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The Indian PHWR

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Abstract

The nuclear power program in India at present is based mainly on a series of Pressurized Heavy Water Reactors (PHWRs). Starting from Rajasthan Atomic Power Station comprising two units of 200 MW_e Canadian designed PHWRs in 1973, the program has come a long way with 12 PHWR units in operation and 6 units under construction which includes 2 units of 540 MW_e PHWRs. Narora Atomic Power Station commissioned in 1991 marked major indigenization and standardization of PHWR designs. The choice of PHWRs in the current stage of India's Nuclear Power Plants program is based on long-term objectives in the right available resources and infrastructure. These reactors use natural uranium as fuel and heavy water as moderator and coolant. The nuclear power stations in India are generally planned as twin-unit modules, sharing common facilities such as service building, spent fuel bay, etc.

This paper brings out the key features of the design of Indian PHWR, highlighting the areas of evolution in successive project. Also covered are highlights of the operating experience with these reactors.

1. Introduction

India's current indigenous nuclear power program is based on the Pressurized Heavy Water Reactor (PHWR). With the exception of the two Boiling Water Reactor units at Tarapur (which is India's first Nuclear Power Plant), all other operating nuclear power plants in India are based on PHWR. Table 1 lists the nuclear power plants in operation and under construction with their capacities and year of commencement of operation. There are today 12 PHWR units in operation representing an installed capacity of 2550 MW_e, and 6 units under construction, including 2 units of 540 MW_e. The 18 PHWR units in operation and under construction together represent an installed capacity of 4510 MW_e.

1.1. Rationale for selection of PHWR for India's current Nuclear Power Plants program

The choice of PHWR in the current stage of India's Nuclear Power Plants program is based on the long-term objectives and availability of resources and infrastructure.

Self-reliance and energy security have been among the important objectives of the Indian Nuclear Power Program right from its inception. This, together with the existence of modest resources of uranium and vast resources of thorium in the country led to the adoption of the three stage program (thermal reactors—plutonium-based fast breeders—^{233}U/thorium-based breeders) and choice of PHWR for the first stage of the program.

The features of PHWR that favored this choice are:

- Use of natural uranium as fuel, which obviates the need for developing fuel enrichment facilities.
- High neutron economy made possible by use of heavy water as moderator, which means low requirements of natural uranium both for initial core as well as for subsequent refueling. Also fissile plutonium production (required for stage 2 of the program) is high, compared to Light Water Reactors.
- Being a pressure-tube reactor, with no high pressure reactor vessel, the required fabrication technologies were within the capability of indigenous industry.
- The technology for production of heavy water, required as moderator and coolant in PHWR, was available in the country.

The Indian PHWR design has evolved through a series of improvements over the years in progressive projects. Such improvements have been driven by, among others, evolution

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Table 1
List of nuclear power plants in India—in operation and under construction

Sr. no.	Station/plant	Units	Type	Status	Year of commercial operation	Rated capacity (MW_e)
1.	Tarapur Atomic Power Station (TAPS)	TAPS-1,2	BWR	Operating	1969	2 × 160
		TAPS-3,4	PHWR	Under construction	2005, 2006	2 × 540
2.	Rajasthan Atomic Power Station (RAPS)	RAPS-1,2	PHWR	Operating	1973, 1981	RAPS-1: (150); RAPS-2: (200)
		RAPS-3,4	PHWR	Operating	2000	2 × 220
		RAPP-5,6	PHWR	Under construction	2007, 2008	2 × 220
3.	Madras Atomic Power Station (MAPS)	MAPS-1,2	PHWR	Operating	1984, 1986	2 × 220
4.	Narora Atomic Power Station (NAPS)	NAPS-1,2	PHWR	Operating	1991, 1992	2 × 220
5.	Kakrapar Atomic Power Station (KAPS)	KAPS-1,2	PHWR	Operating	1993, 1995	2 × 220
6.	Kaiga Generating Station (KGS)	KGS-1,2	PHWR	Operating	2000	2 × 220
		KGS-3,4	PHWR	Under construction	2007	2 × 220
7.	Kudankulam Atomic Power Station (KK)	KK-1,2	LWR (VVER)	Under construction	2007, 2008	2 × 1000

in technology, feedback from experience in India and abroad, including lessons learnt from incidents and their precursors, evolving regulatory requirements and cost considerations. Valuable experience gained in design, manufacture, construction, operation, maintenance and safety regulation has enabled continual evolution, improvement and refinement in the PHWR concept in a progressive manner.

The first PHWRs at Rajasthan Atomic Power Station #1 and 2 were of Canadian design (based on Douglas Point). Work on these was taken up with Canadian cooperation. For the first unit, most of the equipment were imported from Canada, while for the second unit a good amount of indigenization was achieved. At the next station, Madras Atomic Power Station, a number of changes in design were adopted due mainly to site conditions. The import content in these and subsequent plants was reduced to 10–15%. When design work for the third PHWR station, Narora Atomic Power Station #1 and 2, was taken up in early 1970s, major modifications were incorporated with the objective of upgrading the designs in line with the internationally evolving safety standards, and to cater to the seismic environment at the site. Narora Atomic Power Station design was the first opportunity to apply India's operating experience with PHWRs, including aspects such as ease of maintenance, in-service inspection requirements, improved constructability, increased availability and standardization. Some of the new designs incorporated in Narora Atomic Power Station were with the objective of serving as stepping stones for the design of subsequent larger 540 MW_e PHWR.

Some of the significant design improvements made in Narora Atomic Power Station included adoption of an integral calandria (reactor vessel) and end shields assembly, two independent fast acting reactor shutdown systems, a high pressure Emergency Core Cooling System, and a double containment with suppression pool.

Subsequent to Narora Atomic Power Station, Kakrapar Atomic Power Station #1 and 2, Kaiga Generating Station #1 and 2 and Rajasthan Atomic Power Station #3 and 4 saw further improvements leading to standardizations in design and layout for 220 MW_e PHWRs.

The following sections bring out the key features of the design of Indian PHWR, highlighting the areas of evolution in successive projects. The outline of the design of current 220 MW_e PHWR station is given in the accompanying Table 2. In common with the PHWRs/CANDU reactor worldwide, this design is characterized by use of natural uranium dioxide as fuel, with heavy water as moderator, and heavy water at high pressure/temperature in a separate circuit as coolant.

The reactor consists of a low-pressure horizontal reactor vessel ('calandria') containing heavy water moderator at near ambient pressure and temperature. The calandria is pierced by a large number (306 in 220 MW_e and 392 in 540 MW_e PHWR) of pressure tubes, which contain the fuel bundles, and through which pressurized heavy water coolant circulates. The calandria houses all reactivity and reactor shutoff devices in the low-pressure low temperature environment.

2. Indian PHWR design and its evolution

2.1. Siting and plant layout

As part of siting, following zoning requirements exist around all Indian Nuclear Power Stations:

(i) An 'exclusion zone' of minimum 1.5 km radius from the reactor center is established. No public activity is allowed in this area, which is fenced and is under control of the nuclear power station.
(ii) A 'sterilized zone' up to 5 km radius around the plant is established where the growth of population is restricted to enable implementation of emergency measures. In this area, only natural growth of population is permitted. Any new settlement of activity, which may cause population influx, is restricted.
(iii) 'Emergency planning zone' up to 16 km radius from the plant where availability of transportation network, means of communication are ensured for any emergency situation.

Table 2
Design features of Indian standard PHWR −220 MW_e (operating)

	Design data
A. General	
(i) Rated output thermal	756 MW_t
(ii) Rated output electrical	220 MW_e
(iii) Fuel	Natural UO_2
(iv) Moderator and reflector	Heavy water
(v) Coolant	Heavy water
(vi) Type	Horizontal pressure tube
B. Reactor	
(i) Calandria shell	Horizontal stepped cylinder welded to extensions of end shield
(ii) Calandria shell material	SS-304L
(iii) End shields	Cylindrical box type structure integral with calandria shell
(iv) End shield material	SS-304L
(v) Calandria tubes	
(a) Quantity	306
(b) Material	Zircaloy-2
(vi) Pressure tubes	
(a) Quantity	306
(b) Material	Zirconium–2.5% niobium alloy
C. Core and fuel	
(i) Fuel material	UO_2
(ii) Cell array	Square
(iii) Lattice pitch	22.9 cm
(iv) Fuel form	Pellet
(v) Fuel bundles per channel	12 (10.1 in active core)
(vi) Fuel element per bundle	19
(vii) Geometry	1-6-12 concentric
(viii) Total no. of bundles in reactor	3672
(viiii) Clad material	Zircaloy-2
(ix) Clad thickness	0.41 mm
(x) Length of bundle	495 mm
(xi) Outside diameter of bundle	81.7 mm
(xii) Weight of bundle (nominal)	16.4 kg
(xiii Weight of uranium per bundle	13.4 kg
(xiv) Core radius	300 cm
(xv) Core length	500 cm
(xvi) Average linear fuel rating	35.3 kW/m
(xvii) Peak linear fuel rating	50.2 kW/m
(xviii) Maximum clad temperature	301 °C
(xix) Maximum center line temperature	1780 °C
(xix) Average fuel discharge burn-up	6700 MW days/t uranium
D. Moderator	
(i) Moderator type	Heavy water
(ii) Total quantity	136000 kg
(iii) Quantity in calandria	110000 kg
(iv) Heat produced in moderator	38.1 MW
(v) Outlet temperature from reactor (maximum)	63 °C
(vi) Inlet temperature to reactor	44.2 °C
(vii) Heavy water flow to each heat exchanger	264.7 kg/s
(viii) Normal moderator level in calandria	96% full tank

Table 2 (*Continued*)

	Design data
E. Primary coolant	
(i) Type	Heavy water
(ii) Quantity	100000 kg
(iii) Total flow	1734 kg/s
(iv) Pressure (outlet header)	8.532 MPa
(v) Channel inlet temperature	249 °C
(vi) Channel outlet temp.	293 °C
(vii) No. of primary circulating pumps	4
(viii) Type of pumps	Centrifugal
F. Reactor shutdown systems	
(a) Primary shutdown system	14 mechanical rods
(i) Material	Cadmium sandwitch in SS
(ii) Worth of PSS	36 mk
(b) Secondary shutdown system	12 hollow tubes inside calandria
(i) Material (poison)	Lithium penta-borate solution
(ii) Worth of SSS	32 mk
(c) Liquid poison injection system	Borated heavy water
G. Reactor control devices	
(a) Regulatory rods	
(i) Purpose	Regulation of power
(ii) Number of regulating rods	4
(iii)Material	Cobalt in aluminum tubes
(b) Absorber rods	
(i) Purpose	Xenon over-riding
(ii) Number of absorber rods	8
(iii)Material	Cobalt in aluminum tubes
(c) Shim rods	
(i) Purpose	Power reduction (setback)
(ii) Number of shim rods	4
(iii)Material	Cadmium sandwiched in SS
H. Steam generators	
(i) No. of steam generators	4
(ii) Type	Vertical U tube with integral steam drum (mushroom-shaped)
(iii) Material	Incoloy-800
(iv) Steam to turbine	3.923 MPa
(v) No. of tubes per SG	1834
(vi) Total steam flow to turbine	370 kg/s
(vii) Steam temperature	251 °C
(viii) Maximum moisture content	0.25%
I. Process water system	
(a) Active process water system	
(i) Purpose	To remove heat from moderator, endshield, calandria vault water, etc.
(ii) Active process water coolant	Demineralized light water closed loop system
(iii)Heat exchangers	Plate type
(b) Active process water cooling system	
(i) Purpose	To remove heat from active process water system through plate type heat exchangers
(ii) Active PW cooling water	Raw water from reservoir/canal
J. Turbine	
(i) Type	Tandem compound (one HP-5 stages and double flow LP-5 stages)

Table 2 (*Continued*)

	Design data
(ii) Speed	3000 rpm
(iii) Rating	235 MW_e
(iv) Steam flow	370 kg/s
(v) Generator	One direct coupled, hydrogen cooled rotor, 50 Hz

Typical plant layout for Indian PHWRs is based on a twin-unit concept. Fig. 1 shows the layout of the 540 MW_e PHWR Station (Tarapur Atomic Power Plant #3 and 4). Most plant systems including all systems important to safety are unitized to enable independence of each unit. Some of the principles adopted in the layout are outlined below:

(i) The plant areas are separated into nuclear and non-nuclear islands. Two unit module in the nuclear island has been so laid out that it is possible to implement proper access control measures to enforce single point entry in the radiation zones and follow radiation zone philosophy.
(ii) All safety related systems and components have been appropriately grouped and placed in structures of appropriate safety class.
(iii) The seismic class of structures is commensurate with the seismic class of equipment housed in them.
(iv) All safety related structures are so located as to provide safety against the low trajectory turbine disintegration missile.
(v) The auxiliary power supply (both electrical and controls) is divided into two redundant groups. Each of these groups are divided into safety related and non-safety related. Redundant groups of safety related equipments are separated from one another by fire barriers of appropriate rating. Physical and electrical isolation is provided between safety related and non-safety related systems. Segregation of electrical and control cables is done following relevant codes.
(vi) The stack is located near reactor building and service building so that the duct length carrying activity is minimized.
(vii) The choice of the type of condenser cooling water system is between "once-through" and "closed loop" and is site dependent. At all coastal locations and at certain inland locations where abundant water is available, "once-through" system is adopted. At most inland locations, "closed cycle" system with natural draft cooling towers is adopted. In addition, an induced draft cooling tower is provided for cooling process water. The provision for emergency water storage for make up to induced draft cooling tower for long-term cooling under emergency condition is ensured.

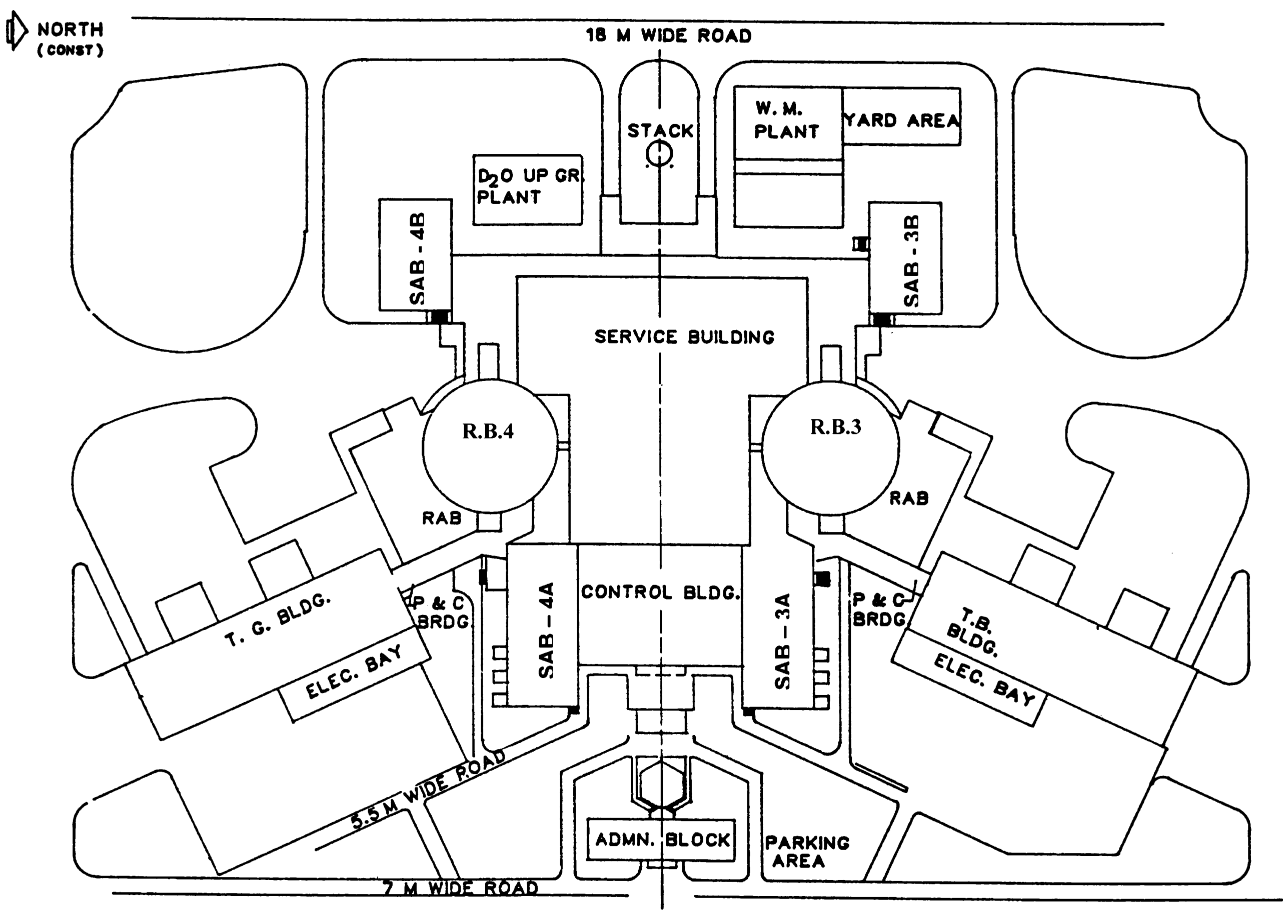


Fig. 1. Site layout of 540 MW_e PHWR (TAPP—3 and 4).

(viii) Accessibility, maintainability and ease in construction is given due consideration while preparing the plant layout.
(ix) Special provision of adequate space adjacent to the reactor building and turbine building is made to enable maneuvering of high capacity outdoor mobile crane during erection and handling of heavy equipment such as reactor components, steam generators and generator station units.
(x) For each station, in addition to the main control room, a supplementary control room is provided in service building, which can be used to perform essential safety functions in case of main control room becoming unavailable. The sensors, power supply and controls of the supplementary control room are independent of the main control room.
(xi) Some of the systems provided in the twin-unit station are shared. These systems include Spent Fuel Storage Bay, Fire Water System and Compressed Air System.

2.2. Reactor components

2.2.1. Calandria-end shield assembly

In the original design adopted in Rajasthan Atomic Power Station and Madras Atomic Power Station, there was a dump tank located underneath the reactor vessel or ‘calandria’. Reactor shutdown was achieved by fast dumping of moderator from the calandria into the dump tank through a system of S-shaped dump ports located at bottom of the calandria. From Narora Atomic Power Station onwards, a new scheme of reactor shutdown systems was adopted allowing dump tank to be eliminated, and considerable simplification of the calandria design (refer Section 2.3).

The design of the two end shields located at two ends of the reactor was also modified. The end shields limit the radiation dose in the vaults (fuelling machine vaults) adjoining the reactor vault; they also support and locate the calandria tubes and primary coolant channel assemblies in which the fuel resides. In the original design, the end shield (about a meter thick) consisted of 30 mm thick slabs of steel shrunk fitted into a steel shell, with water passages in between. These were modified from Narora Atomic Power Station onwards where the slabs were replaced by steel balls which were filled into the end shield at site. The weight of the fabricated end shield to be transported came down to almost half (at 60 t). In the original design of end shield (Rajasthan Atomic Power Station #1 and 2 and Madras Atomic Power Station #1) the material of construction was 3.5% Ni-steel. However, it was found that Nil Ductility Transition Temperature crossed the operating temperature within a short period of operation. While the stability of the end shields in this condition is assessed in detail; from the second unit of Madras Atomic Power Station onwards, the end shield material has been changed to SS-304L, which is immune to radiation embrittlement due to fast neutrons.

In current design (Narora Atomic Power Station onwards) (Fig. 2), the calandria and two end shields constitute an integral assembly, supported from the reactor vault walls, unlike earlier designs wherein the calandria and end shields were separately suspended by support rods. This design allows common tube sheet between calandria and end shield, simplifies alignment requirement between calandria tubes and end shield lattice tubes, and is more suited to conditions at seismic site.

2.2.2. Reactor vault

From Narora Atomic Power Station onwards, the calandria is housed in a steel lined vault filled with light water which serves as shielding as well as provides cooling in the vault. This is an improvement over earlier design which required air and water cooled thermal shields in the vault, and cooling coils in the vault concrete. Absence of air in the vault eliminates the production of Argon-41. Water filled vault also provides a heat sink in case of severe accident.

2.2.3. Coolant channels

In the original design, the coolant tube material was cold worked zircaloy-2. This material undergoes degradation due to hydrogen embrittlement, caused by pickup of hydrogen in reactor environment, which precipitates as hydride. The life of the zircaloy-2 tubes is severely restricted due to accelerated corrosion of this material after about 10 years of full power operation, which leads to enhanced hydrogen pick-up rates. Under these conditions, the ‘leak-before-break’ criteria may not be met. This calls for en-masse replacement of the channels at this stage.

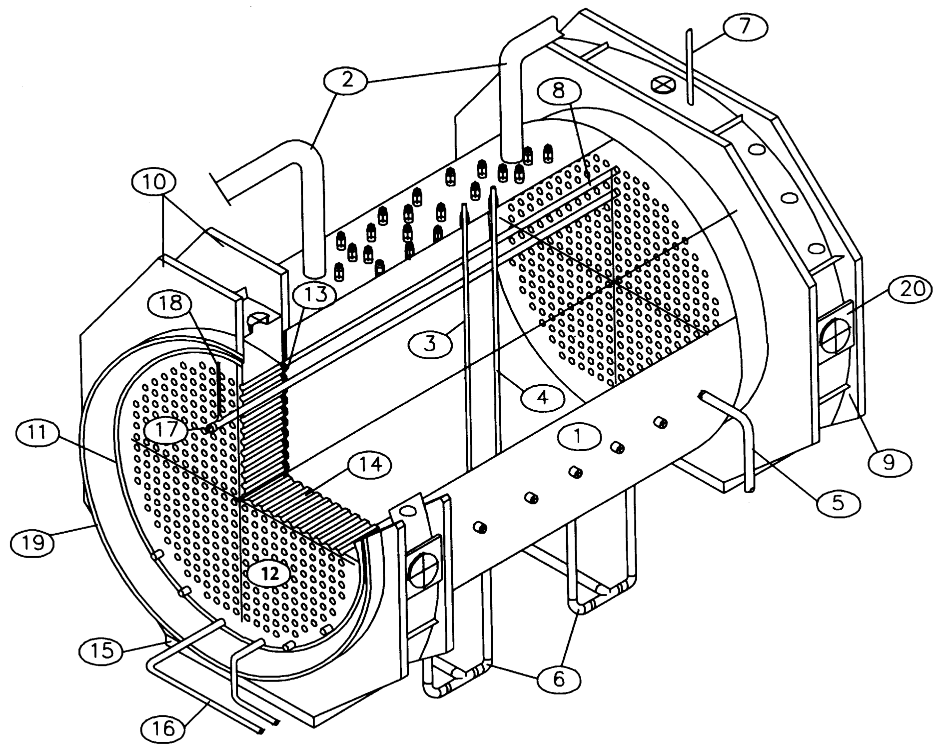
From second unit of Kakrapar Atomic Power Station onwards, the coolant tube material was changed to zirconium–2.5% niobium alloy, which has low pick-up rates of hydrogen. This is a high strength material and required development of ‘zero clearance rolled joints’ to keep residual stresses to lower levels and prevent delayed hydride cracking. This material has been used in all reactors constructed later and will also be used in en-masse channel replacement of existing units after about 10 years of full power operations.

An elaborate program of pre-service inspection, periodic inspection, in-service inspection (discussed later) and post-irradiation examination has also been initiated and is now being implemented to monitor the condition of coolant channels.

In PHWRs, the pressure tube containing fuel and hot pressurized coolant is separated from the calandria tubes (operating at ambient temperature), by garter spring spacers.

Contact between pressure tube and calandria tube needs to be prevented, as such contact leads to reduction in the local temperature in the contact region of the pressure tube, making it susceptible to blister formation due to hydride precipitation and subsequent failure due to cracking. With garter springs in place, as per design, contact between pressure tube and calandria tube is prevented throughout the life of the coolant channels. However, if the garter springs shift significantly from their design positions, the contact time can get reduced, thereby reducing the life of the coolant channel.

In the original design of garter springs which is adopted upto Kakrapar Atomic Power Station #1, loose fitting garter springs are used. During initial commissioning (hot conditioning) when the channels do not carry the fuel load, some of these garter springs were found to move. Techniques were developed to detect the location of the garter springs as well as to relocate them back to their design positions. This exercise was carried



1.	CALANDRIA SHELL	**2.**	OVER PRESSURE RELIEF DEVICE
3.	SHUT DOWN SYSTEM #1	**4.**	SHUT DOWN SYSTEM #1
5.	MODERATOR INLET	**6.**	MODERATOR OUTLET
7.	VENT PIPE	**8.**	COOLANT CHANNEL ASSEMBLY
9.	END SHIELD	**10.**	END SHIELD SUPPORT STRUCTURE ASS'Y
11.	MAIN SHELL ASS'Y	**12.**	TUBE SHEET F/M SIDE
13.	TUBE SHEET CAL SIDE	**14.**	LATTICE TUBE
15.	END SHIELD SUPPORT PLATE	**16.**	END SHIELD COOLING INLET PIPES
17.	END FITTING ASS'Y	**18.**	FEEDER PIPES
19.	OUTER SHELL	**20.**	SUPPORT LUG

Fig. 2. Integral assembly of calandria and end shield (cut-away view of reactor).

out for Narora Atomic Power Station #1 and 2 and Kakrapar Atomic Power Station #1 before initial fuel loading. In subsequent reactors, i.e. Kakrapar Atomic Power Station #2 onwards, the design is changed to have tight fitting garter springs and it was found that they did not move during commissioning.

From Narora Atomic Power Station onwards, the gap between pressure tube and calandria tube is filled with carbon-di-oxide. A continuous monitoring of moisture ingress into this gas is done, which is an indication of leak in the pressure tube/calandria tube.

2.3. Reactivity control: regulation and shutdown

Normal control of power in the reactor is done by the reactor regulating system which controls the position of control rods (220 MW_e) or water level in zone control compartments (in 540 MW_e PHWR). The system is made highly reliable by incorporating redundancy in the control loop as well as in the sensors and the control devices. Apart from normal control, reactor regulating system has an additional feature called 'setback' which can effect auto fast reduction of reactor power (@ $0.5\%\ s^{-1}$) when called for, as sensed by selected plant parameters. In 540 MW_e PHWR, there is yet another feature called 'stepback' which effects auto step reduction in power when demanded by the control system. Table 3A gives the reactivity devices and their worths for 220 and 540 MW_e PHWRs.

Reactor shutdown function is performed by systems totally independent of reactor regulating system. As mentioned before, in early reactors (Rajasthan Atomic Power Station and Madras Atomic Power Station) reactor shutdown was achieved by dumping the moderator from calandria into a dump tank located below it, by equalizing the gas pressure in the two vessels. From Narora Atomic Power Station onwards, this feature was eliminated and two new fast acting and independent shutdown systems of diverse design were adopted. Each of these systems is separately and independently capable of safely terminating any reactivity transient from any operating state of the reactor. The reactivity transients considered include those from a large loss of coolant accident, which result in the fastest reactivity addition rate in a PHWR due to coolant voiding in the core. The total required worth ("depth") of the shutdown system is governed by appropriate combinations of reactivity effects which include complete

Table 3A
Reactivity devices and worths for 220 and 540 MW_e PHWRs

Purpose	RAPS/MAPS	NAPS/KAPS/KGS	540 MW_e
1. Control and regulation	Central adjusters	Regulating rods	Zone control components (ZCCs)
Material	SS/Co	SS/Co	Light water
Number	2 × 2	2 × 2	14
Worth	4 mk	4–5 mk	7 mk
2. Xe over-ride	Corner adjuster	Absorber rods	Adjuster rods
Material	SS/Co	SS/Co	SS
Number	4 × 2	4 × 2	17
Worth	8 mk	8 mk	12 mk
3. Power reduction	Central adjuster + moderator level	Regulating rods + shim rods	Control rods + zone control components
Material			(SS-Cd-SS)
Number		2 × 2	4
Worth		6 mk	11 mk

voiding in the core, cool down of fuel, coolant and moderator; as well as decay of xenon poison in a prolonged shutdown. Table 3B gives the shutdown devices and their worths for 220 and 540 MW_e PHWRs.

The two shutdown systems are:

Shutdown System-1 (also called SDS-1, primary shutdown system, or PSS): consisting of mechanical shutoff rod mechanisms. Whenever a reactor trip signal is received, an electromagnetic clutch in each mechanism is de-energized, releasing a solid cadmium absorber element, which drops into the core under gravity, initially assisted by spring thrust. Shutdown System-1 is the primary means of quickly shutting down the reactor in an accident.

Shutdown System-2 (also called SDS-2, secondary shutdown system, or SSS): consisting (in 220 MW_e PHWR) of vertical empty tubes in the reactor core into which liquid poison (lithium pentaborate solution) is injected when required. When triggered by a trip signal, fast-acting valves between a high pressure helium tank and the poison tanks open to pressurize and inject the liquid poison into the reactor. In 540 MW_e PHWR, injection of the liquid poison is into the bulk of moderator via perforation in horizontal tubes in the calandria.

In addition to the above, in 220 MW_e PHWR, a slow-acting liquid poison injection system is provided for long-term subcriticality, to cater for slow reactivity effects such as xenon decay.

Each of the two shutdown systems has sufficient capacity to perform its safety function, i.e. to provide the required negative reactivity rate and depth, assuming a specified number of elements (one or two shutoff rods in Shutdown System-1 or one poison tube/bank of tubes in Shutdown System-2) is inoperable. The system actuation is fail-safe with respect to power or air failure.

2.3.1. Fuelling scheme

In a 220 MW_e PHWR operating at full power, about one channel is required to be refueled per day. To achieve radial flux flattening for maximum power output from the core, a differential burn-up scheme is adopted. This is done by judicious adjustment of the relative refueling rates in the different core regions. Fig. 3 gives the equilibrium core configuration with different burn-up zones for a standard PHWR (Narora Atomic Power Station onwards).

Eight bundle shift scheme is adopted to minimize fuel failures due to power ramps. This scheme also provides required

Table 3B
Shutdown devices and worths for 220 and 540 MW_e PHWRs

Purpose	RAPS/MAPS	NAPS/KAPS/KGS	540 MW_e
1. Shutdown			
SDS #1	Moderator dump	PSS (rods)	SDS1–shutoff rods
Material		SS–Cd–SS	SS–Cd–SS
Number		14 rods	28 rods
Worth		36 mk	72 mk
SDS #2		SSS	SDS–2
Material		Lithium pentaborate injection in tubes	Liquid poison—$GdNO_3$ injection in moderator
Number		12 tubes	6 nozzles
Worth		32 mk	≈300 mk
2. Hold down	Moderator dump	ALPAS (8 ppm boron addition) From KGS onwards LPIS (8 ppm boron)	

ALPAS: automatic liquid poison injection system; LPIS: liquid poison injection system.

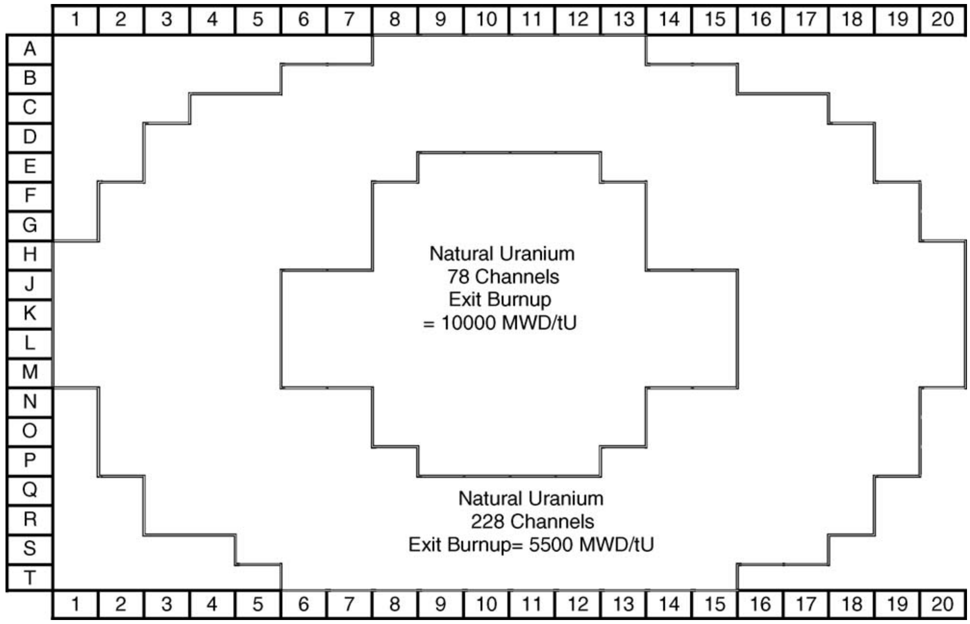


Fig. 3. Equilibrium core configuration with different burn-up zones for a standard PHWR.

reactivity gain per channel refueled as well as optimizes the fuelling machine utilization and availability.

2.4. Fuel

The fuel bundles for Indian PHWRs are short cylindrical assemblies. In each coolant channel assembly of 220 MW_e reactors, there are 12 fuel bundles; whereas each coolant channel assembly of 540 MW_e reactor has 13 fuel bundles. On-power bi-directional fuelling is carried out with the aid of two fuelling machines, one each at either end of the coolant channel. Pressure tubes containing a string of short length fuel bundles and the on-power refueling permit flexibility in choosing fuel designs and in-core fuel management parameters to maximize fuel utilization. A defective fuel can also be identified and removed from the reactor while it is in operation.

The different fuel bundle designs prepared for use in 220 MW_e reactors and that planned to be used in 540 MW_e reactors and the salient design changes made in the fuel design over the years are discussed below.

2.4.1. Nineteen-element design

The 19 element fuel bundles are used in the 220 MW_e reactors. A typical fuel element consists of a stack of sintered natural uranium dioxide fuel pellets, cylindrical in form, placed inside a zircaloy fuel tube and sealed at both ends by end plugs. Nineteen such elements arranged in concentric rings are assembled together by welding them to an end plate on each side to form a bundle. The bundle length is 495 mm and the weight is 16 kg. The different components of the fuel bundles are shown in Fig. 4.

Skewed split spacers in this design maintain gap between the fuel elements. Short bearing pads on outer elements as shown in Fig. 4 are used to provide desired gap between fuel bundle and pressure tube. The spacers and bearing pads are attached to fuel elements by spot welding.

In order to overcome the fuel failure, if any, induced due to power ramps or stress corrosion cracking of zircaloy and in general to increase the potential of the fuel element to resist the power ramp failures, graphite coating on the inside surface of the sheath is carried out.

Apart from above changes, changes in end cap design, end plate design, material technical specifications, fabrication processes used have been made from time to time to improve design, fabrication or quality control of fuel.

2.4.2. Thorium dioxide/depleted uranium fuel bundles

The thorium dioxide fuel bundles are used in the initial core for initial flux flattening to achieve full power operation in initial phase of reactor operation. The selection of the number of bundles and their locations in the core is based on the detailed physics and fuel management studies of the reactor.

Another alternative to achieve flux flattening in initial core is by use of depleted uranium fuel bundles (0.5–0.6% ^{235}U). Depending upon the requirement 10–40% of initial core can be loaded with depleted fuel bundles.

2.4.3. 37 Element fuel bundle

Thirty-seven element fuel bundle designed for 540 MW_e reactors is an extension of the closed packed 19-element fuel bundle. In this design, one more ring of elements has been added. All the elements are of a small diameter of 13 mm. The bundle has been designed to generate a bundle power of about 1 MW. This design has been studied for physics, fuel management and safety aspects.

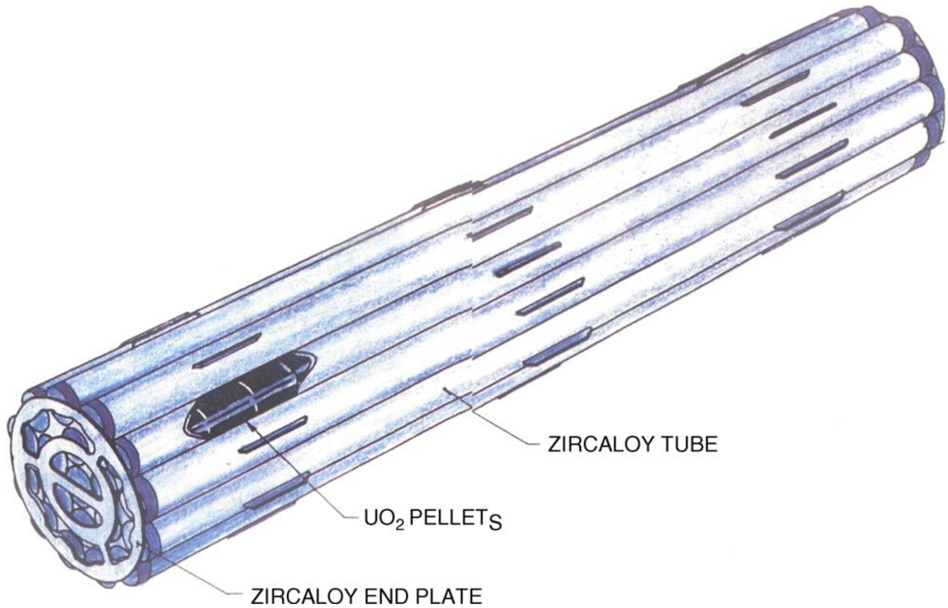


Fig. 4. Nineteen-element fuel bundle for 220 MW_e reactor.

2.4.4. Fuel performance

Fuel performance can be gauged by the fuel failure rate and also by the iodine activity in the coolant. Close monitoring of the I-131 activity in coolant system is done in operating reactors. Even though technical specifications for operation permit 100 μCi/l of I-131 in coolant, a significantly low alert level of 10 μCi is adopted by operation.

On the basis of the fuel performance experience at earlier plants Rajasthan Atomic Power Station #1 and 2 and Madras Atomic Power Station #1 and 2, changes have been incorporated in fuel design, manufacturing and reactor operating guidelines from time to time. So far about 250,000 fuel bundles of 19-element design have been irradiated in the 12 operating PHWRs. These bundles have performed well. The overall failure rate of all the units has been 0.096% in the year 2002–2003.

2.4.5. Detection of failed fuel

Concentration of iodine isotopes 131, 132, 133 and 134 in the heat transport system are measured on a routine basis. In case of a fuel failure, a first order estimate of the nature of defect in fuel and the coolant contamination caused by it is made by systematic analysis of these isotopes. However, a particular channel with a failed fuel bundle is detected during reactor operation by the delayed neutron detection system, which is provided for all Indian PHWRs. The suspected failed fuel is removed from the channel on power.

2.5. Fuel handling

The on-power fuelling operations are carried out by two remotely controlled fuelling machines, operating at each end of a fuel channel. While new fuel bundles, from one-fuelling machine are inserted into a fuel channel (in the direction of coolant flow), irradiated fuel bundles from that channel are received into another fuelling machine clamped at another end. Either machine can load or receive fuel. The fuelling machines receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The entire fuel handling operation is automated. The irradiated fuel is transported under water to the Spent Fuel Storage Bay for long-term storage of the fuel. This bay is common for two unit stations and is having its own cooling and recirculation system.

Over the years, fuel handling system has seen major evolutions in carriage design, fuel transfer system and controls in progressive stations.

In Rajasthan Atomic Power Station/Madras Atomic Power Station, fuelling machine head is supported on a mobile type carriage, which moves on rails. Narora Atomic Power Station onwards, fixed column with vertically moving bridge design has been adopted. Trolley with the fuelling machine head moves horizontally on the bridge. This design is comparatively more stable under various upset conditions including seismic.

In Rajasthan Atomic Power Station/Madras Atomic Power Station, the fuelling machine head magazine housing is of SS-403 welded construction. The welding of large sized SS-403 forgings could not be carried out indigenously. Up to Madras Atomic Power Station, magazine housing was forged and welded abroad. From Narora Atomic Power Station onwards, for ease of manufacture, bolted construction was adopted. Separate forgings are imported and are indigenously machined.

In Rajasthan Atomic Power Station/Madras Atomic Power Station, there is a long hold up of fuelling machine head on the fuel transfer port since the transfer of fuel from fuelling machine head to airlocks in a pair at a time. Narora Atomic Power Station onwards, fuel transfer system accepts all the eight spent fuel bundles into the transfer magazine, which acts as a transient storage. After transfer of eight bundles, the fuelling machine head is free to resume operation on the next channel. Moreover, this concept also allows simultaneous transfer of spent fuel from the fuelling machine head into the transfer magazine and loading of new fuel into the fuelling machine head from the transfer magazine.

While earlier fuel handling controls employed hard wired system, from Narora Atomic Power Station onward, computerized control system has been provided. This has resulted in flexibility and better man–machine communication made possible by messages on computer display screen. Changes in auto logic can be effected by software program modifications. Manual mode operations can still be done in the event of failure of computerized system. Digital panel meters are employed for more accurate read-outs. Hard wired IC-based logic is employed only for interlocks to ensure safety in manual mode and provide additional safety in auto mode.

A separate calibration and maintenance facility to test various sub-assemblies like ram assembly, separators, B-ram drive, various process devices and control equipment has been introduced in Kaiga Generating Station and in 540 MW_e units also. This is specifically meant for performance testing after major maintenance.

Rajasthan Atomic Power Station #2 onwards indigenous manufacture of major assemblies like fuelling machine head, carriage, fuel transfer system, doors, etc., has been going on from 1991. Indigenous manufacture of B-ram and other ball screws, oscillating type shaft seals, etc., has also been successfully done.

From Kaiga Generating Station onwards, tank in tank concept is used in the design of Spent Fuel Storage Bay of 220 MW_e reactors, so that any leak in the liner can be collected into the interspace.

2.6. Primary heat transport system

The primary heat transport system uses heavy water under forced circulation in a figure-of-eight loop. Figs. 5 and 6 show the schematic of the system.

The earlier units of Rajasthan Atomic Power Station #1 and 2 and Madras Atomic Power Station #1 and 2 have a total of eight branches of primary coolant pumps and steam generators with four parallel branches on each side of the reactor. Each of the branches have three isolating valves making a total of 24 isolating valves. The steam generators are of hair pin design type with 10 or 11 (at Madras Atomic Power Station) hair pins making one boiler with common steam drum. Tube material is monel. The main circuit of subsequent 220 MW_e units (Narora Atomic Power Station onwards) have undergone major changes with number of branches reduced to four with two branches on each side. The Steam Generator design has been changed to mushroom type. This design has the advantage of a single integral unit with provisions of manholes on primary head to enable tube in-service inspection (ISI). Similarly, manholes and hand holes are provided for the secondary side to enable inspection as well as tube sheet cleaning. The tube material has been changed to Incolloy 800. The primary coolant pumps of larger size and rating with lower speed of 1500 rpm instead of 3000 rpm of earlier pumps and three mechanical seals backed up by stationary seal with air purge arrangement has resulted in comparatively better performance and equipment reliability.

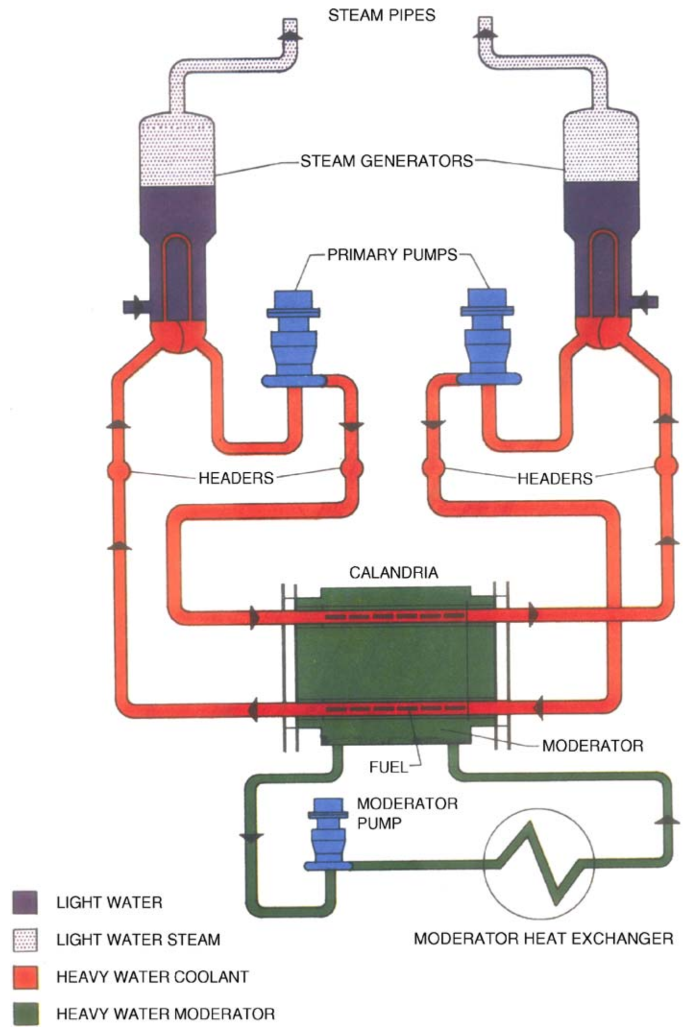


Fig. 5. PHWR simplified flow diagram.

For Narora Atomic Power Station/Kakrapar Atomic Power Station, the main circuit branches have two isolating valves each. Further design review led to a total elimination of these valves in units of Kaiga Generating Station onwards. Thus, for the latest 220 MW_e units Primary Main System is valveless. Elimination of these valves along with general reduction in the number of components has helped to decongest the layout in the pump room, facilitating better maintenance approachability, less maintenance and lesser dose uptake.

For 540 MW_e units, the main circuit has a total of four steam generators and primary coolant pumps each, is valveless with a two-loop concept. The two-loop concept has been adopted to minimize the inventory loss and in turn minimize core positive void coefficient and minimize the enthalpy release to the containment under loss of coolant accident.

During the evolution of main circuit design, effort has been made at minimizing the weld joints and in turn the in-service requirement of such joints. This has been achieved by adopting integral forging for the headers with pull out nozzles, main circuit piping with hockey stick combination for large pipe spools. The Steam Generator shells have been also specified as ring forging to eliminate longitudinal joints.

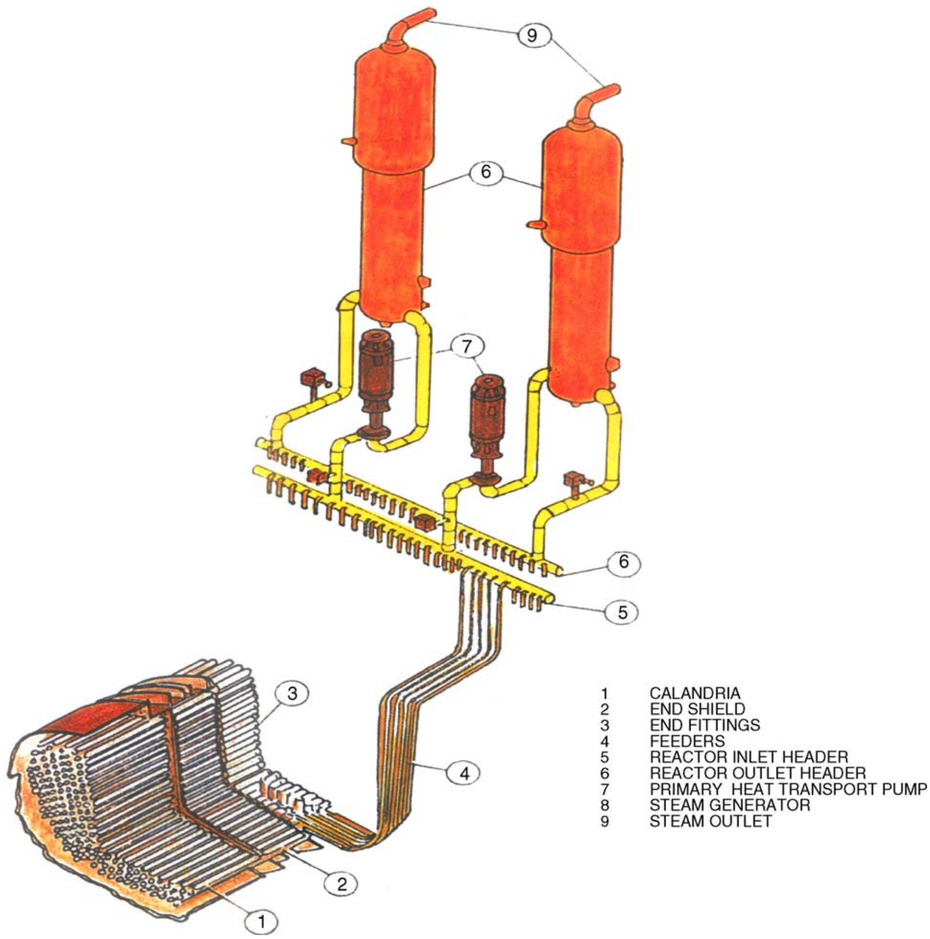


Fig. 6. Perspective view of steam generator and PHT system (one bank).

The primary heat transport system pressure control in 220 MW_e units is based on the 'feed and bleed' concept. During normal operation, the average pressure of both outlet headers is controlled at a set pressure of 8.53 MPa to keep the system 'solid', i.e. no bulk boiling is permitted. Operational transients such as turbine trip, load rejection, boiler feed pump trip, reactor power setback, etc., cause 'swell' or 'shrinkage'. The feed or bleed control valves actuate to counter these swells/shrinkages. Though the system is 'solid' there is some finite compressibility of the system, which keep the pressure changes within a fairly narrow band. The feed is provided by two feed control valves located on the discharge side of primary pressurizing pumps, which take suction from storage tank. Two bleed control valves are connected to reactor inlet header and hot bleed needs to be depressurized, cooled and returned back to storage tank via purification system.

In 540 MW_e PHWR, a pressurizer has been introduced for primary heat transport system pressure control, while feed and bleed is retained for inventory control.

Following improvements have been evolved in the design of primary heat transport system to minimize the radiation dose uptake:

- Cobalt free hard facing, i.e. Colmonoy in place stellite has been utilized for all the valves.
- Scope of in-service inspection has been minimized by reducing the number of weld joints.
- Bellow seal valves have been utilized extensively for smaller size valves to minimize heavy water leaks and necessity to replace packing.
- Concepts of double packed valves and inter packing leak-off utilized for larger size stem packed valves. Graphoil packing with live loading have been utilized.
- Concept of double gasketted joints with flexitalic gaskets with inter gasket leak-off have been utilized.

2.7. Emergency Core Cooling System (ECCS)

In the original design of earlier PHWRs (Rajasthan Atomic Power Station, Madras Atomic Power Station), the cold heavy water available in the moderator system was used for emergency injection into the primary heat transport system for postulated loss of coolant accident conditions. From Narora Atomic Power Station onwards, the system was modified to incorporate high pressure emergency injection, and to delink the system from moderator system. In current 220 MW_e PHWR, the Emergency Core Cooling System incorporates:

(a) High pressure heavy water injection.
(b) Intermediate pressure light water injection.

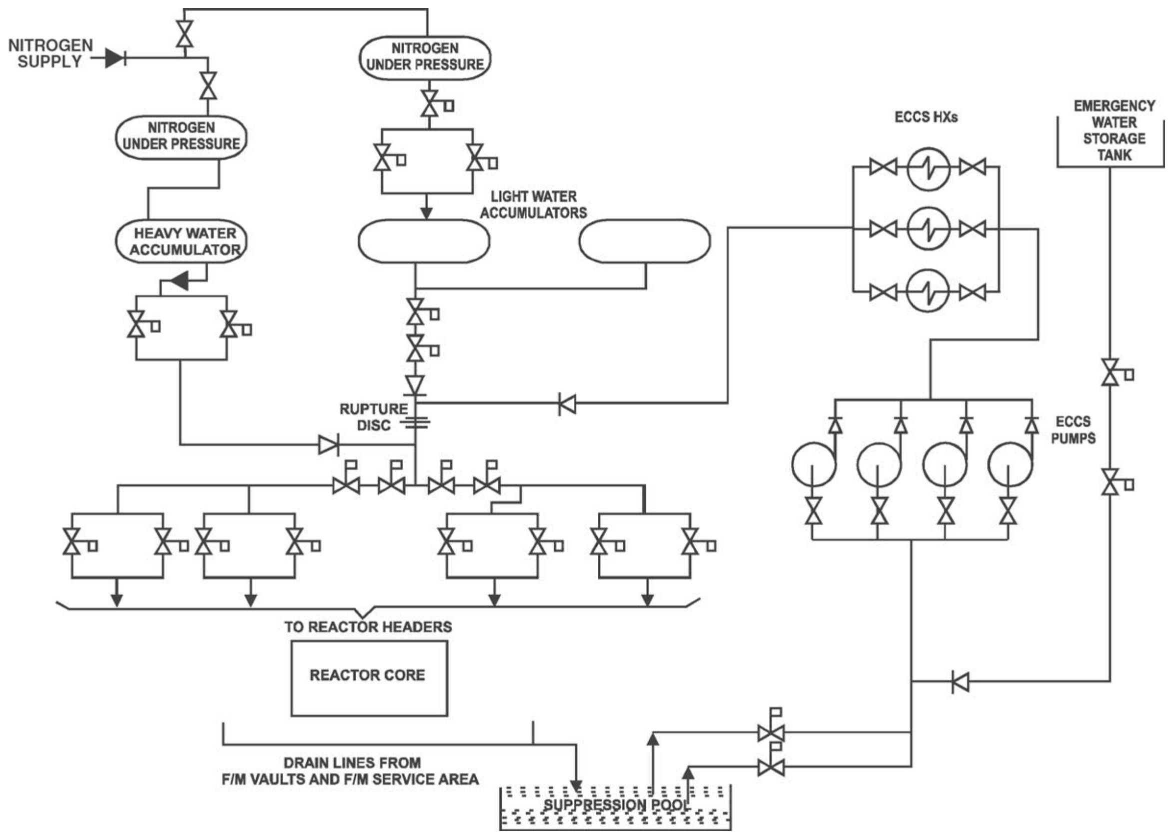


Fig. 7. Emergency Core Cooling System schematic.

(c) Low pressure long-term recirculation.

The high pressure heavy water injection is provided by a system of accumulators containing heavy water pressurized by a nitrogen gas tank. Intermediate pressure light water injection is provided by a system of two accumulators, with a pressurized nitrogen gas tank. When the primary heat transport system pressure falls further, low pressure light water injection occurs, followed by recirculation provided by pumps. The Emergency Core Cooling System pumps take suction from the suppression pool and this water is cooled in the heat exchanger (by active high pressure process water). Fig. 7 shows the schematic of Emergency Core Cooling System.

The provision of a heavy water accumulator for the initial high pressure injection permits its use during certain non-loss of coolant accident transients to make up fast shrinkage in the primary heat transport system, without downgrading the primary heat transport system heavy water.

Injection of Emergency Core Cooling System water takes place through two of the four headers, which are selected depending on the size and location of the break. For small breaks, or breaks on the outlet header side, in which flows continue in the normal direction, injection takes place into the reactor inlet headers; for large breaks on the inlet header side, which result in reversal of flow in half of the core, injection is directed to the inlet and outlet header on the side away from the break, so as to assist the flow direction. Selection of the headers takes place automatically, based on a ΔP signal between the headers, indicating the flow direction.

All actions upto and including the establishment of long-term recirculation from suppression pool are automatic.

In 540 MW_e PHWR, the high pressure injection is from light water accumulators. A simple scheme of injecting light water into all reactor headers (instead of selection scheme) has been adopted.

Very small breaks (e.g. instrument line break) within the capacity of the primary heat transport system pressurizing (or feed) pumps can be handled without involving the actuation of Emergency Core Cooling System. The spilled heavy water is collected through appropriately located drain lines, into a tank, and pumped back to the primary heat transport system storage tank after cooling it in a heat exchanger and after passing through purification circuit.

2.8. Containment

Current Indian PHWRs use a double containment principle. The annular space between the primary and secondary containment envelopes is provided with a purging arrangement to maintain a negative pressure in the space. This arrangement significantly reduces the ground level releases to the environment during accidents where there is a radioactivity release into the primary containment. Fig. 8 shows the schematic of containment.

The containment structures are of concrete. The primary containment is a pre-stressed concrete structure, consisting of a perimeter wall topped by a pre-stressed concrete dome. The

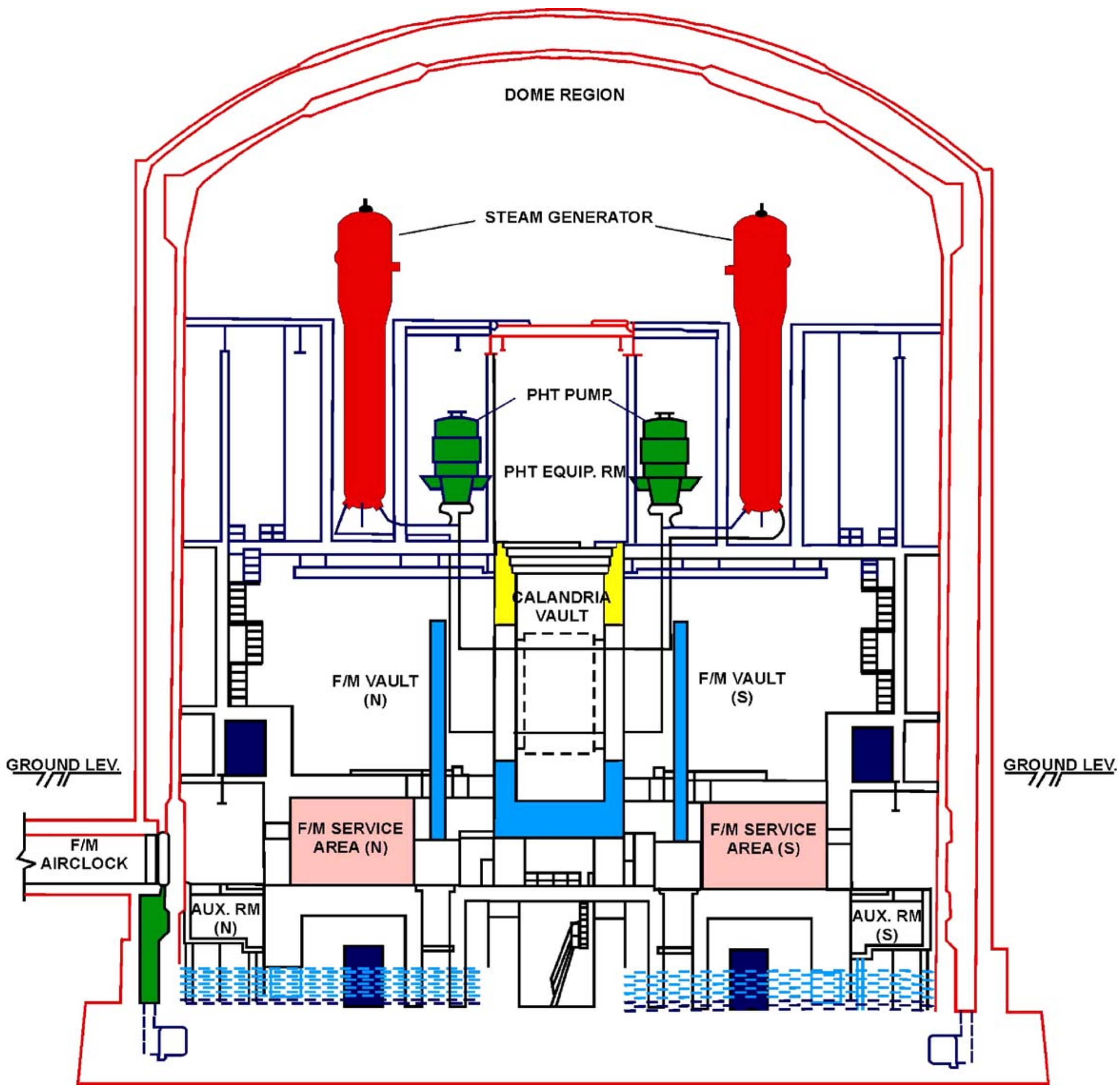


Fig. 8. Schematic on 540 MW_e PHWR containment (RB sections).

outer, or secondary containment envelope is a reinforced concrete cylindrical wall topped by a reinforced concrete dome. The primary containment uses epoxy coating as liner for added leak-tightness and ease of decontamination. Because of the use of double containment, further enhancement in leak-tightness by incorporating a steel liner is not considered necessary.

Automatic isolation of the containment is initiated in the event of (i) pressure rise or (ii) activity build-up in the containment.

A pressure suppression system incorporating a suppression pool is used for limiting the peak pressure in the containment following loss of coolant accident or a main steam line break. The primary containment building is divided into two accident-based volumes, volume V1 (‘drywell’ having high enthalpy) and volume V2 (‘wetwell’), separated by leak-tight walls and floors and connected through a vent system via a suppression pool containing water in the sub-basement. During a loss of coolant accident or main steam line break, the pressure rise in volume V1 will cause steam–air mixture to flow via the vent system to the suppression pool, where the steam will condense, and the air will escape into volume V2.

The pressure suppression system is an entirely passive system and does not perform any function during operational states. In addition to pressure suppression, the suppression pool water forms part of the long-term recirculation mode of emergency core cooling.

To cool down and thereby depressurize the containment following an accident in as short a time as possible, a system of building air-coolers, distributed at various locations in volume V1 of the containment building are used. These coolers are used during normal operation as well. The coolers are supplied from an assured process water supply; their fan motors are driven by power supplies backed by diesel generators (on-site electric power supply). Like all other containment-related engineered safety features, these coolers are designed to work in post-loss of coolant accident containment environment conditions.

In order to achieve further depressurization at low pressures (say below 3.923 kPa) which may be difficult to achieve by cooling alone, there is a provision for controlled gas discharge to the stack via filters. This system can also be used up to a maximum containment overpressure of 39.23 kPa for delayed containment venting if warranted due to accumulation of non-condensables during the post-accident phase. Usually operation of this system is not envisaged before 48 h following an accident.

For post-accident clean-up of the atmosphere in the containment, two systems are used:

(a) *Primary containment filtration and pump-back system*: In this system, air flow is recirculated within the primary containment through charcoal filters, to perform containment atmosphere clean-up operation on a long-term basis after an accident. Significant reduction in the concentration of iodine in the primary containment would be effected over a period of time, so that by the time the controlled gas discharge system is operated, say after 48 h, the associated stack releases will be low.
(b) *Secondary containment filtration, recirculation and purge system*: This system provides multi-pass filtration and mixing by recirculation within the secondary containment space, and also maintains negative pressure within it. The negative pressure maintained in the secondary containment space brings the net ground level release down to very low values.

The loads for which the containment structure is designed include normal and construction loads, abnormal loads (including pressure and temperature loads resulting from the design basis loss of coolant accident/main steam line break) and extreme environmental loads (wind and earthquake). The seismic design is based on a dynamic analysis for the safe shutdown earthquake. Combinations of loss of coolant accident and safe shutdown earthquake occurring simultaneously are considered.

2.8.1. Early evolution in containment design

The containment system of Indian PHWRs has seen several modifications in successive project. The first Indian PHWR (Rajasthan Atomic Power Station) reactors have a single containment envelope of reinforced concrete cylindrical section and a pre-stressed concrete dome. In Rajasthan Atomic Power Station, pressure suppression in the containment is provided by a dousing system where water stored in a very large tank, at the topmost floor of the containment establishes a curtain of water in the path of releasing steam during postulated loss of coolant accident, by actuation of fast-acting valves, to limit the building pressure. In Madras Atomic Power Station and all subsequent reactors a pressure suppression system is used where the released steam–air mixture is led to a large body of water (suppression pool) stored at the bottom of the containment. In Madras Atomic Power Station, a partial double containment was used with primary containment of pre-stressed concrete, and secondary of rubble masonry. From Narora Atomic Power Station onwards, all reactors are of full double containment design. In Narora Atomic Power Station and Kakrapar Atomic Power Station, for effective separation of light water and heavy water areas, it was decided to keep the light water part of steam generators out of the primary containment with provision of a sealing bellow. Accordingly, a flat roof primary containment with four holes for steam generators was used. To limit pressure rise in secondary containment due to steam line break in these reactors, blow-out panels to outside atmosphere are used. However, subsequently based on experience with erection of Steam Generator bellows, it was decided to change the design for further projects. Accordingly, from Kaiga Generating Station onwards, the primary containment is extended in height to house the steam generators entirely, and is capped by a dome. This design also eliminates the need for secondary containment blow-out panels.

2.9. Moderator system

Moderator circulation and cooling system is designed as a dual loop so that in case of any problem in one, circulation and hence cooling can be maintained through healthy loop. As brought out earlier, Rajasthan Atomic Power Station/Madras Atomic Power Station design uses moderator dumping for shutting down the reactor. Therefore, in these reactors, moderator from the dumped tank is sprayed on to the calandria tubes for their post-shutdown cooling. From Narora Atomic Power Station onwards, calandria remains filled after reactor shutdown and thus this requirement does not exist. In addition, on account of change in the system design Narora Atomic Power Station onwards; feature of moderator pump-up to fill back calandria before restarting is eliminated.

In Rajasthan Atomic Power Station/Madras Atomic Power Station design, moderator system inventory is utilized to supply to the primary heat transport system also as part of Emergency Core Cooling System in case of accident involving breach in primary pressure boundary. From Narora Atomic Power Station onwards, Emergency Core Cooling System is made entirely independent and does not have any link with the moderator system.

From Narora Atomic Power Station onwards, another design feature is introduced which will restrict fall of moderator level below the topmost row of the calandria tubes in case of accident involving loss of moderator inventory. Before this, reactor tripping and fast cool down of primary heat transport system also gets triggered automatically. These design provisions ensure that calandria tubes are always submerged in moderator and continue to get cooled.

The pumps for moderator circulation system in Rajasthan Atomic Power Station/Madras Atomic Power Station/Narora Atomic Power Station are of mechanical seal type. As a design improvement, in Kakrapar Atomic Power Station these pumps are of canned rotor design thereby eliminating seal leakage and associated tritium activity. However, in order to meet the schedule of on-going projects; Kaiga Generating Station #1 and 2 and Rajasthan Atomic Power Station #3 and 4 are provided with mechanical seal type pumps. The future plants are proposed to be provided with canned rotor pumps.

One of the requirements of keeping moderator at low temperature is to ensure that it will function as heat sink in the event of postulated accident involving loss of coolant accident with simultaneous unavailability of Emergency Core Cooling System. For this purpose, both moderator and process cooling water circulation is ensured by providing their pumps with reliable on-site power supply. The main moderator circulation system is seismically qualified.

2.10. Process water system

Process water system is used to pickup heat from various process loads and transfer it to the ultimate heat sink. Since, this system picks up heat with potentially active systems, process water system acts in a closed loop to avoid any radioactivity release to the atmosphere.

From Narora Atomic Power Station onwards, backup is provided from Fire Water System to some of the critical process water loads to ensure continued cooling of these loads in case of failure of process water. A surge tank is provided in the system to take care of system leakages.

The location of the loads to be serviced by the process water system governs the pumping requirements of this system. Depending upon this, process water system is divided into two sub-systems, namely, high pressure and low pressure. Also, depending upon the potential radioactivity of the serviced loads, process water system is also classified into active and non-active sub-systems. All these classifications also mean that the number of systems serviced by one sub-system are limited, thereby any failure of one sub-system will not lead to non-availability of process water to all the serviced loads. Further, to enhance system availability; process water system pumps are provided with reliable on-site power supply and are seismically qualified.

2.11. Control and instrumentation

In spite of the usual conservative approach to adoption of newer technologies, control and instrumentation is an area of nuclear reactor design, which has changed most during past 50 years. The explosive growth in the field of analog and digital electronics induced this metamorphosis.

The instrumentation in the 1960s when Tarapur Atomic Power Station #1 and 2 were built, was mostly pneumatic and the actuators were mainly hydraulic or pneumatic. These systems were extremely rugged but required periodic maintenance because of wear and tear caused by mechanical movement. Reactor control and protection system of Rajasthan Atomic Power Station #1 and 2 and Madras Atomic Power Station #1 and 2 was made using conventional discrete analog circuits. The process instrumentation was more or less still pneumatic.

With Narora Atomic Power Station the scene started changing rapidly. Many pneumatic transmitters were replaced by electronic transmitters. The magnetic amplifiers were replaced by solid state amplifiers, large number of discrete components were replaced by just one or two integrated circuits. The computerization of control systems was attempted for systems having sequential logic and large data acquisition. The fuelling machine control and channel temperature monitoring were the systems where initiation of computerization was carried out. Intel 8085-based microprocessor boards were used for triple redundant reactor regulating system of Narora Atomic Power Station. Control room computer system was another step towards centralization of acquisition of control room data with improved human–machine interface. The introduction of digital systems (computers) did away with the problems of analog circuit design like drift, poor noise, etc.

By the time Kakrapar Atomic Power Station #1 and 2 were being designed we were already in the 1980s and more steps were taken towards utilizing the flexibility of computer-based systems. The boldest of the steps was to computerize the alarm generation function. The system called Programmable Digital Comparator System was designed using 8086-based boards. The radiation data acquisition system was another system computerized for Kakrapar Atomic Power Station #1 and 2.

Till Kakrapar Atomic Power Station #2, the computerization was on block-by-block basis without any change in the functional requirements of corresponding analog systems. The requirement specification focused on input–output behavior only; thus the capabilities of microprocessors were under utilized and there was scope for improved system architecture leading to execution of optimized algorithms and improvement in reliability and availability of the system. Systems with Dual Processor Hot Standby architecture and Multinodal architecture were evolved for Kaiga Generating Station #1 and 2 and Rajasthan Atomic Power Station #3 and 4. The Dual Processor Hot Standby based systems were applied for major process and reactor control application. Use of token bus-based local area network for information gathering and display in programmable logic controllers for the plant control systems and in Dual Processor Hot Standby–Reactor Regulating System and Dual Processor Hot Standby–Process Control System provided valuable insight into the development of networked systems. Microprocessor-based single loop controllers were also introduced.

Upgradation of old projects was done in decade of 1990s and lot of old systems was reengineered with computer-based systems. Smart transmitters were used for the first time in second unit of Rajasthan Atomic Power Station.

With year 2000 came the awareness about the amount of computerization already done and the realization of possible hazards. The challenge of year 2000 was successfully handled and also helped increase in awareness about embedded system and other indirect uses of computers. Regulators also became aware of the new task of licensing the computer-based systems. The designers mainly at Bhabha Atomic Research Centre, Electronic Corporation of India Limited and Nuclear Power Corporation of India Limited successfully handled the challenge of getting the computer-based systems of Kaiga Generating Station #1 and 2 and Rajasthan Atomic Power Station #3 and 4 verified and validated.

In design of Tarapur Atomic Power Station #3 and 4 reactors, the new concepts used in instrumentation and control systems include—incore detectors for reactor power and power distribution monitoring, control and protection, use of fiber optic cables for noise free transmission of signals and process control related information, use of Ethernet in control and monitoring systems, introduction of information gateways and reactor power display system, introduction of user interface for Multiple Input Alarm System, which otherwise is a fully hardwired system, use of online trip parameter system testing and test and monitoring of protection systems. Operator interface with menu driven screens for control action and system information was also introduced.

2.12. Safety analysis

As part of initial licensing as well as periodic renewal of license, in depth safety analysis for the plant is carried out; which involves both the deterministic and probabilistic approaches. While the safety analysis of the new plant is based on the system design; any periodic review will focus on the changes carried out in the plant since the last review and changes in the analysis methodology.

The safety analyses for licensing purposes is based on conservative assumptions and is carried out as per the guidelines agreed with the regulatory authorities.

The safety assessment of PHWRs has traditionally followed some unique rules, which have become standard practice. These include (i) consideration of multiple failures involving a postulated initiating event (PIE) coincident with unavailability of a mitigating safety system. (ii) Ignoring of the first trip parameter in shutting down the reactor. These rules are over and above the usual safety analysis assumptions, e.g. taking no credit for offsite power supply, considering single failure in each mitigating system, etc.

Beyond design basis events are addressed through offsite emergency measures. For each station, a detailed offsite emergency preparedness manual is prepared. This manual is prepared in association with local government authorities who have to play an equally important role in the management of beyond design basis accident situations.

3. Operating experience

Nuclear Power Corporation of India Limited has undertaken to make nuclear energy a safe, economically viable and environmentally benign source of electricity generation in the country.

As mentioned before, India started building nuclear power plants in the 1960s and continued to add more units progressively to meet the growing electricity demand in the country. Today, India has 14 operating nuclear power units with a total installed capacity of 2770 MW_e. Starting with a humble beginning of commissioning the first unit at Tarapur in 1969 to commissioning of four units at Kaiga and Rajasthan in the year 2000, India has made significant progress in this field.

Old units		New units			
Madras Atomic Power Station, Unit-1	Madras Atomic Power Station, Unit-2	Kaiga Generating Station, Unit-2	Rajasthan Atomic Power Station, Unit-3	Kaiga Generating Station, Unit-1	Rajasthan Atomic Power Station, Unit-4
∼1122 days	∼810 days	550 days	381 days	172 days	161 days

With an accumulated experience of more than 200 reactor years of safe operation, Indian nuclear power plants now rank among the best in the world. Having demonstrated the capability to design, construct and operate the nuclear power plants, performance improvement has been the major thrust area for the last few years at Nuclear Power Corporation of India Limited. As a result, the annual average capacity factor of Indian Nuclear Power Plants steadily improved from 60% in 1995–1996 to 90% in 2002–2003. Relevant performance details are given in Figs. 9 and 10. As per rankings put out by Candu Owners Group (COG) for the calendar year 2002, the performance of Nuclear Power Corporation of India Limited was ranked as first among the leading operating reactors groups in the world in terms of gross capacity factor (Ref. COGnizant, vol. 8, issue 2, February 2003). Also Kakrapar Atomic Power Station #1 was ranked as the best operating unit for the year 2001–2002, in terms of gross capacity factor (Ref. COGnizant, vol. 8, issue 1, January 2003).

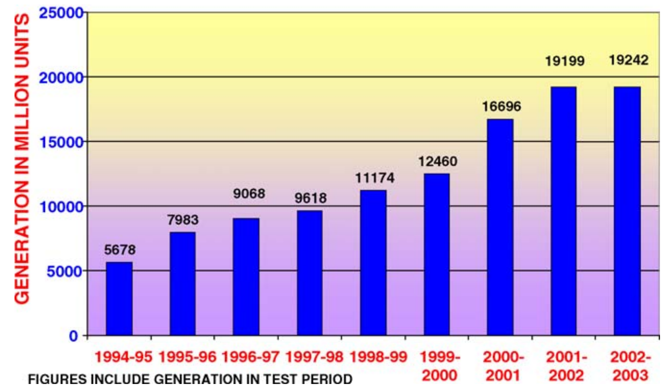


Fig. 9. Operating performance of NPCIL.

Various important factors that have significantly contributed towards improved performance of the nuclear power plants in India are briefly dealt with in the following paragraphs.

3.1. Commissioning

Commissioning time is the time taken from hydro testing of the heat transport system to commercial operation of the plant. Expertise is achieved learning from the past experience and taking corrective actions in commissioning the new plants in a short time. Kaiga Generating Station #1 and Rajasthan Atomic Power Station #4 were the last 220 MW_e plants to have been commissioned. Commissioning time of these plants was considerably less than that of earlier plants as is evident from the data given below:

3.2. Quality assurance and peer reviews

Safety and reliability of a nuclear power plant are of utmost importance requiring constant vigilance by all the concerned

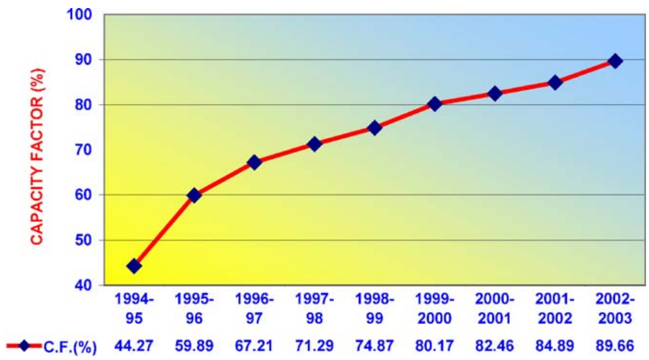


Fig. 10. Performance of NPCIL.

agencies. A good quality assurance program implemented through the stages of planning, execution and verification brings about a quality performance. To achieve this, best quality measures were introduced during all stages of operation and maintenance. This has helped in creating a safe and productive work culture at Nuclear Power Corporation of India Limited, resulting in continued improvement in the overall performance of its plants.

Self-assessment and peer reviews are the other well known tools to identify strengths and weaknesses, which help in taking steps to achieve even higher standards of operational performance. To this end, most of the Indian Nuclear Power Stations have been subjected to international peer reviews conducted by teams of experts from the World Association of Nuclear Operators, a non-governmental body based in Tokyo, Japan. Apart from this, all Indian Nuclear Power Stations have also obtained ISO-14000 certification for environmental excellence. Improved operating performance and cost reductions have been natural spin-offs of the efforts made in living up to the international standards and achieving ISO certification.

3.3. Operating experience feedback

The operating experiences feedback from other nuclear power plants and some times from non-nuclear power plants helps in the solution of existing problems. A well defined mechanism was established at each Indian Nuclear Power Plant for review of operating experience feedback available from Indian as well as international sources in the form of unusual occurrence reports, incident reports, and industrial safety reports. An in depth study of these reports resulted in identifying the areas of improvements applicable to individual stations. These were then vigorously followed up for necessary modifications. The important sources of information include International Atomic Energy Agency, World Association of Nuclear Operators, Candu Operators Group and our own operating plants.

The proactive measures thus taken, in addition to preventing recurrence of events that happened elsewhere, have enhanced the safety and good performance of the plant.

3.4. Root cause analysis

The root cause analysis is performed to determine appropriate corrective actions for the removal of the root cause and other observed deficiencies. This is utilized for improving and developing procedures and preventing events.

The station management at all Indian Nuclear Power Station has over the years resorted to various tools of root cause analysis to take advantage of findings which in turn have resulted in many design/procedural improvements.

3.5. Improved maintenance practices

Most of the structures, systems and components of nuclear power plants are continuously under varying influence of material degradation due to neutron irradiation, dynamic stresses, thermal fatigue, creep, corrosion, erosion, wear, vibration, etc. The maintenance of systems, structures and components in nuclear power plants plays an important role in assuring their safe and reliable operation.

Excellent performance of the operating plants has been achieved mainly because of the improved maintenance practices adopted at Nuclear Power Corporation of India Limited. Some of the important features of these practices are as follows:

- Implementation of predictive and condition based maintenance.
- Implementation of computerized maintenance management system.
- Deployment of special manipulators and tooling.
- Deployment of latest maintenance strategies.
- Post-maintenance testing.
- Conduct of maintenance audits.
- Development of maintenance performance indicators and their trending.
- Root cause analysis of failures and improvements based on lessons learnt.
- Standardization of procedures.

The use of mock-ups and adequate training to improve the skills of personnel has paid rich dividends by way of radiation dose reduction and lower costs of maintenance apart from the improvement in station performance.

3.6. Outage management

One of the most important factors contributing towards improving the capacity factor of a power station is effective management of planned and unplanned outages. Although there is no ‘refueling outage’ requirement because of on-power fuelling, reactor shutdowns are still required for periodic in-service inspection, surveillance, tests and maintenance activities. Completing all the maintenance jobs that have been accumulated during operation, and carrying out modifications that are required for better operation and maintenance in a short time is the key for better outage management. To ensure this, lot of attention has been paid where continuous efforts to reduce the time taken for planned outages have yielded excellent results. Planned outage periods of PHWR stations have come down from 60 days earlier to around 20 days now.

3.7. Training and qualification

The standard of performance of a work force is decided by the quality of training that is imparted to it. Also, having qualified personnel is a must to ensure safe and reliable operation of the nuclear power plants. The responsibility of looking after the training needs of Nuclear Power Corporation of India Limited personnel has been entrusted to well-established Nuclear Training Centers located at Rajasthan, Madras and Kaiga Atomic Power Stations and training sections at all power stations.

The training generally starts with an induction training (class room and field training) and then 'on the job' training. The procedure outlines the eligibility criteria, training requirements and administrative norms for verification of competence, assessment methodologies, retraining requirements and re-qualification procedures.

The importance given to training and qualification by Nuclear Power Corporation of India Limited has resulted in a steady improvement in the capacity factor of its plants in a safe manner.

4. Safety performance

4.1. Environmental releases

4.1.1. Gaseous wastes

The gaseous wastes comprise tritium, fission product noble gases such as xenon and krypton, argon-41, iodine and radioactive particulates. While tritium is generated as an activation product of deuterium in the primary heat transport system and moderator system containing heavy water, the fission product noble gases, iodine and particulates are generated in the fuel. These gaseous wastes are filtered through a pre-filter, absolute filter, charcoal filter and discharged through a 100 m tall stack. A micro-meteorological laboratory is attached with each nuclear power station. The laboratory collects, among other things, data on wind speed and direction continuously. From these data, the dispersion factors are calculated at different downwind distances from stack.

4.1.2. Liquid wastes

In PHWRs, the liquid wastes originate mainly at the following points:

- Personnel showers, active laundry, etc. (potentially active).
- Heavy water upgrading plant, reactor building sump, heavy water clean-up rooms (tritiated waste).
- Laboratories, decontamination center, etc. (active chemical waste).

Liquid waste generated at the plant is collected in tanks at the Liquid Effluent Segregation System, which is located in the service building. Subsequently, the waste is pumped to the Treatment and Disposal System of the Waste Treatment Plant, which is equipped with facilities for chemical treatment, purification by ion exchange, evaporation, etc. Treated liquid waste is discharged to the water body after adequate dilution with condenser cooling water/blow down water. In some plants, nearly 70% of the low active liquid waste is discharged to atmosphere through a Solar Evaporation Facility. As a result, waste volume reduction by a factor of up to 100 is achieved.

4.1.3. Monitoring of radioactive effluents

4.1.3.1. Gaseous waste. The radioactive gaseous waste is discharged through a stack. The discharged gaseous waste is sampled, analyzed for radioactivity and the data is maintained on line by Stack Monitoring System.

A multinozzle probe located in the stack delivers a representative sample of gaseous effluents to the detectors of fission product noble gases, iodine and particulate activity. Fission product noble gas is detected by sodium iodide detector kept inside the sample chamber. Iodine is trapped in a charcoal filter and detected by sodium iodide detector. Particulate activity is trapped by an absolute filter and detected by plastic scintillators. Tritium activity is absorbed in gas washing bottles. Analysis of the water sample in liquid scintillation spectrometer indicates tritium activity. The monitoring system of radioactive effluents has standby detectors and is powered by a reliable source. The monitors provide information on instantaneous activity release rate and cumulative release.

4.1.3.2. Liquid waste. Liquid waste after treatment, sampling and monitoring is diluted with condenser cooling water/blow down water and discharged to the environment water body through a single point. A continuous sampling provision is made at a point downstream of effluent injection in the discharge channel after thorough mixing has taken place. The collected samples are analyzed in the laboratory.

4.1.4. Impact assessment

An Environmental Survey Laboratory is located at each site in the country and operated by an independent organization, viz. Health Physics Division of Bhabha Atomic Research Centre. The primary aim of the environmental monitoring program is to demonstrate compliance with the radiation exposure limits set for members of the public. This requires detailed measurement of radioactivity content in a number of environmental matrices. The samples are selected on the basis of potential pathways of exposure. The number and type of samples and sampling frequency can be site specific depending on the nature of the operations, utilization of the local environment, existence of population clusters, etc. On an average 2500 samples are collected and analyzed per year by Environmental Survey Laboratory at every site. The radioactivity content in the environmental matrices is assessed using sophisticated radiochemical analysis. From the mean values of the concentration in different food items, the internal dose resulting from intake of these food items is computed. External dose due to release of noble gases is computed based upon released quantity of these noble gases and micro-meteorological data at the site. Continuous evaluation of the meteorological data is carried out for this purpose. The sampling program covers a distance of about 30–40 km from the installations and includes aquatic, atmospheric and terrestrial samples.

The dose to population at the fence resulting from nuclear power plant operation considering all pathways, i.e. submersion, inhalation and ingestion is about 1.5% of the authorized dose limit which is a small fraction of the natural background radiation. The doses at further distances are still lower.

4.2. Significant events and feedback for improvements

All events at operating plants having a bearing on safety are subjected to multi-tier review in utility as well as in regulatory body, and lessons learnt from them are incorporated into designs and operation.

All significant events at plants are categorized as per the International Nuclear Event Scale (INES) of IAEA. Since the inception of this categorization in 1992, there has been one event (Narora Atomic Power Station fire incident in 1993) at level 3, and four at level 2. All others have been below this, with most at level 0, i.e. below scale.

Selected major significant events that have been experienced are described below.

4.2.1. Fire incident in Narora Atomic Power Station

The fire in Narora Atomic Power Station #1 (220 MW_e PHWR) in March 1993 was initiated by sudden failure of two turbine blades, with the resulting imbalance vibrations leading to rupturing of hydrogen seals and lube oil lines, culminating in fire. The fire spread to several cable trays, relay panels, etc., in a short duration. The control room operators responded by tripping the reactor by manual actuation of primary shutdown system after 39 s from the incident and also initiated fast cool down of the reactor by opening (2× 10% capacity) Atmospheric Steam Discharge Valves on the main steam line.

As the fire had spread through the generator bus duct in the turbine building into the Control Equipment Room where fire barriers had given way, the power cables as well as control cables from the source were damaged. Hence, even though the power sources were available, neither power supply from Grid, the diesel generators or from the batteries were available. Therefore, a complete loss of power supply in the Unit occurred in about 7 min of the incident, which resulted in an extended Station Blackout lasting for a period of 17 h. During the blackout, the core cooling was maintained by thermo siphoning in the primary side with steam generators on secondary side fed by fire water as heat sink. There was no radiological impact of the incident either on the plant workers or in the public domain. The major fire could be put out in about 1 h 30 min. Based on the degradation of defense-in-depth of the engineered safety features during the incident, Atomic Energy Regulatory Board classified this event at Level 3 of the International Nuclear Event Scale.

The detailed investigation and reviews following this incident, both by the regulatory body and by the utility, resulted in several modifications and improvements in various areas covering design, operation and administrative and surveillance practices. One study was with regard to the susceptibility of the existing design and layout of Narora Atomic Power Station to common cause failure, mainly due to fire as an initiating event. Consideration was given to preventive measures for avoiding common cause failures, as well as to need for additional mitigative measures for assured core cooling, etc., in situations such as station blackout.

The review, initially carried out for Narora Atomic Power Station, was subsequently extended to cover all other operating stations. The design provisions of nuclear power plants under construction at that time (Rajasthan Atomic Power Station #3 and 4 and Kaiga Generating Station #1 and 2) were also checked against the recommendation of this study, and where required remedial actions were taken.

Based on the lessons learnt from this incident, key modification implemented included (a) rerouting of cables for station power supplies through diverse routes for redundant trains, (b) strengthening of fire barriers to ensure localization of fire zones, (c) increased provisions to ensure control room habitability, (d) TG system related improvement in design and operation to minimize blade failure and (e) improvements in Station Black Out handling capability.

4.2.2. Leak from PHT system in Madras Atomic Power Station (Unit-2)

In March 1999, during annual shutdown of Madras Atomic Power Station #2, the moderator system was also under shutdown for maintenance. During this period, in-service inspection of the channels was in progress. One of the shutdown cooling pumps was operating. The core configuration was such that shutdown cooling circuit of the other bank was isolated and one out of four steam generators of that bank was connected. During the in-service inspection job, leak from one channel started, which was plugged in about 30 min. For this much duration, the shutdown cooling pump was stopped to minimize heavy water leakage. Since the plug used during in-service inspection operation was not repositioned properly, it was decided to replace this plug by normal seal plug so as to restart shutdown cooling. However, this effort was not successful and the special plug was re-installed with persistent leak. Subsequently, shutdown cooling pump was restarted after 2.5 h of stoppage. Later, the normal seal plug was put back during which period the shutdown cooling pumps were again kept stopped for around 18 minutes. Inventory in the reactor core was made up by periodic addition of fire water to the heavy water storage tank during this period.

This incident was reviewed with respect to acceptability of following considering shutdown cooling pump stoppage:

(i) PHT temperature rise and calandria tube rolled joint temperature.
(ii) Decay heat removal through thermo siphoning.
(iii) Discharge rate from the leaky channel.

The review of the incident has resulted into the following recommendations:

- Even in the event of leak from coolant channel end fitting, operation of the shutdown cooling pump away from the leakage location can be continued.

- Adequate time (around 20 min) is available for fire-water addition to heavy water storage tank.
- In the prolonged shutdown situation, thermo siphon cooling through shutdown cooling heat exchangers can also be resorted to, although it is recognized that this path is less effective than thermo siphon through steam generators.
- In the absence of any flow through primary heat transport system, decay heat removal after prolonged shutdown is possible by making the heat loss paths through moderator, end shield cooling, feeder cabinet and fuelling machine vault air available.

4.2.3. Flood incident at Kakrapar Atomic Power Station

Heavy rains at Kakrapar Atomic Power Station together with non-operation of weir control for adjoining water pond caused the flooding at the plant. Many equipments got flooded in the turbine building due to flood water entry through removed cover on the tunnel connecting to the turbine building basement. During the incident, Unit-1 of Kakrapar Atomic Power Station was under long shutdown whereas Unit-2 was under commissioning stage. During the incident offsite power supply failed and for core cooling, on-site power supply was used. As process cooling water pumps in turbine building also got submerged, fire water, which is a backup to process water, was used for core cooling through shutdown cooling system.

The event was analyzed in depth and corrective measures were taken not only at Kakrapar Atomic Power Station but at other stations and projects also. Some salient recommendations are:

(i) All the entry points of pipe tunnels/cable trenches to the basement of all buildings having safety related equipment are sealed with multiple barriers.
(ii) The openings (like manholes, ventilation points) are either raised above design basis flood level or are sealed.
(iii) A number of administrative measures (like maintenance/inspection of flood prevention measures) and review/modifications of flood protection plans/emergency operating procedures.

5. Upgradation/ageing management

All operating PHWRs have an ageing management program in place to monitor and correct the effects of ageing. The major elements of the Program are:

(a) Identification of systems and components important to safety for which ageing needs to be considered.
(b) Understanding dominant ageing mechanisms in the selected components.
(c) Monitoring of ageing, i.e. detecting component degradation before failure.
(d) Timely mitigation/replacement.

Some features of the PHWR with respect to ageing management are:

Table 4
Indian PHWR fuel channels

Reactor	In-service date	Pressure tube material	Garter spacers no./fit
RAPS-1	1971	Zircaloy-2	2/loose
RAPS-2			
Original	1981	Zircaloy-2	2/loose
Refurbished	1998	Zirconium–niobium	4/tight
MAPS-1 and -2	1984	Zircaloy-2	2/loose
	1986	Zircaloy-2	2/loose
NAPS-1 and -2	1991	Zircaloy-2	4/loose
	1992	Zircaloy-2	4/loose
KAPS-1	1993	Zircaloy-2	4/loose
KAPS-2 and later	1995 onwards	Zirconium–niobium	4/tight

- Other than equipment inside the calandria vault, all other key equipments are amenable to monitoring, refurbishing and replacement. The equipment inside the calandria vault (calandria, end shields and moderator system piping inside the vault) consists of low pressure, low temperature systems made of stainless steel which are resistant to radiation-induced or other degradation. Failure, if any, in these systems would not be of a catastrophic nature and would not pose a radiological risk to the public; however, there could be an economic penalty since the equipment is hard to access.
- By appropriate design specification and material selection, a minimum design plant life of 30–40 years has been ensured for all major equipment, other than zircaloy-2 pressure tubes. For the latter, it is recognized that replacement is required after 12–15 years. Zircaloy-2 pressure tubes are used in PHWRs up to Kakrapar Atomic Power Station #1. Subsequent units use zirconium–niobium, which has a longer working life.

5.1. Fuel channels

As indicated in Table 4, current PHWRs (Kakrapar Atomic Power Station #2 onwards) have pressure tubes of zirconium–2.5% niobium with four tight fitting spacers. The material for these tubes is quadruple melted to control trace elements for improved fracture toughness; the initial hydrogen concentration specified for this material is <5 ppm.

For the earlier Pressurized Heavy Water Reactors, in which zircaloy-2 pressure tubes are used along with two or four loose spacers, a well-structured assessment and life management program is in place which includes in-service inspections, post-irradiation examination as well as maintenance/replacement of channels as required.

Table 5 summarizes the life management strategies for pressure tubes of the PHWRs.

A high priority element of this program is prevention of formation of unacceptable hydride blisters. Monitoring and repositioning of displaced spacers is therefore an important part of the program.

En-masse replacement of coolant channels with those of the current design is being done in these early units. This activity

Table 5
Life management strategies for pressure tubes of Indian PHWRs

Ageing mechanisms	Potential consequences	Units*	Monitoring and corrective actions
Irradiation-enhanced deformation (wall thickness, i.d., axial creep)	Deformation exceeding design limits	RAPS, MAPS, NAPS	Monitor deformation in sample channels during in-service inspection Periodic axial repositioning of creep stops to accommodate elongation
		KAPS onwards	En-masse replacement of channels before elongation exceeds allowance provided in channel bearing length Monitor deformation in sample channels during in-service inspection Periodic axial repositioning of creep stops to accommodate elongation
Delayed hydride cracking initiating from hydride blister at pressure tube/calandria tube contact	Pressure tube rupture or failure of leak before break	RAPS, MAPS, NAPS, KAPS-1	Monitor pressure tube/calandria tube gap or sag profile, and spacer positions in selected channels during in-service inspection Identify channels which could have pressure tube/calandria tube contact from combination of vibration diagnostic techniques, calculations based on ISI data on spacer locations and direct inspection of selected channels Compute hydrogen pickup and blister depth at contact location, based on conservative data and assumptions Quarantine/replace channel with unacceptable computed blister depth Reposition spacers in channels in which inspection has shown shifting and in which blister has not yet exceeded acceptable depth En-masse channel replacement based on assessment of hydrogen content and contact time
		KAPS-2 onwards	Periodic inspection (normal) No special action required: 4 tight fit spacers ensure no pressure tube/calandria tube contact
Delayed hydride cracking initiating from stress concentration (e.g. from service induced flaw)	Failure of leak before break	All units	Volumetric/surface examination of sample channel during base line inspection, in-service inspection For any detected flaw exceeding calibration standard, evaluate cause, effect on pressure tube integrity; repair/quarantine/replace channel if required
Changes of tube properties during operation	Failure of leak before break		Material surveillance: post-irradiation examination of selected channels from lead units; scrape samples
Reduction in fracture toughness (critical crack length) due to hydrogen pickup + irradiation.	Failure of leak before break		Operating procedure for avoidance of cold pressurization Replacement whenever hydrogen levels are unacceptable

MAPS: Madras Atomic Power Station; NAPS: Narora Atomic Power Station; KAPS: Kakrapar Atomic Power Station.
* RAPS: Rajasthan Atomic Power Station.

is taken up at an appropriate stage for each unit (after 8.5–12 full-power-years of operation) based on:

(i) predicted time for pressure tube and calandria tube contact, and consequent blister growth;
(ii) axial elongation exceeding allowances available in the channel bearings; and
(iii) deterioration in fracture toughness due to hydrogen/deuterium pickup, which reduces the margins on meeting the leak-before-break criteria.

The operating procedure for startup and shutdown of the plants avoids cold pressurization of the pressure tubes. This reduces the probability of delayed hydride cracking, and also increases the probability of leak before break.

For detection of leaks in pressure tubes, in Narora Atomic Power Station onwards, an annulus gas system is used. In reactors prior to Narora Atomic Power Station (i.e. Rajasthan Atomic Power Station and Madras Atomic Power Station), where the annulus between pressure tube to calandria tube is open to the calandria vault atmosphere, the leak detection system is based on detection of moisture in the calandria vault atmosphere. The sensitivity of this leak detection method is assessed to be adequate to satisfy the leak before break criterion.

The pressure tube life management program ensures that the margins available for safe operation of the pressure tubes continue to be adequate at all times.

5.2. Other backfits—improvement in Rajasthan Atomic Power Station #2 during retubing

As part of the replacement of the pressure tubes in Rajasthan Atomic Power Station #2, significant upgrades were performed to bring the safety systems to current standards. These are listed below.

(1) Containment dousing system
The original containment dousing system at Rajasthan Atomic Power Station incorporates a flow modulating feature whereby dousing flow varies in proportion to the velocity of steam–air mixture flowing in the vicinity of the dousing curtain. This velocity in turn is expected to depend on the discharge flow rate from a postulated loss of coolant accident. This scheme has been modified to a simpler optimized one wherein the modulating feature has been done away with, and a fixed dousing flow rate has been set which can cater for all loss of coolant accident break sizes.
(2) Emergency Core Cooling System
In the original design, Emergency Core Cooling System is provided by circulation of moderator heavy water using the normal moderator circulation pumps at relatively low pressure (0.58 MPa and below). In the modified design, Emergency Core Cooling System has been enhanced by adding a high pressure accumulator injection system capable of supplying at a pressure of 5.4 MPa.
(3) Supplementary control room
A supplementary control room has been added at Rajasthan Atomic Power Station away from the main control room. The instrumentation in this room has separate cables and power supplies independent of the control room. The feature provided ensures that the supplementary control room along with local panels/controls has the capability to perform essential safety functions (safe shutdown and decay heat removal as well as monitoring of plant safety status) independently of the main control room.
(4) Minimizing instrument air in-leakage into containment
During postulated loss of coolant accident conditions, requiring boxing up of containment, the continued inleakage of instrument air into the containment results in its gradual repressurization. To minimize the instrument air in-leakage, a design modification has been made whereby valves requiring continued air supply under loss of coolant accident conditions have been provided with a separate air supply, so that the main instrument air supply to the reactor building can be cut-off during containment box-up conditions. This air cut-off is envisaged to be a manual action about 1 or 2 h into the accident.
(5) Augmentation/improvements in station electric power supplies
The backup power supply to the Class III buses, which originally was based on 2× 100% capacity diesel generators, has been augmented by adding a third diesel generator set. This diesel generator, having a diverse cooling system (air cooling) from the existing ones, is located above the maximum anticipated flood levels that might arise from postulated failure of the upstream dam.

Re-routing of cables for power supplies to safety related loads has been carried out for improved segregation of redundant trains of supplies.