Status report 105 - Indian 700 MWe PHWR (IPHWR-700)

Overview

Full name	Indian 700 MWe PHWR
Acronym	IPHWR-700
Reactor type	Pressure Tube Type Reactor
Coolant	Heavy Water
Moderator	Heavy water
Neutron spectrum	Thermal Neutrons
Thermal capacity	2166.00 MWth
Gross Electrical capacity	700.00 MWe
Design status	Under Construction
Designers	Nuclear Power Corporation of India Limited (NPCIL)
Last update	01-08-2011

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 18 PHWR units and 2 BWR units. In addition, the construction of 4 units of 700 MWe has commenced. Government sanction is available for 4 more 700 MWe units.

This report presents information for the 700 MWe Indian PHWRs.

Table 1 - 700 MWe PHWR Projects in India

Serial Number	Project	Unit	Status	Rated Capacity (MWe)
1.	Kakrapar Atomic Power Project	KAPP 3,4	Under construction	2 x 700
2.	Rajasthan Atomic Power Project	RAPP-7,8	Under construction	2 x 700

KAPP-3,4 units are being constructed at Kakrapar Atomic Power Station, where 2x220 MWe units are operational. RAPP-7,8 units are being constructed at Rajasthan Atomic Power Station, where 6 units (RAPS-1 to RAPS-6) are operational.

The 700 MWe PHWR design retains the features of standardized Indian PHWR units, which include: two diverse and fast acting shutdown systems, double containment of reactor building, water filled calandria vault, integral calandria - end shield assembly, Zr-2.5% Nb pressure tubes separated from respective calandria tubes by 4 tight fit garter springs per coolant channel with inter space between pressure tube and calandria tube filled with carbon dioxide (which is recirculated) to monitor pressure tube leak by monitoring dew point of carbon dioxide.

The 700 MWe PHWR design includes some features, which are introduced for the first time in Indian PHWRs. These systems and design features are termed as First of A Kind (FOAK). These include partial boiling at the coolant channel outlet, interleaving of primary heat transport system feeders, passive decay heat removal system, regional over power protection, containment spray system, mobile fuel transfer machine, and a steel liner on the inner containment wall. Provisions are also made in design to handle severe accident scenario.

Summary technical data of Indian PHWR 700 MWe unit (Typical) is given in the appendix.

The detail system design of Indian PHWR 700 MWe unit is covered in ensuing sections, the salient design features are given below in Table-2:

Table 2 - Salient Design Features of IPHWR 700

Serial Number	**System**	**Design features/rationale**
1.	Siting	In addition to conventional siting parameters, the plant is sited considering 1. Geological, seismological, meteorological and hydrological conditions 2. Population distribution around site 3. Away from air corridor 4. Requirement of exclusion zone, sterilized zone and emergency planning zone around site.
2.	Layout	1. Twin unit module, with main plant buildings unitized. Some of the services are located in buildings shared by both the units. 2. Safety related buildings and structures are located in low trajectory turbine missile free zone. 3. 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. 37 element fuel bundles are of 0.5 m length and each pressure tube contains 12 fuel bundles.
6.	Reactor Core	The reactor consists of an 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 392 coolant channel assemblies. The fuel bundles are contained in pressure tubes, also called coolant tubes, of zircaloy 2.5% niobium. The 392 pressure tubes are arranged in 286 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 which 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, which includes • Liquid zone control system – for 14 zone • 17 cobalt/stainless steel adjuster rods to provide xenon override capability and • 4 cadmium sandwiched stainless steel control rods for quick power reduction. • Moderator liquid poison addition system (MLPAS) 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 under operating and accident conditions and maintain long term sub criticality margin. Shutdown System-1 (SDS-1) consists of 28 mechanical rods of cadmium sandwiched between stainless steel elements. SDS-1 is designed to come first in case of reactor shutdown demand. The shutdown System-2 (SDS-2) injects liquid poison which mixes with moderator inside calandria.
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 the fuelling machine and fuel transfer equipment, spent fuel is transferred to spent fuel storage bay for long term cooling. A FOAK mobile transfer machine (MTM) is used for transferring fresh and spent fuel to and from the fuelling machine.

9.	Primary Coolant System	The primary heat transport (PHT) 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 two reactor inlet headers, from where coolant is directed to coolant channels through 196 inlet feeders. The coolant from the coolant channels flows to two reactor outlet headers through 196 outlet feeders. The core is divided in two loops with feeders connected in interleaved fashion. 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 266^{o}C at inlet to 310^{o}C at the outlet. Limited boiling to the extent of 3% is allowed near coolant channels outlet. The heat transport system pressure is maintained at 100 kg/cm^{2} at the outlet headers with the help of a pressurizer which rides over both the loops of primary heat transport system. During LOCA condition, the loops automatically get isolated from each other, thereby limiting consequences to one loop. 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. In order to conserve water inventory on the secondary side of steam generators, in case of station blackout, a passive decay heat removal system is provided to passively cool and re-circulate the secondary inventory. 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. This system is capable of valving in at full PHT system pressure and temperature. Emergency core cooling system is provided to remove core heat following loss of coolant accident. This system operates in two phases incorporating high 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 inventory addition and recovery system is provided.
10.	Moderator System	Heavy water in calandria, maintained below 76^{o}C by a circulation and cooling system. The moderator system equipments are provided with onsite power supply.
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 inner walls of the primary containment are lined with 6 mm steel liners. 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 and thereby reducing the containment pressure, to clean up the containment atmosphere and for post accident controlled discharge.

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 all postulated off-normal 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:

1. 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 pumps	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 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.

2. Pattern of coolant flow rates through coolant channels is compatible with pattern of heat production in each channel of the reactor core. Feeders are sized so as yield same boiling length and similar quality at channel exit in all the channels.
3. Concept of interleaving of the feeders to minimize the void reactivity addition due to loss of coolant accident (LOCA)

4. Boiling of the coolant at the coolant channel exit is allowed to the extent of 3% quality.
5. Controlled pressure at the reactor outlet headers to maintain temperature pressure, and quality within specified limits.
6. Incorporation of pressuriser to provide 'cushion' for ease of pressure control and to reduce duty on feed and bleed system.
7. Concept of two loop system to minimise the energy release during loss of coolant accident
8. Over pressure relief to protect the PHT pressure boundary.
9. Addition of coolant to and removal from the system in order to control the coolant inventory in the main circuit.
10. Pressure control of the PHT system by feed and bleed system when pressuriser not in service.
11. Layout of equipment to permit natural circulation of coolant for decay heat removal.
12. Pressurised accumulators for light water injection followed by a recirculation phase for emergency core cooling in case of Loss of Coolant accident (LOCA). Provision of simultaneous all header injection by ECCS in case of LOCA.
13. Inventory addition and recovery system to make up PHT inventory in case of small leaks [within the capacity of primary pressurizing pump (PPP)].
14. Control of dissolved gases in reactor coolant.
15. Purification and chemistry control of coolant.
16. Provision for supply of high pressure heavy water to the fuelling machines.
17. Accessibility of all components during shutdown and accessibility of some during operation for limited duration.
18. Provision for header level control for maintenance of steam generator, primary coolant pump and other boundary valves.
19. Heavy water leakage collection from potential leak points in the system.
20. Study of corrosion coupons in Autoclaves (during hot-conditioning / PHT decontamination).
21. Sub micron filters are used in PHT system to reduce the activity load resulting in less collective dose consumption.
22. Constant boiler pressure program scheme for steam generator pressure control.

2.2 Reactor Core and Fuel Design:

This pressurized heavy water reactor uses heavy water as moderator and coolant and natural uranium dioxide as fuel. Reactor consists of integral assembly of two end shields and a calandria with the latter being submerged in the light water filled vault. Fuel bundles are contained in 392 Zr-2.5%Nb pressure tubes, arranged in a square lattice of 286 mm 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.

Around each coolant 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, annulus gas system is intended to detect leaks in the coolant tube/calandria tubes. Axial shielding from 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 fitting and individual feeder pipes to headers at both ends of the reactor. The bulk of the space available in the calandria, i.e. the one available 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. Besides these, two fast acting independent shutdown systems are provided as a part of protection system. Both these systems are fast acting and independent with adequate capability to suppress and terminate fast reactivity transients under various operating and accident conditions and bring the reactor to safe shut down state. Shutdown system no. 1 (Shut-off Rod Mechanism) consists of 28 shut-off rods with cadmium sandwiched stainless steel elements and normally acts first during a reactor trip. The shutdown system no. 2 (SDS#2) injects gadolinium nitrate solution into the moderator when actuated. SDS#2 has independent set of trip parameters, sensors, trip settings and signal processing for actuation. The worth of these shut down systems are adequate, to take care of long term effects such as xenon decay.

The reactor generates about 2296 MW of total fission power out of which 2166 MW is delivered to the primary coolant.

The fuel bundle consists of 37 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, 12 and 18 elements in different rings. Each element contains a 480 mm long stack of sintered natural UO_2 pellets in a thin zircaloy-4 clad coated with 3 to 9 micro-meter thick graphite on inside diameter with end caps welded at both ends. 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 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 clad 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 inside surface of the clad to act as a lubricant between fuel element's pellets and clad and also act as barrier for fission products to avoid their direct contact with clad. The above features will reduce the pellet clad mechanical and chemical interaction (between fission products and clad), 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 D2O 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 number of equipment 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, Shielding Plug and Fuel Locator. Other major equipment of fuel handling system which performs the task associated with channel refuelling are fresh fuel handling equipment and the equipment required for transferring the irradiated fuel to storage pool. The movement of these mechanisms is achieved either by D_2O, oil, H_2O, air or electric operated actuators. Process systems provide the operating fluid at controlled pressure, temperature, flow and direction to enable precise and controlled movement and forces of various actuators. As the equipments 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, the fresh 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 fresh 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 fresh fuel loading is from upstream to downstream in the direction of coolant flow.

Fuel Handling System design in 700 MWe PHWR has evolved from 220 MWe and 540 MWe PHWR design.

Experience gained over the years in all aspects of reactor design, fabrication, operation and maintenance has been utilised in developing the new design for 700 MWe PHWR. In this design, greater emphasis is given to minimise the requirements of operation and maintenance efforts. For this, the number of equipment especially those which operate in water medium or are submerged under water is minimised. Also, enhanced automation of the irradiated fuel handling operations in the storage pool is incorporated. Some of the salient design features of the 700 MWe PHWR fuel handling system are as follows:

In 700 MWe PHWR, the fuelling machine design incorporates the features of both the 540 MWe and 220 MWe PHWR to make it lighter and shorter. For example, the snout assembly is of modular design and is similar to that in 540 MWe design. Whereas, the ram assembly is ball screw driven as in standardized 220 MWe type. A new concept of snout level draining has been incorporated to facilitate dry transfer operation at the fuel transfer port, thus eliminating the need of heavy water in the fuel transfer system. Fuelling machine head is mounted on a bridge which provides required movement to facilitate its clamping on the channels across the reactor face.

The fuel transfer system is of totally new design. It is adopted to eliminate heavy water from the system, reducing the number of equipment, simplifying the operation and for reducing maintenance requirement. It is achieved by locating the fuel pool closer to the reactor building, thus eliminating the need for shuttle transport system and by incorporating a light water based mobile transfer machine common for both sides of the reactor. Mobile transfer machine is located inside a shielded room and moves on a floor mounted guide rails. It is used to receive irradiated fuel from north and south fuelling machines and transport it up to inner containment wall, for its further transfer to fuel storage pool. In order to eliminate heavy water from fuel transfer system, transfer of irradiated fuel from fuelling machine to mobile transfer machine takes place in air i.e. dry transfer operation. During this operation, both fuelling machine and mobile transfer machine are drained below the snout level and irradiated fuel bundles are transferred one pair at a time by moving it above the water level to align with the port. Further, automation of irradiated fuel loading operation in storage tray inside the fuel pool is provided to eliminate manual handling of irradiated fuel bundle by tongue type long handled tool. This is to minimise the personnel occupancy in the generally hot and humid environment around storage pool areas.

Irradiated fuel storage pool is designed as tank-in-tank type with inner tank SS lined and with provision of leak detection. It eliminates the possibility of contamination of ground water due to any leakage of pool water. Automation of irradiated fuel loading operation has enabled the use of high density storage tray and hence the increased irradiated fuel storage density in the pool.

Fuel handling control system is designed to perform all the operations in auto mode of operation from the main control room. The manual and safety logic are implemented using field programmable gate arrays (FPGAs) to reduce the wiring density. Hardware logics are duplicated to take care of single component failure. Redundant cable routing is done to ensure availability of the control system. Manual and auto control consoles are separate and independent PC based display system is provided for human-computer interface.

In order to carryout maintenance on various fuel handling equipment, subassemblies and components, a separate maintenance facility is provided. It has several test rigs for testing and calibration of the critical subassemblies and components. In addition, a rehearsal facility is provided in the fuelling machine service area which is used for rehearsing the refuelling 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:

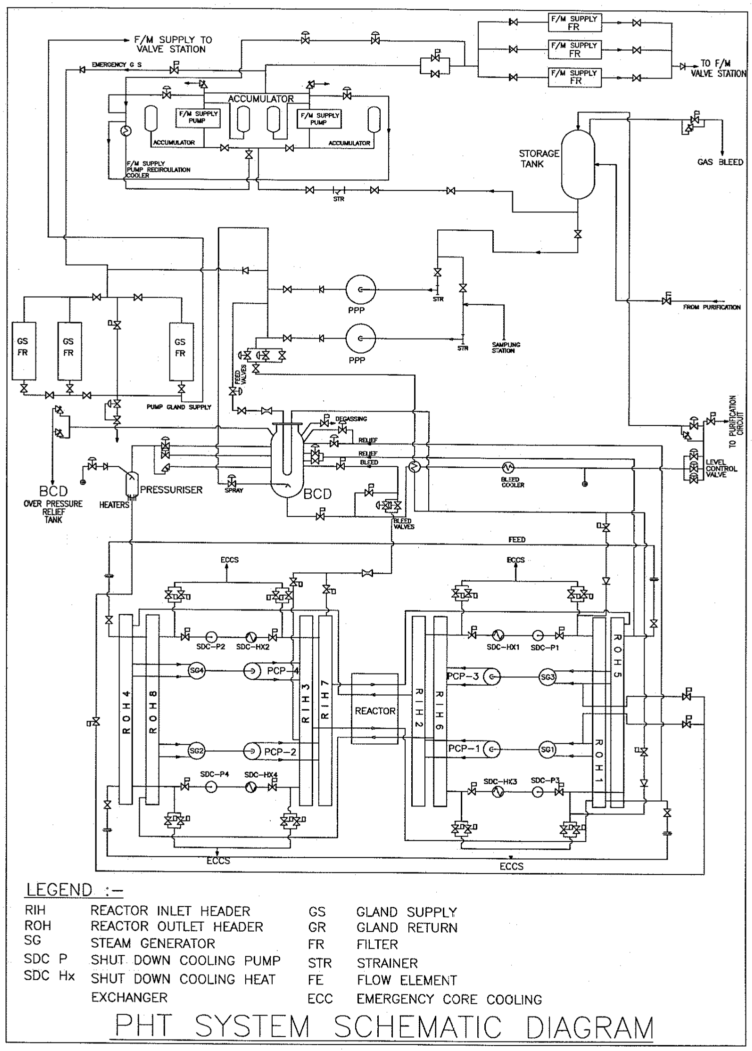

Figure 1: Primary Heat Transport System Schematic

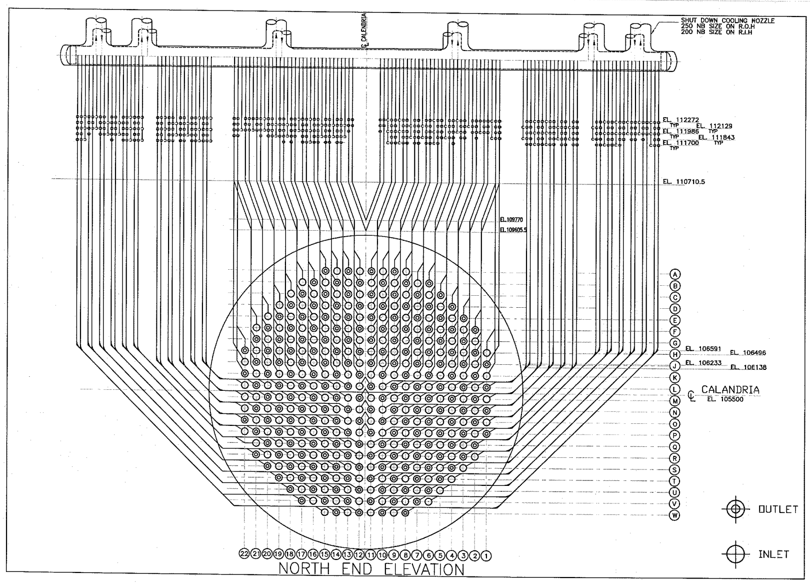

Figure 2: Primary Heat Transport System Feeders and Headers Schematic

2.4.1 Primary Coolant Pump (PCP)

Primary coolant pumps (PCPs) circulate coolant through the reactor core. The PHT main circuit has four PCPs with two pumps in each loop. PCP is a single stage, double volute, vertically mounted, single suction double discharge centrifugal pump.

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 vapour seal to prevent leakage of heavy water vapour past the shaft. Besides this, a stationary backup seal is provided which can be energized after the shaft is made stationary, to prevent any escape of heavy water to reactor building (RB) atmosphere in case of gross seal failure. The stationary backup seal can also be used for reducing differential pressure across the third mechanical seal when sealing system is boxed up under pressure with shaft in stationary condition.

PCP motors are connected to 6.6 kV class-IV electric power supply. The motors are provided with air brakes to prevent slow forward turbining of the tripped pump in 1-0 mode of pump operation. 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. Two SGs are provided in each of primary loop. The Steam Generators are vertical mushroom type design without integral pre-heater. They are recirculation type heat exchangers, having inverted U tubes for primary flow. The shell side is designed for a 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 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 two pipes, passes through the tube bundle and enters the primary coolant pump via one pipe. The feed water at 180oC enters the steam generator through two feed water nozzles provided diametrically opposite in the steam drum. The feed nozzles are joined to a circular feed water distribution header through a goose-neck piping connection. Auxiliary feed water nozzle is connected to feed water nozzles which provides emergency feed water supply to the distribution header.

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 and feed water from ring header flows down to enter the tube bundle through the flow distribution plate 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 100 mm NB, 80 mm NB, 65 mm NB and 50 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' including feeders of ‘L’ to ‘P’ rows for

11th and 12th columns have runs vertically upwards while feeders connecting channels in rows L through W 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. Two shut down pumps per loop are provided. The functions of the system are listed below:

1. The system is designed for long-term decay heat removal at or below 55^{o}C to facilitate inspection and maintenance of PHT system equipment.
2. Enables PHT System to be cooled down from 150^{o}C to 55^{o}C or below and depressurize, if required, for maintenance purpose.
3. Maintain header level (of heavy water) such that steam generators, primary coolant pumps and/or primary pressure boundary isolation valves can be opened up for in-service inspection and maintenance.
4. Provides flow for purification of the primary coolant when main PHT circuit is shutdown and in depressurized state.
5. The system is capable of valving in at full PHT system pressure and temperature.

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.

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

The 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 equal to 5 to 6 half lives (35-42 seconds) ensures N^{16} activity decay. 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 reactor normal operation as well as during shutdown.

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 radiations.

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 $5^{o}C$.

2.5.4 Spent Fuel Storage Bay Cooling System

The 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 temporarily 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 function of the purification system is 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 the reactor and also for storage of one full reactor charge under emergency conditions. The cooling water scheme is designed to meet the cooling requirement of the bays during normal as well as during full reactor core unloaded condition.

2.5.5 Annulus Gas Monitoring System (AGMS)

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 -20 ^{o}C. 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 -5 ^{o}C, 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 has been 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).

2.6 Reactor Operating Modes:

i. Normal operation

Operation of a plant within specified operational limits and conditions: This includes startup, power operation, shutting down, shutdown state, maintenance, testing and refueling.

ii. 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.

iii. 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 $55^{o}C$.

iv. 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 reactors. 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 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.

1. 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.

2. 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.
3. 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.
4. Complementary design features and use of non safety systems is envisaged at the fourth level of defence in depth
5. 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 Site evaluation Committee (SEC), Project Design Safety Committee (PDSC) or 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 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

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	Liquid zone control system
2.	Xenon override	Adjuster rods
3.	Power Reduction	Control rods

Reactor shutdown is achieved by two diverse and fast acting shutdown systems. The shutdown systems are so designed that the first shutdown system is preferred mode of shutdown.

Serial Number	Purpose	Device / Equipment
1.	First Shutdown System	Shutdown System#1
2.	Second Shutdown System	Shutdown System #2

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 Secondary ii) Hot Shutdown Condition Primary Secondary iii) Cold Shutdown Condition Primary Secondary	 Primary coolant pumps Steam generators (SG) fed by main boiler feed pumps Primary coolant pumps Steam generators fed by main/ auxiliary boiler feed pumps Shutdown Cooling Pump Process Water in Shutdown Cooling heat exchangers.
Under accident condition i) Station Blackout Primary Secondary ii) Loss of Coolant Accident	 Thermosyphoning Re-circulating steam generators secondary side inventory after cooling in passive decay heat removal system (PDHRS). As a backup fire water supplied from pumps independent of station power supplies can also be injected into PDHRS or steam generators after their depressurization. Through Emergency core cooling system (ECCS) i. High pressure H_2O injection (Accumulator) ii. 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 (with steel liners) and secondary containment of reinforced concrete. Both are of dome shape.
Engineered Safety Features	Containment spray system – for containment cleanup and cooling 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.

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. To improve availability of moderator, calandria vault water and end shield water heat sinks under severe accident condition these systems are provided with a provision where fire water can be manually hooked up to these systems from outside the reactor building.

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

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.

3.7 Seismic Design Considerations

The seismic design is incorporated by classifying SSC under three categories

i. 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.*

ii. 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.

iii. 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 measures

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.

The provisions made for proliferation resistance include:

- Installation of equipment (such as cameras, bundle counters, core discharge monitors etc.),
- Reliable power supply,
- Lighting arrangements in the vicinity of cameras and
- Appropriate embedded parts for cable routing.

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 Frame work

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 reactor is utilised to produce near saturated steam (0.25% wet) in steam generators (SGs). The steam from steam generators is transported to steam turbines by main steam system. Four steam generators are located inside the reactor building. Four steam headers come out of the reactor building and led to turbine building to supply steam to steam turbines.

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 SSE. This portion is structurally and functionally separated from the non-safety portion by means of an anchor and main steam isolation valve (MSIV). The safety related portion of the steam lines comprises of main steam safety valves, atmospheric steam discharge valves (ASDVs) and main steam isolation valves.

Three numbers of pilot operated main steam safety valves have been provided on each main steam header. All twelve valves on four headers put together have 150% full power steam discharge capacity.

Two ASDVs are provided on each main steam header, totaling to eight ASDVs on four headers put together. ASDVs are rated for 80% full power steam discharge capacity that is all the ASDVs put together have 80% full power steam discharge capacity. Electric motor actuated MSIVs are provided on each main steam header and are automatically operated in case of steam line rupture inside reactor building.

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 and loss of coolant accident.
- Helps in limiting containment pressure in case of main steam line break inside containment.

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 valves).

6.2 Feed Water and Condensate System

Engineering of Turbine-Generator (TG) & Auxiliaries and Secondary Cycle System in 700 MWe PHWRs is under progress and detail features and parameters of the system will be available only after completion of engineering activity.

6.3 Turbine-Generator and auxiliaries

Engineering of Turbine-Generator (TG) & Auxiliaries and Secondary Cycle System in 700 MWe PHWRs is under progress and detail features and parameters of the system will be available only after completion of engineering activity.

Electrical and I&C systems

7.1 Electrical systems

The electrical power supply system for typical 2 x 700 MWe Indian PHWR power station 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 and half breaker switching scheme is adopted for 400 kV switchyard.

Start up power for each reactor unit is derived from the 220 kV switchyard through two numbers of start-up transformers (SUT). Two main cum transfer switching is adopted for 220 kV switchyard. All Transformers are located in Transformer yard.

7.1.2 Station Auxiliary Power Supply System (SAPSS)

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

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 (i) from 220 kV grid through two numbers of start-up transformers (SUTs) and (ii) from the terminals of the main generator through two numbers of Unit transformers (UT) 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.

This system has two voltage levels at (i) 6.6 kV, 3 phase supply and (ii) 415 V, 3 phase supply.

Class IV, 6.6 kV system consists of four numbers of buses. Each of the Class IV, 6.6 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 6.6kV/433V, 3 phase auxiliary transformers.

2. Emergency Power Supply System

Emergency Power Supply System is subdivided into two divisions i.e. Division-I & Division II. Safety related loads are duplicated with 100% standby capacity and they are connected on Division-I or Division-II such that operating and standby loads are connected to different divisions. Electrical Power Supply Equipments of Division-I and Division II are physically separated with Division-I equipments are in Station Auxiliary Building-A (SAB-A) & Division-II equipments are in SAB-B to reduce risk of common cause failure (like fire). Electrical power supply equipments for safety related process cooling system is located in Safety Related Electrical House (SREH).

A single failure criterion is considered while designing for emergency power supply systems. Equipments located in both the 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 (1) Class III, (2) Class-II and (3) Class-I power supplies. These Power Supplies feed all the safety system loads of the unit and also some of the non-safety system loads.

a. AC, Class-III Emergency Power Supply System

AC power supply system are normally fed from Class-IV power supply system and backed up by four 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 four diesel generator sets is adequate to meet the system demands under all conditions. Two DGs are located in SAB-A & other two DGs are located in SAB-B.

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

i. 6.6 kV, AC, Class-III System

Four numbers of Class III 6.6kV buses are provided in each unit. Two buses are grouped in sub-division-IA & sub-division-IB under Division-I and other two buses are grouped in sub-division-IIA & sub-division-IIB under Division-II for distributing the redundant loads. In each division, buses are provided with tiebreakers. Manual inter divisional ties are provided to extend supply from one division to other division.

Emergency transfer (EMTR) logic provided, automatically restores power supply to the affected 6.6 kV Class-III bus either by closing circuit breaker to the adjacent 6.6 kV, Class-III bus or by starting emergency DG sets and closing the DG set circuit breaker after checking for all required conditions. Once power supply is available to Class-III bus, 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 bused to avoid DG overloading.

ii. 415V, AC, Class-III System

Two buses are located in SAB-A and other two buses are located in SAB-B. Another Four buses are located in Safety Related Electrical House (SREH). All buses are fed from Class-III 6.6 kV system through 6.6kV/433V Class-III auxiliary transformers.

Two buses are grouped into sub-division-IA & sub-division-IB under Division-I and other two buses are grouped into sub-division-IIA & sub-division-IIB under Division-II. Each division is provided with one standby transformer. In the event of failure of main auxiliary transformer, transfer to standby auxiliary transformer is done automatically.

Four buses are provided in safety related electric house (SREH) and fed from auxiliary transformers for loads of safety related pump house (SRPH) like IDCT fans, feed water pump motor etc. Two buses are grouped under Division-I and other two buses are grouped under Division-II. In the event of failure of one auxiliary transformer, transfer to other bus in the same division is done automatically.

b. 415 V AC, Class-II, Uninterruptible Power Supply System

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. 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. Variable voltage & frequency starters are provided for fuelling machine (F/M) supply pump motor starting on UPS system.

There are two Class-II Power Supply buses (consisting 2x100% redundant UPS, DC Switchgear and power battery). One Class-II bus is associated with Division-I and is located in SAB-A. Other Class-II bus is associated with Division- II and is located in SAB-B.

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 Division, two 220 V DC sub-division have been provided. Each 220 V DC is provided with one ACVR and one battery bank. A common standby ACVR is also provided for both the sub-divisions within a division. All ties between the subdivisions and division-I and division-II are normally kept open.

Each control battery bank is designed to supply emergency lighting loads connected to its associated bus under Station Blackout condition (SBO). While one of the batteries is under test/maintenance, the loads on the corresponding bus will be fed from other bus in the same Division.

7.2 I&C Systems

7.2.1 Control Power Supply

Control Power system consists of Main Control Centre Power Supply (MCCPS) system and Back – Up Control Centre Power Supply (BCCPS) system.

The Main Control Centre Power Supply system provides uninterrupted power supply with utmost reliability and tolerable voltage variation, frequency variation under normal & off normal condition. It consists of Class-I 24V DC power supply for relays, indicating lamps, solenoid valves, hand controllers, input to transmitter power supplies etc.,Class-I 220V DC power supply for DC to DC converters, radiation monitors etc. and Class-II 240V AC, single phase, 50 Hz supply for selected loads. Interruption time of 100 ms is allowed in class-II 240V AC control power supply loads. MCCPS feeds power to all plant instrumentation and control loads except channel-G, H & J which are located in Back-up Control Centre (BCC).

I. **Class-I 24V DC MCCPS**

Class-I, 24 VDC MCCPS System consists of 3 numbers of 24 V DCPS for supplying safety critical / safety related loads and two numbers of 24 V DCPS for supplying Non safety related loads . Input supply to rectifier is taken from 415V AC, 3 phase, Class-III system having Division-I & Division-II. Division-I is fed through plant generator unit transformers and Division-II is fed from grid through start-up transformers.

24 V Direct Current Power Supply (DCPS) 1, 2 & 3 are provided for supplying control power to Safety Related (SR) channels and 24 V DCPS 5&6 supply Non Safety Related (NSR) loads. 26 VDC battery banks are provided for each main DCPS. 24 V DCPS 4 is standby DCPS 1, 2 & 3 and DCPS 7 is standby to DCPS 5&6. In case of failure of any one main DCPS the standby DCPS is automatically switched to replace the faulty DCPS. The switching logic is provided with electrical interlocking so as to ensure the standby DCPS replaces only one main DCPS at a time. Two 24V DC Standby Battery Chargers (SBCs) taking input from 415V AC Class-III buses are also provided for initial charging, boost charging and equalize charging of battery. SBCs also supply the load in emergency condition of failure of main DCPS. The positive and negative buses of the 24V DC control circuits are insulated from ground.

II. **Class-I 220V DC MCCPS**

220V DCPS 1, 2 & 3 are provided for supplying control power to SR channels and 220 DCPS 5 & 6 supply NSR loads. 220V DC battery banks are provided for each DCPS. In case of failure of main control power supply, 220V DC power supply is ensured by battery. 220 V DCPS 4 is standby to DCPS 1, 2 & 3 and 220 V DCPS 7 is standby to DCPS 5&6. In case of failure of any one main DCPS the standby DCPS is automatically switched to replace the faulty DCPS. The switching logic is provided with electrical interlocking so as to ensure the standby DCPS replaces only one main DCPS at a time. Two 220V DC standby battery chargers 65521-SBC-1 & 2 taking input from 415V AC Class-III buses are also provided for initial charging, boost charging and equalize charging of battery. SBCs also supply the load in emergency condition of failure of main DCPS. The positive and negative buses of the 220V DC control circuits are insulated from ground.

III. **Class-II 240VAC MCCPS**

Very few control loads, which cannot be supplied from Class-I 220V DC power supplies and require 240V AC supply, are fed from 240V AC Class-II control power supply system. Class-II 240V AC control power supply system consists of 3 nos. of 220V DC to 240V AC Inverters for supply of safety critical / safety related/non-safety related loads. The system consists of one standby 240V AC inverter for safety critical / safety related loads. Input supply to Inverter is taken from Class-I 220V DC system. 240V AC, single phase Inverters 1, 2 & 3 are provided for supplying SR channels and INV 5&6 supply NSR loads. 240 V AC INV - 4 is standby to INV 1, 2 & 3 and INV 7 is standby to INV 5&6. In case of failure of any one main Inverter the standby Inverter is automatically switched to replace the faulty Inverter. The switching logic is provided with electrical interlocking so as to ensure the standby Inverter replaces only one main Inverter at a time. Similar to 24/220V DC control circuits, the positive and negative buses of the 240V AC control circuits are insulated from ground.

IV. **Back-up Control Centre Power Supply (BCCPS) System**

The Back-up control centre power supply system provides uninterrupted power supply with utmost reliability and tolerable voltage variation, frequency variation under normal & off normal condition. It consists of Class I 24V DC & Class II 240V AC for supply of safety critical 24V DC and 240V AC loads. It feeds power to all plant instrumentation and control loads of channel-G, H, & J which are located in Back-up Control Centre (BCC).

V. **Class-I 24V DC BCCPS**

24V DC BCCPS System consists of 3 numbers 415V AC to 24V DC rectifiers for supply of safety critical / safety related loads. Input supply to rectifier is taken from 415V AC 3 phase class III system. 24 V DC DCPS-1, 2 & 3 are provided for supplying SR Channels. In case of failure of main control centre power supply, 24V DC power supply is ensured by battery. 24 V DCPS-4 is standby to DCPS 1, 2 & 3. In case of failure of any one main DCPS, the standby DCPS is automatically switched to replace the faulty DCPS. The switching logic is provided with electrical interlocking so as to ensure the standby DCPS replaces only one main DCPS at a time. Two 24V DC standby battery chargers which are common for both main and backup control power supply batteries are provided for initial charging, boost charging and equalize charging of battery. The positive and negative buses of the 24V DC control circuits are insulated from ground.

VI. **Class-I 240V AC BCCPS**

240V AC BCCP system consists of 3 numbers 220V DC to 240V AC Inverters for supply of safety critical / safety related loads. The system also consists of 1 standby Inverter for supply of safety critical / safety related loads. Input supply to Inverter is taken from 220V DC, class-I Electrical power supply system having Division-I & Division-II. 240V AC single phase Inverters 1, 2 & 3are provided for supplying SR Channels. 220V DC battery banks of electrical power supply system are provided as back-up to 240 V AC Invertors. 240 V AC INV-4 is standby to INV 1, 2 & 3. In case of failure of any one main Inverter the standby Inverter is automatically switched to replace the faulty Inverter automatically through static switch at the outage of inverter. The switching logic is provided with electrical interlocking so as to ensure the standby Inverter replaces only one main Inverter at a time. Time taken by static switch to supply the power to load is 15 to 20 milli seconds. Similar to 24V DC control circuits, the positive and negative buses of the 240V AC control circuits are insulated from ground.

7.2.2 Control Room

In 700 MWe plant compact computerized Main Control Room is provided. Special attention is paid to human factor engineering. MCR comprises of sitting console based operations with computerized operating procedure, advanced fault diagnostics and intelligent alarm system. Plant overview panels and limited hardwired backup control panels (e.g. safety parameter I&C displays) are also provided. In international comparison, the MCR design is close to generation-3 control rooms.

A separate Back up Control Room (BCR) is 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.

7.2.3 Reactor Protection and Safety Systems

In order to protect the plant against 'common mode' incidents such as fires and internally generated missiles that could affect many safety systems at the same time, the safety systems of 700MWe reactors are located in two distinct, physically separate areas.

The following safety systems are provided:

- Shut Down System#1 (SDS#1)
- Emergency Core Cooling System (ECCS)
- Shut Down System#2 (SDS#2)
- Containment Isolation System (CIS)

They meet the requirements of reactor shutdown, remove decay heat from the fuel subsequent to shutdown, prevent any subsequent escalation of failure(s), minimise the escape of radioactivity, and provide necessary information to the operators for assessment of the state of the plant.

I&C equipment of SDS#1 and ECCS are located in Main Control Room (MCR) area and SDS#2 and CIS are located in Backup Control Room (BCR) area. Triplicated channel philosophy with 2 out of 3 coincidence logic is used for each safety system. This permits one channel to be tested without affecting plant operation. It also allows one faulty channel to be put in a safe state. It facilitates inter-channel comparison among the signals and alerts the operator in case of inconsistency.

I&C of each safety system is designed to achieve unavailability of 10^{-3} yrs/year or less and is qualified by deterministic safety analysis supplemented by Probabilistic Safety Analysis (PSA).

i. <u>Shut Down System #1 and Shut Down System#2 I&C</u>

The two Reactor Protection Systems (RPS#1& RPS#2) of SDS#1and SDS#2 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. Each shutdown system is independently capable of reactor shutdown with adequate shutdown margin to keep the reactor shutdown for prolonged period of time and under cold condition.

SDS # 1 consists of 28 Shut-off rods (SS-Cd-SS sandwiched rods -SR) grouped into two banks, each bank consisting of 14 rods. These rods are poised and kept at parked out (out of the core) position against the initial acceleration spring by electromagnetic clutch which are normally energised. On a reactor trip signal, the clutches are de-energised causing Shut-off rods to drop inside the reactor core under gravitational force. The rod grouping in two banks is to reduce electrical load on single circuit and for group wise withdrawal during plant restart.

SDS#2 is a fast acting liquid poison injection system. It consists of six poison tanks connected to horizontal liquid poison injection units located inside the calandria, The liquid poison tanks are isolated from a high pressure helium storage tank through a group of six fast acting valves. These valves are distributed in three channels and are connected in series parallel configuration emulating two out of three logic ladders such that opening of valves belonging to any two channels leads to quick injection of poison in moderator by helium pressure.

The neutronic trip parameters are monitored by Out-core Neutron Alarm System #1&2 and Regional Overpower Protection System #1□ for RPS#1/RPS#2. Neutronic instrumentation channels are based on Ion Chambers and SPNDs. Ex-core Ion Chambers for low power trip and rate-log trip are provided on East side of Calandria for RPS#1 and on the West side for RPS#2. In core flux detectors i.e SPNDs have been provided in vertical flux units for RPS#1 and horizontal flux units for RPS#2 for regional overpower trip.

The process trip parameters are monitored by computerized Process Alarm System (PAS#1) for RPS#1 and hard-wired Process Alarm System (PAS#2) for RPS#2. The trip parameter sensors, transmitters, amplifiers, trip alarm units and trip processing channels are triplicated and designated as channels D, E & F for RPS#1 and G, H & J for RPS#2.

Relay based logic is used to provide necessary trip logic and logic interlocks as well as for SDS #1/SDS#2 actuation command. A channel trip is indicated & annunciated when any of the RPS#1/RPS#2 channel trip parameters crosses the trip set point. The systems follows triplicated channel philosophy with general coincidence logic used for the logic channel and global 2/3 coincidence logic for actuation of SDS#1/SDS#2 final control elements i.e., Electromagnetic clutches for SDS#1 and fast acting valves for SDS#2. This scheme improves system reliability and permits on-line testing of one channel at a time.

Each channel of RPS#1/RPS#2 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 systems called Test and Monitoring systems # 1&2 are used for monitoring health of RPS#1&2 and online testing of trip parameters and SDS#1&2 performances during their actuation.

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.

There are 2x100% trains of ECCS high pressure injection from light water accumulators followed by 2x100% trains of low pressure long term re-circulation by ECCS pumps. When LOCA is detected, the isolation valves between the N2 tank and light water accumulators open and high pressure is exerted on the light water tank and ruptures the rupture disc, allowing water to flood into the reactor headers as a first stage of water injection. The spilled water gets collected in the ECCS sump and the water from ECCS sump is re-circulated through the reactor, continuously, through out its mission time through long term recirculation phase of ECCS.

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

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

Poised status monitoring of ECCS is provided on control panel through indicating lamps and hardwired indicators. A computer based ECCS Test Facility is provided for on-line testing of active equipments (valves and pumps). The status monitoring of ECCS equipment, sensors and 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 buildings are of double containment design. The inner containment called Primary Containment (PC) and the outer containment is called Secondary Containment (SC). PC houses all the piping & equipments of nuclear systems and is designed to withstand the over pressure and high temperature under LOCA/MSLB conditions. This reduces the ground level release of radioactivity by way of hold up and dilution of leakage from the inner PC. In order to restrict the airflow from SC to the outside atmosphere, SC is kept under slightly negative pressure than the atmospheric pressure and PC is kept under slightly negative pressure than the SC pressure.

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 high enthalpy area. The increased pressure in the containment is selected as the primary parameter for sensing the LOCA / MSLB conditions. The primary containment ventilation exhaust duct activity signal is taken as a backup to the containment pressure high signal for the containment isolation logic.

Containment isolation logics:

Triplicated differential pressure sensors are used for sensing the Primary Containment pressure. Signals from these pressure sensors are wired into a triplicated logic which calls for Reactor Building Isolation when Primary Containment Pressure exceeds the set limit. This logic is known as high containment pressure logic.

The high containment pressure logic is backed up by Reactor Building Exhaust Activity Very High Logic for the purpose of Containment Isolation. 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. The system follows triplicated channel philosophy with global 2/3 voting logic 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 & ducting penetrating the containment structure and the Main and Auxiliary Air Lock doors.

Also 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 panel if required.

On-line operator initiated testing of the logics and actuation devices is provided through hard-wired means as well as through computer based test and monitoring system. System health status is available in MCR and BCR

Spent fuel and waste management

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 safe guard the environment. High specific activity low volume tritiated liquid waste is discharged through stack after diluting with huge ventilation flow.

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), Reactor Auxiliary Building (RAB) 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, CO_2 based Annulus Gas Monitoring system to eliminate Ar^{41} 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

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

The above three activities reduces 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

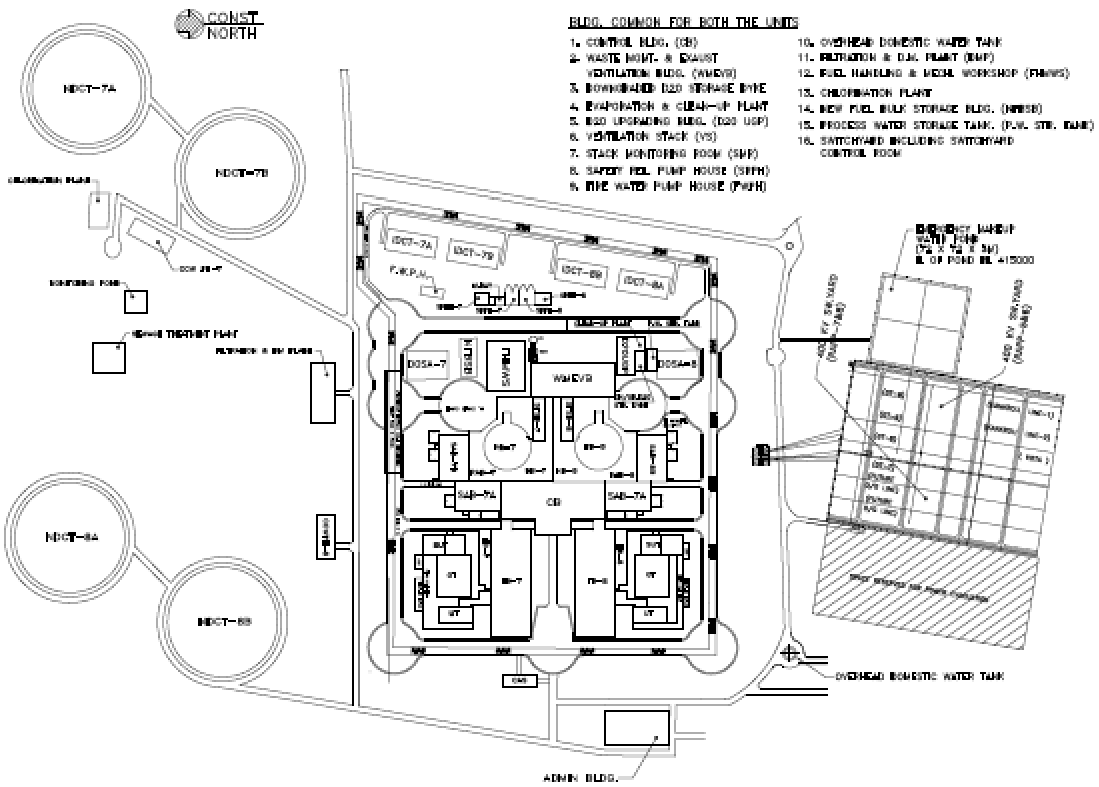

Figure 3: Plant layout

The main plant layout of 700 MWe PHWRs has been developed on the basis of twin unit concept.

In case of 700 MWe units, a single nuclear building (NB) consisting of reactor building (RB) at centre encompassed by reactor auxiliary building (RAB), housing reactor auxiliary system, spent fuel system and part of facilities in service building of earlier PHWR layout. All these facilities are founded on a common raft. Spent fuel bay is located very near to outer containment wall. Route for loading of fresh fuel and unloading of Spent Fuel, is minimised by eliminating Spent Fuel Transfer Ducts and introducing Tray Loading Bay, just outside outer containment wall (OCW). The control room and control equipment rooms of both units are located in the same floor, which also cater for unitized operation. A separate backup control room has been provided for each unit, in the nuclear building diametrically opposite to the main control room, to avoid common cause failure. Two independent access paths from Main Control room to back-up control room through Zone-1 areas are identified.

The principal features of plant layout for the Nuclear Power Station consisting of 2 units of Indian PHWRs 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 and has been shared only some of the common facilities 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 NB, SABs, Safety related cooling towers 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
- 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:
    - Enforce single point entry in the radiation zones.
    - 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 has been followed. The containment system consists of an inner (primary) containment enveloped by an outer (secondary) containment. The annulus between the inner and outer containments is kept at a slightly negative pressure with respect to the atmosphere so as to minimise ground level activity releases to the environment during an accident condition.

Cut-outs are provided in the containment domes for handling of steam generators. These are closed after the steam generators are erected.

Major plant buildings and facilities provided therein for 700 MWe unit, are as follows:

Reactor Building (Nuclear Building)	Reactor, Reactor Process Systems
Reactor Auxiliary Building (Nuclear Building)	Emergency Core Cooling, Equipment, PHT and moderator Purification system, Heavy Water Storage Tanks, Vapour Recovery System (Dryers).
	Spent fuel storage, inspection and receiving bays and shipping flask loading bay.
	Backup control room, chemical and bio-assay laboratories and TLD issue counter, source calibration room, radiation emergency centre, change room, showers.
Control Building	Control room, control equipment room, EDP centre, Cl. III & Cl. IV Compressors & Chillers, ABFP
Station Auxiliary Buildings A & B	Emergency Power Systems, Class-III DG sets, Class-II UPS systems, Class-I batteries, ACST.
Turbine Building	Turbine Generator, Secondary cycle system and Electrical Class-IV power supplies.
Waste Management Plant	Active process water system heat exchangers, Liquid effluent segregation system, decontamination centre, resin fixation etc.

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 for operation. 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.

10.2 Reliability

Successful and proven technologies 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 and 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.

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 1.0×10^{-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

Setting up of nuclear power projects in India in about 5 years has been demonstrated with the help of 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 and 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) to first criticality. 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.

The design of 700 MWe is being carried out in an integrated engineering environment using state of art tools like 3D plant modeling, use of Prodok software (for electrical and C&I drawings) .

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.

Development status of technologies relevant to the NPP

11.1 NPCIL Thermal-hydraulic Test Facility.

NPCIL Thermal-hydraulic Test Facility (NTTF) is a full-elevation thermal-hydraulic loop modeled on the prototype 700MWe PHWR. Experiments can be conducted in this test loop simulating the PHT conditions of the prototype reactor to understand the themo-siphon behavior, accident & post-accident scenarios, performance of the proposed mitigative systems and generate data base for validation of in-house developed computer analysis codes for prediction of steady state and transient behaviors of PHWR.

11.2 Fuelling Machine Snout Level Control.

Snout level control is considered for Fueling Machine (FM) of 700MWe PHWR because snout level draining facilitates dry transfer of spent fuel bundles between FM head and light water filled Mobile Transfer Machine across fuel transfer port. The water level in fueling machine is to be maintained within a band to avoid mixing up of heavy water with light water as well as keeping the fuel bundle submerged in the water. To ensure the same, a snout level control system is employed. The level of water in the tank is controlled by the flow rate through weir connected to the outlet piping.

11.3 Three-Pitch SPNDs

The SPNDs being used for in-core neutron flux measurement in PHWRs are based on Vanadium-Cobalt combination. Cobalt SPNDs have high burn-up rate and low service life. Cobalt being a hard gamma emitter, disposal of these SPNDs after reactor use adds to high solid waste inventory. Besides this, the single-pitch cobalt SPNDs used in smaller PHWRs can cause spurious trips during online refueling of larger reactor cores. To circumvent these problems, 3-pitch SPNDs with inconel emitter and collector is designed for use in 700MWe PHWR.

11.4 Containment Cable Penetration Assembly.

Containment wall penetrations of Indian PHWRs use epoxy resins and heat-shrink sleeves assembled in-situ for sealing against containment air leak. These require frequent maintenance against aging degradations. An improved penetration assembly based on ferrule fittings for individual conductors is considered for the 700MWe PHWRs. These factory-built assemblies are welded to containment liner plates to ensure positive sealing against containment pressures even in accident conditions.

11.5 Hydrogen Recombiner Test Facility

Hydrogen mitigation technology based on Passive Catalytic Recombiner Device has been developed. This will be tested under dry & wet environments before deployment in the NPPs. For this purpose, Hydrogen Recombiner Test Facility (HRTF) is set up. HRTF is an instrumented test facility simulating different reactor containment environments under controlled conditions.

11.6 Passive Decay Heat Removal System (PDHRS)

The System has been incorporated in 700 MWe PHWR design. Prior to incorporation of this system, experiments have been carried out by setting up scaled simulation loop which had full-elevation steam generator and the required power scaling. PDHRS condenser is capable of removing 3% decay heat.

11.7 Containment Spray System

In 700MWe PHWRs, suppression pool is replaced with containment spray system for removal of fission products from the containment atmosphere after any major events, such as LOCA. The proposed containment spray system design has been validated and optimized by parametric variation of nozzle design and nozzle angles. Efficacy of the system as well as the reach & coverage of spray have been physically verified on containment mock-up of required size.

11.8 Steam Generator Lancing and Imaging

A hydro-lancing equipment is developed to carry out in-situ lancing of steam generators. The system incorporates necessary optical aids to visually inspect SG internals. The system has since been successfully deployed in IPHWRs.

11.9 One Way Rupture Disc (OWRD)

One Way Rupture Disc (OWRD) which has a very large differential pressure hold capacity in the reverse direction has been designed for the ECCS of 700 MWe PHWR. This would have the potential advantage of eliminating isolation valves and/or minimizing plant down time on account of inadvertent rupture during plant transients.

Deployment status and planned schedule

Serial Number	Project	Unit	Expected date of criticality
1.	Kakrapar Atomic Power Project	KAPP 3,4	KAPP-3: December 2014 KAPP-4: June 2015
2.	Rajasthan Atomic Power Project	RAPP-7,8	RAPP-7: December 2015 RAPP-8: June 2016

Conclusion

Appendix: Summarized Technical Data (Typical)

General plant data		
Reactor thermal output	**2166 MWth**	
Power plant output, gross	**700 MWe**	
Power plant output, net	**630 MWe**	
Power plant efficiency, net	**29 %**	
Mode of operation	**Base load**	
Plant design life	**40 Years**	
Plant availability target	**90 %**	
Seismic design, SSE	**0.214 g**	
Primary Coolant material	**Heavy Water**	
Secondary Coolant material	**Light Water**	
Moderator material	**Heavy Water**	
Thermodynamic Cycle	**Modified Rankine**	
Type of Cycle	**Indirect**	

Non-electric application	NA	
Safety goals		
Core damage frequency	10^{-5} / Reactor Year	
Large early release frequency	10^{-6} / Reactor Year	
Occupational radiation exposure	20 mSv / Year (Average in any consecutive 5 years)	
Operator Action Time	30 minutes, generally	
Nuclear steam supply system		
Steam flow rate at nominal conditions	3844 t/h	
Steam pressure/temperature	44 kg/cm^2(g)/256.3^{o}C	
Feedwater flow rate at nominal conditions	--	
Feedwater temperature	180.0 ^{o}C	
Reactor coolant system		
Primary coolant flow rate	28.9 x10^6 kg/h	
Reactor operating pressure	100 kg/cm^2(g) nominal	
Core coolant inlet temperature	266.0 ^{o}C	
Core coolant outlet temperature	310.0 ^{o}C	
Mean temperature rise across core	44.0 ^{o}C	
Reactor core		
Active core height	594 cm	

Equivalent core diameter	638.4 cm	
Average linear heat rate	27.4 kW/m	
Average fuel power density	235 MW/m^3	
Average core power density	12.1 MW/m^3	
Fuel material	Natural UO_2	
Cladding tube material	Zircaloy-4	
Outer diameter of fuel rods	1.308 cm (cladding OD)	
Rod array of a fuel assembly	37 elements arranged in 4 concentric rings	
Number of fuel assemblies	4704 fuel bundles in 392 channels.	
Enrichment of reload fuel at equilibrium core	Natural Uranium (0.7% U-235)	
Fuel cycle length	12 months	Average residence time at 75% capacity factor
Average discharge burnup of fuel	7000 MWD/t	
Burnable absorber	--	
Control rod absorber material	Cadmium sandwiched in SS	
Soluble neutron absorber	Natural boron (during initial phase)	
Reactor pressure vessel (Calandria Vessel)		
Inner diameter of cylindrical shell	7800 mm	
Wall thickness of cylindrical shell	32 mm	

Total height, inside	6 m	
Base material	Austenitic SS-304L	
Design pressure/temperature	Internal: 0.85 kg/cm^2(g) + hydrostatic head of moderator External: 1.6 kg/cm^2(g) + hydrostatic head of Calandria Vault water Design temperature: $100^{o}C$	The hydrostatic head of moderator varies between 0 (at top) and 7.8 m (at bottom). The hydrostatic head of Calandria Vault water varies between 13 m (at bottom) and 5 m at top.
Transport weight	40 t	
Fuel channel		
Number	392	
PT inside diameter	103.4 mm	
Core length	5.940 m	
PT material	Zr - 2.5% Nb Alloy (Cold Worked)	
Steam generator		
Type	Mushroom type with Integrated Steam Drum	
Number	4	
Total tube outside surface area	3370 m^2	

Number of heat exchanger tubes	2489	
Tube outside diameter	19 mm	
Tube material	Incoloy-800	
Transport weight	210 t	
Reactor coolant pump		
Type	Vertical Centrifugal, Single Stage	
Number	4	
Head at rated conditions	221 m	
Flow at rated conditions	2.44 m^3/s	
Pump speed	1490 rpm	
Pressurizer		
Total volume	45 m^3	
Steam volume: full power/zero power	--	
Heating power of heater rods	1.37 MW	
Moderator system		
Moderator volume, core	310 t	
Inlet/outlet temperature	53.0 °C / 76.0 °C	
Primary containment		
Type	Prestressed Inner Containment Wall (ICW), Reinforced Outer Containment Wall (OCW)	

Overall form (spherical/cylindrical)	Cylindrical Double Containment with Dome	
Dimensions (diameter/height)	ICW: 49.5 m diameter 53.1 m height OCW: 54.74 m diameter 55.565 m height	
Design pressure	1.6 kg/cm^2(g)	
Design leakage rate of PC	1.0 (Vol%/Day for ICW) at design pressure	
Is secondary containment provided?	Yes	
Residual heat removal systems		
Active/passive systems	Active: Shutdown Cooling System Passive: Through natural circulation through SGs and Passive Decay Heat Removal system	
Safety injection systems		
Active/passive systems	Emergency Core Cooling System	
Turbine		
Type of turbines	--	
Number of turbine sections per unit (e.g. HP/MP/LP)	--	
Turbine speed	3000 rpm	
HP turbine inlet pressure/temperature	--	

Generator		
Type	--	
Rated power	--	
Active power	700 MWe nominal	
Voltage	21 kV	
Frequency	50 Hz	
Total generator mass including exciter	--	
Condenser		
Type	Surface Condenser	
Condenser pressure	0.1 kg/cm^2 abs	
Feed water pumps		
Type	Centrifugal	
Number	3 x 50%	
Head at rated conditions	--	
Flow at rated conditions	--	

Technical data

General plant data

Reactor thermal output	2166 MWth
Power plant output, gross	700 MWe

Power plant output, net	630 MWe
Power plant efficiency, net	29 %
Mode of operation	Baseload
Plant design life	40 Years
Plant availability target >	90 %
Seismic design, SSE	0.214
Primary coolant material	Heavy Water
Secondary coolant material	Light Water
Moderator material	Heavy water
Thermodynamic cycle	Modified Rankine
Type of cycle	Indirect

Safety goals

Core damage frequency <	1E-5 /Reactor-Year
Large early release frequency <	1E-6 /Reactor-Year
Occupational radiation exposure <	0.020 Person-Sv/RY
Operator Action Time	0.5 Hours

Nuclear steam supply system

Steam flow rate at nominal conditions	1067 Kg/s
Steam pressure	4.31 MPa(a)
Steam temperature	256.3 °C
Feedwater temperature	180.0 °C

Reactor coolant system

Primary coolant flow rate	8028 Kg/s
Reactor operating pressure	9.81 MPa(a)
Core coolant inlet temperature	266.0 °C
Core coolant outlet temperature	310.0 °C
Mean temperature rise across core	44.0 °C

Reactor core

Active core height	5.94 m

Equivalent core diameter	6.384 m
Average linear heat rate	27.4 KW/m
Average fuel power density	12.3 KW/KgU
Average core power density	12.1 MW/m^3
Fuel material	UO2
Fuel element type	Fuel rod
Cladding material	Zircaloy-4
Outer diameter of fuel rods	13.08 mm
Rod array of a fuel assembly	37 fuel elements arranged in 4 concentric rings
Number of fuel assemblies	4704
Number of fuel Elements in fuel assemblies	37
Enrichment of reload fuel at equilibrium core	0.7 Weight %
Fuel cycle length	12 Months
Average discharge burnup of fuel	7000 MWd/Kg
Control rod absorber material	Stainless steel-clad cadmium mixture
Soluble neutron absorber	Boron

Reactor pressure vessel

Inner diameter of cylindrical shell	7800 mm
Wall thickness of cylindrical shell	32 mm
Design pressure	0.26 MPa(a)
Design temperature	100 °C
Base material	SS304L
Total height, inside	6000 mm
Transport weight	40 t

Steam generator or Heat Exchanger

Type	Mushroom type with integrated steam drum
Number	4
Total tube outside surface area	3370 m^2
Number of heat exchanger tubes	2489
Tube outside diameter	19 mm
Tube material	Incoloy 800
Transport weight	210 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	221 m
Flow at rated conditions	2.44 m^3/s

Pressurizer

Total volume	45 m^3
Heating power of heater rods	1370 kW

Moderator system

Moderator volume, core	280.4 m^3
Inlet temperature	53.0 °C

Primary containment

Type	Prestressed ICW Reinforced OCW
Overall form (spherical/cylindrical)	Cylindrical part and hemispherical dome
Dimensions - diameter	49.5 m
Dimensions - height	53.1 m
Design pressure	0.157 MPa
Design leakage rate	1.0 Volume % /day

Secondary containment

Overall form (spherical/cylindrical)	Cylindrical double containment with dome
Dimensions - diameter	54.74 m
Dimensions - height	55.57 m

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

Turbine speed	3000 rpm

Generator

Active power	700 MW
Voltage	21 kV
Frequency	50 Hz

Condenser

Type	Steam Surface
Condenser pressure	9.8 kPa

Feedwater pumps

Type	Horizontal, centrifugal
Number	3