ADVANCED HEAVY WATER REACTOR (AHWR)

BARC (India)

Overview

Full name	Advanced Heavy Water Reactor
Acronym	AHWR
Reactor type	HWR
Coolant	Light Water
Moderator	Heavy Water
Neutron Spectrum	Thermal Neutrons
Thermal Capacity	920 MW(th)
Electrical Capacity	304 MW(e) gross 284 MW(e) net
Design Organization	Bhabha Atomic Research Centre
Last update	11-06-2013

1. Introduction

1.1. Full and abbreviated name of the nuclear plant design

Full and abbreviated name: Advanced Heavy Water Reactor (AHWR)

1.2. Summarized historical technical basis

The first phase of the Indian nuclear power programme is based on natural uranium fuelled, heavy water moderated pressure tube type reactors commonly designated as Pressurized Heavy Water Reactors (PHWRs), also known as CANDUs for such reactors of Canadian origin. Eighteen out of twenty Indian nuclear power reactors under operation, and four out of seven Indian nuclear power reactors under construction, at the beginning of April 2013, are PHWRs. The first two of these reactors, Rajasthan units –1 and –2 are similar in design to the Canadian Douglas Point reactor. Rajasthan–1 was built at Rawatbhata in India with Canadian collaboration. This reactor started commercial operation in November 1972. Subsequently, the construction of Rajasthan–2 and design and construction of all subsequent Indian PHWRs was done indigenously in India. The design of Indian PHWRs has progressively been improved and augmented to take into account the feedback from national as well as international experience with such reactors. A large infrastructure was set up at Bhabha Atomic Research Centre (BARC), Mumbai to facilitate research, design, and development in several areas relevant to PHWRs. These areas include: materials technologies, critical components and new systems, reactor physics, thermal hydraulic and safety analysis codes, testing and qualification of reactor systems and equipment, and design and development of systems for in-service inspection and ageing management.

The AHWR, being a pressure tube type heavy water moderated reactor, makes use of the PHWR specific technologies pertaining to pressure tube and low pressure moderator based design. These technologies are already developed and successfully demonstrated internationally. There are, however, several significant differences, between the PHWR and the AHWR. These differences are mainly related to the use of thorium based fuel with negative void coefficient of reactivity, the use of boiling light water in natural circulation mode as coolant, and incorporation of several passive safety features aimed at achieving a grace period of seven days and elimination of the need for emergency planning beyond the plant boundary. The concept of the reactor was developed in early nineties. Its basic design, and experimental development in areas required to establish feasibility of the basic design, have been completed at BARC. Several major experimental facilities have been set-up, and some others are under construction to produce additional data. This also include a critical facility, with a capability to simulate the AHWR core lattice and fuel configurations, and a full height integral test loop to simulate the Main Heat Transport (MHT) system of the AHWR.

The design and development of this reactor has been fully funded by the Government of India, Department of Atomic Energy (DAE). This work is mainly carried out at BARC, a constituent unit of DAE.

1.3. Summarized design features and rationale **(safety philosophy, applications, high-level design characteristics etc)**

The Indian Advanced Heavy Water Reactor (AHWR) is designed and developed to achieve large-scale use of thorium for the generation of commercial nuclear power. This reactor will produce most of its power from thorium, with no external input of uranium-233 in the equilibrium cycle.

The AHWR is a 300 MWe, vertical, pressure tube type, boiling light water cooled, and heavy water moderated reactor. The reactor incorporates a number of passive safety features and is associated with a closed fuel cycle having reduced environmental impact. At the same time, the reactor possesses several features, which are likely to reduce its capital and operating costs.

Important Safety Features of the AHWR

- Slightly negative void coefficient of reactivity.
- Passive safety systems working on natural physical laws.
- Large heat sink in the form of a Gravity Driven Water Pool with an inventory of 8000 m^3 of water, located near the top of the reactor building.
- Removal of heat from core by natural circulation of coolant.
- Injection of cooling water by Emergency Core Cooling System directly inside fuel cluster
- Two independent shutdown systems.
- Passive poison injection into the moderator in the event of non-availability of both the shut down systems due to wired system failure or malevolent action.

Some Distinctive Features of the AHWR

- Flooding of V1 volume (space containing high enthalpy reactor systems) in case of LOCA
- Core catcher with bottom flooding
- Elimination of high-pressure heavy water coolant resulting in reduction of heavy water leakage losses, and eliminating heavy water recovery system.
- Elimination of major components and equipment such as primary coolant pumps and drive motors, associated control and power supply equipment and corresponding saving of

electrical power required to run these pumps.

- Shop assembled coolant channels, with features to enable quick replacement of pressure tube alone, without affecting other installed channel components.
- Replacement of steam generators by simpler steam drums.
- Higher steam pressure than in PHWRs.
- Production of 2400 m^3/day of demineralised water in Desalination Plant by using steam from LP Turbine.
- A design objective of no exclusion zone on account of its advanced safety features

1.4. Summary level technical data

TABLE 1 SUMMARY LEVEL TECHNICAL DATA OF THE AHWR

General plant data		
Reactor thermal output	920	MWth
Power plant output, gross	304	MWe
Power plant output, net	284	MWe
Power plant efficiency, net	30.9	%
Mode of operation	Base load / load following mode	
Plant design life	100	years
Plant availability target	90	%
Seismic design basis	OBE & SSE	
Primary Coolant	Boiling light water	
Secondary Coolant	Light Water	
Moderator	Heavy water	
Thermodynamic Cycle	Rankine	
Type of Cycle	Direct	
Non-electric application	Desalination - 2400 m^3/day	
Safety goals		
Core damage frequency	5.46E-8	/RY
Occupational radiation exposure	0.02	Per Person-Sv/RY
Operator Action Time	7	days
Nuclear steam supply system		
Steam flow rate at nominal conditions	408	kg/s
Steam pressure/temperature	7 / 558	MPa(a)/K
Feedwater flow rate at nominal conditions	408	kg/s
Feedwater temperature	403	K
Reactor coolant system Core inlet		
Primary coolant flow rate	2145	kg/s
Reactor operating pressure	7	MPa(a)

Core coolant inlet temperature	532.5	K
Core coolant outlet temperature Average exit quality	558 19	K %
Mean temperature rise across core	25.5	K
Reactor core		
Active core height	3.5	m
Average linear heat rate	10.8	kW/m
Average core power density	10.1	MW/m^3
Fuel material	(Th, ^{233}U)MOX and (Th, Pu)MOX	
Cladding tube material	Zircaloy-2	
Outer diameter of fuel rods	11.2	mm
Rod array of a fuel assembly	54	Pin cluster
Number of fuel assemblies	452	
Enrichment of reload fuel at equilibrium core	Ring 1: (Th, ^{233}U)MOX/3.0 Ring 2: (Th, ^{233}U)MOX/3.75 Ring 3: (Th, Pu)MOX/ 4.0 (Lower half) 2.5 (Upper half)	Wt %
Average discharge burnup of fuel	38	MWd/kg
Burnable absorber (strategy/material)	NIL	
Control rod absorber material	Boron carbide (B_4C) packed in Stainless Steel tubes placed inside absorber assembly	
Soluble neutron absorber	Gadolinium nitrate solution	
Calandria		
Inner diameter of cylindrical shell	6900	mm
Wall thickness of cylindrical shell	32	mm
Total height, inside	5000	mm
Base material	SS 304L	
Design pressure/temperature	0.127 / 373	MPa/K
Transport weight	50	t
Fuel channel		
Number	452	
Pressure Tube (PT) inside diameter	120	mm
Core length	3.5	m
PT material	Zr-2.5%Nb	
Steam generator		
Type	Steam drums with gravity separation	
Number	4	
Other components		
Reactor coolant pump (if applicable)	Not Applicable	
Pressurizer (if applicable)	Not Applicable	m^3

Moderator system		
Moderator volume, core	117	m^3
Inlet/outlet temperature	323/348	K
Primary containment		
Type	Pre-stressed concrete	
Overall form (spherical/cylindrical)	Cylindrical with dome	
Dimensions (diameter/height)	52/75.5(54.5 above Ground)	m
Design pressure/temperature	0.274/410	MPa(a)/K
Design leakage rate	< 0.1	volume % /day
Is secondary containment provided?	Yes	
Residual heat removal systems		
Active/passive systems	Active : Condenser Passive : Isolation Condenser in GDWP	
Safety injection systems		
Active/passive systems	Passive : Emergency Core Cooling System	

2. Description of nuclear systems

2.1. General

AHWR is a land-based nuclear power station. The reactor is designed to produce 920 MW of thermal power, generating 300 MW(e) (gross), and 2400 m^3/day of desalinated water. The plant can be configured to deliver higher desalination capacities with some reduction in electricity generation. AHWR based plant can be operated in base load, as well as in load following mode. The target lifetime load factor and availability factors for AHWR are 80% and 90% respectively.

2.2. Main characteristics of the primary circuit

A vertical cross-sectional view of the AHWR reactor block is shown in Fig. 1, and a simplified schematic diagram of the AHWR based nuclear power plant is given in Fig. 2.

The reactor core is housed in calandria, a cylindrical stainless steel vessel containing heavy water, which acts as moderator and reflector. The calandria, located below ground level, contains vertical coolant channels in which the boiling light water coolant picks up heat from fuel assemblies suspended inside the pressure tubes. The coolant circulation is driven by natural convection through tail pipes to steam drums, where steam is separated for running the turbine cycle. The four steam drums (only one shown for clarity), receive feed water at stipulated temperature to provide optimum sub-cooling at reactor inlet. Down-comers, four from each steam drum, bring the flow to a circular inlet header, which distributes the flow to each of the 452 coolant channels through individual feeders.

Inside the calandria, a zircaloy–4 calandria tube surrounds each of the pressure tubes, to provide an annulus, which separates the cold moderator from the hot pressure tube. An annulus leak monitoring system is incorporated to detect any leakage from either pressure tube or calandria tube.

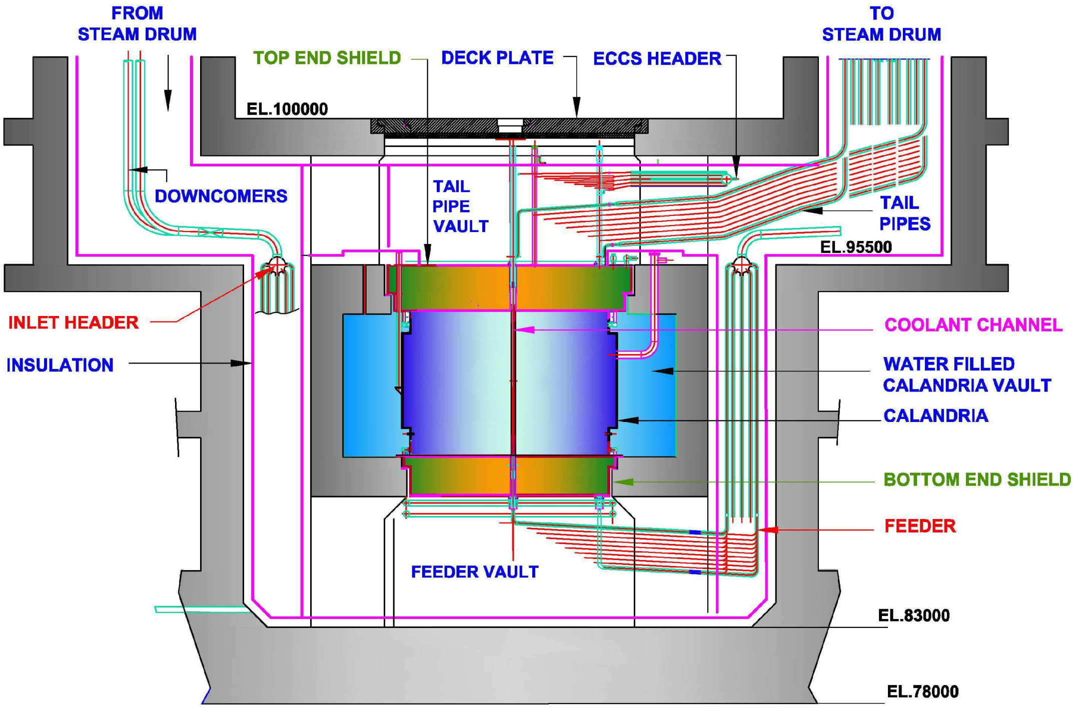

FIG. 1. AHWR Reactor Block.

During shutdown, passive valves establish communication of steam drums with the isolation condensers submerged inside a 8000 m^3 capacity gravity driven water pool (GDWP) for decay heat removal, under hot shut-down condition. The pool acts as a heat sink for passive decay heat removal system.

In case of loss of coolant accident (LOCA), the emergency core cooling system (ECCS) with four independent circuits (only one is shown for clarity) is actuated in passive mode, as the depressurisation of the MHT system progresses, following a LOCA.. ECCS operation consists of high pressure injection system using accumulators and a low pressure injection system using GDWP as source of water, in a sequential manner. This can provide core cooling for at least 7 days by ECCS water flooding feeder and tail pipe vaults in V1 volume.

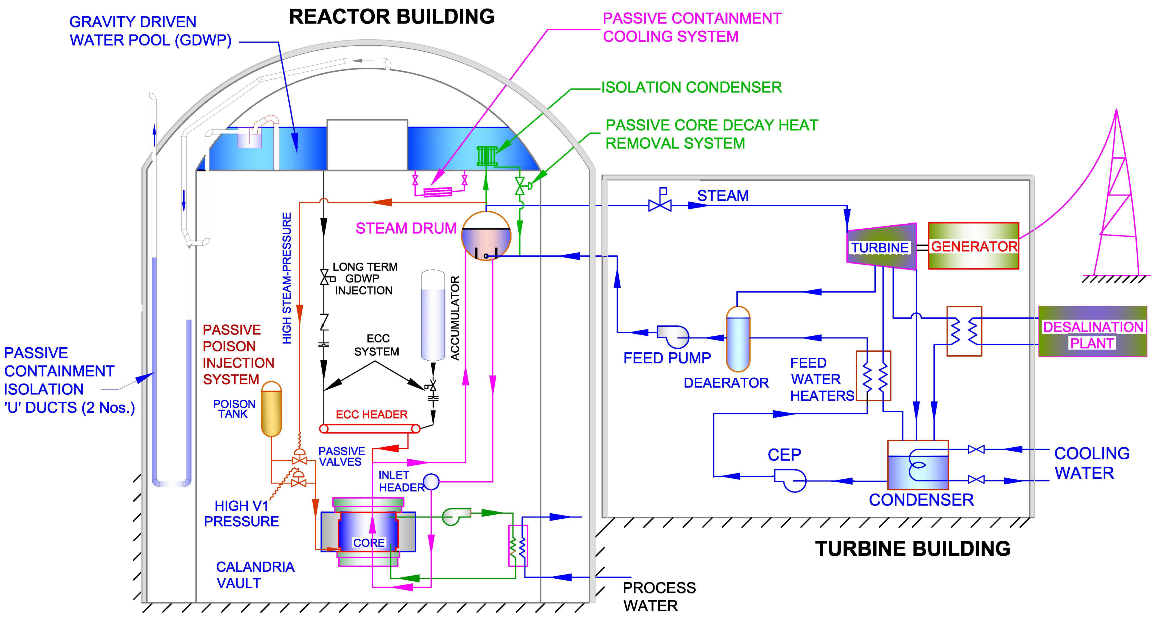

FIG. 2. Simplified schematic of AHWR.

The Reactor Protection System comprises two independent fast acting shutdown systems. Shutdown System–1 (SDS–1) is based on mechanical shut-off rods with boron carbide based absorbers in 37 lattice positions, providing a total negative reactivity worth of 74 mk with all rods inserted, and a worth of 51 mk with two maximum worth rods not available. Shutdown System–2 (SDS–2) is based on a liquid poison injection into the moderator. In addition, a pressurized addition of poison, passively driven by steam pressure, takes place in the event of over pressure in the MHT system. In addition, for long-term sub-criticality control, there is a provision to add boron to the moderator.

2.3. Reactor core and fuel design

The core consists of total 513 lattice locations arranged in square pitch of 225 mm. There are 452 coolant channel assemblies, 8 absorber rods, 8 regulating rods, 8 shim rods and 37 shut off rods in the core.

The circular fuel cluster of AHWR (Figure 3) contains thirty (Th,^{233}U) MOX pins and twenty-four (Th, Pu) MOX pins, along with a displacer rod at the centre. The inner ring of 12 pins has a ^{233}U content of 3.0% by weight and the middle ring of 18 pins has 3.75% ^{233}U. The outer ring of (Th,Pu)MOX pins have average of 3.25% Pu. The lower half of the active fuel will have 4.0 % Pu and the upper part will have 2.5 % Pu.

A ferrule type spacer design, while giving minimum resistance to coolant flow, offers an option to reconstitute the plutonium pins, thus enhancing fuel burn-up and power from thorium. The central structural tube along with holes in it, allows direct injection of ECCS water into the fuel cluster.

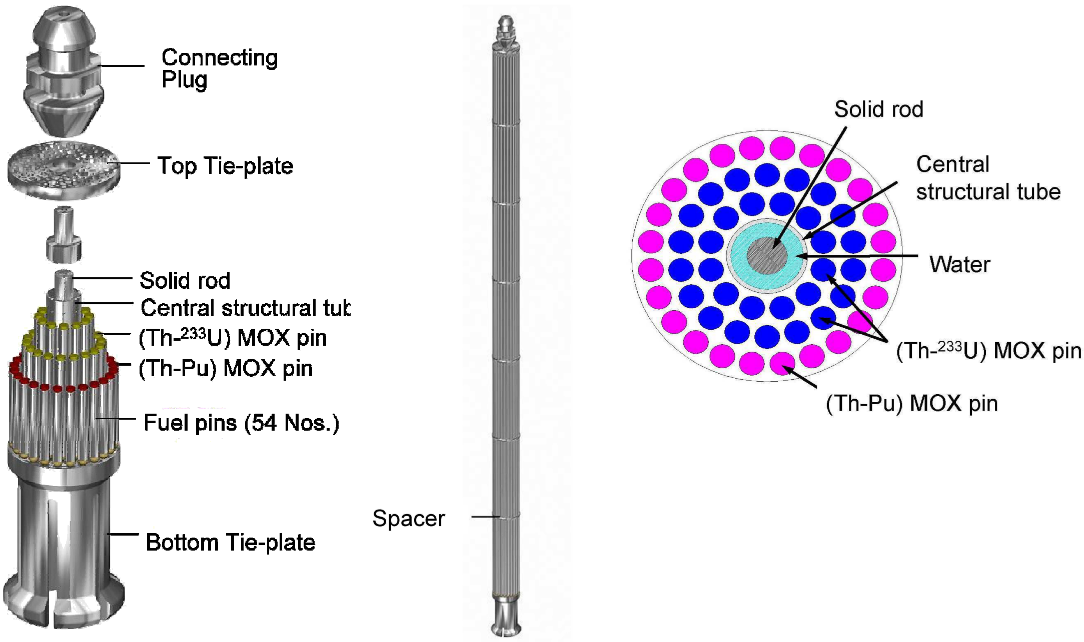

FIG. 3. AHWR fuel cluster.

2.4. Fuel handling systems

Provision is made for on-line refuelling by the fuelling machine which is located on top of the reactor block. The Fuel Storage and Handling System is required to replace spent fuel cluster with new fuel cluster in the reactor core to maintain the requisite reactivity for reactor operation. New and spent fuel clusters are stored in Fuel Storage Bay located in the fuel building. Each new fuel cluster is brought from fuel building to Temporary Fuel Storage Bay (TFSB) located inside reactor building using an Inclined Fuel Transfer Machine (IFTM). The shielded fuelling machine picks the fuel cluster from TFSB and loads into the pre-selected coolant channel. Each spent fuel cluster is removed from the channel by the fuelling machine and is transferred to the temporary fuel storage bay from where it is taken to the fuel building using inclined fuel transfer machine. To get optimum burn up from the fuel, reshuffling of the fuel clusters within the reactor core is carried out using the fuelling machine. Refuelling and transfer operation is carried out remotely from the main control room of the reactor.

2.5. Primary circuit component description

Following are the main components of primary circuit.

(1) Coolant Channel Assembly: Channel Assembly consists of Zr 2.5% Nb pressure tube in the core region extended above the top end shield and below the bottom reflector region by top and bottom stainless steel end fittings respectively. The end fittings are attached to pressure tube by rolled joint. Pressure tubes in the core region house the fuel assemblies and the hot coolant flows past it removing the heat from the fuel. It is located inside a Zircalloy-4 Calandria tube, which is rolled to the tube sheets of top and bottom end shields. The annular gap between pressure tube and calandria tube filled with CO_2 gas serves as thermal insulation between hot coolant and cold moderator. Coolant enters the channel through the feeder coupling of bottom end fitting, flows past the fuel assembly, and goes out through the tail pipe welded to top end fitting to the steam drum.

(2)Tail pipes : There are 452 tail pipes connecting the corresponding coolant channel to the steam drums. Tail pipes supply steam water mixture to the steam drum from the coolant channel.

(3) Steam Drum: Steam drum is a, horizontally mounted, cylindrical vessel closed at both ends by the torispherical heads. The two-phase steam-water mixture produced in the reactor core enters steam drums through tail pipes connected to the coolant channels. The feed water enters the steam drum through a sparger, which runs along the length of the steam drum and located in the space between the partition plates. Steam is taken out from each steam drum through outlet nozzles located on the top of the steam drum.

(4) Downcomer: MHT water flows from each steam drum to the inlet header through downcomers. Four downcomers are connected to the bottom of the steam drum shell in between the partition plates.

(5) Inlet Header: The header is the largest pipe in the MHT system. To facilitate core submergence even in case of a postulated rupture of the header, the header is installed 2.0 m above the active core. The header is connected to the 16 downcomers through forged nozzles. Water from the downcomer enters the header from the top through the 16 nozzles. Water leaves from the header through 452 nozzles to the feeders.

(6) Feeder Pipes: The austenitic stainless steel feeders from the Inlet header are connected to the lower end of the martensitic stainless steel bottom end fittings by feeder couplings with self energising metal C-rings as sealing element. Feeder pipes supply cooling water to the coolant channel from the Inlet header

2.6. Auxiliary systems

Shut Down Cooling System

The Shut Down Cooling System (SDCS) is capable to remove the decay heat and cool down the Main Heat Transport System (MHTS) of AHWR.

The primary mode of decay heat removal and cool down of MHTS is Main Steam Condenser. Normal cool down from 558 K to 423 K is achieved by main steam condenser or passively by Isolation Condensers (if the main steam condenser is not available). During hot shut down condition, steam generated due to decay heat is fed to main steam condenser. This steam, after condensation, will be pumped back to the steam drum by feed water pumps. In case of non-availability of main steam condenser, a passive system is available to perform this function by diverting the steam flow to Isolation Condensers (ICs) by operation of passive valves. The ICs, submerged in GDWP, will dissipate the heat to the GDWP.

Cooling of the MHTS from 423 K to 328 K and maintaining it at 328 K during prolonged shutdown is achieved by MHTS Purification Coolers. Selection of cold temperature as 328 K has been arrived from consideration of maintenance of MHTS equipment.

The Gravity Driven Water Pool (GDWP) recirculation and cooling system

The Gravity Driven Water Pool (GDWP) is a large pool of water containing 8000 m^3 of water inventory, located in the dome region of the reactor building, to cater to the cooling requirements of different systems during various reactor conditions. The GDWP consists of eight interconnected compartments.

The GDWP is provided for

1. Removal of core decay heat during shutdown (Hot & cold shutdown) by condensation of steam flowing through Isolation Condensers (ICs) (submerged in GDWP)
2. Low pressure coolant injection directly into the core (after the Loss of Coolant Accident (LOCA) for 3 days
3. Vapour suppression during LOCA
4. Removal of containment heat by condensation of steam from the air-steam mixture across the Passive Containment Cooler (PCC) tubes (during and after LOCA).

The GDWP recirculation and cooling system, is provided for cooling the GDWP inventory catering to above requirements as well as recirculating, filling and draining the water in each compartment of the GDWP. It consists of four heat exchangers, four pumps, filters, ion exchangers and a chemical addition tank to maintain the water chemistry.

2.7. Operating modes

Start Up

A separate cold start up procedure for the reactor is required to avoid low power flow oscillations. As per this start-up scheme, boiling in the MHTS is permitted only at 7.0 MPa. To keep the system in single phase during cold start up, pressure in the steam drum is maintained at a value higher than the saturation pressure corresponding to core exit temperature. The desired pressure in the steam drum is maintained by supplying steam from the start-up boiler.

Full Power Operation

In this operational state, all MHT auxiliary systems like pressure control, over pressure protection, feed and steam system, MHT purification system shall be in normal operating mode. While IC system, ECCS, GDWP and all safety systems shall be in readiness to operate (if required on demand).

Under normal operating conditions, the pressure in the steam drum is maintained at 7.0 MPa. The steam flow rate is about 408 kg/s. The feed water with the same mass flow rate enters the steam drum at 403 K. The primary circulation flow rate will be of the order of 2145 kg/s maintaining the average steam exit quality at about 19%.

Operation at different power levels

The natural circulation flow rate depends on the operating conditions such as the operating pressure, power and feed water flow rate and its temperature. The natural circulation flow rate has also been predicted at partial power conditions considering the feed water temperature variation and its effect. The flow rate increases with power, the increase being steep at low powers compared to that at higher powers. The required feed water temperature above 50% full power is 403 K considering the sub-cooling requirement to maintain adequate stability margin. However, at reactor power levels less than 50% full power, the feed water temperatures need to be reset to different values.

Hot Shutdown and cold shut down state

These modes are covered in section 2.6.

2.8. Standard Fuel cycle

A closed nuclear fuel cycle is to be used for the AHWR reactor. Thorium, ^{233}U and plutonium will be recovered from the spent fuel. The recovered thorium and ^{233}U will be recycled back as Th-233U MOX fuel and reprocessed plutonium will be stored, which will be later used as fuel for a fast breeder reactor (FBR). The plutonium requirement for the reactor will be met by reprocessing of the spent fuel of PHWRs. A schematic of the fuel cycle for AHWR is given in Fig. 4. The fuel cycle facilities (fabrication and reprocessing) for AHWR will be co-located with the reactor at the same site.

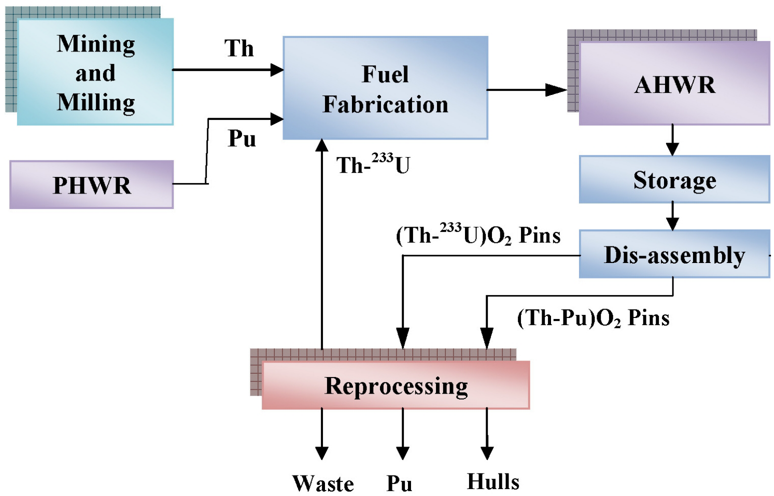

FIG. 4. Fuel cycle for AHWR.

2.9. Alternative fuel options

AHWR can be used for diverse fuel cycle options including once through and closed fuel cycles. AHWR is also optimised to achieve high burn up with LEU-Thorium based fuel in AHWR300-LEU. The design provides for inherent safety characteristics through achievement of required reactivity coefficients.

AHWR300-LEU is fuelled by (Th-LEU) MOX fuel. The equilibrium core has 444 fuel channels. The fuel cluster is similar as explained in section 2.3. The content/fraction of LEU in (Th-LEU) MOX fuel is different for the different rings of fuel pins in the fuel cluster. The inner ring fuel pins have 30wt% LEU and the middle ring fuel pins have 24wt%. Two fuel pins of the inner ring also have 5wt% Gd_2O_3 in the MOX fuel as an integral burnable absorber to suppress the excess reactivity of the cluster. The outer ring of fuel pins have (Th-LEU) MOX with 18wt% LEU in the lower half of the fuel stack and 14wt% LEU in the upper half of the fuel stack. The equilibrium core cluster will give average discharge burnup of 60 MWd/kg. The other thermal hydraulic parameters and design of various systems remains similar.

First AHWR to be constructed as technology demonstration plant is expected to be AHWR300-LEU.

2.10. Spent nuclear fuel and disposal plans if any

Reprocessing and recycling of both fissile and fertile materials is adopted to facilitate low consumption of fuel resources. A closed fuel cycle leads to reduction of radio-toxicity of waste to a lower extent. Use of thorium leads to reduction in generation of minor actinides. The use of thorium as fuel practically leads to elimination of the generation of minor actinides from a non-plutonium bearing fuel of AHWR. This advantage is especially evident in AHWR-300LEU design.

2.11. Examples of energy systems with NPPs of this kind, if any

There are no similar plants. This reactor is of newer design and is of innovative type as per the definition of IAEA. However, similarities exist vis a vis BWR/PHWR are moderator system, direct steam system, cooling systems for end shields & calandria vault, etc.

3. Description of Safety concept

3.1. Safety concept and design philosophy and licensing approach

The emphasis in design has been to incorporate inherent and passive safety features to the maximum extent, as a part of the defence in depth strategy. AHWR design provides a grace period of 7 days for absence of any operator or powered actions in the event of accident. The main objective has been to establish a case for elimination of the need for planning for evacuation in case of accident scenario in the plant. This is achieved through various passive and active safety systems designed to mitigate consequences of Design Basis Accidents (DBA) and features to avoid escalation of DBA to severe accident.

3.2. Provision for simplicity and robustness of the design

Incorporation of several passive safety features, low power density in the core, good thermal characteristics of the thoria based fuel, and large coolant inventory are some of the important provisions for simplicity and robustness in design.

3.3. Active and passive systems as well as inherent safety features; indication of whether the system is the main or backup system

Passive Safety Systems and Inherent Safety Features

The main features in these categories are listed below:

Inherent safety features

- Negative void coefficient of reactivity.
- Natural circulation driven heat removal during normal operation and hot shutdown.
- Double containment system.
- Four independent ECCS trains.
- Direct injection of ECCS water into the fuel cluster.

Passive safety systems

- Passive injection of high pressure and low pressure emergency core coolant through the use of one-way rupture disks and non-return valves.
- Passive containment isolation, following a large break LOCA, with a water seal.
- Passive shutdown by injection of poison in the moderator by use of system steam

pressure in case of failure of wired systems of SDS-1 and SDS-2.

- Passive containment cooling system

3.4. Defence-in-depth description

Some major highlights of the AHWR design, structured in accordance with the various levels of defence in depth are brought out below:

Level 1: Prevention of abnormal operation and failure

(A) Elimination of the hazard of Loss of Coolant Flow:

Heat removal from the core under both normal full power operating condition as well as shutdown condition is by natural circulation of coolant. This eliminates the hazard of a loss of coolant flow.

(B) Reduction of the extent of overpower transient:

The characteristics of AHWR design, which help to achieve this target, are as follows:

- Slightly negative void co-efficient of reactivity.
- Low core power density.
- Negative fuel temperature coefficient of reactivity.
- Low excess reactivity.

(C) Continuous monitoring of plant state:

The condition of all important equipment items and components will be continuously monitored on line. For example, the annulus leak monitoring system is incorporated to monitor any postulated leakage from either the pressure tube or the calandria tube.

Level 2: Control of abnormal operation and detection of failure

The characteristics of AHWR design, which help achieve this objective, are as follows:

- An increased reliability of the control system achieved with the use of high reliability digital control using advanced information technology.
- Increased operator reliability achieved with the use of advanced displays and diagnostics using artificial intelligence and expert systems.
- Large coolant inventory in the main coolant system.

Level 3: Control of accidents within the design basis

The following features contribute to the achievement of this objective:

- Increased reliability of the ECC system, achieved through passive injection of cooling water (initially from an accumulator and later from the overhead GDWP) directly into a fuel cluster through four independent parallel trains.
- Increased reliability of a shutdown, achieved by providing two independent shutdown systems, one comprising the mechanical shut off rods and the other employing injection of a liquid poison into the low pressure moderator. Each of the systems is capable of shutting down the reactor independently.
- Further enhanced reliability of the shutdown, achieved by providing a passive shutdown device operated by steam pressure for injection of poison in case of

extremely low probability case of failure of both shut down systems.

- Increased reliability of decay heat removal, achieved through a passive decay heat removal system, which transfers the decay heat to GDWP by natural circulation.
- Large inventory of water inside the containment (about 8000 m^3 of water in the GDWP) provides a prolonged core cooling meeting the requirement of grace period.

Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents

The following features contribute to the achievement of this objective:

- Use of moderator as heat sink.
- Presence of water in the calandria vault.
- Flooding of reactor cavity following a LOCA.

Level 5: Mitigation of radiological consequences of significant release of radioactive materials

The following features help in passively bringing down the containment pressure and in minimizing any releases from the containment following a large break LOCA:

- Double containment
- Passive containment isolation
- Vapour suppression in GDWP
- Passive containment cooling

3.5. Safety goals (core damage frequency, large early release frequency and operator grace period)

For AHWR, the goal for the frequency of severe core damage can be set at least one order of magnitude lower compared to the goal for new reactors of present generation, i.e., it should be less than or equal to 10^{-6} per reactor per year. As the reactor employs passive heat removal systems, this goal appears to be reasonable and achievable. Peak cladding temperature value greater than or equal to 1473 K is considered to lead to core damage in Level-1 PSA study that is carried out for AHWR. Similarly, a value of 10^{-7} per reactor per year can be set as a goal for large early release frequency.

The point value for the core damage frequency is predicted by BARC to be less than 1.0e-07 per reactor year. This value is about two orders of magnitude lower than the value specified for the current generation reactors.

3.6. Safety systems to cope with Design basis accidents and severe accidents (beyond design basis accidents)

A major objective of the design of AHWR has been to provide a capability to withstand a wide range of postulated events without exceeding the specified fuel temperature, thereby maintaining fuel integrity. The safety analysis of AHWR has identified an exhaustive list of 43 postulated initiating events. The events considered are categorized as follows:

- Decrease in coolant inventory (Loss of Coolant Accidents)
- Increase in coolant inventory
- Increase in heat removal

- Increase in system pressure / Decrease in heat removal
- Decrease in coolant flow
- Reactivity anomalies
- Start-up shutdown transients
- Multiple failure events
- Failure of wired shutdown systems and other BDBAs
- AHWR specific events (Defuelling, refuelling of AHWR channel)

Safety analysis included the analysis of 4 transients due to failure of wired systems of SDS-1 and SDS-2 and reactor shut down effected passively by injection of poison in the moderator by usage of system steam pressure.

Actual calculations indicate that in none of the design basis accident sequences mentioned above the fuel clad temperature exceeds 1073 K.

For the purpose of containment design, a double-ended guillotine rupture of the 600 mm diameter inlet header has been considered as the design basis accident. A large number of other accident scenarios would conventionally fall within the category of beyond design basis accidents (BDBA). However, even in these cases, including a case of a NPP blackout together with failures of both independent fast acting shut-down systems (SDS–1 and SDS–2), predictions by BARC are that none of the acceptance criteria for design basis accidents as indicated above has been violated.

3.7. Provisions for safety under seismic conditions

The AHWR structures, systems and equipment are being designed for high level and low probability seismic events such as operating basis earthquake (OBE) and safe shutdown earthquake (SSE). These are also called S1 and S2 level earthquake respectively. Seismic instrumentation is also planned in accordance with the national and international standards.

3.8 Provisions for scenarios relevant to Fukushima

The response of AHWR was analysed by postulating several scenarios relevant to the Fukushima event. The analysed scenarios included long term station blackout (LSBO), station blackout with partial and complete loss of all heat sinks. AHWR design is found to be robust for LSBO as well as LSBO with partial loss of heat sink. Incorporation of core catcher imparts robustness to deal with LSBO with complete loss of heat sink. To maintain the integrity of the last barrier (i.e. containment) for radioactivity release to the public domain, it is found necessary to add hydrogen mitigation system in the primary containment. During LSBO, the GDWP water inventory is maintained below boiling point for seven days. Subsequently, boiling of the inventory causes pressurisation of containment. This necessitates periodic venting of the primary containment through a filtered system to avoid pressurization. Other measures that would help in such scenarios are provision of passive depressurization of AHWR, passive union between V1 and V2 volume, provision of fire water injection into MHTS, passive moderator and end shield cooling systems, and core catcher.

3.9. Probabilistic risk assessment

BARC expects that the probability of unacceptable radioactivity release beyond the plant boundaries will be less than 1×10^{-7}/RY.

A level-1 PSA has been performed in India for the AHWR considering only internal Initiating Events (IEs), full power operation state with reactor core as the source of radioactivity release. It will be extended to include external events as well. Since the study was performed during the design stage of the AHWR, the component reliabilities have been used either from generic sources or from the operating experience from other Indian reactors. Small Event Tree and large Fault Tree methodology forms the major framework for this analysis. Fault tree analysis has been extensively employed for system modeling while accident sequence propagation has been modeled using event tree approach. Apart from this, deployment of state-space approach using Markov diagram has been a special feature of this study. In order to evaluate the failure frequencies of process systems and unavailabilities of safety systems, the software package ISOGRAPH has been employed.

Reliability analyses of various process systems and safety systems have been carried out in India. The Core Damage Frequency (CDF) was found to be around 5.46e-08/RY. Uncertainty analysis has also been carried out taking into consideration the variability in component failure parameters. This analysis was carried out using ISOGRAPH. Log normal distribution has been considered for the components failure parameters. The 95% value for CDF was found to be 8.13e-07/RY and 99 %confidence value was found to be 1.05e-06/RY.

3.10. Emergency planning measures

One of the important design objectives for AHWR is to eliminate the need for any intervention in the public domain beyond the plant boundaries as a consequence of any postulated accident condition within the plant.

The results of level-1 PSA are helpful in defining test and maintenance schedules of various systems and components and in the preparation of emergency operating procedures (EOPs).

4.0. Proliferation resistance

4.1. Technical features to facilitate implementation of safeguards

Some of the important technical features of AHWR, which reduce attractiveness of its spent nuclear fuel material for use in any clandestine nuclear weapons programme, are as follows:

- The content of fissile plutonium in discharged fuel is very low.
- Radiation field from ^{233}U is very high due to the presence of ^{232}U.
- In the equilibrium condition, a high fraction of ^{234}U (up to about ten percent) will exist along with ^{233}U in the fuel.

The three factors indicated above also contribute to the prevention and discouragement of the diversion of AHWR based nuclear material for any clandestine nuclear weapons programme.

The technical features that prevent or discourage the production of weapon grade material in AHWR are:

- A self-sustaining design with respect to ^{233}U.
- Operation of reactor with low excess reactivity.

Provision for nuclear material accounting is an inherent part of the AHWR based nuclear fuel cycle, as has been the practice followed in the entire Indian nuclear programme. High gamma activity in the fresh as well as reprocessed AHWR fuel is expected to facilitate its verification with high efficiency and reliability.

5.0 Safety and Security (physical protection)

5.1 Features against human-induced malevolent external impacts and insider actions.

The Passive Poison Injection System (PPIS) is an additional system in AHWR to fulfil the shutdown function during a low probability event of failure of wired shutdown systems (i.e. Anticipated Transient without Scram (ATWS) case of both the shutdown systems SDS-1 and SDS-2 failure condition). PPIS injects the liquid poison into the moderator passively by system fluid pressure during such transients to shutdown the reactor. This situation may arise due to human-induced malevolent action caused by insider threat or compromise of functioning of both shut down systems.

The physical protection system is integral part of plant layout of AHWR. The plant is divided into nuclear island and administrative island. The nuclear island consists of main buildings like reactor building, fuel building, turbine building etc. The administrative island consists of administrative buildings, training centres, workshops and electrical switchyard etc. The plant layout is designed with a dual-layered security arrangement to provide enhanced physical protection to nuclear island. Nuclear Island is isolated from administrative facilities by double-wire fencing with additional security arrangement. The double fencing also provides for electronic surveillance and independent roads for patrolling by security personnel.

6.0 Description of turbine-generator systems

6.1 Steam and feed water system

The primary function of the steam and feed system is to transfer heat produced in the reactor core to turbine for production of electrical power. Steam and feed System forms an interface between the Main Heat Transport System and the Ultimate Heat Sink (Sea Water), and provides means for heat removal at various reactor-operating conditions.The steam and feed system consists of steam mains, turbo-generator and auxiliaries, condensing system, condensate and feed water heating system, steam dumping and relief systems and on line full condensate flow purification.

At normal full load a flow of 408 kg/s, 0.25% wet saturated steam, at pressure of 6.83 MPa from steam drum is delivered to the turbine throttle. After expansion in the High Pressure (HP) Turbine, steam is exhausted at pressure of 1.025 MPa. with wetness of 14.32%. Then it passes through external moisture separator reheater (MSR). Moisture separator reduces the wetness to 0.5%. Subsequently the steam is reheated in bled steam reheaters (BSR) to temperature of 503.2K.The steam enters the low pressure double flow turbine at 503.2 K temperature and pressure of 0.956 MPa, where it expands to a condenser back pressure of 8.3 kPa. The steam is condensed in a surface condenser and condensate at approx. saturation temperature (315.7 K) is taken out from the condenser. Heat is recovered by the condensate from the steam air ejector inter-after condenser and turbine gland steam condenser. There are 2 nos. of LP heaters in series LP heater-1 and LP heater-2. Feed water is heated up to 384.3 K at the outlet of LP heaters.

Feed water is further heated up to 403 K in a direct contact type deaerator heater by drains from MSR and steam extraction from LP turbine. Feed water return to the steam drum through two feed pumps. Feed water flow is controlled by control valve station. After the deaerator, feed water temperature is controlled by Temperature Control Heater by utilizing the live steam according to the variable set point of feed temperature with power.

7.0 Electrical and I&C systems

7.1 Power supply systems

AHWR electrical power supply system shall comprise of off-site supplies and on-site supplies and associated distribution systems. While evacuation systems are the part of offsite supplies, the onsite power supplies are further divided into normal power supply system and emergency power supply systems. The emergency power supply systems are only considered as safety related systems.

The emergency power supply systems are classified into class III, class II and class I according to type of power supply (AC or DC) and permissible interruption period. The normal power supply systems are classified as class IV.

Class-IV Power supply: Alternating current power supply to auxiliaries which can tolerate prolonged interruption without affecting safety of reactor is classified as Class-IV.

Class-III Power Supply: Alternating current power supply to auxiliaries which can tolerate short interruptions (up to 2 to 3 minutes) is classified as Class-III power supply.

Class-II Power Supply: Alternating current power supply to auxiliaries which require un-interruptible or very short interruption (up to few hundred milliseconds) in power supply is called Class-II power supply.

Class-I power supply: Direct current power supply to loads which require uninterruptible direct current power.

7.2 Safety related electrical systems

The various unit auxiliary loads to be connected to the electrical power supply system can be divided into two broad categories: Safety related loads and Non-safety related loads. While non-safety related loads can endure prolong power failure without affecting the safe operation of plant, and are generally connected to normal power supply system. The safety related loads depending on their criticality have defined limitation of duration of power failure to ensure safe operation of plant. The emergency power supply system ensures electrical power with specified quality to safety related loads which can take care of safe operation or shut down of plant after a postulated initiating event.

8.0 Spent Fuel and Waste management

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

AHWR has average burn up of 38 000 MWd/T. Flexibility to adopt different fuel cycles like AHWR300-LEU and feasibility of closed fuel cycle enhances utilisation of fuel resources.

8.2 Provisions for minimum generation of wastes at the source

The basic philosophy of minimizing the generation of radioactive waste at source and environmental discharges on ALATA (as low as technically achievable) basis shall be followed.

8.3 Provisions for acceptable or reduced dose limits

The siting, design of the plant, the operating procedures and the radiation protection programme are intended to ensure that the radiation exposures of the plant personnel and the members of the public resulting from the plant operation are controlled so as to be As Low As Reasonably Achievable (ALARA) and to comply with the dose limits prescribed by Atomic Energy Regulatory Board (AERB).

The reactor building is double containment cylindrical concrete structure roofed by two segmental concrete domes. The double containment structure besides containing and controlling release of effluents also provides necessary radiation shielding to the surroundings from the radiation sources present in the reactor building during normal operation. The reactor is located in reactor building. The Calandria vault of thick heavy concrete walls provides enclosure and shielding for the reactor vessel. The vault is filled with light water under circulation to act as a heat sink as well as radiation shield. The top and bottom circular ends of the Calandria are shielded by end shields containing steel balls and water mixture.

During normal operation and maintenance, effluents of different chemical and radiochemical nature will be generated. These effluents shall be segregated and collected at source for safe and effective waste management.

9.0 Plant layout

9.1 General

The design of plant layout has significant influence in achieving overall nuclear power plant performance and safety. The plant layout of AHWR is optimized for meeting various functional needs, as well as safety, radiation zoning, operation and maintenance, piping and cabling requirements, erection and construction requirements, transportation, and access and security considerations. Floor plans of various buildings have been optimized based on the equipment and systems within, on functional needs, space utilization, radiation zoning, accessibility, serviceability, maintenance, transportation and ventilation aspects, and on access philosophy. The present site layout is optimised for a single unit in space of 16 Hectares located at a typical coastal site.

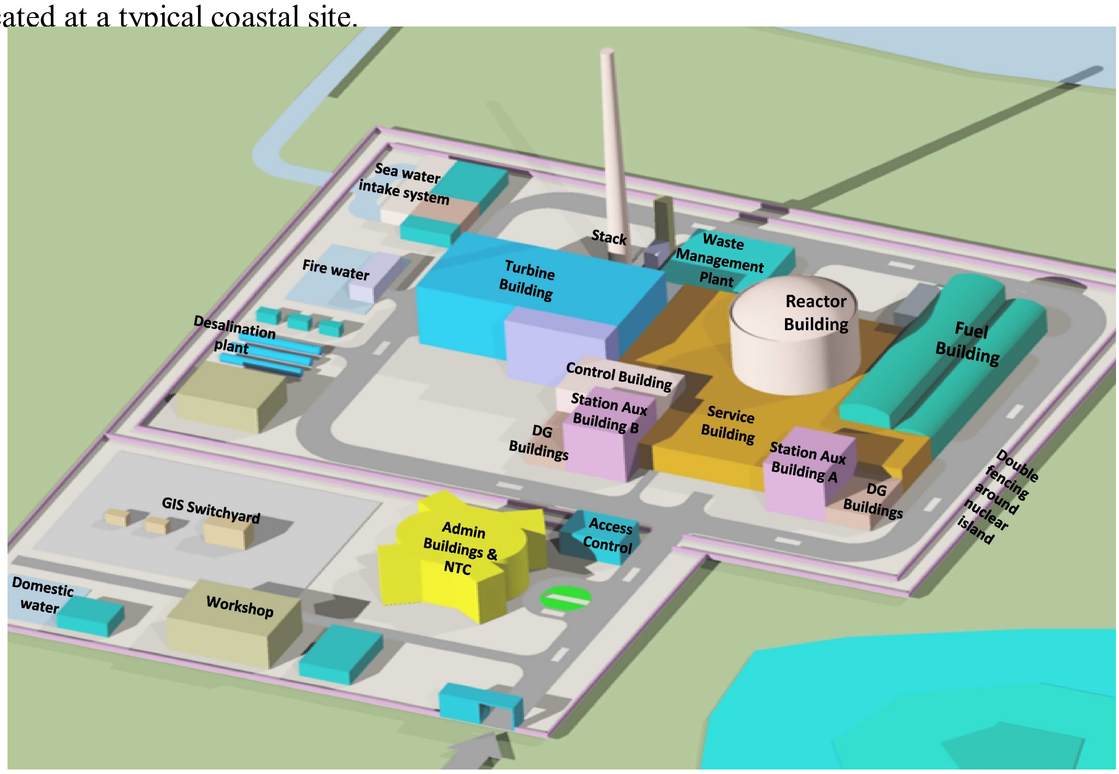

FIG. 5. Proposed layout of an AHWR based nuclear power plant for a coastal site.

9.2 Buildings and structures

The general plant layout and general location of the power station complex are indicated in Fig. 5. This figure shows the general arrangement of various buildings and structures provided at AHWR site. The main plant buildings contain Reactor Building, Control Building, Backup Control Building, Service Building, Turbine Building, Station Auxiliary Buildings A & B, Fuel Building, Waste Management Building, DG Buildings, GIS switchyard, Fire Water Reservoir, Emergency Water Reservoir, Work-shop, Ware house, Desalination Plant, Condenser cooling water & Service Water pump-houses, water intake and discharge structures, Administrative Building, Nuclear training centre, Stack & Radiation Monitoring Rooms, etc.

The containment serves basically the following functions:-

- It houses the reactor, the primary coolant and moderator systems and other systems connected with steam generation.
- It provides adequate shielding to restrict the level of radiation at site within limits.
- It contains the radioactive release in the event of a postulated loss of coolant accident (LOCA) in the reactor. In this function the most severe loadings result from what is defined as a postulated design basis events (DBE) involving rupture in the reactor primary coolant system. The resulting mixture of air and steam at high pressure and temperature is required to be contained to such an extent as to minimise the leakage from the containment and keep it below the acceptable level.

Service Building provides nuclear and conventional service facilities like workshops, ventilation intake, compressors and decontamination areas as well as laboratory services of health physics, chemical and instrumentation to power station complex. Service Building serves to control access nuclear island. A passage for a heavy-duty truck is provided for access to the reactor building for the shipment of heavy equipment inside reactor building. Service building also houses reactor auxiliary systems and purification ion exchangers of various systems.

The main control building housing Safety Group–1 C&I hardware is functionally and physically independent from the backup control building housing Safety Group–2 C&I hardware.

Station Auxiliary Building A & B provide station electrical services and house electrical equipment and maintenance area. These buildings are also connected to DG buildings. Grouping philosophy is also adopted for all concerned functions and buildings.

A fuel building is provided for storage of fresh and spent fuel. It is connected to reactor building through inclined fuel transfer machine. It is located so as to facilitate easy fuel transfer from co-located fuel cycle facility.

Turbine Building accommodates turbine generator, condenser and associated steam cycle equipment required for generation of electricity from steam cycle of the plant. GIS type switchyard considering a coastal site and compact layout is proposed. A Desalination Plant is also provided to produce desalinated water required for different processes and systems of the plant and drinking water requirements. It is placed adjacent to the turbine building. Sea water intake and discharge system comprises of condenser cooling water system and service water pump houses, and network of intake and discharge tunnels and associated systems.

DG Building accommodates equipment and supplies for DG sets to provide class-3 power supplies to plant facilities. Four DG buildings have been provided at physically separated location and fuel tanks. Waste Management Building accommodates features required for segregation, collection, treatment, conditioning, storage and safe disposal of liquid and solid radioactive disposable waste.

10.0 Plant Performance

10.1 Provision for reduced capital and construction costs

The advanced safety features and low capital cost of this reactor could facilitate its easy deployment in those developing countries that have limited financial and technological resources.

The main design features of this reactor leading to reduced capital cost per MW(e) are:

- Elimination of main coolant pump and associated other equipment.
- Substitution of steam generators with steam drums of simple design.
- Avoiding the use of costly heavy water in the main coolant system.
- Elimination of heavy water recovery and tritium management systems.
- Shop-assembled coolant channel assemblies.

The reactor has two important provisions for reducing O&M costs. The first one is the elimination of potential for heavy water leakages from the main coolant system (such as in conventional PHWRs) as heavy water exists only in the low pressure moderator system. This saves the recurring cost for heavy water make up. The second important provision is the achievement of an easily replaceable coolant channel design. This not only eliminates the need for a long shutdown for replacement of the coolant channel as the design life of the pressure tube comes to an end, but also enables the re-use of end-fittings and other channel appendages saving considerable cost of their unnecessary replacement too.

11.0 Deployment status and planned schedule

11.1 Information on research and technology development status

TABLE 2. RESEARCH AND TECHNOLOGY DEVELOPMENT STATUS

MAIN OBJECTIVES	ENABLING TECHNOLOGIES	STATUS OF DEVELOPMENT
Use of thorium based fuel	Nuclear data for nuclides important for the thorium cycle	A critical facility is under operation for physics related experiments.
	Remote fuel fabrication technologies	Several options examined; technologies are developed on a demonstration scale
	Three stream reprocessing of fuel containing Pu, Th and U.	Laboratory level process development done
	Dry reprocessing of fuel	Early studies in progress
Negative void coefficient	Tight lattice pitch	Feasibility demonstrated
Optimum use of passive systems for core heat removal	Natural circulation driven main coolant system	Several ongoing and future experimental programmes. Large experimental facility - Integral Test Loop and AHWR Thermal-Hydraulic Test Facility
	Isolation condensers	
	Large passive heat sink within containment	
	Passive valves	R&D in progress
Low peaking factor	Graded enrichment	Already a part of fuel manufacturing technology

MAIN OBJECTIVES	ENABLING TECHNOLOGIES	STATUS OF DEVELOPMENT
		development
Increased burn-up of fuel	Provision for reconstitution of fuel	Though not a prerequisite for initial operation of AHWR, the technology is planned to be developed and demonstrated
	Laser isotopic denaturing of zirconium	Not a prerequisite for operation of AHWR. A future R&D programme has been planned
	Provision for on-line refuelling	Large experience with PHWRs exists
Reduced O&M costs	Easily replaceable coolant channels	Rolled joint detachment technology developed
	Nuclear desalination to provide demineralised water	Technology demonstrated
Enhanced safety following LOCA	Passive ECCS (Emergency Core Cooling System) of enhanced effectiveness	Large experimental facility (Integral Test Loop)
	ECCS injection directly into fuel	Ongoing experimental programme
	Passive containment isolation	Large scale demonstration planned in a major facility under construction
	Core submergence	Passive feature, no R&D required
	One way rupture disk	R&D in progress
	High reliability non-return valve	R&D in progress
Additional features to achieve low core damage frequency	Steam driven poison injection	Planned to be demonstrated in a large facility

11.2 Companies/Institutions involved in RD&D and design

Bhabha Atomic Research Centre, Trombay, Mumbai. *INDIA*

12. Deployment Status and Planned Schedule

12.1 Estimate of an overall time frame within which the design could be implemented

The construction of AHWR is expected to commence in 2015. Evaluation of probable sites is under progress.

12.2 Information on main RD&D and licensing stages and their duration

The R&D for AHWR is fully supported by Government of India. The design and development of this reactor is mainly done by BARC. The basic design of the reactor and detailed design of its major nuclear systems have been completed. The research, design, and demonstration

(RD&D) for AHWR has been and is being performed at the BARC with the financial support of the Government of India. The Nuclear Power Corporation of India Ltd. (NPCIL) has completed a peer review of the design in September 2003. The Indian Atomic Energy Regulatory Board (AERB) has carried out a pre-licensing safety appraisal of AHWR. Subsequently, the regulatory clearances for different stages of construction, starting from plant siting and procurement of long delivery major equipment, will be progressively sought.

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