective measures in order to avoid protective actions,

- control of protection functions that are implemented to mitigate the consequences of a design basis event, up to reaching a controlled, stable state following the detection of a design basis event,

- control and monitoring of post-accident functions that are implemented to mitigate the consequences of a design basis event, and to bring the plant from the controlled state down to the safe shutdown state,

- control of functions that are implemented in order to mitigate the consequences of a beyond design basis event.

7.2.1 Basic Architecture and Level Structure m

The different roles of the systems and plant requirements lead to an I&C architecture level structure as follows:

- level 0: Process Interface,

- level 1: System Automation Level,

- level 2: Unit Supervision and Control Level,

- level 3: Site Management Level.

The systems and equipment of Level 3 are not discussed in this document. (Examples of Level 3 systems would be business and financial networks, and related site management systems).

Different general design requirements correspond to these levels. All I&C systems are designed to withstand the environmental conditions that exist in the rooms or locations where they are installed so that they properly perform their design functions. Additionally, Level 2 systems are user-friendly designed.

Classification

The I&C functions and equipment are categorized as safety-related (F1 functions), quality-related, and non-safety-related according to their importance to safety. All portions of the I&C systems and equipment needed to perform a given I&C function are classified according to the highest class functions they must perform.

Diversity

The I&C architecture is designed to distribute diverse I&C functions into an appropriate number of different safety I&C systems to avoid common-cause failures and thus to meet the required probabilistic targets. Two complementary types of diversity are implemented in the design in order to reduce the risk of common-mode failures as defined below.

- functional Diversity -- Functional diversity provides two separate I&C functions based upon two different methods of detecting a condition in order to initiate the same type of protective action,

- equipment Diversity -- Equipment diversity consists of providing two different hardware platforms in order to preclude a common-mode failure taking out a function.

The separate I&C systems have adequate independence and diversity features to minimize the risk of common mode failures (hardware and software) in accordance with plant probabilistic targets.

Description of the I&C Architecture

The I&C architecture, shown in Figure 7-2 fulfils the operational, licensing, and safety design goals to operate the plant and perform protective functions.

The unit supervision and control level (Level 2) consists of the work stations and panels of the Main Control Room, the RSS, the TSC, as well as the Process Information and Control System (PICS) and Safety Information and Control System (SICS), which act as interfaces between the MMI and the automation systems.

The system automation level (Level 1) consists of the following main systems:

- protection System (PS),
- safety Automation System (SAS),
- process Automation System (PAS),
- priority and Actuator Control System (PACS),
- reactor Control, Surveillance, and Limitation System (RCSL).

The process interface (Level 0) comprises the interface with the sensors, actuators, and switchgear.

The instrumentation is categorized according to the specific nuclear requirements for the measurements listed below.

- process instrumentation,
- radiation instrumentation,
- accident instrumentation,
- in-core instrumentation,
- ex-core instrumentation,
- flux mapping system,
- RPV level instrumentation,
- rod position measurement,
- loose part and vibration monitoring,
- seismic instrumentation,

- hydrogen detection system,
- advanced boron instrumentation.

Operators use the workstations and the plant overview display in the Main Control Room to operate and monitor the plant. Signals from and to the workstations and the plant overview panel are processed by the Process Information and Control System (PICS).

In the event of a design basis event, all functions necessary to reach the controlled state[2] (namely “F1A” functions) are initiated by the Protection System (PS). The functions to reach the safe state[3] (namely “F1B” functions) are either automatically generated in the Safety Automation System (SAS) or manually initiated and processed by PICS and SAS.

If PICS fails, the operators use the Safety Information and Control System (SICS) together with PS, SAS and PACS for unit control and safe shutdown.

If one of the Level 1 systems, PAS, RCSL, PS, or SAS, fails, the remaining I&C systems in the other lines of defence are sufficient to shut down the plant and keep it in a safe shutdown state.

In the event that the Main Control Room is not accessible, the plant is monitored and controlled from the RSS, making use of PICS. The SAS, PAS, RCSL and PS remain available when operation is performed from the RSS.

Depending upon the different tasks of the I&C functions, contradictory commands could be given by the different I&C functions to particular actuators. Therefore, general priority rules are established so that any potential command will be assigned to a defined priority level.

The following general rules are applied for all actuators in the plant:

- higher classified functions have priority over commands from lower classified functions. The order of priority is: (1) F1A function, which has priority over (2) F1B function which has priority over (3) non-safety related functions (F2 function and non classified functions),
- the order of priority between different categories of I&C functions within the quality related class is: (1) highest priority for control of design basis accidents and design extended conditions, then (2) limitation function, and at the lowest level, (3) limitation of operating condition,
- the principal order of priority within each I&C category in all classes is: (1) highest priority for component and system protection, then (2) automatic action, and at the lowest level, (3) manual action. Automatic control functions may be switched off if the process conditions allow.

[2] The **controlled state** is defined as a state where the fast transient resulting from a DBC-1 to DBC-4 event is finished. The plant is stabilized and where the core is sub critical, the heat removal is ensured in the short term, the core coolant inventory is stable and activity releases remain tolerable.

[3] The s**afe shutdown state** is defined as a state following a DBC-1 to DBC-4 event where the core is sub critical, the decay heat is removed durably and activity releases remain tolerable.

7.2.2 Man-Machine Interface

Due consideration is given to human factors at the design stage, taking into account operation, testing, and maintenance requirements. The general aim is to minimize the likelihood of operator error and lower the demands on the operator. For this purpose, appropriate ergonomic design principles are implemented and sufficiently long grace periods are made available to the operators. The necessary time duration depends on the complexity of the situation to be diagnosed and the actions to be taken (i.e., in the Main Control Room, in the RSS, or locally).

Sufficient and appropriate information supplied by the I&C systems provide a means for clearly understanding plant status during normal operation, design basis events, and beyond design basis events, and for evaluating the effects of actions taken.

The MMI facilities are subdivided into the following main items:

- the central, permanently staffed Main Control Room,
- the RSS staffed on demand, if the Main Control Room becomes inaccessible,
- local control stations staffed on demand,
- the TSC.

7.2.3 Main Control Room

During all plant conditions (except if the Main Control Room becomes inaccessible) the plant is supervised and controlled from the Main Control Room. The Main Control Room is equipped with essentially identical operator workplaces consisting of PICS-driven screens (video display units) and soft controls.

These operator workplaces provide for the following staffing arrangement:

- two of the operator workplaces are staffed during normal plant operations,
- a third operator workplace is staffed during plant states requiring increased operating and monitoring tasks (e.g., refuelling period, post accidental conditions),
- a fourth operator workplace is used by the shift supervisors,

Additional monitoring and control equipment in the Main Control Room include:

- the plant overview panel consisting of several large PICS-driven screens that provides overviews of plant status and main parameters,
- the safety control area with the SICS displays and controls enable operation during accidental conditions and is a back-up in case of unavailability of PICS,
- fire detection and fire fighting controls and site closed circuit TV monitoring screens.

7.2.4 Remote Shutdown Station

If the Main Control Room becomes inaccessible, the operators supervise and control the plant from the RSS.

The RSS is equipped with:

- manually-actuated switches for disconnecting all the Main Control Room equipment that may generate component actuation of the Level 1 systems and placing the RSS workstations in the control mode. Technical and administrative precautions prevent spurious or unauthorized actuation of this function,
- operator workplaces consisting of PICS driven screens (video display units) and soft controls that are of the same type and provide the same functionality as those in the Main Control Room. The operators can bring the plant to safe shutdown state and monitor plant conditions from these operator workplaces,
- communication equipment for maintaining communications with other plant personnel.

7.2.5 Technical Support Center

The TSC is used by the technical support team in the event of an accident. The additional staff in the TSC analyzes the plant conditions and supports post-accident management. The TSC is equipped with PICS screens that have access to plant information. No process control function is available in the TSC. Appropriate communications equipment is also provided in the TSC.

7.2.6 Description of I&C Systems

Protection System

The PS is the main I&C line of defence and performs the automatic Class F1A functions that are needed to bring the plant to a controlled state if a design basis event occurs.

The PS performs its safety functions to ensure:

- the integrity of the RCPB,
- the capability to shut down the reactor and maintain it in a safe shutdown condition,
- the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures.

Several examples of PS actions include tripping the reactor, actuating containment isolation, actuating emergency core cooling, initiating ATWS mitigation actions, and performing EFW system protection and control (i.e., actuation, isolation, and pump run-out protection).

The PS is a digital I&C system, located in dedicated cabinets in the Nuclear Island. The system is implemented in four divisionally separate trains, each with its own Class F1 power source.

Additionally, each PS cabinet has redundant power supplies for its electronics. The PS is functionally independent of all other I&C automation systems, thereby ensuring that the failure of one of those systems does not prevent the PS from performing its safety functions. The system can process Class F1A (and lower classified) functions.

The primary function of the PS is to initiate automatic reactor trips, actuate engineered safety features, and start their support systems. The PS reactor trip function utilizes voting logic in order to screen out potential upstream failures of sensors or processing units. The PS manages all permissions used in the logic for the execution of the automatic protective actions and also provides the capability to perform manual actions on safety systems.

Safety parameter information is provided by the PS for the SICS and the PICS and gathers signals from various Class F1A process sensors and components. The process variables, initiation signals, and actuation signals of the PS are displayed to the plant operators by the PICS and the SICS. Safety interlocks are incorporated in the PS, defeating manual initiation and resetting automatic functions by the PICS or SICS if the process conditions do not warrant manual actuations.

Connections with other I&C systems are implemented through isolated channels to maintain complete independence. The PS is a self checking system which is able to perform continuous self-diagnosis for many conditions and alert the plant operators to unusual conditions or internal failures. The PS also includes equipment dedicated to periodic testing and maintenance.

Safety Automation System

The SAS is devoted to automatic control, manual control, and measuring and monitoring functions needed to bring the plant to a safe shutdown state. It provides for the following:

- post-accident automatic and manual control as well as the monitoring functions needed to bring the plant from a controlled state to the safe shutdown state,
- I&C functions related to the support systems of safety systems if they do not change their status during an event,
- the automatic initiation of I&C functions, to prevent a spurious actuation that could result in design basis accidents,
- functions preventing radioactive releases.

The SAS performs its functions via a closed loop control, a sequencing control, a combination control, data acquisition, a drive control, and alarm generation and processing. The SAS is a digital I&C system that receives process data from plant instrumentation and switchgear; process data from the PS; and some control signals from the SICS and the PICS. The SAS sends actuation signals to PACS modules or switchgear and monitoring signals to the SICS and PICS. It can perform F1B (and lower classified) functions.

Priority and Actuator Control System

The PACS consists of individual modules that control one safety actuator only. These individual modules may be common to different defence lines as they are using the same actuators and thus have to fulfil the high availability and reliability requirements against common cause failures.

For the Class F1A actuators that also support functions of a lower class, the PACS performs drive monitoring; priority management between control signals of different safety classes; actuator specific commands for essential component protection; and drive actuation. The priority of actuation signals coming from the PS, SAS, or PAS is managed by the priority function of the PACS. Actuators shared by different safety classifications are always controlled in the direction required by the actuation signals of the I&C system having the highest safety classification, even if a lower safety classification system simultaneously requires another direction.

In addition to handling these priority functions, the PACS performs the drive control functions, including drive actuation, drive monitoring, and essential component protection (such as equipment overloads and trips).

Reactor Control, Surveillance and Limitation System

The role of the RCSL is to implements the automatic functions, manual actions and monitoring functions that are needed to control and limit the main reactor parameters, like:

- reactor power,
- power density,
- reactivity.

They include the following types of functions:

- closed loop control functions,
- automatic limitation functions for improvement of plant availability,
- automatic limitation functions to ensure the Limiting Conditions of Operation (LCO),
- LCO surveillance (alarm) functions.

The RCSL is distributed in the four divisions for the sensors acquisition units and in the division 1 and 4 for the processing and actuation of the rods.

Although the RCSL functions are Class F2 functions, the technology that is implemented is the same as the one implemented for the PS. The aim is to group PS and RCSL within a consistent equipment package so as to facilitate data communication and reduce the number of cables between each other.

Process Automation System

The PAS implements the automatic control, manual control, and monitoring functions that are classified as F2 or non-classified functions. The PAS is a digital I&C system that performs functions such as normal feed water control, normal steam system control, and normal support systems control.

The functions of the PAS are monitored and controlled by the plant operators via the PICS. If the PICS becomes unavailable during normal plant operation, the limited functions of the PAS which are required to maintain the plant in a stable state can be monitored via the SICS.

Process Information and Control System

The PICS is a Level 2 system and is used in all plant conditions by the operators to monitor and control the plant. The PICS is a Class F2 system. It employs computers, video display units, and soft controls.

It has access to all Level 1 systems and presents information to the plant operators at the following MMI devices and locations:

- screens for monitoring and control at the operator workstations in the Main Control Room,
- screens for monitoring at the shift supervisor's location,
- large screen or projected video display for the plant overview display in the plant Main Control Room,
- screens for monitoring and control in the RSS,
- screens for monitoring in the TSC,
- printing stations and information recording/archiving stations.

The PICS displays alarms in the event of abnormalities in processes or systems and provides guidance to the operators in order to perform the appropriate corrective actions. It can perform non-safety related functions, as well.

The PICS design considers the quality of displayed information and ease of operation of actuator controls. Compliance with recognized ergonomic principles and navigation techniques between various operation displays are key factors in ensuring efficient and error-free operation.

Safety Information and Control System

The SICS is also a Level 2 system and is used by the Main Control Room operators in the event of unavailability of PICS (due to PICS-internal failure or other causes). In this case, the operators use SICS to monitor and control the plant for a limited time in steady state power operation and bring the plant to and maintain it in shutdown state (non-safety related functions), if PICS cannot recover within 2 to 4 hours.

In design basis events, with PICS unavailable, SICS makes it possible to:

- monitor the safety functions of the plant, especially the automatic protective actions and post-accident functions,
- manually control the post-accident functions necessary to bring the plant from the controlled state to the safe shutdown state,
- monitor and manually control the support systems of safety systems needed for post-accident control.

SICS is a Class F1B system.

In the event of an accident with PICS unavailable, the member of the operating crew who is in charge of monitoring the actions of the shift team uses the SICS to obtain information.

When the SICS is not needed, its controls are deactivated to reduce the risk of spurious actuations due to any possible hazards or internal equipment failures within the SICS.

Communication

Each I&C system manages its own internal exchanges (including data exchange between divisions) without using external resources, when reasonable. Data is exchanged between the different I&C systems primarily through standard network or standard exchange units connected to the corresponding system networks. The design of the communication between the systems assumes the same failure mechanisms as the I&C systems (single failure criterion, preventive maintenance, hazards and common mode failure).

7.2.7 Instrumentation

The instrumentation is organized according to the nuclear specific requirements.

The classification of the instrumentation channels follows the classification of the safety functions to which they feed signals. The instrumentation is designed to monitor process variables with sufficient accuracy and response time in all plant states in which they are needed. Self-monitoring and self-testing are implemented, as well as remote calibration and testing.

Process Instrumentation

Process instrumentation includes the sensors and transducers needed to measure thermal-hydraulic process parameters such as pressure, temperature, flow, and level. Appropriate connections are provided for isolation, maintenance, and testing of sensors and transducers.

Radiation Monitoring

The radiation monitoring system detects ionizing radiation and radionuclide transport. Its purpose is to monitor and document when prescribed values have been detected, and in the case of deviations from these values, to record the deviation and initiate countermeasures

This functionality is achieved through:

- continuous monitoring of radioactivity in the plant systems by means of fixed measuring instruments (process monitoring),
- continuous monitoring of radioactivity of the plant environment and work areas by means of fixed measuring instruments (area monitoring).

The measuring instruments are designed to function properly under normal operating conditions as well as during and after accident situations.

The measuring points consist of radiation detectors connected to electronic transducers which convert the measurement into digital information signals. These digital information signals are sent to the Process and Information Control System (PICS) and the SICS which perform the monitoring functions.

Additionally, the radiological instrumentation feeds signals to the central radiological computer system that processes radiological data, displays information, and generates alarms. The central radiological computer system also captures data from the meteorological system and calculates propagation of radioactivity into the plant environment.

The safety classification of each measuring point is assigned based on the I&C function that uses its measurements. Redundancy requirements of this system are met by using at least two mutually independent instrument arrangements in a parallel configuration.

In addition to providing information signals to PICS and SICS, the radiation monitoring system also interfaces with other main I&C systems.

The power supply for the radiation detectors and transducers is provided by the UPS. Pumps and heating systems used for the radiation monitoring system are supplied by the emergency power system.

In-Core Instrumentation

The fixed in-core instrumentation (see Figure 2-8 and Figure 2-9) measures the radial and axial neutron flux distribution in the core and the radial temperature distribution at the core outlet.

The neutron flux signals are used for the following functions:

- control of axial power distribution,
- core surveillance for maintaining LCO,
- core protection.

The core-outlet thermocouples continuously measure the fuel element outlet temperatures, provide signals for core monitoring after a LOCA, and provide additional information on radial power distribution and local thermal-hydraulic conditions.

Ex-Core Instrumentation

The excore instrumentation consists of the following instrumentation channels:

- one instrumentation channel group for the source range,
- one instrumentation channel group for the intermediate range,
- one instrumentation channel group for the power range.

The total neutron flux range to be covered amounts to approximately 9 to 10 decades up to 150% rated reactor power.

The organization and overlap of source range, intermediate range and power range channels is shown in Figure 7-3.

Flux Mapping Instrumentation

The flux mapping instrumentation is an aero ball measuring system (see Figure 2-7). Relative local neutron flux density is periodically measured by introducing stacks of balls made of vanadium-alloyed steel (aero balls) into the core

The measured flux density provides a means for:

- calibrating the fixed neutron in-core and ex-core detectors,
- verifying fuel loading conformity and effect of burn-up,
- detecting anomalies.

Reactor Pressure Vessel Level Instrumentation

The RPV water level measurement provides information of the RPV water inventory in all plant operating conditions up to core melting. The RPV water level information is a safety related parameter of first importance and can be used in post-accident management.

Rod Position Measurement

The rod position measurement instrumentation provides information on the current position of all reactor control rods

Control rod position measurements are provided in all plant modes of operation and are required for:

- surveillance of representative core parameters,
- closed loop control,
- LCO surveillance,
- limitation functions.

Rod position measurement instrumentation consists of one measurement circuit for each RCCA.

Each measurement circuit is made up of the following components:

- sensor,
- power supply module,
- processing module,
- cables.

Rod position measurements are classified as Safety Class F1A. These measurements are utilized as inputs to the PS and to the RCSL enabling those systems to perform their safety-related functions.

The 89 RCCAs are divided into four banks spanning the reactor core. Banks 1, 2, and 3 contain 22 assemblies each, while bank 4 contains 23, and includes the center assembly. The position measurements from each bank are routed to the corresponding Safeguard Buildings 1, 2, 3, and 4. This system of division constitutes the basis of the four-fold redundancy classification of the rod position measurement instrumentation.

The rod position measurement sensors are located inside the Reactor Building above the reactor vessel head and are designed to withstand the environment at that location.

Loose Parts and Vibration Monitoring

The loose parts and vibration monitoring instrumentation consists of accelerometers, sensors, transducers, and processing for:

- detecting loose parts located in places where their effect may result in severe damage with serious consequences on the availability or the safety of the plant,

- monitoring vibrations of the RCS, RPV, internals, and RCPs for early detection of abnormal behaviour of large components.

Seismic Instrumentation

The seismic instrumentation consists of eight triaxial accelerometers and associated processing equipment. These instruments measure and process the accelerations generated by an earthquake. The data are used to evaluate the seismic level and make the appropriate decisions depending upon whether or not the earthquake exceeds a predetermined level. Recording capabilities are provided.

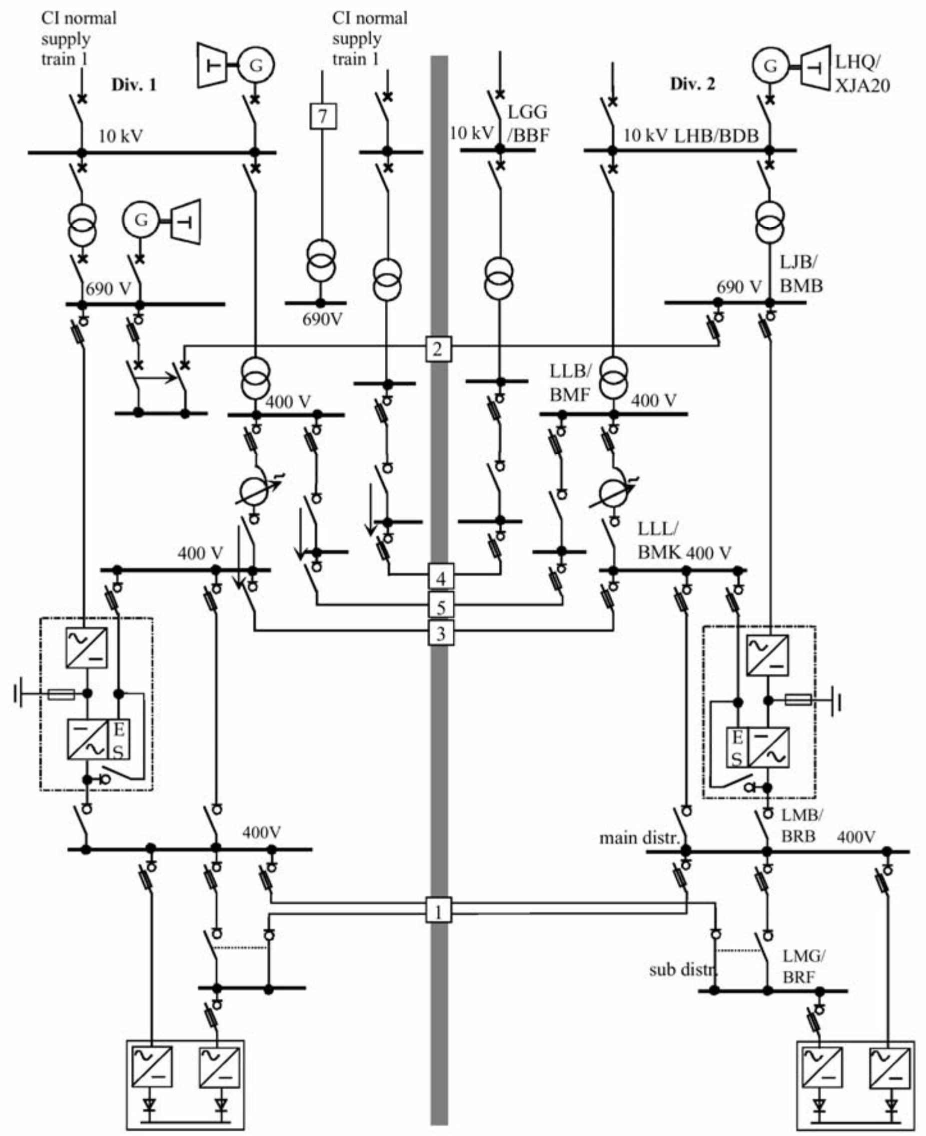
Hydrogen Detection System

Hydrogen may be produced as a consequence of an accident. The hydrogen detection system assesses the efficiency of the Combustible Gas Control System (hydrogen recombiners) inside the reactor building.

Advanced Boron Instrumentation

This instrumentation provides measurements that are used to monitor the boric acid concentration and detect an inadvertent supply of diluted water to the RCS.

FIGURE 7- 1: ELECTRICAL SINGLE LINE DIAGRAM



This diagram shows divisions 1 and 2. Divisions 3 and 4 are similar

FIGURE 7- 2: OVERALL I&C ARCHITECTURE

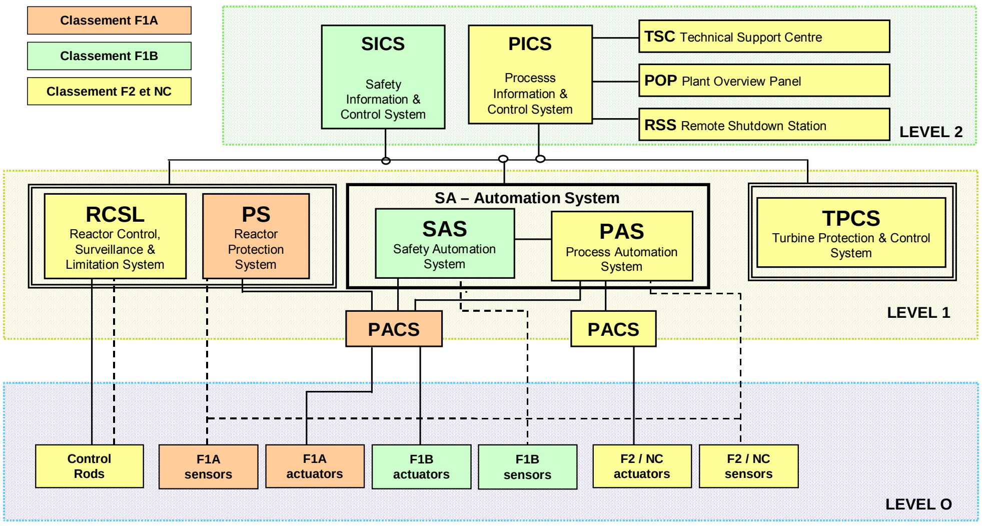
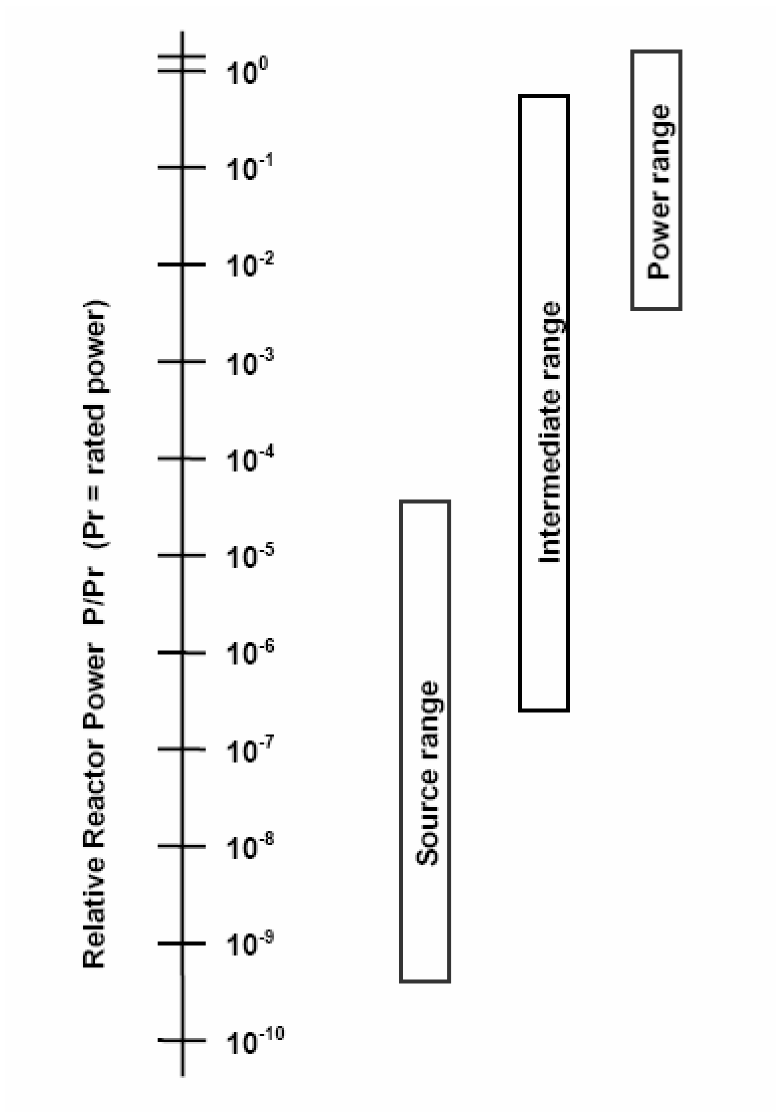


FIGURE 7- 3: MEASURING RANGES OF EX-CORE INSTRUMENTATION



8.0 AUXILIARY SYSTEMS

8.1 Fire protection systems

The EPR fire protection design basis is focused to protect the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection design basis is based on the concept of defence-in-depth.

Relative to fire protection, defence-in-depth is achieved when an adequate balance of each of the following elements is provided:

- fire prevention,
- rapidly detecting fires that do occur,
- promptly controlling and extinguishing fires that do occur, thereby limiting fire damage,
- providing an additional level of protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

The fire protection features of the EPR are capable of providing reasonable assurance that, in the event of a fire, the plant will not be subjected to an unrecoverable incident.

Two separate safe shutdown systems provide ongoing fire protection capabilities to meet the following performance criteria in the event that one train has been become inoperable:

- <u>Reactivity Control</u>: Reactivity control shall be capable of inserting negative reactivity to achieve and maintain sub-critical conditions. Negative reactivity insertion shall occur rapidly enough such that fuel design limits are not exceeded.
- <u>Inventory and Pressure Control</u>: With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling the coolant level such that subcooling is maintained.
- <u>Decay-Heat Removal</u>: Decay-heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel to maintain a safe and stable condition.
- <u>Vital Auxiliaries</u>:Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that systems are capable of performing their required nuclear safety function.
- <u>Process Monitoring</u>: Process monitoring shall be capable of providing the necessary indication to assure these criteria have been achieved and are being maintained.

Additionally, the fire protection system design basis ensures that radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be As Low As Reasonably Achievable (ALARA) and shall not exceed applicable regulatory limits.

8.1.1 Fire Area Functions and Definition

Individual fire areas are established to confine fires to their area of origin and prevent fires from spreading to adjacent fire areas.

Structural barriers, consisting of walls, doors, windows, floors, ceilings, and ventilation dampers, are used to separate the fire areas.

According to the ETC-F (EPR Technical Code for Fire Protection), 4 fire areas are defined:

- **Fire and radioactivity confinement area**: a fire and radioactivity confinement area is a fire area in which a fire could lead to radioactive releases, and in the absence of measures intended to avoid their spread outside the fire area, to a dose greater than the maximum permissible dose for workers and for the population. In addition to the confinement of fire, this area ensures the control of radioactive releases.

- **Access area**: to protect the means of egress for plant personnel and access for fire brigade.

- **Safety fire area**: a safety fire area is a fire area designed to safeguard the safety trains against a common mode failure.

- **Unavailability limitation fire area**: an unavailability limitation fire area is a fire area established when the fire load density of a room is greater than 400 MJ/m² in order to limit Plant Unit unavailability and facilitate action from the fire brigade. It may form part of a safety fire area or be independent of any safety fire area.

8.1.2 Fire Area Boundaries

Fire area boundaries are established to separate the fire areas. This fire resistance rating is determined in relation to the design basis fire defined within the fire area, available fire-fighting facilities and installation conditions, and guarantees that a fire occurring inside cannot spread to the outside, or, if it arises outside, cannot spread inside. Wall, ceiling, structural components, thermal insulation, radiation shielding, and sound-proofing materials used are noncombustible.

8.1.3 Fire Barrier Openings and Penetration Seals

Openings through fire barriers for pipe conduit and cable trays are sealed with non-combustible materials to provide a fire resistance rating equal to that required by the barrier itself, and qualified in accordance with design criteria.
Penetrations in fire barriers are provided with fire-rated door assemblies or rated fire dampers having a fire resistance rating consistent with the designated fire resistance rating of the barrier.

8.1.4 Security for occupants

Design measures are taken into consideration to ensure the protection and safe egress of occupants not intimate with the initial fire development and to provide a high level of assurance for the survivability of occupants intimate with the initial fire development.

8.1.5 Fire Detection and Alarm System

The general fire detection network is designed to ensure:

- rapid detection of an incipient fire,
- location of the fire source,
- monitoring of the fire progression,
- triggering of the alarm and in some cases, actuation of fire dampers and smoke dampers of the smoke control system.

Fire detectors are selected and installed in accordance with ETC-F. Preoperational and periodic testing of pulsed line-type heat detectors is performed to demonstrate that the detector transmit frequencies used will not affect the actuation of protective relays in other plant systems. The detectors give audible and visual alarm and annunciation in the control room. Where zoned detection systems are used in a given fire area, local solutions are provided to identify the detector zone that has actuated. Local audible alarms sound in the fire area and are distinctively unique so they will not be confused with any other plant system audible alarms.

Primary and secondary power supplies are provided for the detectors and for electrically operated control valves for automatic fire suppression systems.

8.1.6 Fire Water System

An underground firewater loop is provided around the EPR site to furnish anticipated water requirements. The loop is connected to a reliable fire protection water distribution system. The underground yard loop distributes fire water to the site hydrants and other water-based fire protection systems.

The fire protection water supply is of adequate reliability, quantity, and duration. These requirements will be satisfied with the following features:

- a fire protection water supply consisting of not less than two separate supplies, and
- a fire protection water supply providing a minimum flow rate for 2 hours based on 240 m^3 for manual hose streams plus the largest design demand of any sprinkler or fixed water spray system(s) in the power block. The fire water supply is capable of delivering this design demand with the hydraulically least demanding portion of the fire main loop out of service.

The separate water tanks are interconnected so that fire pumps can take suction from either or both. A failure in one tank or its piping will not allow both tanks to drain.

The water inventory of the tanks is assumed to be recovered within 8 hours.

The fire water system incorporates two redundant 100-percent capacity fire pumps. A motor-driven jockey pump is used to keep the fire water system full of water and pressurized, as required. Fire pumps are provided to ensure that 100 percent of the required flow rate (largest system demand plus 120 m^3/h corresponding to 2 hose streams allowances) and pressure are available even when assuming failure of the largest pump or pump power source.

Individual fire pump connections to the yard fire main loop are provided and separated with sectional valves between connections. The capability to isolate portions of the yard fire main loop for maintenance or repair is also provided without simultaneously shutting off the supply to both fixed fire suppression systems and fire hose stations.

A method of automatic pressure maintenance of the fire protection water system is provided and is independent of the fire pumps. A means is provided to immediately notify the Main Control Room or an alternate location (continuously staffed) of the operation of the fire pumps.

8.1.7 Automatic Sprinkler Systems

The fire protection systems are designed to detect, control, and/or extinguish possible fires throughout the facility. Fire suppression systems are selected based on the type of hazard(s) in the fire area, the impact on operation, and the potential for release of the suppression agent.

In fire areas where the use of a water-based suppression system is the preferred means of suppression, wet-pipe sprinkler systems are provided (with possible exception concerning individual fire department connections for each sprinkler system). Due to the fortified and protected design of the facility, it is not practical to provide individual fire department connections for each sprinkler system. Since the sprinkler systems are supplied by the plant's highly reliable fire protection water supply, individual connections are not necessary.

Each system is equipped with a water flow alarm and all alarms from any fire suppression system will provide an audible alarm in the control room or other suitable constantly attended location.

Automatic sprinkler systems are installed at the following areas:

- four extinguishing systems for the EDGs, which provide coverage for the service tank area, emergency diesel area, and area for oil cooler and auxiliary equipment,
- two extinguishing systems for the SBO diesel generators, which provide coverage for the service tank area, SBO diesel area, and area for oil cooler and auxiliary equipment,
- diesel Fire Pump Room.

8.2 Steam Generator Blow down System

The SGBS maintains the necessary quality of the water/steam cycle in conjunction with the nuclear sampling system. The radioactive and chemical characteristics of secondary-side water are kept within permissible limits during all plant operating conditions. SG secondary water is removed continuously, because the secondary side may contain impurities in the form of corrosion products, condenser in-leakage, or primary-to-secondary leakage contaminants. After treatment, the blow down water is normally recycled to the condenser, except for exceptional cases when it is discharged to the liquid waste discharge system.

Figure 8-1 shows a flow schematic of the SGBS.

The operational functions of the SGBS are to:

- maintain the contaminants and minerals produced by phase separation in the SGs within predetermined limits by blowing down a part of the flow,
- expand and cool the blow down water from the SG so it can be returned to the water/steam cycle via the blow down demineralising system (the resin in the demineralizers must be protected from high temperature),
- ascertain the quality of SG water by continuous monitoring of the blow down water, enabling fast detection of SG tube leaks,
- provide partial or total draining of the SG secondary side,
- provide for the bubbling of the SG secondary side using the Nuclear Island Nitrogen Distribution System to mix the sampling area to get a homogeneous sample, especially when chemical reagent is injected during cold shutdown,
- reprocess secondary side samples coming from the Nuclear Sampling System.

The safety functions of the SGBS are to:

- ensure activity retention in the affected steam generator in case of a SGTR,
- avoid steam generator overfilling and subsequent liquid release in the event of a SGTR,
- ensure containment isolation.

During normal plant operation, 1% of total feedwater is blown down. During a special operating configuration, three SGs could be isolated so that they do not have any blowdown. Should this scenario develop, the maximum continuous blowdown flow rate of the fourth would be about 2% of the feedwater flow to the remaining SG.

SG blowdown is automatically isolated in the event of a SGTR and also on the activation of emergency feedwater flow.

Each blowdown line is fitted with:

- two SG secondary side isolation valves that are located in the Reactor Building, and represent the first level of SG isolation. One of these valves is located on the SG common hot leg blowdown pipe and the other valve is located on the SG cold leg blowdown pipe,
- one SG secondary side isolation valve, representing the second level of SG isolation, is located in the Reactor Building.

These valves ensure activity retention in case of a SGTR; prevent a partial and simultaneous draining of two SGs; and also provide the capability to isolate three of the four SGs in order to have a higher blowdown flow rate for the fourth SG.

The high pressure portions of the SGBS, including the SGBS flash tank, are located in the Reactor Building. The arrangement also keeps the piping lengths short between the SG and the flash tank.

Each SG blowdown stream is routed to its dedicated flash valve which adjusts the blowdown flow rate and permits an expansion of the blowdown. The four flash valves are attached to the flash tank which provides liquid/gas phase separation.

The SG blowdown is cooled to maintain the required operating conditions for the demineralizers of the SG blowdown demineralising system. A regenerative heat exchanger, located downstream of the blowdown flash tank and cooled by main condensate, cools the flashed liquid to the required temperature. The blowdown flash tank and blowdown cooler are located in the Reactor Building.

The cooled blowdown condensate is treated in the SG blowdown demineralising system which consists of filtration, cationic bed demineralization, and mixed bed demineralization.

The radioactive and chemical characteristics of SG blowdown are monitored by the nuclear sampling system and plant radiation monitoring system. Sampling is provided at the SG outlet on the cold and hot leg blowdown lines (because the sludge concentration may be different in the hot and cold legs) upstream of the SG isolation valves. Therefore, sampling the SG secondary side and measuring its chemical and radioactive characteristics is possible even if the SGs are isolated. Secondary side samples that come from the nuclear sampling system are re-injected upstream of the SG blowdown demineralising system.

8.3 Reactor Boron and Water Make-up System

The RBWMS supplies makeup water and boric acid to the RCS via the CVCS to control the boron concentration in the RCS during slow reactivity changes

The system performs the following functions:

- supplies boric acid and/or deaerated and demineralised make-up water in a ratio equal to the actual concentration in the RCS system via the CVCS for control of the reactor coolant level in the PZR,
- supplies the RCS with boric acid and/or deaerated and demineralised make-up water, via the CVCS system, for control of slow reactor reactivity variations that result from plant start-ups and shutdowns, variations in the neutron-absorbing fission products in the fuel (xenon and samarium effect) during load-follow operation, and fuel burn up and burnable poison burnout,
- batch preparation of fresh 4% boric acid (7,000 ppm boron) enriched in 10B,
- storage of 4% boric acid in two separate tanks,
- supplies the required amount of boric acid for filling and makeup via the Fuel Pool Purification System (FPPS) to the fuel pools and IRWST,
- supplies boric acid filling and makeup (7,000 ppm boron) to both tanks of the EBS,
- as a backup to the safety-related EBS, the RBWMS can be used to add 7,000 ppm borated water to the RCS in the event of an ATWS.

The boric acid mixing tank is used to produce fresh boric acid by dissolving nuclear-grade boric acid powder enriched in ^{10}B in warm demineralised water to obtain a concentration of 4% H_3BO_3. The boric acid feed pump is utilized for initially filling the boric acid storage tanks of the RBWMS and the tanks of the EBS with boric acid having a concentration of 4%. The 4% boric acid is diluted with demineralised water to initially fill the SIS and accumulators, the IRWST, the RCS, and the fuel pools with diluted boric acid of 1,700 ± 100 ppm boron depending on the concentration necessary for refuelling. During normal plant operation, the preparation of H_3BO_3 is only necessary to replace losses and (in rare cases), to replace depleted boric acid.

The requirements for tank inspection and the layout of the Fuel Building lead to dividing the storage capacity for boric acid into two tanks. After the initial filling has taken place, the boric acid (4%) comes from the evaporator column of the Reactor Coolant Treatment System. For operational redundancy, two boric acid pumps with downstream control valves are installed to deliver the required amount of boric acid to the CVCS. The pumps are normally lined up to one tank but are capable of pumping from either tank. The boric acid pumps and control valves are powered from the emergency buses, backed with EDG power.

Excess coolant coming from the RCS is separated into boric acid and demineralised water in the Reactor Coolant Treatment System. The demineralised water is stored in the coolant storage tanks, one of which is always connected to the suction side of the demineralised water pumps. For operational redundancy, two pumps with downstream control valves, each having a capacity of 100%, are installed to provide the appropriate amount of demineralised water to the CVCS. The demineralised water pumps and control valves are also powered from the emergency buses, backed with EDGs.

During normal system operation, both boric acid pumps are lined up to their respective storage tank and both demineralised water pumps are lined up to the Coolant Storage and Supply System. One boric acid pump and one demineralised water pump is designated as the primary pump. The Reactor Control and Surveillance Limitations (RCSL) system provides a signal to automatically start and stop the primary pump in each train. When the level in the VCT decreases to a low limit, automatic make-up is initiated and water at the proper concentration is added to the CVCS. When the VCT level reaches its high limit the system is automatically shutdown.

The major components which contain boric acid are located within the Fuel Building and the demineralised water components are located within the Nuclear Auxiliary Building.

Figure 8-2 shows a simplified flow diagram of the RBWMS.

8.4 Gaseous Waste Processing System

Radioactive fission gases, among them xenon and krypton, are generated in the reactor core. A portion of these gases is released to the reactor coolant if fuel cladding defects occur. Additionally, hydrogen is added to the reactor coolant by the CVCS for the purpose of oxygen control. Since the gases are dissolved in the reactor coolant, they are transported to various systems in the plant as a result of process fluid interchange. Because of the explosive nature of hydrogen in a mixture with oxygen, the amount of these gases in components of the auxiliary systems is controlled.

The GWPS performs the following tasks:

- compensates for the level deviations of the free gas atmosphere in the connected tanks by injecting or accommodating the corresponding gas volume,
- prevents the escape of radioactive gases from the connected components into the building air by maintaining a negative system pressure,
- flushes components in which coolant degasification occurs with nitrogen in order to process the waste gases,
- limits the hydrogen content in the system and in the flushed components to less than 4% by volume and the oxygen content to less than 0.1% by volume in order to prevent the formation of a combustible mixture. (This also prevents absorption of oxygen by the reactor coolant, thereby preventing corrosion in the RCS),
- handles the excess gas flow rates arising from the several systems connected to the GWPS during start-up and shutdown of the plant,
- permits decay of the noble gases to an acceptable radiation level prior to their release to the environment.

The following safety functions are performed by the GWPS:

- containment isolation,
- activity retention and contribution to limiting releases to the environment. The GWPS limits the hydrogen concentration in the connected systems in order to prevent the formation of explosive mixtures and processes radioactive gaseous wastes so as to minimize personnel exposure to radiation. Performance of these tasks will effectively control the release of radioactive gaseous wastes to the environment to limit total radiation exposure to personnel in accordance with the relevant regulations and ALARA standards.

The GWPS is designed for all normal operating conditions of the plant. The different systems connected to the GWPS consist mainly of tanks and vessels that contain a variable volume of free gas. GWPS design criteria are listed below.

- prevent the release of radioactive gases from the connected systems and components into the atmosphere of the Radioactive Waste Building. This is ensured by exhausting gases originating in the RCS and maintaining a sub-atmospheric pressure in the flushing part of the GWPS,
- minimize the discharge of gases to the environment by using a closed loop GWPS in which the flushing gas nitrogen is reused after reduction of the H2 and O2 content,
- contain the radioactive gases (xenon, krypton) for a sufficient decay time and release to the Nuclear Auxiliary Building ventilation system,
- handle the flushing gas flow from the RCS in case of nitrogen flushing in mid-loop operation,
- maintain a positive pressure in the delay beds to optimize the gas storage capability of the delay line,

- use activated charcoal for delaying the noble gases to reduce the necessary component volume of the delay line,

- limit the oxygen concentration in the GWPS to < 0.1% by volume in order to prevent absorption of oxygen by the reactor coolant which could cause corrosion in the RCS,

- limit the hydrogen concentration in the GWPS to < 4% by volume in order to prevent the formation of an explosive gas mixture with oxygen (the limits of flammability of such a mixture are 4% H2 by volume and 5% O2 by volume),

- reduce the hydrogen and oxygen concentration in the flushing gas. For this purpose, a catalytic recombiner is installed in the GWPS.

Components connected to the GWPS are flushed with sufficient quantities of nitrogen to limit the hydrogen concentration below the lower flammability limit of 4% by volume. The flow rate design criteria of the waste gas compressor and of the several flushing lines are based on the maximum hydrogen amount arising from the flushed tanks and vessels as well as the maximum flow rate during the surge gas operation mode. Parallel operation of both waste gas compressors is possible.

The recombiner is designed to handle the full flushing gas flow at a maximum concentration of 4 vol. % hydrogen and 2 vol. % oxygen. The required volume of catalyst depends on the type of catalyst and the maximum gas flow rate.

Although the reaction starts at ambient temperature, the catalyst is heated to about 100°C by the installed heating elements to guarantee efficient operation. This prevents moisture from precipitating onto the catalyst material, which might impair the reaction capability of the recombiner.

The noble gases xenon and krypton are retained in the delay line by adsorption until radioactivity has decayed to a level permissible for release to the vent stack.

The single failure criterion is applied for functions concerning containment isolation.

All electrical loads of the GWPS are connected to the emergency power supply.

8.5 Liquid Waste Processing System

During operation of the nuclear power plant, liquid wastes are produced by system drains, leakage, flushing, and other processes. The EPR has a liquid radioactive waste processing and storage system that performs the collection, short-term storage, processing, and cleaning of the waste streams produced by letdown, drainage, purge, venting, or leakage from systems in the controlled area.

The Liquid Waste Processing System is designed to perform the following tasks:

- activity retention and limiting releases to the environment,

- collect and segregate liquid effluents produced by the RCS, reactor auxiliary systems, reactor cavity and the spent fuel pool, as well as all potentially contaminated liquids produced in the plant such as floor drains, laundry, and decontamination wastes,

- route the collected waste to the storage and processing facilities,
- manage, under administrative and automatic controls, the waste water fed to the wastewater collecting tanks according to the collection, treatment, and discharge capacity of the Liquid Waste Processing System.

Total storage capacity for the system corresponds to the average quantity of liquid effluents produced on a weekly basis.

Oil removal equipment is provided in radioactive liquid waste systems (sumps) prior to subsequent treatment.

The system provides for analysis of the contents of each storage tank and subsequent treatment so that the quality of the treated liquid is acceptable and can be discharged to the environment.

Prior to discharging the processed wastewater to the environment, the following tasks are performed by the system:

- measuring the volumes of the liquid effluents to be released,
- measuring the activity of the liquid effluents to be released,
- determining and recording release rates.

The processed wastewater is monitored during discharge. The discharge line is automatically isolated if an authorized limit is exceeded.

Processed water discharged to the environment via the Liquid Waste Processing/Storage System will comply with local regulations. The liquid waste water can be pumped from the monitoring tank only if the concentration of radionuclides in the monitoring tank is in accordance with local discharge regulations.

In the event of damage to a collection tank or a storage tank, the damage or failure would not lead to a radioactive discharge from the plant due to the following design considerations:

- the room where the tanks are installed can retain the entire volume of the tank,
- the capacity of the other tanks in the system will accommodate the volume of a damaged tank.

In the event of leakage from pipes, the drains collect such leakage and convey it to the sump where the sump pumps are designed to route the leakage into one of the collection tanks.

The Liquid Waste Processing System plant discharge valve is supplied with emergency power.

8.6 Nuclear Sampling and Hydrogen Monitoring System

The nuclear sampling and hydrogen monitoring system provides centralized and local facilities for obtaining liquid and gaseous samples from the primary and secondary circuits, liquid and gaseous waste treatment systems, and from auxiliary systems, to determine the characteristics of these fluids by subsequent measurements and analyses. In addition, the hydrogen monitoring function measures hydrogen gas concentrations in the containment after an accident.

The operational functions of the nuclear sampling and hydrogen monitoring system are to collect liquid and gas samples for analysis, monitoring, and surveillance of the following systems:

- RCS,
- RHRS,
- CVCS,
- coolant purification system,
- spent fuel pool cooling and purification system,
- SIS (including IRWST),
- SGBS,
- boron recycle evaporator,
- reactor coolant gas stripper,
- reactor coolant storage system,
- boric acid storage system,
- GWPS.

The primary elements and parameters that are monitored by this system are:

- boron,
- oxygen,
- hydrogen,
- pH,
- specific conductivity,
- cation conductivity,

- sodium,
- lithium,
- noble gas activity concentration of the reactor coolant.

After an accident, the nuclear sampling system provides the following information:

- boron concentration of the reactor coolant and reactor coolant activity (gamma wide range),
- secondary side activity in case of a SGTR to support the detection of the affected SG.

Samples of the primary coolant can be taken via the sample points located downstream of the LHSI pumps. Gaseous samples of the containment atmosphere are provided by the system to determine the nuclide specific activity concentration. A sample of the IRWST water can be taken by means of the CHRS.

The following safety functions needed to contain radioactive substances are performed by the Nuclear Sampling and Hydrogen Monitoring System:

- RCPB isolation,
- containment isolation,
- provide post-accident information on reactor coolant boron concentration and activity level,
- provide post-accident information on secondary side activity,
- post-accident sampling of containment atmosphere,
- post-accident measurement of hydrogen concentration of containment atmosphere.

System samples are categorized as radioactive liquid samples, slightly active liquid samples, secondary (normally inactive) samples, and gaseous samples. The samples which fall into these different categories and their collection/sampling point(s) are listed below.

Radioactive Liquid Samples

- Liquid samples are drawn from the reactor coolant (including a post-accident situation) at the primary loops, PZR, RHRS, and CVCS (upstream and downstream of the purification unit).
- Liquid samples are taken from the boron recycle system, more precisely from the storage tanks, downstream of the mixed bed filter, and downstream of the boric acid metering pumps.
- Liquid samples are taken from the four trains of LHSI.

Slightly Active Liquid Samples

- Liquid samples are collected from the SIS, in particular from the accumulators.

- Liquid samples are taken from the boron recycle system and the boron makeup system, specifically from the distillates downstream of the after cooler; downstream of the degasified column of the coolant degasification system; and from the boric acid storage tanks.

- Liquid samples are drawn from the Fuel Pool Cooling and Purification System, specifically from downstream of the heat exchangers, and from the Reactor Building and Fuel Building pool purification loops.

Secondary Samples

- Liquid samples are collected from each SG at three different locations: the SG itself near the feedwater nozzle level, and the hot and the cold legs of the SG blowdown system.

- Liquid samples are taken from the SG blowdown system, downstream of each demineralizer and filter.

Gaseous Samples

Movable connections allow the sampling of the gaseous phase of a coolant storage tank and the sampling of the GWPS upstream of the gas dryer; upstream of the recombiners; and upstream and downstream of the delay beds. Gaseous samples can also be taken locally from the vent and drain tanks.

8.6.1 Post-Accident Sampling System

The Post-Accident Sampling System is designed to provide gas and liquid samples of the containment atmosphere following an accident. When a sample is taken from the containment atmosphere, aerosols and iodine are retained in the scrubbing liquid of the pool sampler with a retention efficiency of about 99%.

In the event of an accident, the atmosphere of the containment may include radiolysis products (hydrogen and oxygen), steam, particulates, and radioactive fission products (noble gases, radioactive aerosols, and iodine). The number and amount of isotopes as well as the gas composition in the atmosphere following an accident can provide an indication of the temperature history of the core during the accident and thereby provide information on the physical state of the core. Early information concerning the extent of the damage and the conditions prevailing in the containment after an accident is important for accident management measures.

The scrubbed gas, which contains mainly noble gases, is transported to the gas sampling module for dilution of the sample with nitrogen. After sampling of the noble gases, a sample of the scrubbing liquid is removed from the pool sampler. This liquid, containing aerosols and iodine, is transported by means of pressure pulses through the sample pipes to the scrubbing liquid sampling module for diluting with demineralised water. After dilution, both the gas and the liquid samples are transported to the sampling box where a sample can be taken with a syringe.

All modules, the sampling box, and the local control cabinet are located in the Nuclear Auxiliary Building. To ensure protection of the operating staff while taking a sample in the sampling box, all modules and pipes that may convey highly contaminated fluids and gases are located behind a biological shield.

8.6.2 Hydrogen Monitoring System

The H_2 concentration or gas composition in the containment atmosphere arising from LOCAs or severe accidents is measured at representative points inside the containment. Measurements are displayed and recorded in the Main Control Room. The system is used to assess the hydrogen release and distribution inside containment and the efficiency of the hydrogen reduction system. The data provided by the system are used for accident management measures.

A significant number of measurement points are arranged in the containment for the measurements during accident conditions. The locations of the measuring points are based on the gas distribution during design basis accidents and severe accidents and are tentatively located in the lower and upper SG compartments, the PZR compartment, the PZR valve room, the containment dome, and the IRWST. Due to the different measurement and qualification requirements in severe and design basis accident scenarios, different measurement points may be selected.

The nuclear sampling system has redundant sampling points and containment penetrations. The post-accident sampling system and the hydrogen monitoring system are supplied with emergency power.

8.7 Heating, Ventilation and Air Conditioning Systems

The HVAC systems function is to contain radioactive substances and reduce radioactive releases to the environment for normal operating modes and transients, as well as abnormal events.

As supporting systems, the HVAC and Chilled Water Systems maintain ambient conditions for equipment and personnel within acceptable limits (temperature and fresh air flow rate) to ensure the correct operation of safety-related systems and habitability in the Main Control Room.

The HVAC and Chilled Water Systems are classified into the following four categories.

Systems that contribute to the reduction of radioactive releases

- **AVS**: used during normal operation and accident conditions. The annulus is maintained at sub-atmospheric pressure to collect any leakage through the inner containment. The leakage is filtered, and then vented to the stack.

- **Nuclear Auxiliary Building and Fuel Building Ventilation System**: used only for normal operational service.

- **Containment Purge System**: used only for normal operational service. A part of the exhaust duct and containment penetration is used in combination with the Safeguard Buildings controlled area ventilation system for fuel handling accidents.

- **Safeguard Buildings Controlled Area Ventilation System**: used during normal operation and during plant accident conditions.

- **Radioactive Waste Building Ventilation System**: used only for normal operational service.

- **Access Building Ventilation System:** used only for normal operational service.

Systems which maintain the ambient conditions necessary for safety-related systems and components

- **Reactor Building Ventilation System**: used for normal operational service.

- **Main Control Room Air Conditioning System**: used during normal operation and during plant accident conditions.

- **Electrical Unit of the Safeguard Building Ventilation System**: used during normal operation and during plant abnormal conditions.

- **Diesel Building Ventilation System**: used during normal operation and during plant accident conditions.

- **Service Water Pump Building Ventilation System**: used during normal operation and during plant abnormal conditions.

Chilled water systems to support the safety-related ventilation systems

- **Safety Chilled Water System**: used during normal operation and during plant accident conditions.

Balance of Plant HVAC and Chilled Water System for operational service

- Ventilation/Air Conditioning System for Switchgear Building.

- Ventilation System for Turbine Building.

- Ventilation System for Circulating Water Pump Building.

- Space Heating System.

- Chilled Water System for Balance of Plant.

- Air Humidifying System.

- Ventilation System for Auxiliary SG Building.

- Circulating Water Seal Pit Building Ventilation System.

Systems that contribute to the reduction of radioactive releases filter the exhaust flow using HEPA filters and iodine filters (or recirculation in the containment cooling ventilation system) and release the gaseous waste to the stack.

Systems that maintain the ambient conditions necessary for the safety-related components also maintain ambient conditions within acceptable limits (temperature, humidity, fresh air flow rate, radioactive contamination, and cleanliness) for equipment and personnel access and habitability in the Main Control Room.

Chilled Water Systems to support the safety-related ventilation systems maintain chilled water circulation in the ventilation coils.

Ventilation and filtering systems that reduce the concentration of radioactive substances in the plant atmosphere (thus preventing the spreading of radioactive substances to other plant quarters) or restrict the environmental releases of radioactive substances, are capable of operating at their design conditions in the event of a single failure during normal operational conditions and design basis and accident conditions.

The inlet air filtering system of the Main Control Room and the rooms required for operations during accidents is capable of accomplishing its safety function even in the event of a single failure during operational conditions and accidents.

For safety-related systems that reduce the concentration of radioactive substances in the plant atmosphere and restrict environmental releases of radioactive substances during normal plant operation, the fans are redundant for maximum availability and not supplied by emergency power. The filters are non-redundant because only slow failure modes are assumed (filtration efficiency is checked periodically).

For safety-related systems that reduce the concentration of radioactive substances in the plant atmosphere and restrict environmental releases of radioactive substances during abnormal plant operation, the fans and filters are redundant. The fans are supplied by emergency power sources, if needed.

For filtering of air intake for the Main Control Room, the fans and filters are redundant. Fans are supplied by emergency power sources.

Ventilation systems that ensure ambient conditions for safety systems fulfil the same redundancy requirements as the safety systems they support.

The intake air centers and intake air systems of buildings containing systems important to safety are designed and located so that the spreading of smoke is unlikely.

The intake air of the Main Control Room is filtered for poisonous and contaminated gases and aerosols. The intake air center of the Main Control Room is equipped with a device that can be closed if poisonous or contaminated outside atmospheres are detected.

Emergency power supplies are provided to those systems that are involved in reducing radioactive releases; are used during accidents; maintain the ambient conditions during accidents (necessary for the safety-related components); and are cooling systems that support the safety-related ventilation systems.

8.8 Sampling Activity Monitoring System

The sampling activity monitoring system obtains representative airborne radioactivity samples during normal operation and accident conditions by:

- sampling at a constant flow from the condenser evacuation system exhaust and from other compartment exhaust air ducts,
- sampling the plant stack isokinetically. The stack is the pathway for gaseous radwastes and the exhaust outlet for the ventilation systems.

The safety function performed by the sampling activity monitoring system is to obtain representative samples to determine airborne concentration activity.

The Reactor Building, Fuel Building, Nuclear Auxiliary Building, and mechanical area of the Safeguard Buildings have separate ventilation circuits ending in a common discharge stack. Samples are taken from each circuit and from the discharge stack during normal operation and accident conditions. Samples are also taken from the exhaust air ducts of the Access Building and the Radioactive Waste Building.

The air samples are passed to the sampling and measuring instruments along the shortest path and returned to the exhaust air system. The adsorption losses of gaseous iodine and radionuclides bound in aerosol particles are kept sufficiently small by using suitable routing and materials.

Two separate exhaust air sampling systems, each with two 100% rotary compressor fans and separate sampling lines are provided. These rotary compressor fans are connected to the emergency power system, but not to the SBO diesels.

FIGURE 8- 1: STEAM GENERATOR BLOW DOWN SYSTEM

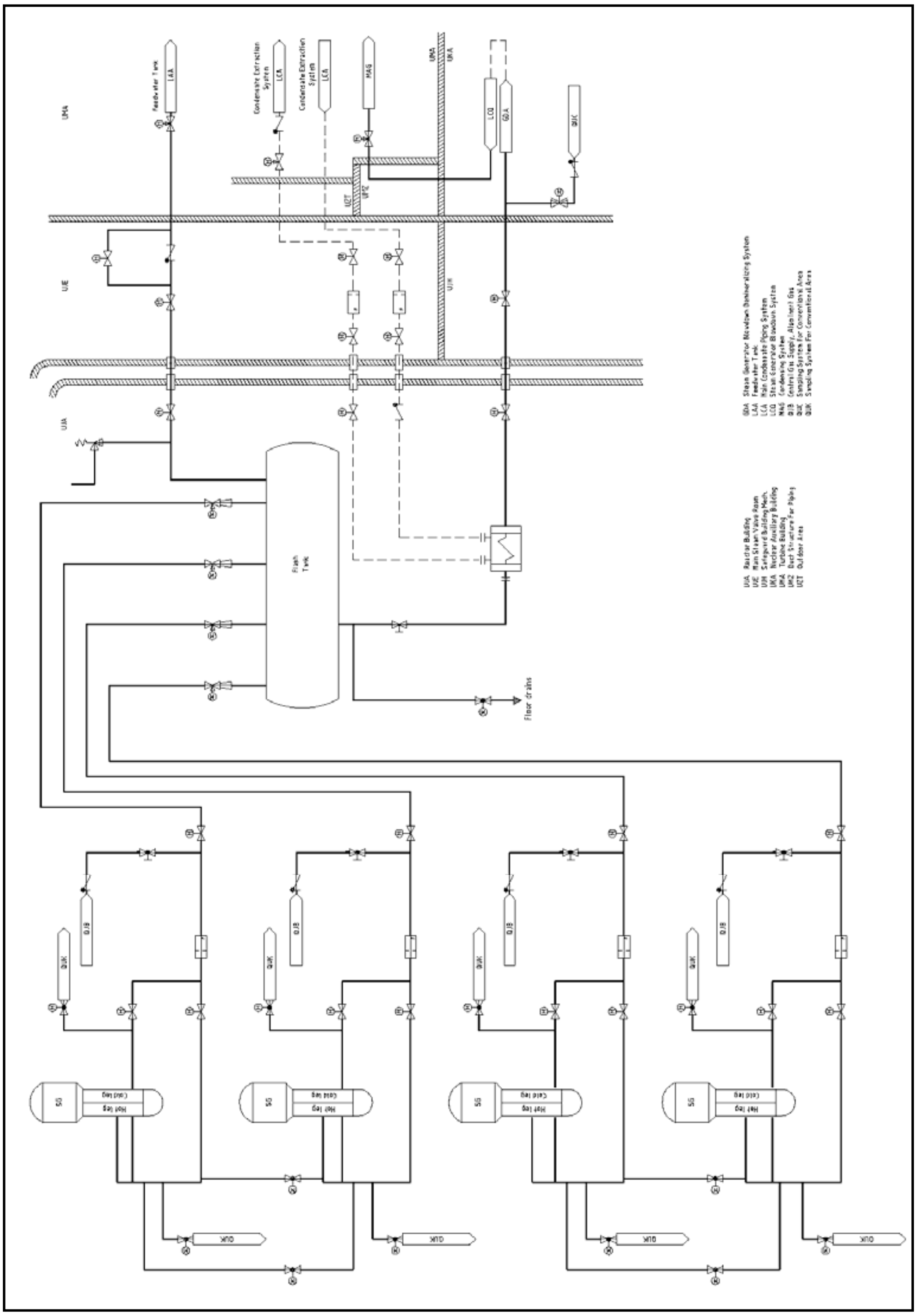
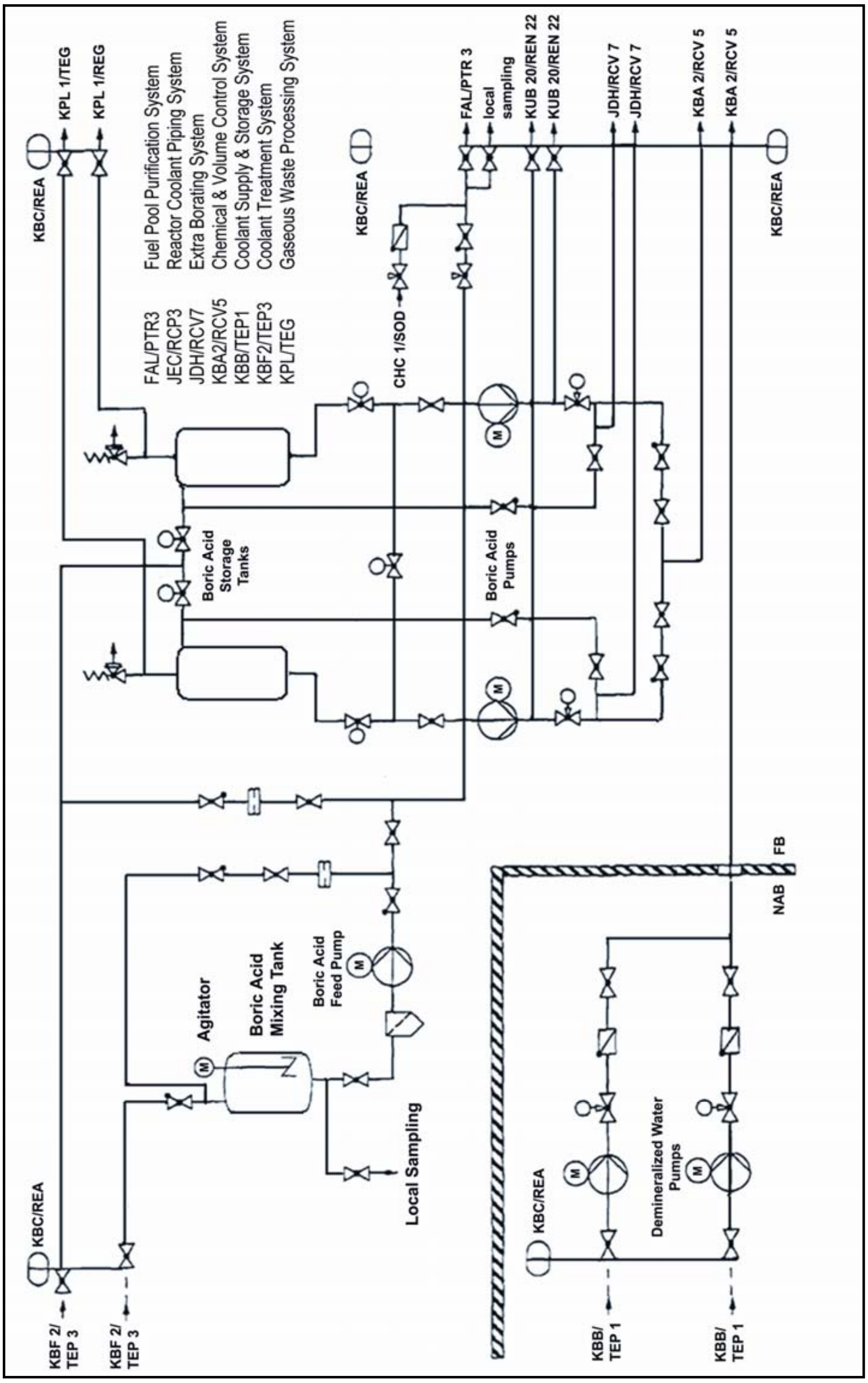


FIGURE 8- 2: REACTOR BORON AND WATER MAKE-UP SYSTEM



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9.0 TURBINE ISLAND DESIGN

9.1 Turbine building

The Turbine Building contains the components of the steam-condensate-feed water-cycle, including the turbine and generator set. The building is divided into two bays: the main bay with the turbine generator set, the condensate system, and the feed water heating system; and the service bay with the deaerator, feed water storage tank, and the feed water pumps.

The steam and power conversion system includes the Main Steam System (MSS), the turbine generator, the main condenser, the feed water system, the feed water storage tank, and other auxiliary systems.

The main condenser condenses the turbine exhaust and transfers the heat rejected in the cycle to the circulating water system. Regenerative feed water heaters heat the condensate and the feed water and return it to the SGs. A feed water storage tank is integrated into this cycle to deaerate and heat the condensate. This tank also provides a buffer volume to accommodate minor system transients.

The following parts of the steam and power conversion system have safety-related functions with respect to RHR:

- EFWS,
- MSS inside the Nuclear Island,
- MFWS inside the Nuclear Island.

The Turbine Building is independent of the Nuclear Island such that internal hazards in the Turbine Building remain confined. The building is located in a radial position with respect to the Reactor Building to provide protection from turbine missile impact.

The Switchgear Building is part of the balance of plant. It contains the power supply and the I&C, and is located next to the Turbine Building. Both of these buildings are designed not to adversely impact the Nuclear Island.

The main circulating water lines are routed in the basement of the Turbine Building. The low pressure (LP) drains cooler; the conventional closed cooling water system; and pumps that require a high suction head are located in the basement. The main condenser is positioned crosswise to the turbine axis and occupies the space below the LP turbine. The turbine oil system with main oil tank, filters, coolers, and pumps is installed in a dedicated fire zone below the turbine generator.

Two vertical moisture separator/reheaters are installed in front of the high pressure (HP) turbine and extend over several floors. Located within the main bay are LP and HP heaters, the vacuum pumps, and the generator auxiliary system. The MFW pumps are arranged at elevation ±0.00 m in the service bay.

Additional LP and HP heaters are arranged within the main bay on an intermediate floor elevation.

The generator bus ducts are located at the intermediate level and cross above the Turbine Building entrance bay and leave the building toward the generator transformer bank.

9.2 Main Steam System

The MSS routes the steam produced in the four SGs to the HP turbine inlet valves.

Each main steam line has a MSIV located just outside the containment. A bypass line with shut-off and control valves is provided around each MSIV for warming the piping system downstream from the cold condition. After pressure balance has been achieved between the secondary side of the SGs and the main steam lines in the Turbine Building, the MSIVs are opened and the warm-up valves are closed.

Overpressure protection on each main steam line is provided by a Main Steam Relief Train (MSRT) and two MSSVs. Each MSRT consists of a Main Steam Relief Isolation Valve (MSRIV) and a downstream MSRV. The MSRIVs are fast opening valves that are normally closed. The MSRVs are normally open control valves. The MSRIVs open quickly in the event of an overpressure transient. The MSRVs allow termination of flow through a stuck open MSRIV.

In addition to steam supply to the main turbine, the MSS supplies backup auxiliary steam for miscellaneous uses such as deaerator pegging. Additionally, the MSS features a non-safety grade turbine bypass to the condenser for operational flexibility.

The safety functions of the MSS are to provide reactivity control, RHR, and containment of radioactive substances.

Figure 9-1 shows a simplified flow diagram of the MSS.

Reactivity Control

The MSS does not directly affect reactivity control, however, the safety-related portion of the MSS indirectly supports reactivity control by isolation of the steam lines in the event of excessive steam flow. An excessive increase in steam flow causes overcooling of the reactor coolant and thus, positive reactivity feedback to the core.

In the event of a steam line break, quick closure of the steam line isolation devices enables the broken line to be isolated to limit the cooldown of the RCS and the energy release, so that the allowable limits specified for the fuel and the design conditions for the RPV and the containment are not exceeded. The isolation devices stop the steam flow (or two-phase mixture) that may flow through them in the normal flow or reverse flow direction.

Residual Heat Removal

The MSS removes residual heat by steam dump to the condenser via the turbine by-pass (if available) or to the atmosphere via the MSRT from the hot shutdown condition until RHRS entry conditions are reached.

In the event of a small or intermediate break LOCA or SGTR, the MSS cools the primary side down to the MHSI pressure by means of the MSRTs (i.e., partial cooldown) or turbine bypass (if available).

Main steam release to the atmosphere is designed so that the fuel temperature remains within specified limits and the RCPB remains within the design conditions.

Heat removal is performed even in the event of a loss of external power combined with a single failure (failure to open one MSRT).

Operational Functions

The MSS supplies main steam to the turbine and all other main steam consumers in the turbine building during normal operation, and removes the residual heat by steam transfer to the condenser during non-power operation. The main steam is transported through four lines from the SGs via the main steam valve stations in compartments on top of the Safeguard Buildings to the main stop and control valves of the HP turbine.

From each of the SGs, the main steam flows in a main steam line out of the Reactor Building via the valve compartment, into the Turbine Building and up to the turbine valves.

A main steam valve station consists of:

- one MSIV,
- two MSSVs,
- one MSRIV,
- one MSRV.

The MSIV is welded into a straight piping section between the containment penetration and a fixed point downstream of the penetration. The MSIV is an oil-pneumatic operated gate valve.

The MSRV is a motor-driven control valve that is welded into the discharge piping downstream of the MSRIV.

The MSRIV is a fast opening, open–shut valve that is welded to the main steam line section between containment penetration and MSIV.

The two MSSVs are spring-loaded safety valves. Each one is welded onto the main steam line section between the containment penetration and MSIV.

The warm-up line incorporates one motor-driven isolation valve and one motor-driven control valve.

Downstream of the MSIVs, pipes branch off the main steam lines to the turbine bypass station. The heating steam for the two steam reheaters is extracted between the main stop and control valves of the HP turbine.

Following expansion in the HP turbine, the steam is dried, reheated and fed to the three double-flow LP turbine cylinders. Each LP cylinder is assigned a condenser in which the steam condenses following expansion in the LP turbine. The heat of condensation is removed by the condenser circulating water system.

The HP and LP turbines comprise a permanently coupled unit with the generator. During start-up or on turbine shutdown, main steam is dumped directly into the condensers via the turbine bypass.

Condensate which collects in the condenser hotwell is pumped through four stages of LP feed water heating and delivered to the deaerator by the condensate pumps.

Feed water is pumped from the deaerator through two stages of HP feed water heating and delivered to the SGs by the feed water pumps. The drains accumulating in the feed water heaters, reheaters, moisture separators, and in drains traps downstream of the third stage of feed water heating are cascaded back and subsequently pumped forward by a drain pump. Drains upstream of the third stage of feed water heating cascade back to the condenser. The feed water control valves are located in the valve compartments and are accessible at all times. A swing check valve is located inside the containment upstream of each SG.

During start-up and shutdown, the SGs are supplied with feed water by means of the SSS.

The required degree of purity of the water in the steam/water cycle is maintained by means of the SGBS. The blowdown water is cooled, cleaned, and returned to the steam/water cycle. The required makeup water for the cycle is conditioned in a demineralising system, stored in the demineralised water storage tank, and fed as required to the cycle.

To ensure heat removal from the RCS via the SGs, three systems for supply of feed water to the SGs are provided, namely the MFWS, the SSS, and the EFWS with the latter being a safety system.

Two additional possibilities for steam dumping are provided with either the condenser or the atmosphere acting as the heat sink. The latter path is designed on the basis of safety considerations.

Under normal operating conditions, there are no detectable radioactive contaminants present in the steam and power conversion system. The system is monitored for increases in radioactivity by means of the main steam line monitors (N16, noble gases), the SGBS, and the activity monitoring system for the condenser evacuation system (non-condensing gases extracted from the condenser).

9.3 Main Feed water System

The MFWS extends from the feed water tank through the feed water pumping system, the HP feed water heaters, feed water isolation valves, and up to the SG main feed water inlet nozzles. During normal power operation, the feed water supply to the SGs is provided by the 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 feed water reliability.

Figure 9-2 shows a flow schematic of the MFWS and SSS.

The MFWS discharges feed water from the feed water tank by the feed water pumping system via the feed water piping system to the SGs. Feed water is heated in two HP feed water heater stages by the turbine extraction steam system. The condensed steam is cascaded back by the heater drains system to the feed water tank. These systems are not required to operate during or after an accident. The system layout ensures that no malfunction of any component or piping of these systems will affect the safe operation of the plant or any system which is important to safety. Only the function of MFWS containment isolation is important to safety. Thus, the portion of the MFWS from the main feed water containment isolation valves and feed water piping system (from the isolation valve inlets to the SG main feed water inlet nozzles) is safety class. The safety requirements of the MFWS are described below.

For accident scenarios, the MFWS participates indirectly in the reactivity control function by closure of the main feed water isolation valve and the full-load and low-load isolation valves so as to prevent an overcooling transient due to SG overfeed.

During normal operation, the MFWS controls the SG supply of feed water at the required flow rate as long as the start-up and shutdown system is available.

To provide containment of radioactive substances in the event of a SGTR, the affected SG is detected and isolated. The MFWS provides isolation of the affected SG during a SGTR by means of the main feed water isolation valve and full-load and low-load isolation valves. Thus, the potential for release of reactor coolant to the environment is minimized.

The single failure criterion is applied to the isolation valves of the MFWS to provide safe isolation of the feed water supply by the MFWS and the start-up and shutdown system.
The isolation valves of the MFWS are provided with emergency power backup so their functions can be performed in the event of a loss of off-site power.

The main feed water piping system from the inlet of the MFWS containment isolation valves up to the SGs fulfils the following safety functions:

- prevention of excessive mass flow by isolation via the main feed water isolation valve and the high and low load isolation valves to prevent SG overfeed,
- retention of radioactivity in the affected SG in case of a SGTR by main feed water isolation,
- control of the feed water supply from the start-up and shutdown system to the SGs.

The feed water piping system supplies feed water from the feed water tank to the SGs during power operation and start-up/shutdown operations. The supply of feed water is provided by the feed water pumps or by the start-up and shutdown pump.

The feed water piping system outside the turbine building up to the SGs conveys the feed water leaving the feed water heating system to the SGs, and controls the SG water level by means of full-load and low-load control valves. The feed water piping system outside the turbine building up to the SGs shuts off the feed water supply in the event of a feed water control malfunction, thus preventing overfeeding of the SGs. The feed water piping system from the inlet of the MFWS containment isolation valves up to the SGs performs the following functions during accidents:

- isolates the SG in the event of feed water line breaks,
- shuts off the feed water supply in case of main steam or feed water line break to prevent containment over pressurization,
- retains the radioactivity in the affected SG in the event of SGTR,
- isolates the SG in case of LOCA to prevent containment bypass,
- prevents depressurization of the unaffected SGs in the event of a non-isolable feed water line break inside the containment,

- prevents depressurization of the SGs in the event of an isolable feed water line break,
- reduces overcooling in the event of a main steam line break.

9.4 Turbine

The turbine generator is of the tandem compound design and consists of a double flow HP turbine and a six-flow low-pressure turbine solidly coupled to a three phase synchronous generator with a directly connected exciter.

No failure of the turbine steam system, lubricating oil system, or other system connected to these systems affects any other systems, components, or structures which are important to safety. Pertinent operating parameters are continuously monitored and alarms are actuated upon violation of specified limits.

Turbine trip is initiated if the integrity of any system or component important to turbine operation is endangered. No failure of rotating parts will impair the capability of the reactor to be shut down safely or of the RCS to be cooled down.

The turbine and its auxiliaries are manufactured, erected, tested and commissioned in accordance with the manufacturer's standard practices and in accordance with applicable codes to ensure high reliability of all systems and the mechanical integrity of the turbine generator set.

There is no radioactivity in this system during normal operating conditions. In the event of SG tube leakage, the small amount of radioactivity which may be present in the secondary system is detected by the main steam activity detectors, the SG blowdown processing system, and the condenser evacuation system.

Shaft integrity of the turbine-generator is maintained under all normal operating modes, transient conditions, and worst-case failure conditions. The worst-case failure is the loss of one last stage blade. The integrity of the rotor train is maintained by an appropriate bearing, bearing casing, and bearing pedestal anchor bolt design. The mechanical design of these components is set by the dynamic excitation forces due to the loss of a single last stage blade. The forces originate either from the impulse of a single last stage blade loss at over speed or from the unbalance excitation (i.e., loss of last stage blade at over speed and subsequent shutdown of the turbine-generator).

FIGURE 9- 1: MAIN STEAM SYSTEM (SAFETY CLASSIFIED PORTION)

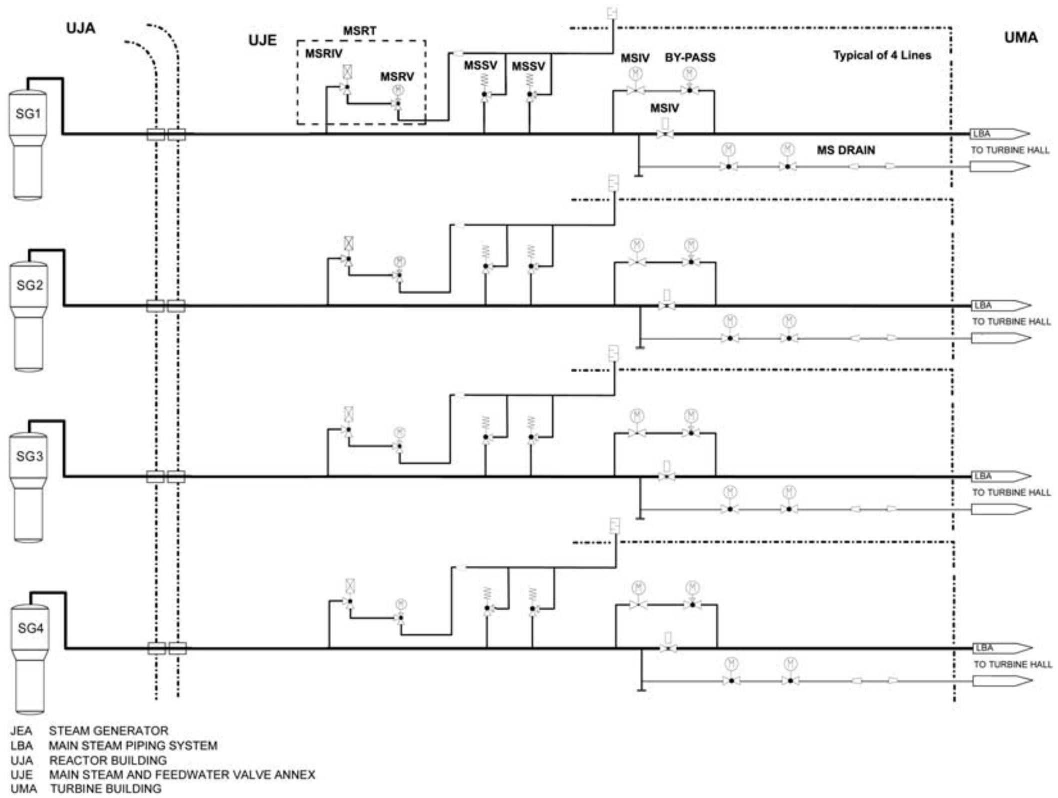
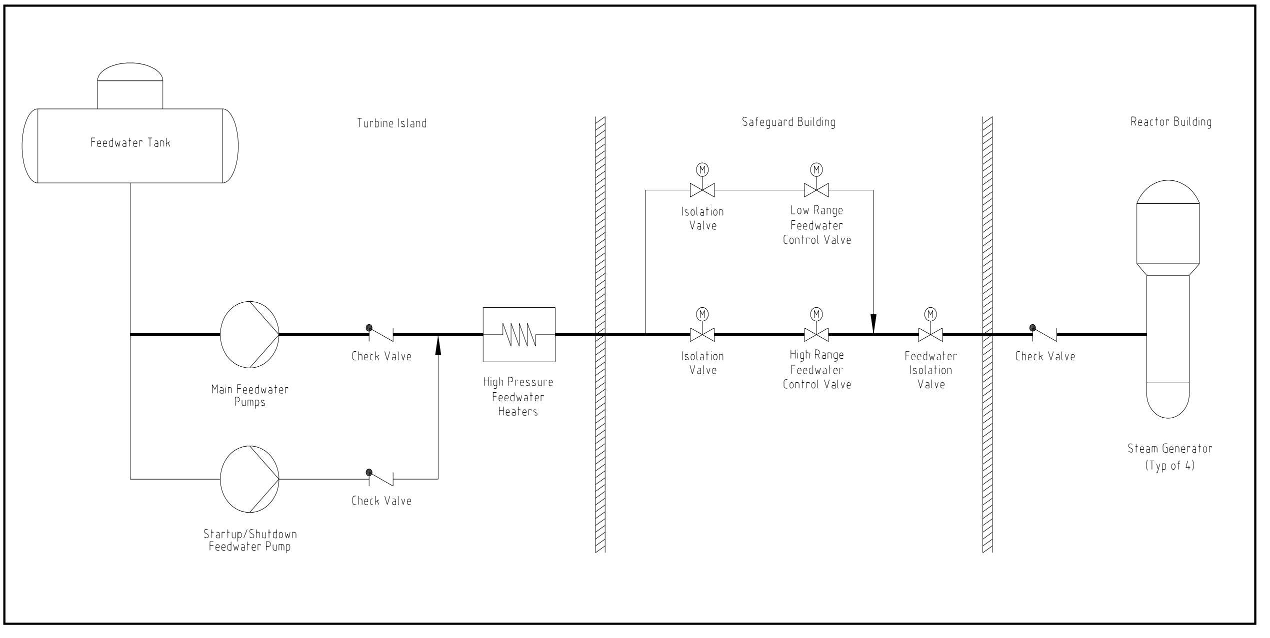


FIGURE 9- 2: MAIN FEED WATER SYSTEM AND START-UP/SHUTDOWN SYSTEM



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10.0 PLANT OPERATION AND REFUELLING

The main operating modes are described below in a chronological sequence from core loading at the beginning of a cycle to core unloading at the end of the cycle. Stretch-out operation is also considered. (Only the primary side operation is described).

10.1 Core Loading

The assumed starting point is the maintenance/refuelling state when all the fuel elements are in the SFP. The reactor vessel head is removed and the RPV internals are stored under water in the internals storage pool, separated from the reactor cavity by a water-tight removable gate. The fuel transfer compartment and the instrumentation lances compartment are flooded with their respective water-tight removable gates removed. The gates separate these compartments from the internals storage pool.

After reclosing the primary components (e.g., SG man ways and RCP seals), the reactor cavity is filled with borated water (at refuelling concentration) taken from the IRWST by one LHSI pump. The gate separating the reactor cavity and the internals storage pool is removed, the transfer tube isolation valve is opened, and the fuel is loaded into the RPV by means of the fuel handling devices (spent fuel mast bridge, transfer tube, and refuelling machine). After fuel loading, the upper internals are installed, the rod control cluster assemblies (RCCAs) are reconnected to the control rod drive lines, and the core instrumentation lances are installed.

10.2 Reactor Coolant System Closing and Filling

After closing the transfer tube, the reactor cavity, the internals storage pool and the transfer pool are drained to the IRWST through the SFP purification pumps and filters. Draining is continued through the CVCS letdown line (using the RHRS/CVCS connection) down to mid-loop level[4]. The RCS level is automatically controlled by the CVCS in order to prevent core uncovery and to provide safe RHRS operation. After the reactor vessel is closed using the multi-stud tensioning device, coolant degassing is performed via the CVCS and connected systems (full flow degasifier). The electrical connections of the control rod drive mechanisms and core instrumentation are installed. During this operation the reactor coolant temperature is controlled by the LHSI/RHRS.

The primary system above mid-loop, including the SG tubes, is cleaned by means of the vacuum pump to minimize the oxygen content of the primary coolant. Then the RCS is fulfilled by borated make-up (degassed water) from the RBWMS through the CVCS charging pumps. The pressure in the gaseous phase is kept low by the vacuum pump.

10.3 Reactor Coolant Heat-up

In parallel with the above operation on the primary side, the secondary side is usually also made available. In particular, the SG secondary side water level is brought to its zero load setpoint.

The primary pressure is raised to approximately 2.5 MPa by switching on the PZR heaters.

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[4] For the EPR, mid-loop actually corresponds to a water level of 75% of the pipe height (3/4-loop).

The maximum rate for the PZR heat-up is 100°C/h. During RCS pressurization, venting of the RCS is performed. After reaching the required pressure, the RCPs are started. This operation is made possible by the prior operation of all necessary auxiliaries (seal injection by CVCS, motor and bearing cooling by CCWS, and electrical power supply). After RCP start-up, the RCS pressure control (via PZR heaters and normal spray) is set in the automatic mode. The power of the four RCPs, in addition to possible decay heat of the fuel, heats up the primary coolant at a prescribed heat-up rate controlled by the LHSI/RHRS up to approximately 120°C. The main steam bypass system and SSS control the heat-up to the hot shutdown condition. The excess volume due to the primary coolant expansion is removed from the RCS by the CVCS letdown line (automatic control of the PZR level) and directed to the primary coolant storage tanks for recycling. In parallel to the coolant heat-up, the primary pressure is brought progressively and automatically to approximately 15.5 MPa by operation of the PZR heaters while ensuring a sufficient subcooling margin.

During this process, at the required pressure or temperature, periodic tests can be performed (e.g., PZR safety valve operability at approximately 4 MPa, main steam valves in hot shutdown) and appropriate systems are made available or configured for higher pressure (e.g., MHSI, SIS accumulators, safety injection, or reactor trip signals interlocked by permissive signals).

10.4 From Hot Shutdown to Power Operation

Before hot shutdown conditions are reached, all safety systems are available. If sufficient negative reactivity margin exists, the reactor coolant is deborated by injection of demineralised water from the RBWMS through the CVCS charging pumps (deboration can start during the last phase of reactor coolant heat-up). The withdrawn primary coolant is stored in primary coolant storage tanks for recycling. The control rods are then lifted out of the core until the core goes critical (it is also possible to go critical by deboration with all rods or some of the rods extracted first). The reactor coolant temperature is controlled indirectly by the main steam bypass (the SG pressure is automatically controlled by the main steam bypass; the thermal power is manually controlled by the rod position). Zero power tests are performed, and the power is then increased to 25% of nominal power. The SG level is controlled by the SSS until 3 to 4% of nominal power is reached, at which point the SGs are fed by the MFWS. While the reactor is between 10% and 25% nominal power, the turbine is started and the generator is synchronized to the main grid, and the power is progressively shifted from the main steam bypass to the turbine-generator set. At this power level, all RCS controls are in the automatic mode and the power is gradually increased up to 100% nominal power. An additional deboration is performed during the first fifty hours of the power generation for xenon build-up compensation.

10.5 Power Operation

When the plant is base-loaded at or near rated power, infrequent planned power changes are made by boration or dilution of the RCS and limited RCCA movement.

In load follow operation, control rods are inserted or extracted by the Temperature and Power Distribution Controls in order to follow short-term reactivity effects (power effects) while long-term effects (Xenon variation) are usually managed by deboration or boration actuation.

10.6 Reactor Shutdown

At the end of the fuel cycle, the power is reduced to zero and the control rods are dropped. The reactor coolant temperature is then indirectly controlled by the main steam bypass. The SG level is controlled by the SSS. Periodic tests can be performed in the hot shutdown conditions and the reactor coolant is borated. The reactor coolant is cooled to approximately 120°C by the main steam bypass with the four RCPs in operation at a prescribed cool down rate. In parallel, the primary pressure is reduced to approximately 2.5 MPa while ensuring a sufficient subcooling margin.

Boration can also be performed during cool down. At approximately 120°C and 2.5 MPa, two RCPs are stopped and two LHSI/RHRS trains are connected and started up to continue RCS cool down. Below 100°C, the last two LHSI/RHRS trains are also connected and started to provide increased cooling capacity. All but one RCP in operation is stopped at 70°C, and the last one is stopped at 55°C when the RCS has been oxygenated and purified. This results in reaching a reactor coolant temperature of 55°C within approximately 16 hours after the reactor is tripped.

The lost volume resulting from the primary coolant contraction is compensated for by the CVCS charging pumps and the RBWMS pumps from the boric acid and demineralised water storage tanks.

After the last RCP is stopped, the PZR pressure is brought to approximately 0.5 MPa by the auxiliary spray. This pressure is controlled by the CVCS letdown pressure control during the final cooling of the liquid phase to 55°C.

10.7 Reactor Coolant System Draining and Opening

The reactor coolant is drained to mid-loop level by the CVCS letdown line through the RHRS/CVCS connection and the drained coolant is sent to the primary coolant storage tanks for recycling. The water level is reliably controlled (automatic level control via CVCS) to prevent core uncovery and provide safe RHRS operation. If the radioactivity content of the primary coolant makes it necessary, the gaseous phase is swept by nitrogen and sent to the GWPS. After a final sweeping with the gas being sent to the stack through the HVAC systems, opening of the RCS is possible.

The electrical connections of the control rod drive mechanisms and core instrumentation are removed. The multi-stud tensioning device is installed and used for vessel head removal.

10.8 Core Unloading

While the vessel head is lifted, the reactor cavity, the internal storage pool and the transfer pool are flooded with borated water taken from the IRWST by one LHSI pump. The core instrumentation is removed and the RCCAs are disconnected. The upper internals are removed to their storage pool, the transfer tube is opened, and fuel handling can start. The reactor coolant temperature is kept below 55°C by LHSI/RHRS trains.

In addition to the decay heat of previously stored fuel elements, the heat load of the SFP in the Fuel Building increases by the decay heat of the unloaded core. Therefore, the second SFP cooling train is operated to keep the SFP below the required temperature.

After complete core unloading, the water-tight gate between the reactor cavity and the internals storage pool is installed and the RCS is drained to the IRWST through SFP purification pumps and is available for in-service inspection and SG tube inspections.

10.9 Stretch-Out Operation

In power operation, the reactivity available for burn-up is compensated by poisoning the coolant with boron. As burn-up increases, the boron concentration is continuously reduced. The end of the cycle is reached when the boron concentration falls to a value approaching zero.

To continue power operation beyond the natural end of cycle, the decrease in reactivity associated with burn-up is compensated by reducing coolant temperature. With the control rods almost fully withdrawn and the turbine admission valves fully opened, the plant power level is determined by the reactivity balance of the reactor and the turbine characteristic.

As there is no longer any reactivity reserve available to ensure a constant average coolant temperature, the average coolant temperature and the reactor power, as well as the main steam pressure decrease steadily. The coolant mass in the RCS is kept constant during power operation. Thus the decreasing coolant temperature leads to a continuously falling PZR level.

When the minimum permissible PZR level is reached, the RCS inventory is increased to allow continuation of the stretch-out mode. This marks the end of the first stretch-out phase. The limit is determined by the requirement to ensure the minimum PZR inventory in the event of a reactor trip. In parallel with the increase of the PZR level to its full-load setpoint, the secondary-side MSRT setpoints are reduced to accommodate the coolant expansion volume in the PZR in the event of a loss of main heat sink when the main steam bypass is not available for main steam pressure limitation.

By means of repeated setpoint adjustments as described above, the stretch-out operating mode exhausts the reactivity reserves.

Stretch-out operation is considered a standard operation for the EPR that can be systematically applied at the end of each fuel-cycle to increase the discharge burn-up. This type of operation is made possible by simple commands from the Main Control Room without the need to manually adjust setpoints. This stretch-out process applies without interruption and the setpoints of controls, limitations, and protections are modified automatically.

10.10 Mid-loop Operation

The RCS is operated at mid-loop in the start-up and shutdown sequences associated with refuelling or maintenance, as follows:

Start-up

- Pool and vessel flange cleaning.
- Coolant degassing and vessel head closure.
- Evacuation of the primary system via the vacuum pump.

Shutdown

- Sweeping of primary system by nitrogen.
- RPV opening.
- Work on RCP seals, if needed.

Mid-loop operation is preferred during evacuation and sweeping of the primary system because it permits draining of the SG, thus making evacuation and sweeping more efficient (filling of SG tubes in the first case and avoiding degassing of water in SG tubes in the second case).

Mid-loop operation is conducted while cleaning the pool and the vessel flange. Since mid-loop automatic level control is available, this cleaning activity is performed preferably at mid-loop, unless radiological exposure while cleaning the vessel flange and pool and during handling of the vessel head proves unacceptable. The subsequent degassing of primary water is performed at the same level as the pool and vessel flange cleaning, either at mid-loop or at flange level.

The RCS loop level control contributes to the control of the RCS water inventory safety function during mid-loop operation at shutdown. The control is based on a comparison between the measured RCS loop level and a reference level and gives a command signal to the CVCS low pressure letdown flow control valve. The RCS loop reference level is determined to ensure a sufficient RCS water inventory for RHRS/LHSI operation and satisfy maintenance requirements. A high or low RCS loop level generates alarms or automatic limitation action, such as closure of the CVCS low pressure letdown flow path in the event of low level. The RCS loop level measurements are also used to actuate protective I&C functions in the event of a failure of the above means. Actuation of safety injection will occur upon a RCS loop low level signal.

The main consequence of loss of level prior to core uncovery is the loss of the RHR function through cavitation of the RHR pump, since the NPSH of these pumps is lower and a vortex can occur at the suction in the hot leg. Cavitation is prevented by:

- reduced RHRS flow velocity,
- an anti-vortex device,
- automatic control of the level in the loop,

In the event of a loss of level at mid-loop operation, three degrees of mitigation are available:

- the RCS level control will intervene using the allowable imbalance between the CVCS charging flow and letdown flow,
- the CVCS letdown line isolation is initiated when the level is low enough for a vortex to be generated,
- the safety-related safety injection signal actuates the MHSI pump and containment is isolated, thus resulting in CVCS letdown line isolation.