Status report 74 - Indian 220 MWe PHWR (IPHWR-220)

Overview

Full name	Indian 220 MWe PHWR
Acronym	IPHWR-220
Reactor type	Pressure Tube Type Reactor
Coolant	Heavy Water
Moderator	Heavy water
Neutron spectrum	Thermal Neutrons
Thermal capacity	754.50 MWth
Electrical capacity	235.81 MWe
Design status	In Operation
Designers	Nuclear Power Corporation of India Limited (NPCIL)
Last update	04-04-2011

Description

Introduction

The Indian Pressurized Heavy Water Reactors (PHWRs) programme consists of 220 MWe, 540 MWe and 700 MWe units. At present India is operating 16 units of 220 MWe and one unit of 220 MWe is under advance stage of commissioning. Two units of 540 MWe are under operation. The design of 700 MWe units is in an advanced stage and as of now government sanction is available for four 700 MWe units.

This report presents information for 220 MWe Indian PHWRs. The operating 220 MWe units are listed below:

Table-1: List of Operating 220 MWe PHWR Units in India

Serial Number	Station / Plant	Unit	Status	Year of Commercial operation	Rated Capacity (MWe)
1	Rajasthan Atomic Power Station	RAPS 1,2	Operating	1973, 1981	RAPS-1 -150
					RAPS-2 - 200

Serial Number	Station / Plant	Unit	Status	Year of Commercial operation	Rated Capacity (MWe)
		RAPS 3,4	Operating	2000	2 x 220
		RAPS 5,6	Operating	2010	2 x 220
2	Madras Atomic Power Station	MAPS 1,2	Operating	1984, 1986	2 x 220
3	Narora Atomic Power Station	NAPS 1,2	Operating	1991, 1992	2 x 220
4	Kakrapar Atomic Power Station	KAPS 1,2	Operating	1993, 1995	2 x 220
5	Kaiga Generating Station	KGS 1,2	Operating	2000	2 x 220
		KGS 3	Operating	2007	220
		KGS 4	Operating	2011	220

The first PHWR units (RAPS-1 and RAPS-2) are of Canadian design (based on Douglas point). Work on these units was taken up with Canadian cooperation. For RAPS-1 most of the equipment were imported from Canada, whereas for RAPS-2 indigenization was achieved, and this unit was commissioned by India. MAPS-1&2 design was evolved from RAPS-1&2, with modifications carried out to suit the coastal site requirement and also introduction of suppression pool to limit containment peak pressure under loss of coolant accident (LOCA) in lieu of dousing tanks in RAPS-1&2. In addition, MAPS-1&2 have partial double containment. This design was further improved and all subsequent PHWR units in India have double containment. Due to observed problems in RAPS-1 end shields, the end shield material in MAPS-2 onwards was changed to austenitic stainless steel (from 3.5% Nickel Carbon steel).

With experience of design and operation of earlier units and indigenous R&D efforts, major modifications were introduced in NAPS-1&2. These units are the basis of standardized Indian PHWR units. The important features introduced in these units include: two diverse and fast acting shutdown systems, double containment of reactor building, water filled Calandria vault, integral Calandria - end shield assembly, zircaloy-2 pressure tube separated from calandria tube by 4 loose fit garter springs per coolant channel with inter space between pressure tube and calandria tube filled & purged with carbon dioxide to monitor pressure tube leak by monitoring dew point of carbon dioxide (In standardized Indian PHWR units, this system is modified to be of recirculation type).

The design of KAPS-1&2 was similar to that of NAPS units. However, material of pressure tubes in KAPS-2 was changed to Zr-2.5% Nb and loose fit garter springs were replaced by garter springs kept tight on pressure tubes (In standardized Indian PHWR units, garter springs are tight fit on pressure tubes).

The design of subsequent units i.e. KGS-1, KGS-2, RAPS-3, RAPS-4, RAPS-5, RAPS-6, KGS-3 and KGS-4 is of standard Indian PHWR design. The major improvements in these designs include valve-less primary heat transport system and a unitized control room concept. In addition, the design of these units included improvements in Control and Instrumentation system and incorporation of computer based systems to match with the advancement in technology.

The older Indian PHWR units have undergone major refurbishment to bring them at par with the latest units. For all old PHWR units, En mass Coolant Channel Replacement (EMCCR) campaign is either completed or in progress. With this, all Indian PHWR units (except RAPS-1) will have Zr-2.5% Nb pressure tubes with 4 tight fit garter springs. Station wise, some of the major modifications carried out in older units are:

<u>RAPS-1 and RAPS-2</u>

1. Incorporation of high pressure emergency core cooling injection;
2. Provision of supplementary control room with local panels/controls;
3. Segregation of essential and non-essential instrument air supply inside reactor building to isolate non essential requirement under LOCA conditions to limit gradual re pressurization of containment;
4. Segregation and rerouting of power and control cables;
5. Ensuring emergency power supply during flood condition, an additional emergency diesel generator is installed above maximum anticipated flood level;
6. Upgradation of fire protection system;
7. Calandria vault dew point monitoring system.

<u>MAPS-1 and MAPS-2</u>

In addition to the modifications listed for RAPS-1 and RAPS-2

1. Replacement of Motor Generator set with static Uninterrupted Power Supply in Class II power supply.
2. Installation of one class-III emergency diesel generator, one Class III air compressor and two fire fighting water pumps above maximum anticipated flood level.

All Indian PHWRs are being subjected to periodic safety review (PSR) and the important issues addressed as per this exercise include:

1. Seismic re-evaluation of old generation PHWRs;
2. Revision of safety analysis;
3. Revision of technical specification for operation;
4. Reduction in collective dose;
5. Optimization of in-service inspection programme.

Summary technical data of the standard Indian PHWR 220 MWe unit (RAPS-3&4) is given in Appendix-A.

The detail system design of standardized Indian PHWRs is covered in ensuing sections, the high level design features of standardized Indian PHWR of 220 MWe are given below in Table-2 (RAPS-3&4 as representative design):

Table-2

Serial Number	**System**	**Design features/rationale**
1.	Siting	In addition to conventional siting parameters, the plant is sited considering - Geological, seismological, meteorological and hydrological conditions, - Population distribution around site, - Away from air corridor, - Requirement of exclusion zone, sterilized zone, and Emergency Planning zone around site.

Serial Number	System	Design features/rationale
2.	Layout	- Twin unit module, with main plant buildings unitized. Some of the services are located in buildings shared by both the units; - Safety related buildings and structures are located in low trajectory turbine missile free zone; - Radiation zones inside operating island.
3.	Safety and Quality Classification	Graded approach as per safety significance of SSCs with applicable requirements of codes, standards, testing. Safety classification : Safety Class 1 to 4 and Non Nuclear class Seismic classification : SSE Category, OBE Category and General category Quality classification : Quality class 1 to 4
4	Reactor Type	Pressurized heavy water reactor using heavy water moderator and primary coolant and natural uranium dioxide as fuel.
5	Fuel	Natural uranium dioxide as fuel with Zircaloy – 4 as cladding. 19 element fuel bundles are of 0.5 m length and each pressure tube contains 12 fuel bundles.
6	Reactor Core	The reactor consists of integral assembly of horizontal cylindrical calandria and two end shields; one at each end of the calandria vessel, The calandria vessel is submerged in light water filled calandria vault. There are 306 coolant channel assemblies. The fuel bundles are contained in pressure tubes, also called coolant tubes of zircaloy 2.5% niobium. The 306 pressure tubes are arranged in 228.6 mm pitch. At each end, pressure tubes are rolled into stainless steel end fittings, which penetrate end shields and extend into fuelling machine vault so as to facilitate on power fuelling. At each end of the end fittings, removable shield and seal plugs are provided. The former provides axial shielding and the latter serves as leak tight mechanical joint. Each pressure tube is surrounded by a concentric calandria tube. The annular gap between pressure tube and calandria tube filled with carbon dioxide serves as thermal insulation between high temperature coolant inside pressure tube and low temperature moderator outside calandria tubes in the calandria vessel. This annulus gas is also used to monitor any leak from pressure tube or calandria tube.
7	Reactivity Control and Shutdown Systems	Reactor regulating system is for power control purposes. This system consists of 4 cobalt/ stainless steel regulating rods for power maneuvering, 8 cobalt/stainless steel absorber rods to provide xenon override capability and 2 cadmium sandwiched stainless steel shim rods for quick power reduction. Automatic liquid poison addition system (ALPAS) is provided to supplement reactor regulating system. Two diverse and fast acting shutdown systems are provided, each having adequate capability to suppress any fast reactivity transient

Serial Number	System	Design features/rationale
		under operating and accident conditions. Primary shutdown system (PSS) consists of 14 mechanical rods of cadmium sandwiched between stainless steel elements. PSS is designed to come first in case of reactor shutdown demand. The secondary shutdown system (SSS) provides fast filing of liquid poison in 12 vertical tubes located inside calandria. The worth of these shutdown system to take care of long term effects like xenon decay is augmented by liquid poison injection system (LPIS), which injects poison directly into the moderator.
8	Fuelling Scheme	On power bi directional refueling with eight bundle shift scheme is achieved by two fuelling machine operating in conjunction at two ends of the reactor. With operation of fuelling machine and fuel transfer equipment, spent fuel is transferred to spent fuel storage bay for long term cooling.
9	Primary Coolant System	The primary heat transport system removes the heat generated in the core through steam generators for normal power operation. Two circulating pumps on each side of the reactor are connected to one reactor inlet header, from where coolant is directed to coolant channels through 153 inlet feeders. The coolant from the coolant channels flows to a reactor outlet header through 153 outlet feeders. The main circuit of primary coolant system is valve less. The nominal temperature rise in the coolant inside the core is 44^{o}C, i.e. from 249^{o}C at inlet to 293^{o}C at the outlet. As no boiling is allowed inside the core, pressure maintained at 87 kg/cm^{2} at the outlet headers provide adequate sub cooling margin. From outlet headers coolant is carried to four steam generators, two in each north and south banks. For protecting the heat transport system against over pressurization, in addition to the protective action of shutting down the reactor, over pressure relief valves are provided. The elevation difference between the core and steam generators provide driving head for hot coolant to flow to steam generators, when primary circulating pumps are not available and the reactor is in shutdown state. To bring coolant temperature below 150^{o}C and maintain the reactor in cold shutdown state, shutdown cooling system is provided. This system through closed loop process water system rejects heat to the atmosphere. Emergency core cooling system is provided to remove core heat following loss of coolant accident. This system operates in three phases incorporating high pressure heavy water accumulators, intermediate pressure light water accumulators and low pressure – long term recirculation system. For catering to smaller leaks in the primary coolant system, a separate system called small leak handling system is provided.
10	Moderator System	Heavy water in calandria, maintained below 70^{o}C by a circulation and cooling system. The moderator system equipment are provided with onsite power supply.

Serial Number	System	Design features/rationale
11	Secondary System	This system provides heat sink for the heat transported from the core. This system consists of steam generators, turbine, condenser and feed water systems. The pressure in the secondary side is limited within permissible values by steam dump valves, atmospheric steam discharge valves and steam relief valves.
12	Containment and associated engineered safety features	Primary containment of pre-stressed concrete is enveloped by secondary containment of reinforced concrete. The annulus between inner and outer containment is maintained at a slightly negative pressure with respect to atmosphere to minimize ground level activity releases to the environment during accident conditions. The ventilation ducts and other lines opening to the containment atmosphere are automatically isolated in case of accident conditions sensing pressure or activity rise inside the containment. The containment is provided with engineered features, which are designed to come into operation after an accident- to cool the containment atmosphere, to limit the peak pressure, to clean up the containment atmosphere and for post accident controlled discharge.

Description of the nuclear systems

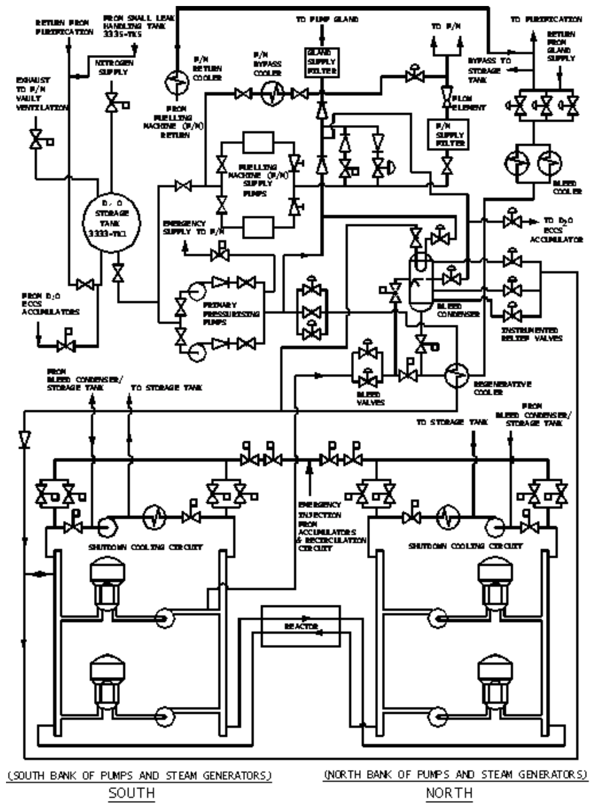

Figure-1: Primary Heat Transport System Schematic

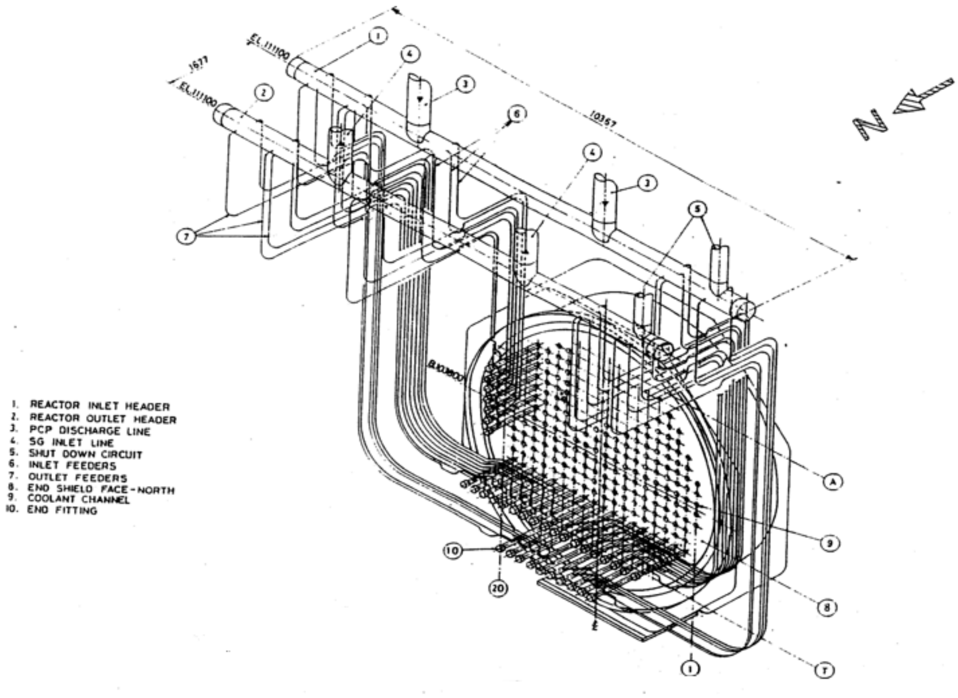

Figure-2: Primary Heat Transport System Feeders and Headers Schematic

2.1 Main Characteristics of Primary Heat Transport (PHT) System

The Primary Heat Transport (PHT) system has been designed with the objective of ensuring adequate cooling of reactor core under all operational states and during and following postulated design basis accident conditions. The system thus ensures that the fuel integrity is protected and radiological consequences are kept as low as reasonably achievable.

The Primary heat transport (PHT) system transports heat produced in the reactor core to steam generators to generate steam, which is fed to the turbine to generate electricity. The transport medium is pressurized heavy water. The principal features, which the system incorporates, are:

i. Continuous circulation of coolant through the reactor at all times by various modes as listed below :

Normal operation	:	By primary coolant pumps (PCPs)
Loss of power to PCPs	:	Initially by inertia of the pump flywheel and later by thermo syphoning (by placing steam generators above the elevation of reactor core).
Shutdown	:	By shutdown cooling pumps and heat exchangers (which are independent of steam generators)
Loss of coolant accident	:	By receiving emergency injection of heavy water and light water from pre charged accumulators while depressurization of primary heat transport system is taking place. After initial supply from accumulators is exhausted, long term core cooling is established by emergency core cooling system (ECCS) recirculation pumps and heat exchangers.

ii. The pattern of coolant flow rates through coolant channels is compatible with pattern of heat production in each channel of the reactor core to result in nearly equal increase in coolant temperature in all channels. This is achieved through four different sizes of feeders and orificing arrangement

iii. Controlled pressure at the reactor outlet headers for maintaining the system coolant subcooled.
iv. Over pressure relief to protect the PHT pressure boundary.
v. Addition of coolant to and removal from the system in order to control the coolant inventory in the main circuit.
vi. Pressure control of the PHT system by feed and bleed system.
vii. Layout of equipment to permit natural circulation of coolant for decay heat removal.
viii. Pressurized accumulators for heavy water and light water injection followed by a recirculation phase for emergency core cooling in case of loss of coolant accident (LOCA). Provision of selective header injection by ECCS in case of LOCA.
ix. Small Leak Handling System (SLHS) to make up PHT inventory in case of small leaks [within the capacity of primary pressurizing pump (PPP)].
x. Control of dissolved gases in reactor coolant.
xi. Purification and chemistry control of coolant.
xii. Provision for supply of high pressure heavy water to the fuelling machines.
xiii. Accessibility of all components during shutdown and accessibility of some during operation for limited duration.
xiv. Provision for header level control for maintenance of steam generator, primary coolant pump and other boundary valves.
xv. Heavy water leakage collection from potential leak points in the system.
xvi. Study of corrosion coupons in Autoclaves (during hot-conditioning / PHT decontamination).
xvii. Variable boiler pressure program scheme for steam generator pressure control

2.2 Reactor Core and Fuel Design

The Pressurized heavy water reactor uses heavy water as moderator and coolant and natural uranium dioxide as fuel. The reactor consists of an integral assembly of two end shields and a Calandria with the latter being submerged in the water filled vault. Fuel bundles are contained in 306 Zr-2.5%Nb pressure tubes, arranged in a square lattice of 22.86 cm pitch. At each end, the pressure tubes are rolled in AISI 403 modified stainless steel end fittings, which penetrate the end shields and extend into the fuelling machine vaults so as to facilitate on power fuelling.

The Calandria is a horizontal vessel containing the coolant channel assemblies, moderator and internal components of various shutdown mechanisms and reactivity control mechanisms. The main shell of the Calandria is stepped down in diameter at each end and known as small shells. Outside diameter and length of the main shell are 6.05 and 4.16 m respectively. Wall thickness of the main and small shells is 25 mm. The Calandria structure is fabricated from Austenitic stainless steel type 304 L. The design, fabrication, inspection and testing is in accordance with ASME section III NB.

The end shields provide shielding to limit the dose rate in the fuelling machine vault to an acceptable level during shut down. Also they support and locate the Calandria tubes and coolant channel assemblies in which the fuel resides. The dead weight of the Calandria – end shield assembly and its contents is transmitted to the concrete vault walls. The end shield is a cylindrical box whose ends are closed by the Calandria side tube sheet (CSTS) and the fueling side tube sheet FSTS). The box is pierced by 306 Stainless Steel Lattice tube arranged in 22.86 cm square lattice. The space inside the End shield is divided into two compartments by a baffle plate. The lattice tube, CSTS and the baffle plate are joined by a single weld joint. The front compartment is filled with water and the rear compartment is filled with water and carbon steel balls. The End shields are designed, fabricated and tested as class II components according to the ASME section III NC.

Around each pressure tube, a concentric calandria tube has been provided with an annular gap. Carbon dioxide gas filled in this gap serves as thermal insulation between the high temperature primary coolant and low temperature moderator. In addition, the annulus gas system is intended to detect leaks in the coolant tube/calandria tubes. Axial shielding to the coolant channel is provided by removable shield plug fitted in the end fittings. At the face of each end fitting, a seal plug is installed which serves as a leak tight mechanical joint and can be removed during fuelling operation. The coolant tubes/channels are connected, via end fittings and individual feeder pipes, to headers at both ends of the reactor.

The bulk of the space available in the calandria, i.e. around the calandria tubes, is filled with heavy water moderator, which is continuously circulated with the help of moderator pumps. On-power fuelling which is a characteristic feature of PHWR, is required on a continuous basis mainly in view of the use of natural uranium fuel.

Reactor control devices are required to regulate the reactor power, to control flux tilt, optimize fuel performance and for the start up process. Two diverse and fast acting shutdown systems supplemented by one slow acting shutdown system are provided as a part of protection system. These shutdown systems terminate fast reactivity transients under various operating and accident conditions and bring the reactor to safe shut down state.

Shutdown system no. 1 (PSS-Primary Shutdown System) consists of Fourteen PSS rods which are grouped into two banks of seven rods each with nearly equal worth. These rods are poised and kept at parked (out of the core) position against the initial acceleration spring by electromagnetic clutches which are normally energized. On a reactor trip signal, the clutches are de-energized causing Shut-off rods to drop inside the reactor core under gravitational force. The grouping of rods in two banks also helps in reducing electrical load on single circuit and to limit positive reactivity insertion during withdrawal during plant restart.

The shutdown system no. 2 (SSS-Secondary Shutdown System) consists of 12 vertical tubes located in the Calandria shell, which are grouped in four banks of nearly equal worth. Each bank has its independent high pressure helium gas tank and liquid poison tank feeding three shut-off tubes. The liquid poison tanks and cover gas atmosphere are isolated from the high pressure helium storage tanks through a group of valves. On a reactor trip signal, these valves actuate and the shut-off tubes are filled with solution of Lithium Pentaborate in heavy water.

The slow acting shutdown system (LPIS- Liquid Poison Injection System) is provided to take care of long term reactivity effects such as xenon decay. This system injects boric acid solution to moderator in calandria.

The reactor generates about 802 MW of total fission power out of which 756 MW is delivered to the primary coolant.

The fuel bundle consists of 19 cylindrical fuel elements of 495.3 mm length, held together by welding the elements to end plates at both ends. The elements are arranged in concentric rings of 1, 6 and 12 elements in different rings. Each element contains a 480 mm long stack of sintered natural UO_2 pellets in a thin zircaloy-4 cladding coated with 3 to 9 micro-meter thick graphite on inside surface with end caps welded at both ends. The pellets are of double dish chamfered type. The elements are separated by spacers attached to the cladding near the mid-plane of the bundle. Inter-element spacers are of the skewed split spacer type. One half of the spacer is attached to each of the neighboring elements such that half spacers contact each other at a skewed angle to reduce any tendency to 'lock' because of vibration. The design of the split spacers is such that the minimum inter-element spacing at the spacer location after maximum anticipated fretting wear will not be less than 0.89 mm. Bearing pads are provided on each element of the outer ring to prevent the fuel sheaths from touching the coolant tube. The 19-element fuel bundle has been designed to generate a bundle power of about 500 kW.

The operating experience in India and that reported internationally indicates that fuel bundles in PHWR may fail during reactor operation due to one or a combination of the following reasons:

1. Damage due to debris in the coolant,
2. Power ramp,
3. Overstraining of cladding due to high bundle power,
4. Manufacturing defect,
5. Handling defects.

Due care is therefore taken in design, manufacturing and commissioning of the reactor and operation to reduce fuel failure to negligible levels. For example the graphite coating has been introduced on the inside surface of the cladding to act as a lubricant between the fuel pellets and the cladding and also act as a barrier for fission products to avoid their direct contact with the cladding. The above effects will reduce the pellet cladding mechanical and chemical interaction (between fission products and cladding), which induce defects due to power ramps. Some fuel failures can be expected during reactor operation which release some fission products to the coolant. To detect and to locate the channel containing failed fuel, iodine monitoring (by PHT system D_2O sampling) and DN (Delayed Neutron) monitoring systems are provided. The channel having a defective bundle will be refueled as and when detected.

2.3 Fuel Handling Systems

On-power refuelling is an integral feature of Indian PHWRs. It involves opening and resealing of the high temperature, high pressure PHT boundary. The tasks associated with the on-power refuelling are performed by Fuel Handling System. It is a dynamic system with a number of components having complex mechanisms operating in different environments and at various temperatures and pressures. On-power refuelling is performed by a pair of

Fuelling Machines working in unison. These machines perform the complex operations of removal and installation of channel plugs namely Sealing Plug and Shielding Plug. Other major equipment of the Fuel Handling System which performs the task associated with channel refuelling are new fuel handling equipment and the equipment required for transferring the irradiated fuel to the storage pool. The movement of these mechanisms is achieved either by D_2O, oil, H_2O, air or electric operated actuators. The Process system provides the operating fluid at controlled pressure, temperature, flow and direction to enable precise and controlled movement and forces of various actuators. As the equipment are located in radiation areas, they are required to be operated remotely in auto mode. This requires use of various sensors and monitoring devices and a complex control system.

During refuelling, one Fuelling Machine is clamped on the upstream end and the other at the downstream end of the reactor channel to be refuelled. Before commencing the refuelling operation, new fuel bundles are loaded into the upstream fuelling machine. After clamping on the reactor channel, the Fuelling Machines remove the various plugs from the channel. Subsequently, the upstream Fuelling Machine loads the new fuel bundles and the downstream Fuelling Machine receives the irradiated fuel bundles.

Both Fuelling Machines can perform the function of loading and receiving of the fuel bundles. During normal operation, the fuel bundles are moved in and out of the reactor channel by push force only. The refuelling operation takes place with the reactor at normal operating pressure and temperature. Normally, 8 bundles refuelling scheme is adopted to refuel the reactor. During refuelling operation, direction of new fuel loading is from upstream to downstream of the channel in the direction of coolant flow.

Fuel Handling System design has evolved from the first generation 220 MWe reactors at RAPS / MAPS to the standardised 220 MWe PHWR at NAPS onwards. Some of the main features of these evolutionary designs are as follows:

2.3.1 Design of Fuel Handling System in First Generation 220 MWe PHWR

In the design of Fuel Handling System of first generation 220 MWe PHWR, the Fuelling Machine Service Area is a large open area located adjacent to Fuelling Machine Vault. The design of Fuelling Machine employs the concept of Carriage supporting the Guide Columns and moving horizontally on the rails fixed on the floor.

The Fuel Transfer System is based on the use of a common system for both sides of the reactor. It incorporates several equipment such as Air Lock, Transfer Arm and Shuttle Transfer System for transportation of irradiated fuel from the reactor building to storage pool for long term under water storage.

In this design, the Safety Interlocks Logic System (SILS) and Sequential Operational Logic System (SOLS) of Fuel Handling components / drives are implemented by making use of discrete logic gates that are hardwired. Operator interface of SOLS is provided through a large number of pushbuttons, lamps, hand switches and meters.

2.3.2 Design of Fuel Handling System in Standardized 220 MWe PHWR

Fuel Handling System in standardised 220 MWe PHWR at NAPS onwards has several evolutionary features which were aimed at improving system availability, enhanced safety and ease of maintenance. Major changes were incorporated in layout and the design of equipment. Some of the main features of the standardised 220 MWe PHWR are as follows:

1. In the first generation 220 MWe PHWR, Fuelling Machine Service Area is a large open area located adjacent to Fuelling Machine Vault. Transfer of Fuelling Machines to Service Area for maintenance results in spread of activity and heavy water vapor from vault into Service Area. The concept of fixed Guide Columns with Bridge moving vertically up and down along the Columns was adopted in standardized 220 MWe PHWR to solve this problem. In this concept, a fully enclosed Fuelling Machine Service Area of smaller size is located below each Fuelling Machine Vault which gets sealed when the Bridge sits on the hatchway connecting the Vault with the Service area. Also, the design based on bridge concept is better suited to withstand high intensity seismic events as compared to RAPS / MAPS Carriage design.
2. Transit equipment called Transfer Magazine was introduced in Fuel transfer system in place of Air Lock and Transfer Arm used in RAPS / MAPS. It can simultaneously load new fuel into the Fuelling Machine and receive the irradiated fuel from the same Fuelling Machine through the exchange mode. It also facilitates the parallel simultaneous operation of refueling by Fuelling Machines on the reactor and transferring of irradiated fuel from the Transfer Magazine to the storage pool through Shuttle Transport Tube. Independent set of Fuel Transfer equipment provided on each side of reactor also enhanced the system availability.

3. From KGS-1&2 onwards, irradiated fuel storage pool is designed as tank-in tank type with inner tank lined and having the provision of leak detection. It eliminates the possibility of contamination of ground water due to any leakage of pool water.
4. In the first generation 220 MWe PHWR design, the Safety Interlocks Logic System (SILS) and Sequential Operational Logic System (SOLS) of Fuel Handling components / drives was implemented by making use of discrete logic gates that are hardwired. Whereas for standardized 220 MWe PHWR, the SILS was implemented using IC based Transistor Transistor Logic (TTL) gates for reducing wiring and increasing the density of logic Printed Circuit Boards (PCBs).
5. In the first generation 220 MWe PHWR, operator interface of SOLS is provided through a large number of pushbuttons, lamps, hand switches and meters. This kind of Human System Interface (HSI) was not operator friendly to quickly diagnose the fault and respond to the problem. Therefore for the subsequent designs, the SOLS is computerized for better HSI, flexibility and control.
6. In order to carryout maintenance on various FH equipment, subassemblies and components, a separate maintenance facility is provided. It has several test rigs for testing and calibration of the critical components. In addition, a rehearsal facility is provided in the Fuelling Machine Service Area. It is used for rehearsing the refueling operations and to qualify the fuelling machine for on-reactor operation, whenever any maintenance is carried out on it.

2.4 Primary Heat Transport Circuit Component Description

2.4.1 Primary Coolant Pump (PCP)

Primary coolant pumps (PCPs) circulate coolant through the reactor core. PHT main circuit has four PCPs. The PCP is vertical single stage type with a radial impeller located inside a volute casing which has an axial bottom entry suction and horizontal radial discharge.

Each PCP is provided with three mechanical seals. Each of these mechanical seals can withstand full system pressure and thus provide a reliable pressure boundary sealing. This feature allows some breathing time for the plant to continue operation with caution before defective seal replacement is planned. The three mechanical seals are further backed up by a vapor seal to prevent leakage of heavy water vapor past the shaft.

PCP motors are connected to 6.6 kV off site electric power supply. Motor is equipped with a flywheel for increasing pump run down time, in case of loss of electric supply to PCP-motor.

2.4.2 Steam Generator (SG)

Steam Generators transfer heat from primary system to secondary system. Four SGs are provided in the PHT system. The Steam Generators are vertical mushroom type design with integral drum and feed water preheater. They are recirculation type heat exchangers, having inverted U tubes for primary flow. The shell side is designed for natural recirculation.

Essential components of SG are : Primary head, tube-sheet, cylindrical shell housing the tube bundle, extended steam drum which accommodates the steam separators and dryers. The primary head is sub-divided into two chambers, viz. Primary inlet chamber and primary outlet chamber. Man-holes are provided in primary head for in-service inspection of tubes and maintenance work including sleeving / plugging of defective tubes. Four hand-hole nozzles are provided a little above the tube sheet in the secondary cylindrical shell for visual inspection of the tube bundle and for tube sheet lancing. The internals necessary for steam drying can be inspected through a manhole in the upper steam plenum. Installation and replacement of steam separators or vanes of steam drier is through the manhole, as required.

Steam Generators are located in two concrete enclosures, one on either side of the reactor core. Thus two SGs on one side of the core are housed in one enclosure which is open at the top.

Hot primary coolant from the reactor outlet header enters the inlet chamber of the SG via one carbon steel pipe, passes through the tube bundle and enters the primary coolant pump via one carbon steel pipe. The feed water at 171^{o}C enters the steam generator through two feed water nozzles i.e. 90% feed enters the preheater portion of the SG above the tube sheet on cold leg side of the tubes and 10% feed in the steam drum.

The U-Tube bundle (Incoloy-800 tubes) is surrounded by a guide shroud. On its upper end (deck plate) centrifugal separators are bolted for separating the steam-water mixture. The shroud and the SG vessel wall form an annular down comer, in which the recirculating water from the separators flows down to enter the boiling region directly above the tube sheet. The separators carry out the coarse separation of the steam- water mixture passing through them. Steam water mixture from the separators is passed through steam dryers. Separators and dryers operate in tandem in the SG giving a final moisture content of less than 0.26% in the steam.

2.4.3 Headers and Feeders

The feeder pipe selected are 65 mm NB, 50 mm NB, 40 mm NB and 32 mm NB carbon steel pipes to SA-333 Grade-6 (with 0.2% chromium minimum). The feeder pipes are I.D controlled to avoid resistance variation in standard pipes. Accuracy in feeder resistance is essential as core flow distribution is sensitive to feeder resistance variation. Headers are made from carbon steel SA-350 Grade LF-2 class-1.

The arrangement of feeders along end-shield face is such that the gap between end fittings is utilized for connecting feeders to end fittings. Feeders connecting channels in rows 'A' through 'K' have runs vertically upwards while feeders connecting channels in rows ‘L’ through ‘T’ have runs horizontally outwards from core. Each of these runs are clubbed in groups called feeder banks. A maximum of ten feeders are grouped in one bank. Feeders and headers are housed in insulation cabinet in each fuelling machine vault.

Each channel is provided with coolant flow rate nearly proportional to its power rating. This is achieved by using suitable feeder pipe sizes in combination with restriction orifices in some of the inlet feeders. This leads to a near identical temperature rise across all channels.

2.5 Residual Heat Removal and Auxiliary Cooling Systems

2.5.1 Shut Down Cooling System

The steam generators provide highly reliable means for removal of core heat during reactor operation. Steam generators are also suitable for cooling down the primary circuit well below the normal operating temperatures efficiently. However, for cooling the system to below 150^{o}C and holding it cold enough for carrying out maintenance work, an independent cooling system is required due to limitation of cooling by steam blowing. The shutdown cooling system is provided for this purpose. There are 2x100% shutdown cooling circuits in the reactor, each having one shutdown cooling pump and one heat exchanger. The functions of the system are listed below:

1. Enables Primary Heat Transport System to be cooled down from 150^{o}C to 55^{o}C and maintain cold shutdown state (long term decay heat removal);
2. Maintains header level such that steam generators, primary coolant pumps and/or primary pressure boundary isolation valves can be opened up for in-service inspection and maintenance;
3. Provides flow for purification of the primary coolant when Main PHT Circuit is shutdown and in depressurized state;

The decay heat is transferred into Active Process Water (APW) system via Shutdown cooling system heat exchanger.

2.5.2 End Shield Cooling System

The End shields cooling system serves to remove nuclear heat generated in end shields and heat transferred from primary coolant across insulating gaps between end fittings and lattice tubes and across support bearings of coolant channels.

Heat to be removed from each end shield is comprising of nuclear heat and heat transferred from PHT system.

The demineralised cooling water from End Shield (ES) will contain N^{16} activity. Therefore, the return lines from ES outlet nozzles up to primary containment have been sized such that a transit time ensures decay of N^{16} activity. Cooling flow is provided for each end shield to remove the design heat load with a temperature rise of about 5^{o}C, which is low enough to keep the thermal stresses in the end shield components, viz. Lattice tubes, tube sheets, inner & outer shell and diaphragm plate within acceptable limits.

The cooling system has been designed adequately to meet cooling requirement under all states of reactor operation.

2.5.3 Calandria Vault Cooling System

The Calandria Vault (CV) Cooling System serves to remove the nuclear heat generated due to attenuation of neutrons and core gamma rays; gamma rays captured in Calandria shell, CV water, CV concrete wall and heat transferred from moderator system. The calandria vault is filled with demineralised water to provide shielding against nuclear radiation.

The design of the cooling system is based on the nuclear heat generation in the Calandria vault water and contribution by nuclear heat generated in calandria shell during full power operation. The cooling circuit is designed to remove the design heat load with a temperature rise of about 5ºC.

2.5.4 Spent Fuel Storage Bay Cooling System

Spent fuel storage bay cooling system serves to remove the decay heat generated in the spent fuel bundles and also to protect personnel from radiation (Beta & Gamma) while storing spent fuel bundles before sending them for further processing. The heat liberated from the spent fuel during its storage in pool water has to be dissipated over a long period before they are sent for final processing. The system water is purified to minimize activity build up and to keep the bay water clean for better visibility required for underwater operations during handling of spent fuel.

The spent fuel storage bay is designed to provide adequately cooled and shielded storage for spent fuel discharged over ten years from each of the reactor and also for storage of one full reactor charge.

2.5.5 Annulus Gas Monitoring System

Carbon di-oxide (CO_2) is circulated through the annuli between Coolant tubes and Calandria tubes. Monitoring of moisture content of CO_2 is done to assess the coolant tube integrity. High purity CO_2 is circulated through all the annuli continuously at a rate of 15 Normal m^3 /hr. The maximum permissible dew point of makeup CO_2 is (-) 20ºC. Provision is made for on-line dew point monitoring. Besides pressure rise in the circuit is also annunciated. Annunciation is given if the dew point of circulating CO_2 rises to (-)10ºC, which indicates leak from one or more of the coolant tubes. Manual shutdown of the reactor is initiated. The annulus gas tubes from individual channels are grouped into several strings and these are grouped in sub groups and a process of elimination identifies the sub group containing the leaky tube, isolating one group at a time. Tritium sampling arrangement from AGMS is provided.

2.5.6 Active Process Water and its Cooling System

Active process water (APW) system removes heat load from various process system heat exchangers like PHT system, Moderator system, End shield cooling system, Calandria vault cooling system etc. The APW system heat load is rejected into atmosphere via induced draft cooling tower (IDCT) in case of an inland site and into sea in case of coastal site.

2.6 Reactor Operating Modes

- Normal operation

Normal operation of the plant is within specified operational limits and conditions. This includes startup, power operation, shutting down, shutdown state, maintenance, testing and refueling.

- Hot shutdown state

Shutdown state of the reactor with primary coolant temperature (inlet to reactor) and pressure close to normal operating condition and the primary coolant pumps (PCPs) running is defined as hot shutdown state.

- Cold shutdown state

State of the reactor when it is maintained sub-critical with specified sub-criticality margin and temperature of the PHT system at inlet to the core is less than 55^0C.

- Guaranteed shutdown state(GSS)

A specified shutdown state of the reactor with sufficiently large reactivity shutdown margin, established by the addition of liquid poison into the moderator to provide positive assurance that an inadvertent increase in reactivity by withdrawal of all other reactivity devices cannot lead to criticality.

2.7 Standard Fuel Cycle

PHWRs use 'Natural' uranium in dioxide form as fuel. During the residence period in the reactor, about 1% of the uranium is burnt. India has limited reserves of uranium and vast reserves of fertile thorium. In view of this, India has adopted a closed end fuel cycle. The nuclear energy policy and consequently the nuclear fuel cycle policy of India is evolved based on this position on fissile and fertile fuel resources. The spent fuel bundles from PHWRs are reprocessed and the depleted uranium and plutonium is planned to be used in fast breeder rectors. A small quantity of reprocessed depleted uranium is recycled in PHWRs also, as given in next section. The Front-End of this cycle like mineral exploration, mining and processing of ore and fuel fabrication; and back end of the cycle, which includes fuel reprocessing, re-fabrication and nuclear waste management are carried out by different units of the Department of Atomic Energy (DAE), Government of India.

2.8 Alternative Fuel Options

Increase in fuel burn up beyond 15000 MWd/TeU using higher fissile content materials like slightly enriched uranium, Mixed Oxide and Thorium Oxide in place of natural uranium in fuel elements used in 220 MWe PHWRs is studied. Due to higher fissile content these bundles will be capable of delivering higher burn up than the natural uranium bundles. The maximum burn up studied with these bundles is 30000 MWd/TeU.

To satisfy specific reactor requirements, apart from natural uranium dioxide fuel bundles, reprocessed depleted uranium dioxide fuel bundles, Slightly Enriched Uranium Bundles (SEU), MOX bundles and thorium dioxide bundles were designed, developed and successfully irradiated in different 220 MWe reactors. Thorium bundles and reprocessed depleted uranium dioxide bundles were used for flux flattening in the initial core such that the reactor can be operated at rated full power in the initial phase. MOX-7 bundle design evolved is a 19-element cluster, with inner seven elements having MOX pellets consisting of plutonium dioxide mixed in natural uranium dioxide and outer 12 elements having only natural uranium dioxide pellets. The SEU bundle design is a 19-element fuel bundle with 0.9% SEU. Studies on reactor physics characteristics like reactor control shut down margin, fuel and other systems thermal-hydraulic and material compatibility have been carried out for each fuel type before taking up actual loading in Indian 220 MWe PHWRs.

3.1 General Safety Principles

Indian PHWRs are designed and operated to achieve the fundamental safety objectives in conformity to regulatory requirements of codes, guides and standards. The licensing process is well established with multi tier review carried out by NPCIL and the Regulatory Body. The well established principle and practice of defence-in-depth and ALARA are followed. In general, following safety principles and practices are applied.

- Defence in Depth;
- Safety systems are designed with requisite redundancy and diversity to achieve specified reliability targets;
- Fail safe design is adopted for systems important to safety;
- Routine testing of systems and safety systems having features so that they can be tested on power;
- Equipment qualification for the systems required to operate under accident conditions;
- Detailed safety analysis using both deterministic and probabilistic methodologies;
- Seismic design of SSCs in accordance with their safety significance;
- Physical and functional separation of items important to safety;
- Safety systems are subjected to a number of commissioning tests.

3.2 Illustration of Defence in Depth

The defence in depth is implemented to provide a graded protection against a large variety of transients, incidents and accidents, including equipment failures and human error within the plant and events initiated outside the plant.

- At first level of defence in depth, regulatory guides are used for the detailed design. Various national and international codes and guides are also referred. The emphasis throughout is to produce a robust design having sufficient safety margins so as to ensure safety under all normal operating conditions throughout the design life. Strict control is exercised during the manufacturing and commissioning processes to assure the reproduction of intended design.
- At second level of defence in depth, systems and procedures are in place to detect abnormal conditions and controlling them so as to minimize deviation from normal operation.
- Safety systems and Engineered Safety Features (ESF) are provided to mitigate the consequences of accidents within design basis e.g. Shutdown systems, Emergency Core Cooling System (ECCS), Containment and associated engineered safety features, etc.
- Complementary design features and use of non safety systems is envisaged at the fourth level of defence in depth.
- Procedures to implement counter measures in public domain in case of offsite release of radioactivity are available for all Indian PHWR units.

3.3 Licensing Process

Major stages identified for authorization for an NPP are Siting, Construction, Commissioning and Operation. The regulatory body adopts a multi-tier review process for safety review and assessment of NPP.

The first level of review and assessment is performed by the Site Evaluation Committee (SEC), the Project Design Safety Committee (PDSC) or the Civil Engineering Safety Committee (CESC), as appropriate. These Committees as a body are comprised of experts in various aspects of NPP safety. The next level of review is conducted through an Advisory Committee on Project Safety Review (ACPSR). This committee is a high-level committee with members drawn from the regulatory body, reputed national laboratories and academic institutions. It also has representation from other governmental organizations and ministries. After considering the recommendations of ACPSR and the first level committee, the regulatory board decides on the authorization.

The multi-tier review process is followed for operating units as well. The first tier of safety review is carried out by the 'Unit Safety Committee' consisting of representatives from the regulatory body and NPP under review and the experts in various aspects of nuclear technology drawn from different institutions. The second tier of safety review of Indian NPPs is by the Safety Review Committee for Operating Plants (SARCOP), which is the apex body to decide on the matters of nuclear safety pertaining to NPPs. The third tier is the regulatory board, which based on the recommendations of SARCOP, considers the major safety issues pertaining to operation of NPPs. The authorization for operation of NPPs is issued for a period of five years. The renewal of authorization is issued based on i) limited safety review of five years of operation and ii) comprehensive review every ten years of operation i.e. Periodic Safety Review (PSR).

3.4 SSC for Fundamental Safety Functions

3.4.1 Reactivity Control

The reactor regulating system is used for normal power maneuvering, including fast reduction of power as a setback action. The devices used for power control purpose are given below.

Serial Number	Purpose	Device / Equipment
1	Control and Regulation	Regulating Rods

Reactor shutdown is achieved by two diverse and fast acting shutdown systems supplemented by a slow acting poison injection system for maintaining long term sub criticality. The shutdown systems are so designed that the first shutdown system is the preferred mode of shutdown.

Serial Number	Purpose	Device / Equipment
1.	First Shutdown System	Primary Shutdown System
2.	Second Shutdown System	Secondary Shutdown System
3.	Long term sub criticality	Liquid Poison Injection System (LPIS)

3.4.2 Core Cooling

Multiple means are provided for core cooling under various plant states. These include main as well as back up systems.

Purpose	**Device / Equipment**
Under normal operating condition	
i) Power operation	
Primary	Primary coolant pumps
Secondary	Steam generators fed by main boiler feed pumps
ii) Hot Shutdown Condition	
Primary	Primary coolant pumps
Secondary	Steam generators fed by auxiliary boiler feed pumps
iii) Cold Shutdown Condition	
Primary	Shutdown Cooling Pump
Secondary	Process Water in Shutdown Cooling heat exchangers. Fire Water backup to Process Water is also provided.
Under accident condition	

Purpose	**Device / Equipment**
i) Station Blackout	
Primary	Thermosyphoning
Secondary	Fire Water injection into steam generators after their depressurization
ii) Loss of Coolant Accident	
	Through Emergency core cooling system (ECCS)
	(I)High pressure D_2O injection (Accumulator) (II)Medium pressure H_2O Injection (Accumulator) (III)Low pressure long term recirculation by ECCS pumps Fire water direct injection to the core is provided as backup.

3.4.3 Containment of Radioactivity

Type	Double containment with primary containment of pre-stressed concrete and secondary containment of reinforced concrete. Both are of dome shape.
Pressure Suppression during accident	Vapor Suppression Pool System
Containment Cooling	Air Coolers
Engineered Safety Features	Primary containment filtration and pump back system – for containment cleanup after accident Secondary containment purge and recirculation system – to maintain negative pressure in secondary containment space Primary containment controlled discharge system - to reduce primary containment pressure on long term basis after initial operation of vapor suppression system and reactor building cooling units.

3.5 Beyond Design Basis Accident Coping Capability

An accident sequence involving loss of coolant with failure of emergency core cooling can lead to a severe accident with failure of maintaining moderator and calandria vault water heat sinks. The Indian PHWR design includes backup such as direct fire water injection into core through emergency core cooling lines. To improve availability of moderator and calandria vault water heat sinks, these system pumps are provided with on site power supplies. In case of station blackout, fire water pumps independent of station electric power supplies are provided to maintain heat sink. Fire water is provided to moderator heat exchangers and ECCS heat exchangers, in addition to direct injection into End Shields.

3.6 Safety Assessment

A comprehensive safety analysis by rigorous deterministic and complementary probabilistic methods is carried out covering the following plant states:

- Normal operational modes of plant,
- Anticipated operational occurrences,
- Design bases accidents,
- During combination of events leading to beyond design basis scenarios including severe accidents.

For all Indian PHWRs deterministic safety analyses are included in the NPP's Final Safety Analysis Report (FSAR). Periodic safety review provides an opportunity to revisit and refine safety analysis. The deterministic safety analysis is available up to the severe accident and is being utilized in conjunction with probabilistic safety assessment in preparation of severe accident management programme. The present scope of probabilistic safety assessment includes Level-1 PSA for internal events at full power and external events. Level-2 PSA and shutdown PSA are completed for one of the 220 MWe Indian PHWR units.

3.7 Seismic Design Considerations

The seismic design is incorporated by classifying SSC under three categories.

3.7.1 SSE Category

SSE category incorporates all systems, components, instruments and structures conforming to safety classes 1, 2 and 3 and are designed for the maximum seismic ground motion potential at site (i.e. SSE) obtained through appropriate seismic evaluations based on regional and local geology, seismology and soil characteristics.

Notes:

Safety class 1 incorporates those safety functions which become necessary to prevent the release of substantial fraction of the core fission product inventory to the containment/environment.

Safety class 2 incorporates those safety functions necessary to mitigate the consequences of an accident which would otherwise lead to release of substantial fraction of core fission product inventory to the environment.

Safety class 2 also includes those safety functions necessary to prevent anticipated operational occurrences from leading to accident condition; and those safety functions whose failure under certain plant condition may result in severe consequences, e.g. failure of residual heat removal system.

Safety class 3 incorporates those safety functions which perform a support role to safety functions in safety classes 1, 2 and 3. It also includes:

- *Those safety functions necessary to prevent radiation exposure to the public or site personnel from exceeding relevant acceptable limits from sources outside reactor coolant system.*

- *Those safety functions associated with reactivity control on a slower time scale than the reactivity control functions in safety classes 1 & 2.*

- *Those safety functions associated with decay heat removal from spent fuel outside reactor coolant system.*

3.7.2 OBE Category

All systems, components, instruments and structures which are to remain functional for continued operation of the plant without undue risk fall under OBE category and the design basis is a lower level seismic ground motion than SSE which may reasonably be expected during the plant life. Exceeding of OBE level seismic event requires a shutdown of the plant and carry out detailed inspection of entire plant prior to startup.

3.7.3 General Category

This category incorporates those systems, structures, instruments and components, the failure of which would not cause undue radiological risk and includes all systems, components, instruments and structures which are not included in SSE or OBE category. The seismic design basis for this category is as prescribed by the relevant Indian standards.

3.8 Emergency Plans

In accordance with different degrees of severity of the potential consequences, emergency situations are graded as:

i. Plant emergency,
ii. Site emergency, and
iii. Off-site emergency

The NPP management is responsible for carrying out remedial measures during plant and site emergency while the state government authorities are responsible for taking actions in public domain to respond to an offsite emergency.

The emergency measures consist of the following

i. Notification
ii. Assessment action during emergency
iii. Corrective actions
iv. Protective measures (countermeasures)
v. Contamination control measure

The following infrastructure exists for Emergency Response:

i. Plant Control Room
ii. Emergency Control Centre
iii. Communication System
iv. Assessment Facilities
v. Protective Facilities

The requisite maintenance of Emergency Preparedness is ensured through training, periodic exercises, review and updating of plans and procedures, internal and external auditing.

Proliferation resistance

India PHWR design facilitates the effective implementation of safeguards. The provisions made for this purpose includes:

i. Installation of equipment (such as Cameras, Bundle counters, Core discharge monitors, etc.),
ii. IAEA equipment control room,
iii. Reliable power supply,

iv. Lighting arrangements in the vicinity of cameras, and
v. Appropriate EPs for cable routing.

These features would ensure achieving safeguards objective that the nuclear material, non nuclear material, equipment, facilities and information specified and placed under safeguards are used only for peaceful purposes.

Safety and security (physical protection)

Physical protection of nuclear facilities and nuclear material against theft and sabotage by individuals or groups has been a matter of national and international concern. Accordingly, security has been given requisite importance during the development of nuclear programme in the country. Over the years, security systems have under gone several changes based on changing threat perceptions and the technological developments. In the present context, security of nuclear installations is of paramount importance, particularly so after the terrorist incident of September 2001. This had brought out a new dimension of the terrorist threat against nuclear sector. Nuclear installations, nuclear material and radioactive sources are now far more focused targets. Therefore, a close review of nuclear security against sabotage and malevolent acts is necessary. This should eventually lead to develop security culture in organizational, national and international levels.

NPCIL has established the environment to create and foster characteristics and attitudes in organization and individuals so that physical protection issues receive attention as warranted by their significance. A multi pronged approach is in place to ensure security of the country's NPPs, which includes the following

5.1 Physical Protection

5.1.1 Screening/ongoing intelligence about employees

All the employees working at NPPs undergo trustworthiness check at the time of initial employment as per approved procedure. The credentials of all the employees are checked through an established procedure. Directive received from government agencies are implemented from time to time.

5.1.2 Physical Protection System

Indian NPPs have following key features of the physical protection program of nuclear power plants:

- Defence in depth using graded physical protection areas
- Intrusion detection
- Assessment of detection alarms, which also distinguishes between false / nuisance alarms and actual intrusions.
- Response to intrusions
- Offsite assistance, as necessary, from local, state and central agencies.

Multi-tier physical barriers with intrusion detection systems are in place with isolation zone. The isolation zones are monitored to detect the presence of individuals or vehicles within the zone so as to allow response to be initiated by a dedicated special response force at the time of penetration of the protected areas.

All points of personnel and vehicle access into protected areas, including shipping or receiving areas, and into each vital area are based on laid down procedures. Identification of personnel and vehicles are made and authorization is checked at all access control points. Access to vital areas is limited to individuals who require such access in order to perform their duties.

5.1.3 National Security Force

Based on the need of the country, a specialized force known as Central Industrial Security Force (CISF) was carved out from existing security forces to provide specialized security to industrial installations of the country. The CISF works under home ministry of Government of India. All NPPs are provided security by CISF personnel who are specially trained to meet the expectations. The mechanism is in place to train and continuously upgrade the knowledge and competence of CISF personnel through training programme and drills, which are conducted regularly.

Suitable linkage is woven in the system to obtain necessary intelligence, which leads to prepare and enhance security cover on requirement basis.

5.1.4 Defence Coverage

Based on location of NPP in the country and security threat perception, the required defence cover is provided.

5.1.5 Regulatory Framework

Over and above, an independent review of nuclear security related issues are in place by national regulators i.e. Atomic Energy Regulatory Board (AERB). Based on perceived threat scenarios and effective regulatory framework, regulator has issued necessary instructions in this regard, which need to be complied by each NPP. The same are followed meticulously.

6.1 Main Steam System

The heat generated in reactors is utilized to produce near dry saturated steam (0.26% wet) in steam generators. The steam from steam generators is transported to steam turbines by the main steam system. Four steam generators are located inside the reactor building. The steam lines from two steam generators are combined to form one main steam header. Hence two steam headers come out of the reactor building and lead to the turbine building to supply steam to the steam turbines.

The main steam system performs safety and non-safety functions. The portion designated to perform safety function has been designed conforming to safety class-2 and seismic category-1. This portion is structurally and functionally separated from the non-safety portion by means of an anchor and main steam isolation valve. The safety related portion of the steam lines comprises of main steam safety relief valves, atmospheric steam discharge valves (ASDVs) and main steam isolation valves (MSIVs.)

Three numbers of spring loaded main steam safety relief valves have been provided on each main steam header. All six valves on both the headers put together have 100% full power steam discharge capacity.

One small and one big ASDVs are provided on each main steam header, totaling to four ASDVs on the both the headers put together. Large ASDVs and small ASDVs are rated for 40% and 10% full power steam discharge capacities respectively, hence all the ASDVs put together have 100% full power discharge capacity. Large ASDVs are ‘fail safe close’ type and the small ASDVs are ‘fail safe open’ type. Electric motor actuated MSIVs are provided on each main steam header and are remote manually operated as and when required.

The safety functions performed by the main steam supply system are:

- Maintain the pressure boundary integrity of steam generators,
- Facilitate crash cool down in case of station black out/loss of coolant accident.

The non safety functions performed by the main steam supply system are:

- Transporting steam to steam turbines,
- Maintain the steam pressure at a preset value during normal operation and transients as per SG pressure programme, either by modulating turbine control valves and / or by modulating turbine bypass system (Steam dump / discharge valves).

6.2 Feed Water and Condensate System

Condensate system is provided to supply the condensate from condenser hot well to the deaerator under all conditions of operation by two Condensate Extraction Pumps, one normally operating and the other stand by. The pumps take independent suction from the hot well. The condensate is passed through gland steam condenser, drain cooler and then through LP heaters before reaching the deaerator. All the LP heaters extract steam from the LP turbine. The deaerator heater, which is a spray cum tray type deaerator, reduces the dissolved oxygen level to acceptable values. The deaerated condensate is stored in the deaerator storage tank having a capacity of 235 m^3. Deaerator storage tank very low level initiates a Reactor trip and the inventory remaining is sufficient for cooling down the PHT up to 150 ^{o} C by running Auxiliary Boiler Feed Pump (ABFP). Along with two CEPs, two condensate transfer pumps are also provided for deaerator make up with on site power supply.

The feed water system supplies the deaerated feed water from the deaerator storage tank to the Steam generator (SG). Three Main Boiler Feed Water Pumps (MBFP) each having 50 % capacity are provided. Each BFP is provided with independent suction from the Deaerator storage tank. The feed water is passed through the HP heater and the SG level control valve station before reaching the SGs. Apart from Main Boiler Feed pumps, two Auxiliary Boiler Feed water Pumps (ABFP) are provided which are connected to the onsite power supply.

6.3 Turbine-Generator and Auxiliaries

Steam turbines for all the 220 MWe PHWRs are configured with one single flow high pressure (HP) turbine and one double flow low pressure (LP) turbine tandem compounded and coupled to 2-pole generator. As the steam at the exhaust of HP turbine is around 12% wet, it is routed through moisture separator and re-heater before it is led to LP turbines.

The materials of construction of the major components are as given below:

- HP turbine casing: Chromium-molybdenum alloy steel considering the operation with wet steam;
- LP turbine casing: Carbon steel considering the super heated steam at the turbine inlet;
- Stationary and running blades: 11-13% martensitic chromium steel considering the combination of various factors like, erosion, corrosion, strength, toughness, ductility, fatigue, notch sensitivity etc.;
- Rotor shafts: Nickel, chromium, molybdenum, vanadium alloy steel. Toughness, strength, fatigue, Nil ductility temperature etc are the main criteria for selecting the above material.

In the arrangement of HP and LP turbine blades possibly all standard configuration e.g straddle type roots, inverted fir tree with side entry roots, pin type roots, continuous and packeted shroud bands, lacing wires & lacing rods, integral shrouds with inserts etc. are used in various stages spanning from HP 1st stage to LP last stage.

The TG sets for the latest plants are designed to operate on a continuous basis in the band of 47.5-51.5 Hz as against 48-51 Hz to improve availability of the units considering prevailing variation in grid frequency.

Considering both, the damage due to water erosion and decrease in efficiency due to enhanced moisture, integral drain arrangement is provided in the casing to remove the excess moisture and drain it out either through casing drains or through extraction lines.

Turbine generators of some plants are provided with hydro-mechanical governing system and some are equipped with electro-hydraulic governing system.

The standard protection features to prevent excessive damage to turbine are provided in all the steam turbines by automatically tripping the turbine on some of the indications like excessive thrust pad wear, low lube oil pressure, high back pressure, high exhaust hood temperature etc.

Also redundant over speed protection systems are provided by means of two independent mechanical trip rings. The turbines of latest plants are provided with electronic protection against over speed.

Non-return valves are provided in almost all extraction lines to close automatically to prevent turbine damage due to over speeding caused by flash over steam from feed water heaters and de-aerator during load throw offs and turbine trips.

The generator is provided with hydrogen cooled rotor, stator core & overhang and water cooled stator conductors. Static excitation system is provided in all the 220 MWe units.

All standard electrical protections are provided in addition to the mechanical protections against low seal oil level, low stator water flow etc.

The turbine generator set is supported by the following auxiliary systems to facilitate the TG operation as per the design intent:

- Gland sealing system: To prevent ingress of air and steam leak through the glands;
- Lube oil system: To reduce the friction by generating hydro-dynamic layer in the bearings and to cool the bearings;
- Control oil system: Derived from lube oil to facilitate the operation of servo-motors of stop and control valves of turbine;
- Turbovisory system: To monitor the thermal and mechanical stability of machine continuously;
- Shaft turning gear: To facilitate slow turning of the shaft train prior to start-ups and shut downs to prevent sagging of the rotor;
- Stator water system: To cool the stator conductors of the generator;
- Generator gas system: To cool the generator rotor conductors and stator core & overhang portion;
- Governing and protection system: To control the steam admission as per the pre-defined programme, to limit the over speed in case of load throw-offs and to prevent the possible turbine damage under abnormal conditions.

7.1 Electrical Systems

The electrical power supply system for the NPP consists of (i) off site power supply system required to evacuate the power generated by the turbine generator to the electric grid through the transmission lines connected to the plant switchyard and provide power supply to unit station auxiliaries and (ii) station auxiliary power supply system which supplies power to unit auxiliaries.

7.1.1 Off-site Power System

The off-site power system consists of (i) 400 kV and (ii) 220 kV switchyards, 400kV and 220kV grids. The electrical power generated by the turbo generator is fed through Isolated Phase Bus Duct (IPBD) and Generator Circuit Breaker (GCB) to the low voltage terminals of the generator transformer which step up the voltage to 400 kV. Power generated by the station is evacuated by 400 kV transmission lines. One & half breaker switching scheme is adopted for 400 kV switchyard.

Start up power for each reactor unit is derived from 220 kV switchyard through one start-up transformer (SUT) having two secondary windings. Two main cum transfer switching is adopted for 220 kV switchyard. All these transformers are located in the transformer yard.

7.1.2 Station Auxiliary Power Supply System (SAPSS)

SAPSS is broadly classified into two categories of power supplies (i) Normal Power Supplies and (ii) Emergency Power supplies depending upon the reliability, continuity and availability of the power supply.

7.1.2.1 Normal Power Supplies

Normal power supply system called as Class-IV power supply forms the main source of power supply to all the station auxiliary loads including loads supplied from emergency power supply system. This system derives power from two different sources of supply (a) from 220 kV grid through start-up transformer (SUT) having two secondary windings and (b) from the terminals of the main generator through unit transformer (UT) having two secondary windings with GCB closed and from 400 kV grid through GT/UT combination with GCB open.

The availability of power from any one of the two sources would be enough to meet the auxiliary power requirement during start-up, normal operation, shut down and Design Basis Event (DBE) of the unit. During off-site power disturbance, the Unit can continue to operate with the turbo-generator supplying the house load.

Class-IV electrical power supply system is classified as non-safety related equipment are located in non safety related buildings (e.g. turbine building, Cooling Water Pump House (CWPH), switchyard, etc.).

This system has two voltage levels at (a) 6.6 kV, 3 phase supply and (b) 415 V, 3 phase supply. Two numbers of 220 V DC control buses with dedicated batteries are provided in switchyard control room.

Class IV, 6.6 kV system consists of four numbers of buses. Each of the Class IV, 6.6 kV buses can derive power either from UT or from SUT. Whenever there is a failure in one of the sources due to fault, 'Auto Transfer' logic will be initiated automatically and supply from the healthy source will be extended to the affected buses.

Class IV, 415 V System consists of six numbers of buses. The 415V, Class-IV power supply is obtained from 6.6 kV, Class IV buses through independent 6.6kV/433V, 3 phase, auxiliary transformers. One common stand by bus is provided for four number of buses located in turbine building. At any time, stand by bus can be connected to one of these four buses manually. One bus is located and feeds loads at CWPH.

7.1.2.2 Emergency Power Supply System

Emergency Power Supply System is subdivided into two groups i.e. Group-A & Group-B. Safety related loads are duplicated with 100% standby capacity and they are connected on Group-A or Group-B such that operating and standby loads are connected to different groups. Electrical Power Supply Equipment of Group-A and Group-B are located and physically separated within Control Building to reduce risk of common cause failure (like fire). Electrical power supply equipment for safety related process cooling system are located in Safety Related Pump House (SRPH). There are no shared emergency power supplies between two units.

A single failure criterion is considered while designing for emergency power supply systems. Equipment located in these buildings are seismically qualified for safe shutdown earthquake condition as per IEEE-344.

Emergency Power Supply System consists of three tier power supply classes i.e. (a) Class III, (b) Class-II and (c) Class-I power supplies. These Power Supplies feed all the safety / safety related system loads of the unit and also some of the non-safety system loads.

7.1.2.3 AC, Class-III Emergency Power Supply System

AC power supply system are normally fed from Class-IV power supply system and backed up by three numbers of 100 % rated, 6.6 kV emergency diesel generator (DG) sets. Auxiliaries connected to this power supply system can tolerate short time power supply interruption (approximately two minutes). Any one out of three diesel generator sets is adequate to meet the safety system requirements under all conditions. DG-1 & DG-2 of a unit are located in independent rooms in DG building. In case of DG-3, it is located in the DG building of other unit.

a. Class III system has two voltage levels viz. (a.i) 3 phase 6.6 kV, and (a.ii) 3 phase 415V.

i. <u>6.6 kV, AC, Class-III System</u>

Three 6.6kV, Class III buses are provided in each unit. One bus consisting of two sub-buses with bus section breaker is provided for Group - A and other bus consisting of two sub-buses with bus section breaker is provided for Group-B. Another 6.6kV, Class III bus with DG-3 is provided as standby for both the groups. Auto / Manual inter group ties are provided to extend supply from one group to other group.

Emergency transfer (EMTR) logic provided automatically restores power supply to the affected 6.6 kV, Class-III buses either by starting its emergency DG sets and closing the DG set circuit breakers after checking for all required conditions or closing circuit breakers to the adjacent 6.6 kV, Class-III bus. Once power supply is available to Class-III buses, automatic load sequence is initiated by the EMTR logic so as to allow transients due to motor starting to die out before starting another motor. EMTR logic also restricts certain loads in case less than adequate DGs have connected to the buses to avoid DG overloading through load shedding logic.

ii. <u>415V, AC, Class-III System</u>

Two buses are provided for Group-A and other two buses are provided for Group-B. All these buses are located in Control Building. Another two buses are located in Safety Related Pump House (SRPH).All buses are fed from Class-III 6.6 kV system through independent 6.6kV/433V, Class-III auxiliary transformers. In the event of failure of any auxiliary transformer, transfer to adjacent bus supply is done automatically

b. <u>415 V AC, Class-II, Uninterruptible Power Supply System</u>

415V AC, 3 Phase, Class-II power supply derives the supply through static uninterruptible power supply (UPS) system comprising rectifier and inverter modules and is connected to Class–III bus. Each UPS is backed up by dedicated battery and provides uninterruptible AC power supply to its connected loads. Battery provides input to inverter in the event of class-III supply failure for specified duration. Reduced voltage starters are provided for fuelling machine (F/M) supply pump motor starting on UPS system.

There are two Class-II Power Supply buses (consisting one UPS, DC Switchgear and power battery). One Class-II bus is associated with Group-A and other Class-II bus is associated with Group-B. Both Class-II Power Supply buses are located in control building.

c. 220 V DC, Class-I, Uninterruptible Power Supply System

220 V DC, Class-I power supply system is normally supplied from Class-III, AC power supply system through Automatic Constant Voltage Rectifier (ACVR). Control batteries are used to back up these supplies. Normally batteries receive float charge from ACVRs, remain fully charged and supply connected DC loads during AC power failure. This system provides 220V DC uninterrupted power to electrical control and protection circuits.

In each Group, two 220 V DC buses are provided. Each 220 V DC bus is provided with one ACVR and one dedicated battery bank. All ties between the buses within a Group are normally kept open.

When one of the batteries/ACVRs in a Group is under test or maintenance, the loads on the corresponding bus will be fed from other bus in the same Group.

7.2 I&C Systems

7.2.1 Control Power Supply

Control Power Supply system is divided into Main Control Power Supply (MCPS) system and Supplementary Control Power Supply (SCPS) system.

7.2.1.1 Main Control Power Supply (MCPS) system

MCPS system supplies control power to Control & Instrumentation (C&I) loads of safety systems, safety related and non safety related loads. It consists of:

a) Class-I 48 VDC MCPS system for relays, indicating lamps, solenoid valves, input to transmitter power supplies etc. 48 VDC MCPS system consists of 415 VAC to 48 VDC rectifiers each backed up by battery, for supply of safety / safety related loads and non safety related loads. The system also consists of standby DC power supply. Input supply to rectifier is taken from 415 VAC 3 phase class III system.

b) Class-II 240 VAC MCPS system for controllers, recorders, computers etc. Class-II 240 VAC MCPS system consists of 415 VAC to 240 VAC single phase Uninterruptible Power Supply (UPS) for supply of safety /safety related loads and non safety related loads. The system also consists of standby AC power supply. Input supply to UPS is taken from 415 VAC, 3 phase class-III system.

7.2.1.2 Supplementary Control Power Supply (SCPS) system

SCPS system supplies control power to C&I loads of safety, safety related and non safety related loads connected to Back up Control Room. It also supplies control power supplies to C&I loads located in Back up Control Room. It consists of:

a) Class-I 48 VDC SCPS system for relays, indicating lamps, solenoid valves, input to transmitter power supplies etc. 48 VDC SCPS system consists of 48 VDC rectifiers, for supply of safety / safety related loads.

b) Class-II 240 VAC SCPS system for computers, monitors, printers etc. 240 VAC SCPS system consists of 240V AC single phase UPS for supply of safety and safety related loads. The system also consists of standby AC power supply.

7.2.2 Control Room

The main control rooms (MCR) in 220MWe plants have conventional control room design, which has evolved from plant to plant with incremental improvements based on the plant design and technology available. These MCR are hybrid control rooms, wherein computer based operator information displays and parameter selection and settings facilities have been provided. The normal operations are mostly carried out from discreet hardwired controls at the panels. The operator consoles located in the center of these control rooms provide facilities for detailed presentation of data in various formats and also provide capabilities for changing operational parameters using Visual Display Unit (VDU) consoles of individual systems. The control panels in the MCR are organized in a plant system-wise manner, with systems associated with various associated functions suitably located nearby for ease of operations. In international comparison, these MCR design is close to generation-2 control rooms.

Control Room is located in the seismically qualified control building (CB). CB location is such that it is not affected from internal missiles from turbine. Unitized Control Room concept is introduced. Illumination and Ventilation is also unit wise. Switchyard control panel which is common to both units is shifted from Control Room to switchyard Control Room. Operator Information Console (OIC) was introduced for mounting of VDUs of various computerized systems. Also, computer room is provided for housing computers. The MCR panels are placed behind the OICs so that the operator can view the indications & annunciations on the MCR panels from his seat. The MCR panels and OICs of both the units are arranged in "L" shape having linear image with the other unit. The use of computerized systems has reduced the density of components on Main Control Room panels. OICs are provided to get information of various systems of the plant on VDUs.

A separate Back up Control Room (BCR) has been provided for each unit. Essential safety functions can be carried out from BCR to bring the unit under safe cold shut down state in case of unavailability of MCR. BCR has been back fitted into older units.

7.2.3 Reactor Protection and Safety Systems

In order to protect the plant against 'common mode' incidents such as fires that could affect many safety systems at the same time, the I&C of safety systems are located in distinct, physically separate rooms / panels in control building (control equipment room).

Each of the safety systems Viz. reactor shutdown systems, emergency core cooling system, containment isolation system are provided with a triplicate channel philosophy with 2 out of 3 coincidence logic. This permits one channel to be tested without affecting normal plant operation. It also allows one faulty channel to be put in a safe state. It facilitates inter-channel comparison among the signals.

i. Primary Shutdown System, Secondary Shutdown System and Liquid Poison Injection System I&C

The safety systems PSS, SSS and LPIS protect the reactor and associated equipment by tripping the reactor, when certain plant parameters exceed their Limiting Safety System Settings (LSSS). The sensors, logic and actuation devices are separate and are not shared with each other or with other reactor control systems. The reactor is brought to a subcritical state by PSS. SSS is actuated automatically either on PSS failure or under Design Basis Accident (DBA-LOCA) condition. For long term sub criticality, worth of PSS / SSS is supplemented by LPIS which acts automatically after a predetermined time delay after operation of PSS/SSS. LPIS is a slow acting system which injects poison directly into the moderator in calandria independent of operation of any process system.

Neutronic instrumentation channels are based on ion chambers. Ex-core ion chambers for out of core measurement , linear power trip and rate-log trip are provided on West side of calandria for PSS and on the east side for SSS. The neutronic trip signals from the respective ion chambers through ion chamber amplifiers dedicated to PSS/SSS are monitored for trip condition in independent hardwired electronic units called Neutron Power Trip Unit (NPTU) of PSS/SSS.

The process trip parameters are monitored by computerized process alarm units called Programmable Digital Comparator System (PDCS-PSS) for PSS and hardwired process alarm units called Multiple Input Alarm System (MIAS) for SSS. The trip parameter sensors, transmitters, amplifiers, trip alarm units and trip processing channels are triplicate and designated as channels D, E & F for PSS and G, H & J for SSS & LPIS.

Relay logic is used to provide necessary trip logic interlocks and for PSS/SSS /LPIS actuation command. A channel and parameter trip is indicated and annunciated when any of the PSS channel trip parameters crosses the trip set point. The systems use triplicate channel philosophy with global 2/3 coincidence logic for actuation of PSS and LPIS final control elements i.e., Electromagnetic clutches for PSS and valves for LPIS. SSS uses local coincidence logic for generating a trip signal and each logic channel directly actuates one set of shutoff valves. This local / global trip coincidence improves system reliability and permits on-line testing of one channel at a time.

Each channel of PSS/SSS/LPIS is testable up to the final 2/3 logic. The electromagnetic clutches and fast acting valves can also be tested without affecting plant operation. Independent computer system called On-Line Trip Parameter Test System (OTPTS) is used for online testing of trip parameters of PSS. Testing of electromagnetic clutches is provided through hard-wired means. On-line operator initiated testing of the logics and actuation devices of SSS and LPIS is provided through hard-wired means.

ii. Emergency Core Cooling System I&C

Emergency Core Cooling System is one of the safety systems provided to mitigate the consequences of Loss of Coolant Accident (LOCA) in the event of a break in Primary circuit pressure boundary.

The I&C of this system has been designed to ensure automatic injection of heavy water, light water and recirculation of water from suppression pool into main PHT system, for cooling the reactor core and maintaining the fuel integrity in case of LOCA.

Triplicate instrumentation is provided for sensing various process parameters. Triplicate hardwired analog comparator system named as Multiple Input Alarm System (MIAS) is used for generating LOCA signal and other contact alarms when process parameters cross the set limits. Relay based logic is used to provide necessary control logics of valves, pumps and ECCS actuation command. 2/3 coincidence logic is used for ECCS actuation.

Poised status of ECCS is monitored on control panel through indicating lamps and hardwired indicators. A computer based ECCS Test Facility is provided for operator to carry out on-line operability and logic testing of equipment (valves and pumps). The status monitoring of ECCS equipment, sensors & circuitry is also available through this test facility.

iii. Containment Isolation I&C

The containment is an envelope around the reactor and associated Nuclear Systems and acts as a barrier in case of an accident involving failure of reactor coolant system and release of radioactivity. The reactor building is of double containment design. The inner containment called Primary Containment (PC) and the outer containment is called Secondary Containment (SC). SC is kept under slightly negative pressure than the atmospheric pressure and PC is kept under slightly negative pressure than the SC pressure. PC is designed to withstand the over pressure and high temperature under LOCA/MSLB conditions.

The primary function of the LOCA / MSLB event instrumentation is to sense LOCA / MSLB conditions well in time to mitigate the consequences. The double ended rupture of Primary heat transport system header / secondary piping would result in a sudden release of high pressure high temperature heavy water / light water into the containment and leads to containment isolation. In case of LOCA, coincident triplicate differential pressure sensors are used for sensing the Primary Containment pressure. Signals from these pressure sensors are wired into a 2 out of 3 coincident logic which calls for Reactor Building Isolation when Primary Containment Pressure exceeds the set limit.

The Reactor Building Exhaust Activity Very High signal is also used for the purpose of Containment Isolation on 2 out of 3 coincident logic. The Reactor Building Exhaust Activity is monitored by 3 Nos. Gross Gamma Monitors mounted on the Primary Containment ventilation Exhaust Duct.

Relay based logic is used to provide necessary logic interlocks for Containment Isolation. RB containment is isolated from external atmosphere by automatic closure of pneumatically operated containment isolation dampers in the ventilation supply and exhausts ducts and all other piping and ducting penetrating the containment structure and the Main and Emergency Air Lock doors. Since hand switches for containment isolation are provided both in MCR panel as well as in BCR panel, even if the Main control room is not accessible, containment isolation is possible from backup control room, if required.

On-line operator initiated testing of the logics and actuation devices is provided through hard-wired means. System health status is available in MCR and BCR.

8.1 Provisions for low consumption of non-renewable sources, including the degree of fuel utilization

The fuel design and operating experience includes natural U fuel bundles, recycled uranium fuel bundles, ThO_2 fuel bundles, MOX fuel bundles and Slightly Enriched Uranium (SEU) fuel bundles. Number of actions have been taken to improve the fuel bundle utilization in the operating PHWRs. Average core discharge burn-ups in the range of 7000 MWD/Te U are achieved in the operating units compared to the design discharge burn-up of 6300 MWD/Te U. This is achieved by improving moderator isotopic purity, increasing uranium weight in fuel bundles, reducing fuel failures and operating with optimum reactivity load. This leads to reduction in annual fuel requirement and also reduces spent fuel discharge.

Alternative fuel cycle schemes to achieve high burnups using MOX, SEU and Thorium bundles are developed and few lead bundles of these varieties are irradiated to higher burnups.

In addition whenever a unit is taken for Enmasse Coolant Channel Replacement (EMCCR), fuelling schemes are updated few months prior to shut down for EMCCR and also low burn up bundles left over in core are recycled from EMCCR unit to other unit within the station, to improve fuel utilization.

India has started analysis and design works for PHWRs using Slightly Enriched Uranium (SEU). This offers higher burn-up and consequently less annual fuel requirement and spent fuel inventory. The core average discharge burn-up increases to 14000 MWD/Te U with 1.1% enrichment. The average discharge burn-up increases with enrichment.

8.2 Provision for minimum generation of waste at the source

It is essential to minimize waste generation at all the stages of a Nuclear Plant Cycle. Waste minimization refers to both i) Waste generation by operational and maintenance activities of plant and ii) Secondary waste resulting from predisposal management of Radioactive Waste. The management of the Effluent is done in an efficient manner by better designs, improved procedure, periodic reviews and above all inculcating the awareness amongst the Waste generators since minimization of waste, at source is the most efficient way to safeguard the environment.

Some of the simple steps followed towards minimization of waste generation are

- Creating awareness for optimum use of water and other resources in active areas;
- Optimum use of Ion exchange columns in the purification system;
- Reducing equipment drains by using better seals, leak free joints and proper monitoring methods;
- Painting the wooden sleepers for easy decontamination;
- Removing packing materials outside the active area;
- Use of high thickness rubber sheets or plastic sheets for ease of Decontamination and reuse;
- Controlling issue of material used in active areas;
- Improving housekeeping;
- Ensuring proper planning of maintenance work;
- Ensuring careful movement of radioactive material;
- Keeping all the protective gears at their designated bins;
- Optimum use of hand gloves.

Pre and HEPA filters are extensively used in the ventilation exhaust system of Reactor Building (RB), Service Building and Waste Management Plant (WMP) and are required to be replaced on attaining the pre-defined differential pressure across the filters. Radiation level on these exhausted filters is generally very low in power stations. Traditionally conditioning through compaction in a drum was carried out before disposing these in earth trenches / RCC trenches of Near Surface Disposal Facility (NSDF). As a step towards waste minimization, Pre filters of ventilation system are removed and washed thoroughly using high pressure jet cleaner from the reverse air flow direction in a controlled area. These are then drip dried and put back to service. About three cycles of re-usage is achieved with this practice. Differential pressure measured in the decontaminated pre filter is at par with the requirement. Liquid waste (potentially active waste) of small volume collected is treated before discharge.

HEPA filters are removed and dipped in a water bath. The metallic filter frame is cut open and the filter media is dismantled. The filter media is collected in a 200 litre drum and compacted using a Baling machine for further volume reduction. Necessary protective wears are used during filter dismantling. The filter frames are decontaminated, scanned, certified by Health Physics Unit (HPU) and sent to stores as inactive metallic scrap. Compacted waste volume generation is 2 M^3 against the 30 M^3 of filter assembly volume.

This approach towards waste minimization has yielded an environmentally benign recycling method, significant cost saving by efficient utilization of expensive engineered barriers of solid waste disposal facility and reduction in cost of filters to be replaced.

Activated charcoal (with Potassium iodide) filters are used in RB, Spent Fuel Storage Bay (SFSB) and control room ventilation system and are meant only for post-accident scenario. Hence they are generally in clean condition. Iodine filter consists of HEPA filter, activated charcoal, Resistance Temperature Detectors (RTD) and wooden & metallic frames. These filters have an active life of about 2 years after which they require replacement. Activated charcoals can be reused for the removal of organic compounds (like oil), odour, colour, etc, from active liquid waste / down graded heavy water. RTDs and metallic frames of this Iodine filters are also reused.

8.3 Provision for acceptable or reduced dose limit

The design of NPP is done with due regard to materials chosen for manufacturing, plant lay out and shielding requirements to meet the specified regulatory requirements of radiation exposures to the occupational workers and to optimize the collective radiation dose to the plant workers. Plant layout is optimized and areas are classified according to the expected radiation levels and potential for incidence of contamination in the area. Materials used in plant systems are selected in such a way that the activation products arising from the base material or the impurity content does not significantly contribute to radiation exposures during operation and also during decommissioning.

At the design stage itself adequate provisions for radiation protection are made in the design of the plant to keep radiation levels in plant areas below design levels. Design radiation levels in the plant areas are based on the area occupancy by the radiation workers. For areas accessible during reactor power operation the maximum design radiation level is 5 µSv /hr for 8 hours per day occupancy and 40 µSv/hr for 1 hour per day occupancy. Provision of ventilation is made such that in full time occupancy areas of the plant, the airborne contamination be maintained below 1/10 DAC.

The NPP is designed to comply with the specifications on design radiation levels in plant areas, maximum radiation dose rates in control room and outside reactor building during accident conditions, design fuel failure targets, limits on concentration for cobalt impurity in reactor materials and features of radiation monitoring systems at NPPs.

The design features, station policies, procedures, organizational arrangements for radiation protection, management commitment to exposure control and the safety culture prevailing are conducive to achieve radiation dose to plant workers as low as reasonably achievable (ALARA).

Based on the operating experience, many design modifications for exposure control, have been incorporated progressively in the NPPs. Some of the design changes such as water filled Calandria Vault Cooling system, CO2 based Annulus Gas Monitoring system to eliminate Ar41 release, valve-less PHT system piping, use of canned rotor pumps and reduction of components in moderator system, use of cobalt-free alloys in in-core components and relocation of equipment from Reactor Building to outside have resulted in significant reduction in exposures.

Radiation Protection Programme during the operation of NPPs comprise of organizational, administrative and technical elements. ALARA measures are applied in exposure control of the plant personnel and the public. The plant management makes adequate review of the implementation and the effectiveness of the Radiation Protection Programme. An effective environmental surveillance programme that provides radiological data to evaluate the impact of operation of the NPP on the surroundings areas of the plant site is established at each NPP.

8.4 Provision for low Spent Nuclear Fuel (SNF) and waste management cost

Spent nuclear fuel inventory is reduced due to

i. increase in fuel burn-up under normal operation;
ii. Updating fuelling plans prior to shutting down the units for EMCCR;
iii. Development of alternative fuel cycle schemes to achieve high burnups.

The above three activities reduce spent fuel discharge and hence low spent nuclear fuel inventory.

In India, Radioactive waste management plants are co-located within the exclusion zone boundary of NPPs to avoid transportation of conditioned solid waste packages. Providing WMP with a compact layout adjacent to Nuclear building / Service building further reduces the cost of transportation of liquid and solid waste. Operating cost is minimized by adopting cost effective methods, like using cement matrix for conditioning solid waste instead of polymer matrix.

Plant layout

9.1 Buildings and Structures

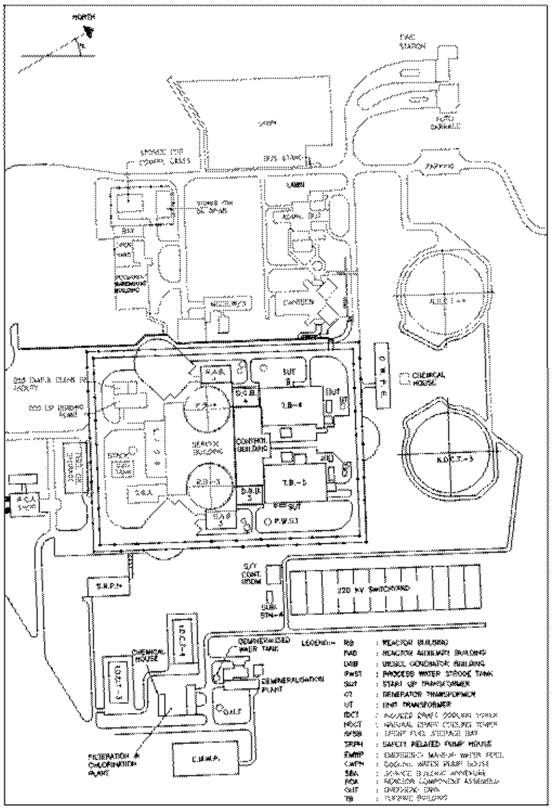

Figure-3: Plant layout

The main plant layout of Indian PHWRs has been developed on the basis of the twin unit concept.In 220 MWe units, the main plant buildings are accommodated in an area 300 m x 200 m (approx). The main plant building consists of two reactor buildings (RB) situated at about 83 m centre-to-centre distance. For each reactor unit, Reactor Auxiliary Building (RAB), Turbine Building, Diesel Generator (DG) Building and Induced Draft Cooling Tower have been provided on a unitized basis, whereas the other buildings/ structures such as Spent Fuel Building, Service Building, Service Building Annex, Control Building, Stack, Stack Monitoring Room, Waste Management Building, Safety Related Pump House, Fire Water Pump House, D_2O Evaporation & Clean-up building, D_2O Upgrading plant and Switchyard are common to the two units. In these common buildings physical separation has been provided between the safety related systems of the two units, so that in the event of an accident in one reactor system, the ability to orderly shutdown, cool down and residual heat removal of the other reactor is not impaired.

The principal features of plant layout for the Nuclear Power Station consisting of two units are generally as follows;

- The layout is based on the concept of independent operation of each unit. As far as possible each unit is independent. Only some of the common facilities are shared for reasons of economy;
- All safety related systems and components are grouped together and placed in separate buildings/structures of appropriate design;

- All safety related structures such as RB, RAB, DG buildings, Safety related cooling tower and pump-houses are protected from Low Trajectory Missiles emanating from turbine. TB is located radial to RB at an angle of 90^0 with reference to construction north-south direction;
- The buildings have been grouped according to their seismic classification in consonance with the classification of the system/ equipment contained;
- Mirror images in equipment layout are avoided to the maximum extent possible for O & M convenience. Adequate measures are taken to avoid human error, such as painting the respective unit areas with different color coding, automatic area announcement on entry and tagging/displaying sign boards near the equipment of different units;
- Reactor Auxiliary Building is located very near to the Reactor Building to avoid long piping lengths;
- A separate Control Building has been provided as a common facility. However, the control room and control equipment rooms located in this building are provided to cater for unitized operation;
- A separate backup Control Room has been provided for each unit;
- Emergency power supply systems such as Diesel Generators, UPS systems and Batteries are separately housed in safety related structures, for each unit;
- Proper access control measures are provided by means of Central Alarm Station (CAS), physical protection fencing and manned gates;
- The two unit module in the nuclear island has been so chosen that it is possible to:

a) Enforce single point entry in the radiation zones,

b) Follow radiation zoning philosophy without undue inconvenience to the operating personnel;

With this concept the total movement of men and materials in the contaminated areas is reduced substantially.

- The location of the upgrading plant has been selected adjoining to the main plant building so as to cut down the locked up D_2O inventory in the pipes and to enable a centralized control by the main plant personnel.

9.2 Containment

Double containment philosophy is adopted to minimise the radioactive releases to the environment. The double containment consists of a prestressed Inner Containment (IC) wall of 600 mm thick with dome and reinforced concrete Secondary Containment (SC) wall of 610 mm thick with dome. The internal diameter of the IC wall is 42.56 m and a gap of 2 m is provided between the two containment walls. The equipment and auxiliaries housed in the Reactor Building are supported on floors which are supported on a cylindrical wall called Structural wall. The IC is divided into two parts, viz volumes V1 and V2. Volume V1 contains high enthalpy systems like PHT system and in the event of an accident has tendency to get pressurised. The low enthalpy systems are housed in volume V2 and as such there is no possibility of excessive pressurization of this volume on account of escape of fluids from the systems. The sealing at the boundaries of the volumes V1 and V2 and the arrangement of vapour suppression pool are so designed that in case of pressurisation of volume V1, the passage of mixture of vapours and the air to volume V2 would be possible only through the suppression pool. The condensation of vapour in the suppression pool reduces the peak pressure in the containment. The inner containment is designed based on the uniform internal over pressure due to Main steam Line Break (MSLB).

Plant performance

10.1 Plant Operation

The commencement of operation of a Nuclear Power Plant (NPP) begins with approach to the first criticality of the station. Before the start of commissioning activities, the station prepares a comprehensive programme for the commissioning of plant components and submits the same for review and acceptance of Regulatory body. The Operation and Maintenance (O&M) department at the station prepares the Technical specification for operation in consultation with the plant designers before the approach to first criticality, based on the inputs from the design and safety analysis. This document which specifies the Operational Limits and Conditions for the station also is subjected to Regulatory review and approval. Once the commissioning activities are completed, the entire plant is handed over for regular operation and maintenance, to the O&M department which already exists at the Site. The units are operated within the limits specified in the technical specifications. To ensure a high degree of quality in operation, all operation persons who are at or above the position of Assistant Shift Charge Engineer (ASCE) are qualified graduate engineers who are trained and licensed as per the licensing procedures approved by Regulatory body. All activities including surveillance testing are performed with approved procedures to minimize errors due to human factors. All operations in the control room as well as in the field are carried out only after adequate pre-job briefing and planning. The station establishes plant configuration control procedures to prevent human errors during outage management, maintenance and implementation of engineering changes.

The station has a well defined organization chart. The chart clearly defines the lines of responsibility and authority to ensure smooth operation as well as safety during start up, normal and abnormal operations. Station Director is the Chief of Station O&M management at site. He has the overall responsibility for the safe operation of the plant and in implementing all relevant policies and radiation protection rules and other instructions and procedures laid down by the operating organization for plant management, and the statutory / regulatory requirements.

The performance of operating Indian PHWRs has improved significantly and an overall availability factor of greater than 90% has been achieved.

10.2 Reliability

Successful and proven technology are employed throughout the plant, including system and component designs, maintainability and operability features, and construction techniques. Vast experience available from Indian PHWR and similar plants elsewhere are extensively used in order to assure the targeted reliability of the station and minimise the risk to the Public, Plant Personnel and Equipment. A high degree of automation has been provided to minimize human error affecting availability / reliability. The safety systems are functionally and physically independent to each other as well as from process systems. The basic safety functions i.e. reactivity control, maintaining continuous core cooling and confining radioactivity are carried out by multiple means. The reactor Protective System design ensures that all the safety functions will be performed reliably while allowing online testing and maintenance of a protection channel without affecting reactor operation. Materials with fire-retardant characteristics only are used in the electrical systems to minimise the probability of fire and the consequences of a fire.

Fuel reliability over the planned lifetime is a primary objective. The fuel bundle design and fabrication have been evolving over the years resulting in many improvements and consequent good fuel performance in the reactors at present. The current fuel failure rate is less than 0.1% in Indian PHWRs.

Another important aspect of the plant reliability is the elimination of human error. A well defined recruitment policy exists which ensures that only highly qualified manpower is inducted for the O&M section of the plant. All plant personnel are given both class room as well as on the job training to perform their duties. Depending on the category of personnel, certain levels of training are fixed, each aimed at imparting definite depth of skills, knowledge and abilities. Training on full scope simulator is mandatory for operation staff, especially for those who are holding licensed positions. The O&M staff has the responsibility of preparing all the station documents required for the plant operation and maintenance.

Since the systems required for safety functions are appropriately designed for Safe Shutdown Earthquake (SSE) condition, their failure during seismic activity is not expected. Similarly adequate defences have been built in the design against flooding, externally or internally generated missiles, fire, etc. Components located in Reactor Building (RB) and required to perform safety functions following accident conditions, are appropriately qualified for the postulated environment. Further the design philosophy ensures that plant conditions associated with high radiological consequences have low probability of occurrence, and plant conditions with high likelihood of occurrence have only small or no radiological consequences. The safety systems are designed to have very high reliability and each safety system is designed to have unavailability target below 1.0 x 10-3 yr / yr. Defence-in-depth concept has been applied to containment of radioactive material, by a series of physical barriers. Provision of periodic testing and inspection of active components in safety systems are possible online.

10.3 Availability Targets

The plant is designed for an average annual availability factor of greater than 90 %, averaged over the life of the plant and accordingly the targets for different types of outages are planned. Indian PHWRs are normally designed to have one planned biennial shut down for about one month duration.

Maintenance programme followed during the life of the plant is a valuable contributor to high plant availability. The maintenance programme is put in place to ensure that (i) Safety Status of the Plant is not adversely affected due to aging, deterioration, degradation or defects of plant structures, systems or components since commencement of operation and (ii) their functional reliability is maintained in accordance with the design assumptions and intent over the operational life span of the plant. The station prepares a preventive maintenance schedule for systems, structures and components. This schedule is modified based on operating experience. In addition, a computer based system for trend monitoring of the important parameters of important equipment is used for predictive maintenance. The preventive maintenance includes periodic surveillance and verification, periodic preventive maintenance and predictive maintenance. Also a periodic In Service Inspection (ISI) programme is available in which plant components and systems are inspected for possible deterioration in safety margins and their acceptability for continued operation of the plant and to take corrective measures as necessary. Systems, Structures and Components (SSC) important to safety of the plant are identified in the In-service Inspection manual, which gives the requirements with respect to (a) areas and scope of inspection (b) frequency of inspection (c) method of inspection and (d) the acceptance criteria. This is further supported by a Performance Review Programme to identify and rectify gradual degradation, chronic deficiencies, potential problem areas or causes. This includes review of safety-related incidences and failures of SSC of the plant, determination of their root causes, trends, pattern and evaluation of their safety significance, lessons learnt and corrective measures taken.

10.4 Construction Management

With rich experience of over 30 years of operation and construction management it is well established that setting up of nuclear power projects in India in about 5 years has been demonstrated with the help of tremendous developments in construction technology, mechanization, parallel civil works and equipment erection, computerized project monitoring and accounting systems. A review of previous PHWR construction experience is performed to assure lessons learned are addressed in current / future design and construction.

By considering the best achieved times for the critical path activities of previous and ongoing projects, even a further reduction of construction time is being aimed at. Definition of Overall Construction time for a Nuclear Power Project is defined as the overall time taken from First pour of concrete (FPC) of Reactor building base raft to commencement of commercial operation. Reactor Building and systems inside reactor building generally define the critical path activity. The other major systems and their buildings such as reactor auxiliary system, Turbine Generator (TG) and Balance Of Plant (BOP) generally are constructed in parallel to achieve the overall schedule. All construction targets are made site specific. Substantial portion of the design work and the regulatory review is completed prior to the First pour of Concrete (FPC) so that no holds are placed during the construction.

Similarly Modularization of equipment packages and structural elements are being pursued for new projects, where it shows a benefit in cost or schedule improvement, subject to preserving space needed for maintenance, testing and other activities requiring access. This includes both in-shop modularization and on-site module assembly in lay-down areas.

The Plant Design, Construction, Operation and Maintenance organizations together develop a detailed overall Project Master Plan prior to the start of construction. The plan encompasses design, procurement, construction and commissioning activities up to the commercial operation. The plan establishes the overall approach and provides a basis for developing and assessing detailed sub-schedules. All schedules are regularly reviewed and monitored to check for compliance with the overall project plan and to identify any deviation requiring corrective action. The project is monitored using quantitative methods appropriate to the particular activity. Schedules are maintained using modern technology (Primavera software, etc.) and methods, and updated as work progresses to realistically reflect the actual work status.

Regular interaction between the construction engineers and the design engineers as well as interdisciplinary design reviews are periodically carried out to successfully implement the constructability requirements at the design stage itself. Standardized component sizes, types and installation details are provided to improve productivity and reduce material inventories.

Technical data

General plant data

Reactor thermal output	754.5 MWth
Power plant output, gross	235.81 MWe
Power plant output, net	210.81 MWe
Power plant efficiency, net	26.5 %
Mode of operation	Baseload
Plant design life	40 Years
Plant availability target >	90 %
Seismic design, SSE	0.2
Primary coolant material	Heavy Water
Secondary coolant material	Light Water
Moderator material	Heavy water
Thermodynamic cycle	Modified Rankine
Type of cycle	Indirect
Non-electric applications	Steam supply

Safety goals

Core damage frequency <	1E-05 /Reactor-Year
Large early release frequency <	1E-06 /Reactor-Year
Occupational radiation exposure <	0.2 Person-Sv/RY
Operator Action Time	0.5 Hours

Nuclear steam supply system

Steam flow rate at nominal conditions	2216.67 Kg/s
Steam pressure	4.03 MPa(a)
Steam temperature	250.6 °C
Feedwater flow rate at nominal conditions	2102.67 Kg/s
Feedwater temperature	171.1 °C

Reactor coolant system

Primary coolant flow rate	221000 Kg/s

Core coolant inlet temperature	249 °C
Core coolant outlet temperature	293.4 °C
Mean temperature rise across core	44.4 °C

Reactor core

Active core height	4.95 m
Equivalent core diameter	4.51 m
Average linear heat rate	28.6 KW/m
Average fuel power density	9.24 KW/KgU
Average core power density	10.13 MW/m^3
Fuel material	Sintered UO2
Cladding material	Zircaloy-4
Outer diameter of fuel rods	15.22 mm
Rod array of a fuel assembly	19 elements arranged in 3 concentric rings
Number of fuel assemblies	3672
Enrichment of reload fuel at equilibrium core	0.7 Weight %
Fuel cycle length	24 Months
Average discharge burnup of fuel	63 MWd/Kg
Control rod absorber material	SS/Co
Soluble neutron absorber	Boric Anhydride

Reactor pressure vessel

Inner diameter of cylindrical shell	5996 mm
Wall thickness of cylindrical shell	25 mm
Design pressure	0.23 MPa(a)
Design temperature	100 °C
Base material	Austenitic SS-304L
Transport weight	21.3 t

Fuel channel

Number	306
Pressure Tube inside diameter	82.6 mm
Core length	5.085 m
Pressure Tube material	Zr - 2.5% Nb Alloy (Cold Worked)

Steam generator or Heat Exchanger

Type	Mushroom type with integrated steam drum and preheater
Number	4
Total tube outside surface area	2037 m^2
Number of heat exchanger tubes	1834
Tube outside diameter	16.0 mm
Tube material	Incoloy 800
Transport weight	110 t

Reactor coolant pump (Primary circulation System)

Pump Type	Vertical, Single Stage centrifugal
Number of pumps	4
Pump speed	1490 rpm
Head at rated conditions	178 m
Flow at rated conditions	0.99 m^3/s

Moderator system

Moderator volume, core	123 m^3
Inlet temperature	43.2 °C

Primary containment

Overall form (spherical/cylindrical)	Cylindrical
Design pressure	0.27 MPa

Residual heat removal systems

Active/passive systems	Active: Shutdown cooling system Passive: Through natural circulation through SGs

Safety injection systems

Active/passive systems	Emergency core cooling system

Turbine

Type of turbines	Tandem compound Horizontal impulse Reaction type
Number of turbine sections per unit (e.g. HP/MP/LP)	2 (1 HP + 1 LP)
Turbine speed	3000 rpm
HP turbine inlet pressure	3.96 MPa(a)
HP turbine inlet temperature	249.66 °C

Generator

Type	Direct coupled, hydrogen cooled rotor
Rated power	264 MVA
Active power	235 MW
Voltage	16.5 kV
Frequency	50 Hz
Total generator mass including exciter	140 t

Condenser

Type	Double Pass Surface Condenser

Feedwater pumps

Number	5
Head at rated conditions	540 m