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Design and development of the AHWR—the Indian thorium fuelled innovative nuclear reactor

R.K. Sinha*, A. Kakodkar

Bhabha Atomic Research Centre, Trombay, Mumbai 400085, India

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Abstract

India has chalked out a nuclear power program based on its domestic resource position of uranium and thorium. The first stage started with setting up the Pressurized Heavy Water Reactors (PHWR) based on natural uranium and pressure tube technology. In the second phase, the fissile material base will be multiplied in Fast Breeder Reactors using the plutonium obtained from the PHWRs. Considering the large thorium reserves in India, the future nuclear power program will be based on thorium–^{233}U fuel cycle. However, there is a need for the timely development of thorium-based technologies for the entire fuel cycle. The Advanced Heavy Water Reactor (AHWR) has been designed to fulfill this need. The AHWR is a 300 MW_e, vertical, pressure tube type, heavy water moderated, boiling light water cooled natural circulation reactor. The fuel consists of $(Th–Pu)O_2$ and $(Th–^{233}U)O_2$ pins. The fuel cluster is designed to generate maximum energy out of ^{233}U, which is bred in situ from thorium and has a slightly negative void coefficient of reactivity. For the AHWR, the well-proven pressure tube technology has been adopted and many passive safety features, consistent with the international trend, have been incorporated. A distinguishing feature which makes this reactor unique, from other conventional nuclear power reactors is the fact that it is designed to remove core heat by natural circulation, under normal operating conditions, eliminating the need of pumps. In addition to this passive feature, several innovative passive safety systems have been incorporated in the design, for decay heat removal under shut down condition and mitigation of postulated accident conditions. The design of the reactor has progressively undergone modifications and improvements based on the feedbacks from the analytical and the experimental R&D. This paper gives the details of the current design of the AHWR.
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1. Introduction

The Indian nuclear power program has been conceived bearing in mind the optimum utilization of domestic uranium and thorium reserves with the objective of providing long-term energy security to the country. One of the essential elements of the Indian strategy is to enhance the fuel utilization using a closed fuel cycle. This entails reprocessing of the spent fuel to recover fissile and fertile materials and its recycle back into the system. Considering this objective, the indigenous nuclear power program in India was initiated with Pressurized Heavy Water Reactors (PHWRs) using natural uranium and heavy water, and based on pressure tube technology. The pressure tube concept, used in PHWRs, has several advantages such as:

- physical separation of the high-temperature high-pressure coolant from the low-temperature low-pressure moderator;
- a high conversion ratio with well thermalized neutron spectrum due to cold moderator;
- low excess reactivity in the core arising out of on-power fuelling;
- a greater flexibility in adopting different refuelling schemes.

India has been operating and developing improved versions of its current generation PHWRs on the basis of operating experience, international trends and indigenous R&D inputs as a first stage.

In the second stage of the Indian nuclear power program, plutonium from the natural uranium-based PHWRs will be used in Fast Breeder Reactors for multiplying the fissile base. Considering the large thorium reserves in India, the future systems, in the third stage of Indian nuclear power program, will be based on thorium–^{233}U fuel cycle. While the initiation of the third

* Corresponding author. Tel.: +91 22 25505303; fax: +91 22 25505303.
E-mail address: rksinha@magnum.barc.ernet.in (R.K. Sinha).

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stage will take place in the future, there is a need for the timely development of thorium-based technologies for the entire thorium fuel cycle. The Advanced Heavy Water Reactor (AHWR) is being developed to fulfill this need.

2. Evolution of the AHWR concept

Thorium is a fertile material and has to be converted into ^{233}U, a fissile isotope. Of the three fissile species (^{233}U, ^{235}U and ^{239}Pu), ^{233}U has the highest value of η (number of neutrons liberated for every neutron absorbed in the fuel) in thermal spectrum. Since ^{233}U does not occur in the nature, it is desirable that any system that uses ^{233}U should be self-sustaining in this nuclide in the entire fuel cycle, which implies that the amount of ^{233}U used in the cycle should be equal to the amount produced and recovered. Thorium in its natural state does not contain any fissile isotope the way uranium does. Hence, with thorium-based fuel, enrichment with fissile material is essential. The large absorption cross-section for thermal neutrons in thorium facilitates the use of light water as coolant. On account of its high cost and its association with radioactive tritium, use of heavy water coolant requires implementation of a costly heavy water management and recovery system. The use of light water as coolant makes it possible to use boiling in the core, thus producing steam at a higher pressure than otherwise possible with a pressurized non-boiling system. With boiling coolant, the reactor has to be vertical, making full core heat removal by natural circulation feasible. The choice of heavy water as moderator is derived from its excellent fuel utilization characteristics. Considering these characteristics, the mainly thorium fuelled AHWR, is heavy water moderated, boiling light water cooled, and has a vertical core.

The future Indian thorium-based reactor systems will be optimized for the thorium cycle. For the AHWR, pressure tube type PHWR technology is selected to take advantage of the vast experience gained and infrastructure developed in the country. It is desirable for the new reactors to incorporate passive safety characteristics consistent with the emerging international trends. The design of the reactor has progressively undergone several modifications and improvements based on feedbacks from the results of analytical and experimental R&D. This paper describes the current design of the AHWR.

3. Overview of the reactor configuration

As already mentioned, the AHWR is a vertical, pressure tube type, heavy water moderated and boiling light water cooled natural circulation reactor (Sinha and Kakodkar, 2003) designed to generate 300 MW_e and 500 m^3/day of desalinated water. The AHWR is fuelled with $(Th–^{233}U)O_2$ pins and $(Th–Pu)O_2$ pins. The fuel is designed to maximize generation of energy from thorium, to maintain self-sufficiency in ^{233}U and to achieve a slightly negative void coefficient of reactivity. An emergency core cooling system injects water directly into the fuel.

The reactor core of the AHWR consists of 505 lattice locations in a square lattice pitch of 245 mm. Of these, 53 locations are for the reactivity control devices and shut down systems. Reactivity control is achieved by on-line fuelling, boron dissolved in moderator and reactivity devices. Boron in moderator is used for reactivity management of equilibrium xenon load. There are 12 control rods, grouped into regulating rods, absorber rods and shim rods of 4 each. The reactor has two independent, functionally diverse, fast acting shut down systems, namely, Shut Down System-1 (SDS-1) consisting of mechanical shut off rods and Shut Down System-2 (SDS-2) based on liquid poison injection into the moderator. There are 30 interstitial lattice locations housing 150 in-core self-powered neutron detectors and 6 out-of-core locations containing 9 ion chambers and 3 start-up detectors. An automatic reactor regulating system is used to control the reactor power, power/flux distribution, power-setback and xenon override. Both for the control rods and the shut off rods, the absorber material, boron carbide, is packed in an annulus within 80 stainless steel tubes. The core map is given in Fig. 1.

The reactor core is housed in a low-pressure reactor vessel called calandria. The calandria contains heavy water, which act as moderator as well as reflector. The calandria houses the vertical coolant channels, consisting of pressure tubes rolled in top and bottom end fittings. The pressure tube contains the fuel cluster. A calandria tube envelops each pressure tube and the air annulus between the two tubes provides thermal insulation between the hot coolant channel and the cold moderator. The calandria tubes are rolled, in the tube sheets of top and bottom end shields of the calandria.

The light water coolant picks up nuclear heat in boiling mode from fuel assemblies. The coolant circulation is driven by natural convection through tail pipes to steam drums, where steam is separated and is supplied to the turbine. A simplified schematic arrangement of the AHWR is shown in Fig. 2.

Four steam drums (only one shown in Fig. 2 for the sake of clarity), each catering to one-fourth of the core, receive feed

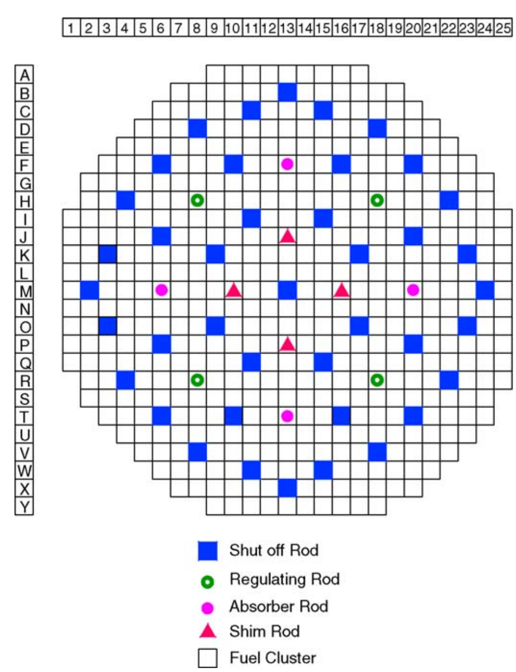

Fig. 1. Core map of the AHWR.

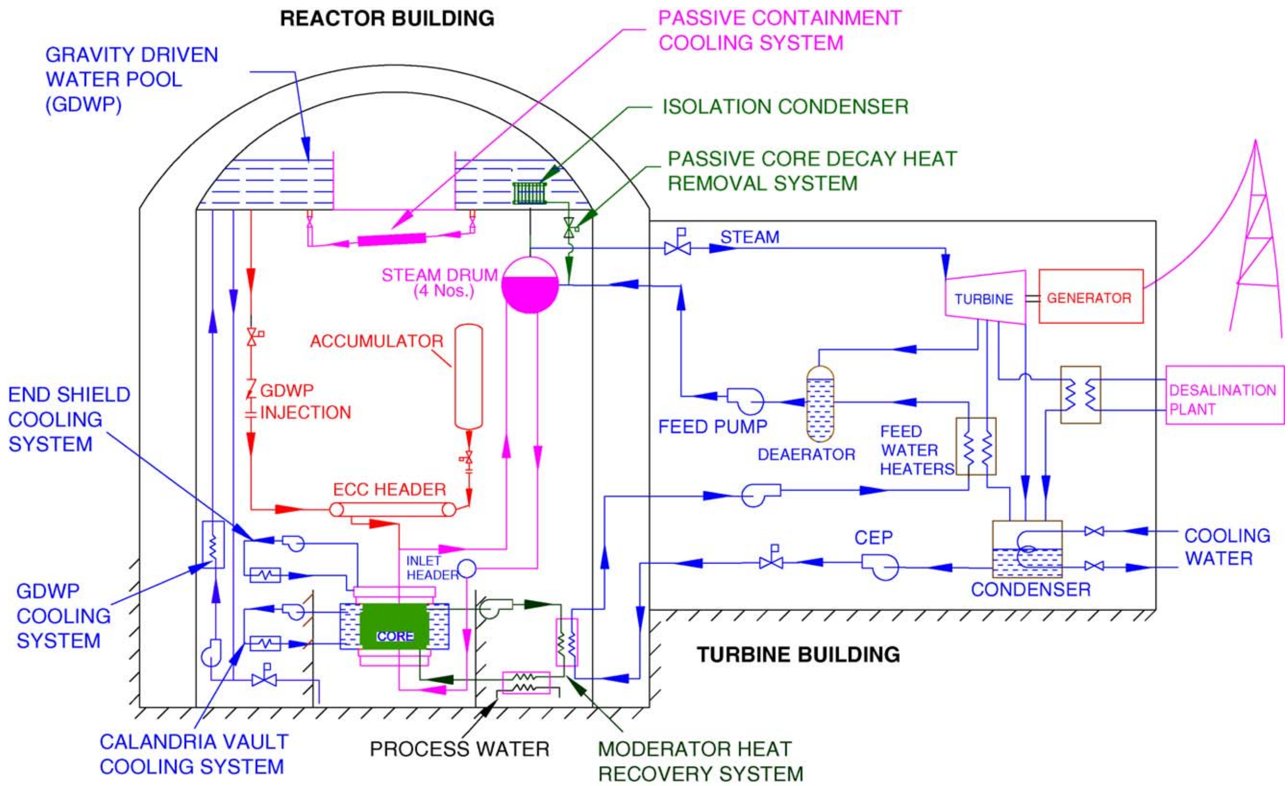

Fig. 2. Simplified schematic arrangement of the AHWR.

water at 403 K to provide optimum sub-cooling at the reactor inlet. Four down-comers, from each steam drum, are connected to a circular inlet header. The inlet header distributes the flow to each of the 452 coolant channels through individual feeders. The AHWR incorporates several passive systems to fulfill several safety functions (Sinha et al., 2000). A 6000 m^3 capacity gravity driven water pool (GDWP), located close to top of the containment serves as a heat sink for several passive systems, besides acting as suppression pool and a source of water for low-pressure emergency core cooling. Achievement of passive shutdown using steam overpressure to provide the driving force and passive cooling of concrete surfaces are some of the other unique passive safety features provided in the AHWR.

A fuelling machine is located on top of the deck plate. The fuelling machine of the AHWR handles the fuel clusters by means of ram drives and snout drive for coupling and making a leak tight joint with the coolant channel. The AHWR has the flexibility to have on power as well as off-power fuel handling. The dimensional details of the core are given in Table 1.

A seawater desalination plant will meet the demineralized water requirements of the reactor and drinking water required at the plant, utilizing the low-pressure steam from the turbine. A provision exists to add to the desalination capacity at the cost of electrical power output.

4. Fuel and fuel cycle

The fuel has been designed to meet the requirement of thermal hydraulics, reactor physics, fuel handling and reconstitution (i.e., replacement of outer ring of irradiated (Th–Pu)O_2 fuel pins with fresh ones). The vertical pressure tube configuration has guided the structural design of the fuel assembly. The fuel assembly is 10.5 m in length and is suspended from the top in the coolant channel. The assembly consists of a fuel cluster and two shield sub-assemblies. These sub-assemblies are connected to each other through a quick connecting/disconnecting joint to facilitate handling.

Table 1
Dimensional details of the core

Total no. of lattice locations	505
Number of fuel channels	452
Number of lattice locations for control rods	12
Number of lattice locations for shut-off rods	41
Lattice pitch (mm)	245
Active core height (m)	3.5
Calandria	
Inner diameter of the main shell (m)	7.4
Inner diameter of the sub-shell at each end (m)	6.8
Length (m)	5.3
Tube material	
Pressure tube	Zr2.5 Nb
Calandria tube	Zircaloy-4
Tube dimension	
Inner diameter/WT of Pressure tube (mm)	120/4
Outer diameter/WT of Calandria tube (mm)	168/2
Reflector thickness (D_2O) axial/radial (mm)	750/600
Moderator temperature (K)	353
Moderator purity (% of D_2O)	99.8

The fuel cluster is a cylindrical assembly of 4300 mm length and 118 mm diameter. The arrangement of pins in the fuel cluster is shown in Fig. 3(a). The cluster has 54 fuel pins arranged in 3 concentric rings around a central rod as shown in Fig. 3(b) (Anantharaman and Shivakumar, 2002). The 24 fuel pins in the outer ring have (Th–Pu)O_2 as fuel and the 30 fuel pins in the inner and intermediate rings have (Th–^{233}U)O_2 as fuel. The innermost 12 pins have a ^{233}U content of 3.0 wt.% and the middle 18 pins have 3.75 wt.% ^{233}U. The outer ring of (Th–Pu)O_2 pins contain 3.25 wt.% of total plutonium, of which the lower half of the active fuel has 4.0% Pu and the upper part has 2.5% Pu (Kumar et al., 1999). Two enrichments have been provided in the outer ring to have favorable minimum critical heat flux ratios.

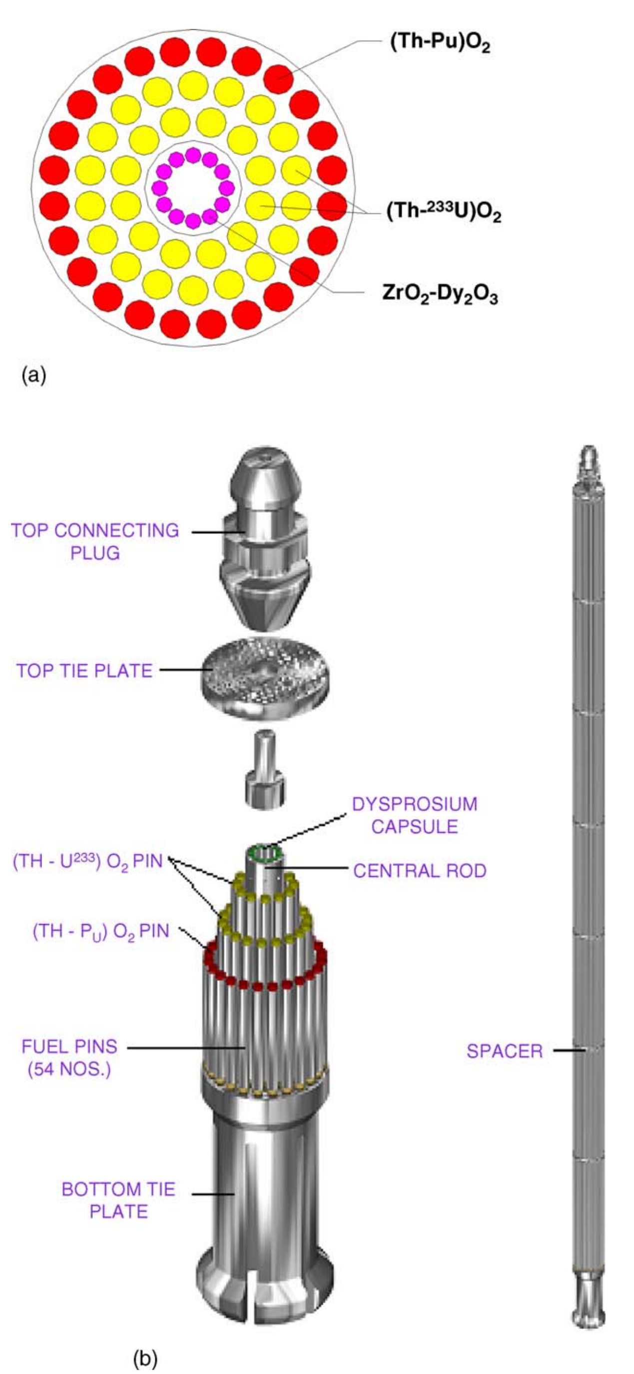

Fig. 3. (a) Cross-section of fuel pins in the cluster and (b) AHWR fuel cluster.

The fuel pin consists of fuel pellets confined in a Zircaloy-2 clad tube. The fuel pin has a pellet stack length of 3500 mm and a plenum volume with a helical spring in it to keep the pellet stack pressed. The fuel pins are assembled in the form of a cluster with the help of the top and bottom tie-plates, with a central rod connecting the two tie-plates. Six spacers along the length of the cluster provide the intermediate pin spacing. The central rod has a tubular construction with holes for direct injection of ECCS water on the fuel rods. It also contains dysprosium capsules containing dysprosium oxide in Zirconia matrix. The design data of the fuel assembly is given in Table 2.

The AHWR fuel cycle is a closed fuel cycle, envisaging recycle of both fissile ^{233}U and fertile thoria back to the reactor (Anantharaman et al., 2000). The currently envisaged fuel cycle time is eight years. This comprises four years for in-reactor residence time, two years for cooling, one year for reprocessing and one year for refabrication. Since the ^{233}U required for the reactor is to be bred in situ, the initial core and annual reload for the initial few years will consist of (Th–Pu)O_2 clusters only. After reprocessing, ^{233}U is always associated with ^{232}U, whose daughter products are hard gamma emitters. The radioactivity of ^{232}U associated with ^{233}U starts increasing after separation. This poses radiation exposure problems during its transportation, handling and refabrication. Hence, it is targeted to minimize delay between separation of ^{233}U and its refabrication into fuel. In view of this, a co-location of the fuel cycle facility, comprising reprocessing, waste management and fuel fabrication plant,

Table 2
Description of the AHWR fuel assembly

Parameter	Value
Number of fuel pins	54
Outer diameter (mm)	11.2
Density (g/cm^3)	9.6
Fuel clad	
Material/thickness (mm)	Zircaloy-2/0.6
Fuel type/number of pins	
Inner ring	(Th–^{233}U)O_2/12
Middle ring	(Th–^{233}U)O_2/18
Outer ring	(Th–Pu)O_2/24
Fuel enrichment (wt%)	
Inner ring (^{233}U)	3.0
Middle ring (^{233}U)	3.75
Outer ring (Pu)	3.25 (average)
Upper half	2.5
Lower half	4.0
Central rod	
Tube o.d./thickness (mm)	36/2
Number of pins/capsule	12
Outer diameter of pin (mm)	6
Material/o.d. (mm)	$ZrO_2 + Dy_2O_3$
Dysprosium (wt%)	3.0
Average discharge burnup (MWd/t)	24,000
Average linear heat rating (kW/m)	10.6
Peak linear heat rating (kW/m)	14.0

with the AHWR has been planned. The ^{233}U-based fuel needs to be fabricated in shielded facilities due to activity associated with ^{232}U. This also requires considerable enhancement of automation and remotization technologies used in fuel fabrication.

The spent fuel cluster, before reprocessing, would undergo disassembly for segregation of $(Th–Pu)O_2$ pins, $(Th–^{233}U)O_2$ pins, structural materials and burnable absorbers. The $(Th–^{233}U)O_2$ pins will require a two stream reprocessing process, i.e., separation of thorium and uranium whereas the $(Th–Pu)O_2$ pins will require a three stream reprocessing process, i.e., separation of thorium, uranium and plutonium. A part of the of reprocessed thorium (45%) may be used immediately in the fabrication of $(Th–^{233}U)O_2$ pins since ^{233}U fabrication is required to be carried out in shielded facilities. The remaining thorium will be stored for sufficient amount of time for the activity to decay to a level at which, it is easier for handling with minimal shielding. The stored thorium will be subsequently used for the fabrication of $(Th–Pu)O_2$ fuel pins.

5. Reactor physics

5.1. Main objectives of the physics design

The physics design of AHWR is carried out to fulfill the following objectives (Srivenkatesan et al., 2000):

(1) maximize the energy from in situ burning of ^{233}U;
(2) achieve a negative void coefficient of reactivity;
(3) achieve greater than 20,000 MWd/t fuel discharge burnup;
(4) minimize, to the extent possible, the initial plutonium inventory;
(5) minimize, to the extent possible, the consumption of plutonium for given energy output;
(6) achieve self-sustenance in ^{233}U;
(7) deliver a thermal power of 920 MW to the coolant.

To achieve these objectives, the physics design has progressively evolved from a seed-blanket core design concept to a core consisting of a single type of cluster called composite cluster, containing both $(Th–^{233}U)O_2$ and $(Th–Pu)O_2$ fuel pins (Kumar, 2000). The main considerations governing the fulfillment of these objectives are discussed in the following sub-sections.

5.1.1. Achieving negative void coefficient of reactivity in both operating and accidental conditions

The cluster design is mainly dictated by the objective of achieving negative void coefficient of reactivity. The void coefficient of reactivity can be made negative by maintaining a harder neutron spectrum in the core. This can be achieved either by changing the properties of the moderating medium or by decreasing the inventory of the moderator (for example, by increasing the cluster size in relation to the lattice pitch). It is also possible to achieve negative void coefficient of reactivity by using a burnable absorber either in the fuel or in isolated pins in an inert matrix. On voiding of the coolant, the thermal neutron flux increases in the cluster, and the neutron flux can be reduced by using a slow burning absorber. In the AHWR, dysprosium is used as a burnable absorber within the cluster at a lattice pitch of 245 mm, to make the void coefficient of reactivity negative for average core burnup.

5.1.2. Achieving a flat radial power distribution

Heat removal through natural convection is an important feature of this reactor. In order to have good thermal hydraulic and neutronic coupling, the radial power distribution has to be flat. This requires the height of the active core to be kept small with respect to the diameter of the core. In view of this, the core height has been chosen to be 3.5 m and the calandria vessel diameter is 7.4 m. There are 505 lattice locations in the core, out of which 452 locations are occupied by fuel and the rest by reactivity devices.

5.1.3. Optimizing the axial power profile for adequate thermal margin

In a typical boiling water reactor with bulk boiling, the axial power profile is bottom-peaked and this increases the thermal margin in the top region of the fuel where the void fraction is high. In AHWR, in order to achieve a desirable axial power distribution for adequate thermal margin, graded enrichment is used along the length of the fuel assembly. This is achieved by altering only the plutonium content in the outer pins without compromising the void reactivity. The lower half of the fuel assembly is loaded with 4.0 wt.% Pu and the upper half with 2.5 wt.% Pu in thorium dioxide.

5.1.4. Achieving self-sustenance in ^{233}U

The objective of achieving self-sustenance in ^{233}U has governed the reactor physics design of AHWR core. The ^{233}U bred in the cluster decides the self-sustaining characteristic of AHWR fuel. With irradiation, the ^{233}U content depletes in the inner $(Th–^{233}U)O_2$ pins and increases in the outer $(Th–Pu)O_2$ pins due to conversion from thorium. The conversion has been maximized by making the spectrum harder, i.e., in an intermediate energy range around 0.2 eV.

5.1.5. Minimization of plutonium make-up requirement

The plutonium pins are placed in the outermost ring of the cluster to minimize the plutonium requirement. The plutonium used as make-up fuel comes from the discharged PHWR fuel. The power from thorium is 60%.

5.2. Reactor physics analyses

The analyses comprise core calculations, using a 3D code for core optimization, for obtaining the optimum fuel discharge burnup, flattened channel power distribution and worth of the reactivity devices.

5.2.1. Physics analysis for the equilibrium core

The reactor physics analysis presented here mainly pertains to the equilibrium core configuration, which consists of the composite type of cluster. Detailed lattice analyses have been performed to calculate the variation of lattice parameters such as the lattice reactivity (k-infinity), the macroscopic cross-sections and the isotopic compositions as a function of irradiation. The pin-wise power distribution across the cluster, reactivity coefficients, and other lattice characteristics are also obtained. The lattice evaluations have been done with WIMSD code system (Askew et al., 1996) and the 69 energy groups WIMSD nuclear data library from the basic data set of ENDF/BVI.8 (IAEA, 2002).

The design features of AHWR for equilibrium core configuration are given in Table 3. The core calculations have been done using 3DKIN and FEMTAVG (Kumar and Srivenkatesan, 1984). The time-averaged simulations have been done to get optimum discharge burnup and flattened channel power distribution for the equilibrium core configuration. The core power distribution has been optimized for a total power of 920 MW_t.

In order to achieve flux flattening, the equilibrium core has been divided into three burnup zones, which are adjusted to get the average discharge burnup of nearly 24,000 MWd/t and the maximum channel power of 2.6 MW_t. The average coolant density in the core is 550 kg/m^3. The code FEMTAVG is coupled to a static thermal–hydraulics code THABNA, and the coolant density as a function of distance from inlet for every channel is calculated. It is seen that the core burnup, power and coolant density distribution converge in three to four iterations and the optimum power distribution is estimated accordingly. The quarter core power distribution, calculated for the average coolant density of 550 kg/m^3 throughout the core, is shown in Table 4. The burnup zones and their exit burnups are also given in Table 4.

Table 3
Physics parameter of AHWR equilibrium core

Parameter	Value
Fuelling rate, annual	
Number of fuel channels	113
Pu (kg)	200
Conversion ratio, ^{233}U	97%
Power from thorium/^{233}U	60%
Peaking factors (maximum)	
Local	1.45
Radial	1.2
Axial	1.64
Total	2.85
Reactivity control	
Boron/gadolinium in moderator	
Control rods (no.)	12 (total of 18.9 mk)
Absorber rods (no.)	4 (total of 7.1 mk)
Regulating rods (no.)	4 (total of 8.1 mk)
Shim rods (no.)	4 (total of 3.7 mk)
Shut Down System-1	41 nos. (total of 80 mk; 46 mk with two maximum worth rods not available)
Absorber material	B_4C pins in SS shell
Shut Down System-2	Liquid poison injection in moderator
Safety parameters	
Delayed neutron fraction, β	0.003
Prompt neutron generation time, Λ (ms)	0.22
Reactivity coefficients, $\Delta k/k$ (°C)	
Fuel temperature	-2.0×10^{-5}
Coolant temperature	$+3.5 \times 10^{-5}$
Channel temperature	$+1.0 \times 10^{-5}$
Void coefficient, $\Delta k/k$ (% void)	-6.0×10^{-5}

The exit burnups of the three zones are 30,000, 23,500 and 20,000 MWd/t. The average discharge burnup is nearly 24,000 MWd/t. The radial and axial peaking factors are calculated to be 1.2 and 1.64, respectively. The limits on power distribution/power are derived from the minimum critical heat flux ratio—MCHFR (CHFR is the ratio of the critical heat flux at any point in the flow channel to the actual flux at that point), and it is a measure of safety margin available for the reactor core. The MCHFR calculated at 20% overpower is 1.67.

The reactivity balance in AHWR is given in Table 5. The equilibrium xenon load is 21.0 mk and the maximum transient xenon load peaking following shut down is 7.0 mk (US$ 1 = 3 mk). This is due to relatively low thermal flux level of 7.0×10^{13} n/cm^2/s. The void reactivity for equilibrium core of AHWR has been calculated as 6.0 mk.

The major postulated initiating events, considered from the point of reactivity changes, are loss of regulation accident and cold-water ingress. Out of these, only loss of regulation accident involves substantial positive reactivity addition. Both the shut down systems of AHWR are capable of independently shutting down the reactor in time.

5.2.2. *The initial core of AHWR*

The ^{233}U, required for the equilibrium core of AHWR will be bred in situ. It is envisaged that there will be a gradual transition from the initial core that will not contain any ^{233}U, to the equilibrium core.

5.2.3. *Recycling of uranium*

With several recycles, the ^{234}U content in uranium increases from 6 to about 12%. It is seen that the reactivity load due to ^{234}U in successive recycling of uranium in the AHWR causes a penalty of about 1500 MWd/t. Fuel cycle calculations have been done to optimize cycle length with respect to the self-sustenance in ^{233}U and other fuel performance characteristics.

5.2.4. *Xenon oscillations*

The possibility of xenon instabilities in the AHWR is reduced considerably due to relatively low thermal flux level along with negative void and power feedback. Only first azimuthal mode, with sub-criticality of 12 mk, is close to the instability threshold in the AHWR. There are four regulating rods, one in each quadrant, to suppress any flux tilt arising due to these azimuthal oscillations.

6. Description of major reactor systems

6.1. *Reactor block*

The reactor block of AHWR consists of calandria, end shields, coolant channels and associated piping, deck plate,

Table 4
Optimized core power distribution

		13	12 14	11 15	10 16	9 17	8 18	7 19	6 20	5 21	4 22	3 23	2 24	1 25
Z	**A**	1.80	1.67	1.57	1.44	1.33								
Y	**B**	0.00	2.06	1.99	1.78	1.63	1.55	1.40						
X	**C**	2.20	2.26		2.20	2.00	2.00	1.71	1.61					
W	**D**	1.92	2.01	2.24	2.17	2.29		2.14	1.80	1.71				
V	**E**	1.54	1.76	2.08	2.35	2.28	2.35	2.24	2.20	1.87	1.71			
U	**F**	***AR***	1.58	2.18		2.36	2.16	2.32		2.20	1.80	1.61		
T	**G**	1.60	1.81	2.11	2.33	2.12	2.01	2.10	2.32	2.24	2.14	1.71	1.39	
S	**H**	2.04	2.13	2.36	2.21	2.03	**RR**	2.01	2.16	2.35		2.00	1.55	
R	**J**	2.34	2.38		2.42	2.11	2.03	2.12	2.36	2.28	2.29	2.00	1.63	1.32
Q	**K**	**SR**	2.44	2.40	2.28	2.42	2.21	2.33		2.35	2.17	2.20	1.78	1.43
P	**L**	2.42	2.28	2.23	2.40		2.36	2.11	2.18	2.07	2.24		1.99	1.56
O	**M**	2.44	2.29	2.28	2.44	2.38	2.13	1.81	1.58	1.76	2.00	2.25	2.05	1.66
	N		2.44	2.42	**SR**	2.32	2.04	1.60	**AR**	1.54	1.90	2.20		1.78

The numbers denote the channel power in MW

AR	Absorber rod	**RR**	Regulating rod	**SR**	Shim rod

Zone		Exit burnup MWd/t
1	(dark grey)	**30000**
2	(light grey)	**23500**
3		**20000**

reactor control and protection systems, and ECCS header with associated piping and main heat transport (MHT) system inlet header. The layout of components in reactor block is shown in Fig. 4. The calandria is housed in a light water filled reactor vault that acts as an effective radiation shield. End shields, sup-

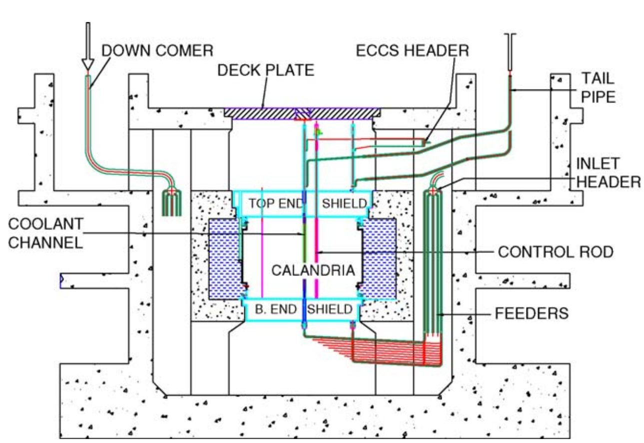

Fig. 4. Reactor block.

Table 5
Reactivity balance in AHWR

Reactor core state	Reactivity (mk)
Reactivity swings	
(1) Cold to hot standby	
Channel temperature (300–558 K)	+2.5
Moderator temperature (300–353 K)	+3.0
Total	+5.5
(2a) Hot standby to full power	
Fuel temperature (558–898 K)	−6.5
Coolant void (coolant density from 0.74 to 0.55 g/cm^3)	−2.0
Total	−8.5
(2b) LOCA from full power (coolant density 0.55–0.0 g/cm^3)	−4.0
(3) Xenon load	
Equilibrium load	−21.0
Transient load after shutdown from full power (peak at about 5 h)	−7.0

ported on concrete structure, are provided at both the ends of the calandria.

6.1.1. Calandria

The calandria is a 5.3 m long cylindrical stainless steel (SS304L) vessel. It houses the reactor core, moderator, reflector, and a portion of the reactor control and protection systems. The central portion of the calandria is called the main shell (7.4 m i.d. × 3.5 m long). Two sub-shells of smaller diameter (6.8 m i.d.) are attached to the main shell at top and bottom with flexible annular plates. The calandria is fully filled with heavy water and is connected to the expansion tank to accommodate volumetric expansion of the moderator. The nozzle penetrations, required for the moderator system, the liquid poison injection system and the expansion tank are provided in the sub-shells of the calandria vessel. The nozzle penetrations for over pressure relief devices are provided in main shell of the calandria vessel to protect the calandria against internal pressure above the design limit occurring during accidental conditions. The nominal inlet and outlet temperatures of the moderator are 328 and 353 K, respectively, and the calandria is designed for 0.05 MPa above the static head due to moderator.

6.1.2. End shields

The end shields are composite cylindrical stainless steel (SS304L) structures, filled with a mixture of steel balls and water and are attached to the top and bottom ends of the calandria by in situ welding. These end shields provide radiation shielding and serve as pressure boundary to the moderator system. The end shields also support and guide the coolant channel assemblies, reactor control systems and protection systems. The vertical calandria tubes are joined to the end shield lattice tubes by rolled joints. Light water is circulated through the end shields to remove the nuclear heat generated.

6.1.3. Coolant channel assembly

The coolant channel houses the fuel assembly with shielding blocks and has suitable interfaces for coupling to the main heat transport system. A suitable interface is provided for coupling the fuelling machine with the coolant channel to facilitate removal of hot radioactive fuel from the reactor and introduction of fresh fuel into the reactor. The coolant channel has features to accommodate thermal expansion, and irradiation creep and growth. The schematic arrangement of the coolant channel with the fuel assembly is shown in Fig. 5. The vertical coolant channel consists of pressure tube, top and bottom end fittings, and calandria tube. The pressure tube, made of zirconium–niobium alloy, is located in the core portion. The core portion is extended with top and bottom end fittings made of stainless steel. The feeder pipe is connected to the bottom end fitting through a self-energized metal seal coupling and this facilitates easy removal. The tail pipe is welded to the top end fitting. The coolant enters the coolant channel at 533.5 K, flows past the fuel assembly and hot coolant flows out as steam–water mixture at 558 K flows out to tail pipes. The annular space between the pressure tube and calandria tube, as shown in Fig. 6, provides a thermal insulation between the hot coolant and the cold moderator. The coolant channel assembly is laterally supported within the lattice tube by two bearings located at the two ends of the top end shield lattice tube. The weight of the coolant channel is supported at the top end shield. An annulus leak monitoring system is incorporated to provide an early warning of a leakage in the pressure tube, as a part of the strategy to meet the leak before break requirement for the pressure tube.

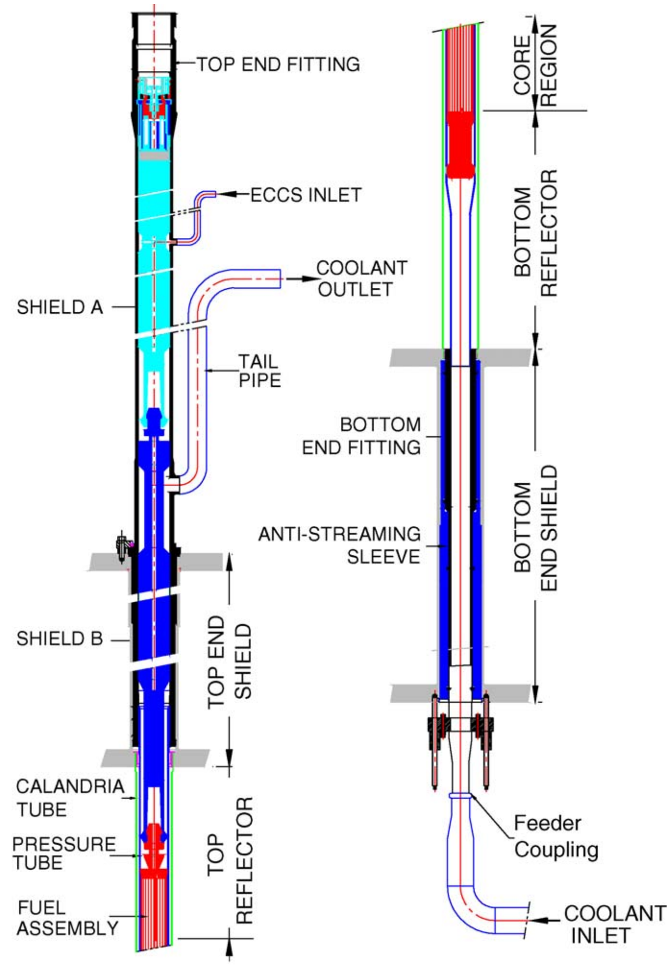

Fig. 5. Schematic arrangement of the coolant channel assembly.

Easy replacement of pressure tubes, as a part of the normal maintenance activity is an important consideration in the design

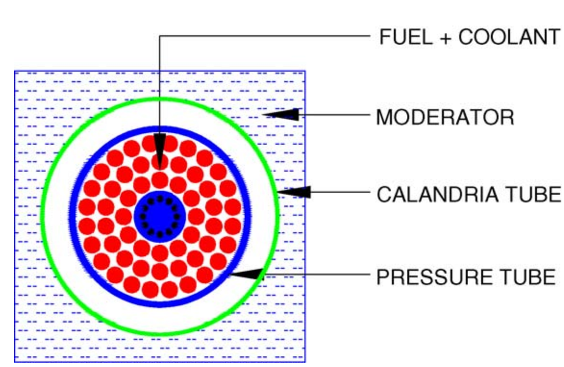

Fig. 6. Fuel cluster in the pressure tube.

of the coolant channel assemblies. The AHWR coolant channel is designed for easy replacement of pressure tubes as a regular maintenance activity without incurring a large downtime of the reactor. This allows the individual coolant channel to be replaced at the end of its design life. A longer life and easy replacement criteria have guided the selection of the pressure tube material and the design of pressure tube, end fittings and the couplings. The pressure tube is provided with an in-built reducer at the bottom end and a thicker walled top end. The pressure tube is detachable from the rolled joint with the top end fitting. The bottom end fitting can be detached from the feeder by de-coupling the bottom metal seal coupling. The bottom end fitting is sized such that it can be removed along with the pressure tube through the bore of the top end fitting after detaching it from the feeder and the top end fitting. Top end fitting is provided with two sets of rolled joint bores. A shop assembled fresh pressure tube with bottom end fitting can be inserted through the bore of the top end fitting and rolled to the fresh set of rolled joint grooves.

6.2. Fuel handling and storage system

The refuelling operation is carried out by a remotely operated fuelling machine moving on rails laid on the reactor top. The fuel handling system mainly consists of a fuelling machine, an inclined fuel transfer machine, a temporary fuel storage block located inside the reactor building and a fuel storage bay located outside the reactor. The temporary fuel storage block comprises fuel port and under water equipment. The fuel port acts as an interface with fuelling machine for charging new fuel and receiving spent fuel. Underwater equipment is used for handling the fuel within the storage block, and for transferring the fuel clusters across the containment walls through an inclined fuel transfer machine. The temporary fuel storage block also caters to buffer storage of the fuel to meet refuelling requirement in case of temporary outage of the inclined fuel transfer machine. The inclined fuel transfer machine transfers the fuel from temporary fuel storage block to the fuel storage bay located outside the reactor building. The fuel storage bay houses new fuel storage area, spent fuel storage area and the handling equipment. The design of the system has been conceptualized and following important concepts have been evolved.

6.2.1. Fuel transfer system to transfer fuel across containment walls

The inclined fuel transfer machine is a tall machine connecting the temporary fuel storage block to the fuel storage bay through the containment walls. A water filled pot containing fuel, guided in an inclined ramp, is hoisted up in the tilting leg and subsequently hoisted down to unload the fuel on the other side. Fig. 7 shows the fuel handling system of AHWR. The concept of the inclined fuel transfer machine is most suitable because of less requirement of space inside the reactor building, on line fuel transfer, small containment penetrations, assured cooling of fuel throughout the transfer and passive containment isolation features.

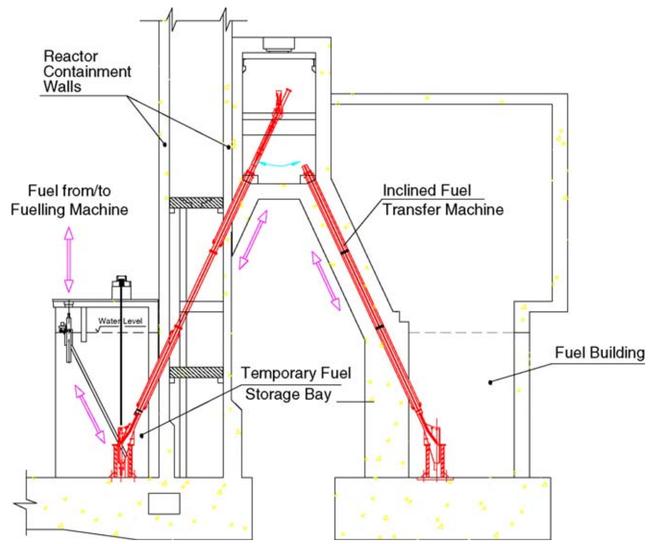

Fig. 7. Fuel handling system of the AHWR.

6.2.2. Fuelling machine

The experience and feedback of fuelling machines of the existing PHWRs and the Dhruva research reactor have been considered for the design of the AHWR fuelling machine. The fuelling machine is a vertical and shielded machine designed to handle the 10.5 m long fuel assembly (Fig. 8). The fuel assembly of the AHWR consists of the fuel cluster, shield A and shield B joined together through collet joints. The fuelling machine moves on the reactor top face to approach any individual coolant channel for carrying out the refuelling operation. The function of the fuelling machine is to remove and insert the fuel assembly. The major components of the fuelling machine are ram assembly, magazine assembly, snout assembly,

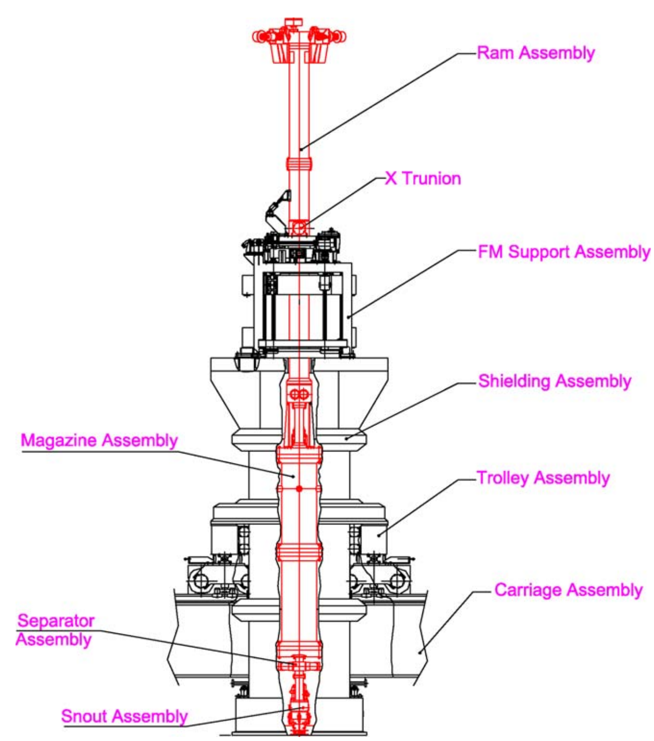

Fig. 8. Fuelling machine of the AHWR.

separator assembly and its trolley and carriage assembly. The snout plug located in the snout assembly makes a leak tight connection with the coolant channel end fitting. The snout assembly clamps the fuelling machine to the end fitting for carrying out the refuelling operation. The seal plug is located at the top of coolant channel and acts as a pressure boundary for the MHT water/steam. The ram assembly consists of three coaxial rams and the outer ram travels up to 7.6 m for removal of the fuel assembly from the coolant channel. The three coaxial rams manipulate the snout plug, seal plug, ram adaptors and shield plugs for their removal, movement and installation. The magazine assembly consists of eight tubes fixed on a rotor and temporarily stores the plugs and fuel cluster. The ram adaptor is hung in the magazine tube and holds the fuel cluster through a collet joint. The separator assembly is required to sense and hold the fuel assembly during its removal and insertion to facilitate joining/disjoining different fuel assembly parts. The fuelling machine head is hung to the fuelling machine support system through a X-trunion located at the ram housing of the ram assembly. The fuelling machine support system is mounted on a shielding assembly. The shielding assembly is supported on a trolley and carriage assembly. The trolley moves in *Y* direction on the rails provided on the carriage assembly. The carriage assembly moves across the reactor top face in *X* direction on fuelling machine rails. The drive is provided by an oil hydraulic system. The fuelling machine is coarse aligned to a particular channel through the trolley and the carriage travels, and fine alignment is by *X* fine and *Y* fine movements provided in the fuelling machine support system. During the refuelling process the fuelling machine clamps with the channel, makes leak tight joint, removes the seal plug, removes the fuel assembly, separates the shield 'A', shield 'B' and fuel cluster, replaces with new fuel and boxes up the channel after completing the reverse sequences of operations. The entire operation of fuelling machine is done remotely.

6.2.3. Fuel storage bay

A storage bay, located in the fuel building adjoining the reactor building, stores the fresh and the spent fuel under water. The storage pool capacity is decided based on the refuelling frequency of 113 fuel clusters (one-fourth of the core) per annum, two years cooling span for the fuel cluster and six months inventory of new fuel clusters. Provision is made to monitor the leakage from the bay. A fail-safe crane and handling equipment are provided in the bay.

6.3. Reactor building

The concept of double containment has been adopted in the design of AHWR reactor building. The containment structures consist of an inner containment wall and dome, forming the primary containment. An outer containment wall and dome form the secondary containment. The inner containment wall and the containment dome are made of prestressed concrete and the outer containment wall and outer containment dome are made of reinforced concrete. There exists an annular space of 5.2 m width between the two containments. The containment structures and the internal structures of the reactor building are founded on a common circular reinforced concrete base raft. The base raft is 4 m thick near the center and 5 m thick near the edges, where the walls are connected.

The AHWR reactor building has a 6000 m^3 capacity circular water tank in the inner containment located at an elevation of 136 m. This large water pool, called gravity driven water pool, is sufficient to cool the reactor for three days following any accident in the plant. The GDWP tank is made of reinforced concrete with a steel liner inside. The pool is supported on the ring beam of the inner containment all along its circumference. In addition to this, two tail pipe towers support it. These tail pipe towers extend right up to the base of the raft. Two steam drums are located within each tail pipe tower at an elevation of 123 m. The GDWP is divided into eight compartments.

7. Passive systems and inherent safety features of AHWR

The AHWR has several passive safety systems for reactor normal operation, decay heat removal, emergency core cooling, confinement of radioactivity, etc. (Bhat et al., 2004). These passive safety features are listed below:

- core heat removal by natural circulation of coolant during normal operation and shutdown conditions;
- direct injection of ECCS water in the fuel cluster in passive mode during postulated accident conditions like LOCA;
- containment cooling by passive containment coolers;
- passive containment isolation by water seal, following a large break LOCA;
- availability of large inventory of water in GDWP at higher elevation inside the containment to facilitate sustenance of core decay heat removal, ECCS injection, containment cooling for at least 72 h without invoking any active systems or operator action;
- passive shutdown by poison injection in the moderator, using the system pressure, in case of MHT system high pressure due to failure of wired mechanical shutdown system and liquid poison injection system;
- passive moderator cooling system to minimize the pressurization of calandria and release of tritium through cover gas during shutdown and station blackout;
- passive concrete cooling system for protection of the concrete structure in high-temperature zone.

The passive and active heat removal paths of the AHWR under various operational states and in LOCA are shown in Fig. 9. The design features of passive systems are described in the following paragraph.

7.1. Passive core heat removal by natural circulation during normal operation

During normal reactor operation, full reactor power is removed by natural circulation caused by thermo siphoning

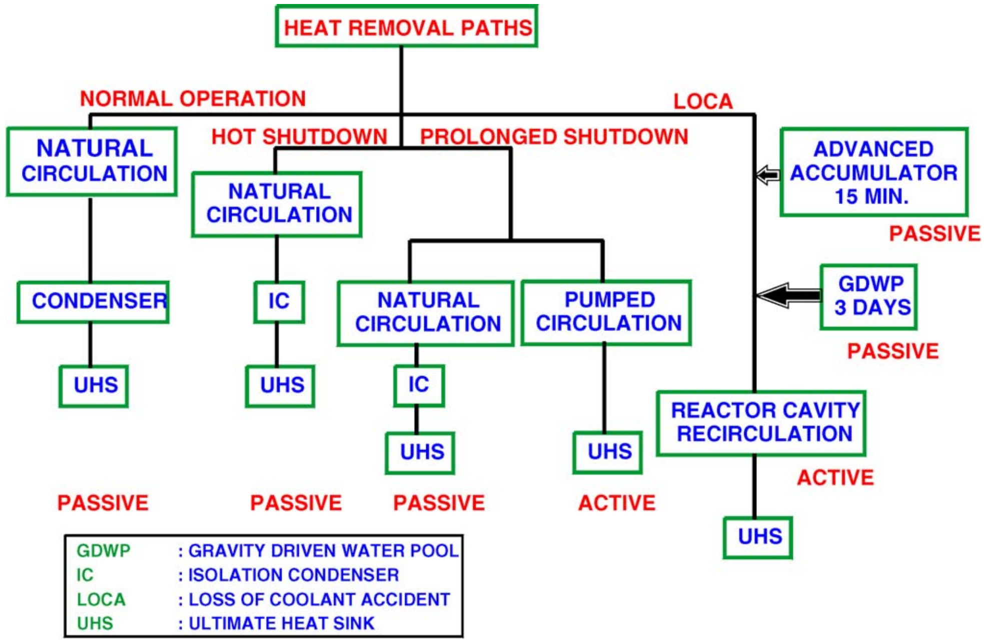

Fig. 9. Heat removal paths of the AHWR.

phenomenon. The main heat transport system transports heat from fuel rods to the steam drums using boiling light water in a natural circulation mode. The necessary flow rate is achieved by locating the steam drums at a suitable height above the core. By eliminating nuclear grade primary circulating pumps and their drives, and control system, all event scenarios initiating from non-availability of main pumps are therefore excluded besides providing economical advantage. The above factors result in considerable enhancement of system safety and reliability. Full power heat removal using natural circulation depends on several associated phenomena. Some details of the analysis and experimental studies are described later in the paper.

7.2. Passive core decay heat removal system

During reactor shut down, core decay heat is removed by eight isolation condensers (ICs) submerged in the gravity driven water pool. The pool acts as a heat sink for passive decay heat removal system. Four isolation condensers are capable of removing 6% full power core heat (decay heat at reactor trip). Passive valves are provided on the down stream of each isolation condenser. These valves get activated at a set steam drum pressure, and establish steam flow by natural circulation between the steam drums and corresponding isolation condenser under hot shutdown. The steam condenses inside the isolation condenser pipes immersed in the GDWP and the condensate returns to the core by gravity (Fig. 11).

The isolation condensers are designed to bring down the main heat transport system temperature from 558 to 423 K. The water inventory in the GDWP is adequate to cool the core for more than three days without any operator intervention and without leading to boiling of pool water.

During normal shutdown, decay heat is removed by natural circulation in the main heat transport circuit and the heat is transferred to ultimate heat sink through main condenser. The isolation condenser system removes heat during non-availability of the main condenser. In case of unavailability of both isolation condenser and main condenser, decay heat is removed by active system utilizing the MHT purification coolers.

7.3. Emergency core cooling in passive mode and core submergence

In the event of a loss of coolant accident (LOCA), four independent loops of ECCS provide cooling to the core for at least 72 h. A high-pressure injection system using accumulators and a low-pressure injection system using GDWP as source of water are passively brought into action, in a sequential manner, as the depressurization of the MHT system progresses, during LOCA.

The ECCS has four accumulators, each connecting to a quadrant of the fuel channels through a one-way rupture disc and a non-return valve. The rupture action due to the depressurization of the MHT system causes the injection of emergency coolant from accumulators. The accumulator houses a fluidic flow control device as shown in Fig. 10. It consists of a vortex chamber with a radial inlet connected to a vertical standpipe open at the top and a small tangential inlet. A large mass of cold water enters quickly into the core in the early stages of LOCA due to flow through the standpipe and tangential inlet during water level higher than standpipe. Later, a relatively small flow is extended for a longer time (∼15 min) through the tangential inlet due to vortex formation in the fluidic flow control device.

The GDWP is also connected to ECCS header by a one-way rupture disc and a non-return valve. The GDWP water is injected to remove the decay heat in the fuel after the MHT system pressure falls below GDWP pressure head.

During LOCA, the water from the MHT system, the accumulators and the GDWP, after cooling the core, collects in the space

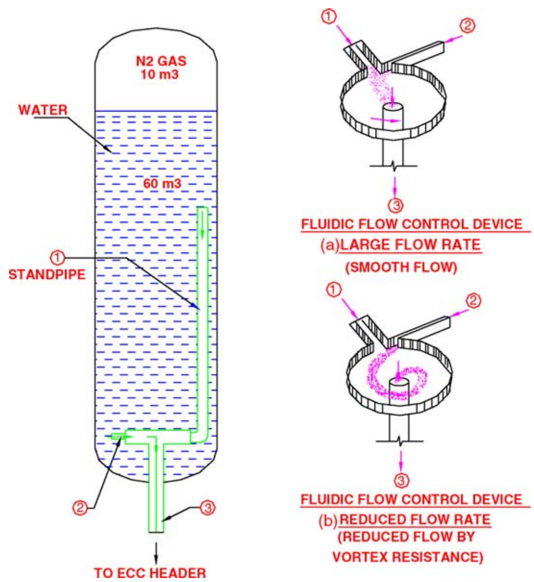

Fig. 10. Accumulator with fluid flow control device.

around the core in the reactor cavity and eventually submerges the core in water.

7.4. Passive containment isolation system

To minimize early large releases following a LOCA, it is necessary to isolate containment following a large break LOCA. To achieve this, a passive containment isolation arrangement has been provided, in addition to the closing of the normal inlet and outlet ventilation dampers (Maheshwari et al., 2004). The reactor building air supply and exhaust ducts are shaped in the form of U bends of sufficient height. In the event of a large LOCA, the containment gets pressurized and the pressure acts on the GDWP inventory and swiftly establishes a water siphon, into the ventilation duct U bends. Water in the U bends acts as seal between the containment and the external environment, providing the necessary isolation between the two. An isolation water tank is provided inside the GDWP to achieve the water seal in minimum possible time. The isolation water tank has a baffle plate with one side connected to V1 volume (volume containing high enthalpy systems) through a vent shaft and other side connected to V2 volume (volume comprise areas having low enthalpy systems) U duct. Due to the differential pressure between two sides of the baffle during the LOCA, the isolation tank water spill in to the U duct and isolate V1 volume from V2. Drain connections provided to the U bends permit the re-establishment of containment ventilation manually when desired.

7.5. Vapor suppression in gravity driven water pool

The GDWP absorbs the energy released in the containment immediately following the LOCA. After a postulated LOCA, the steam released to the V1 volume is directed to the GDWP through a large number of large size vent ducts. The vent ducts opens into the GDWP water, condensing the steam and cooling the non-condensable and reduces the heat load released to the containment.

7.6. Passive containment cooling

The passive containment coolers are utilized to achieve post-accident primary containment cooling to limit the primary containment pressure. The passive containment coolers are located below the GDWP and are connected to the GDWP inventory (Maheshwari et al., 2001). During the LOCA, a mixture of hot air and steam flows over the passive containment coolers. Steam condenses and hot air cools down on the outer tube surfaces of the coolers due to natural circulation of GDWP water inside the tubes providing long-term containment cooling after the accident.

7.7. Passive shutdown on MHT system high pressure

Passive shutdown system injects poison into the moderator by using the increased steam pressure arising out of the failure of wired shutdown systems. The AHWR has two independent shutdown systems, one comprising the mechanical shut off rods (SDS-1) and the other employing injection of a liquid poison in the low-pressure moderator (SDS-2). Both the shutdown systems require active signals for shutdown of the reactor. The scheme of passive shutdown actuates on high steam pressure due to unavailability of heat sink, followed by failure of SDS-1 and SDS-2. The schematic of the passive shutdown on MHT high pressure is shown in Fig. 11. In such an event of pressure rise, pressure opens a rupture disc and pressure is transmitted for opening a passive valve connected to a pressurized poison tank, injecting poison in the moderator to shutdown the reactor. Inadvertent poison injection is avoided by keeping the rupture disc burst pressure above the expected MHT pressure rise during and after reactor shutdown activated by either SDS-1 or SDS-2.

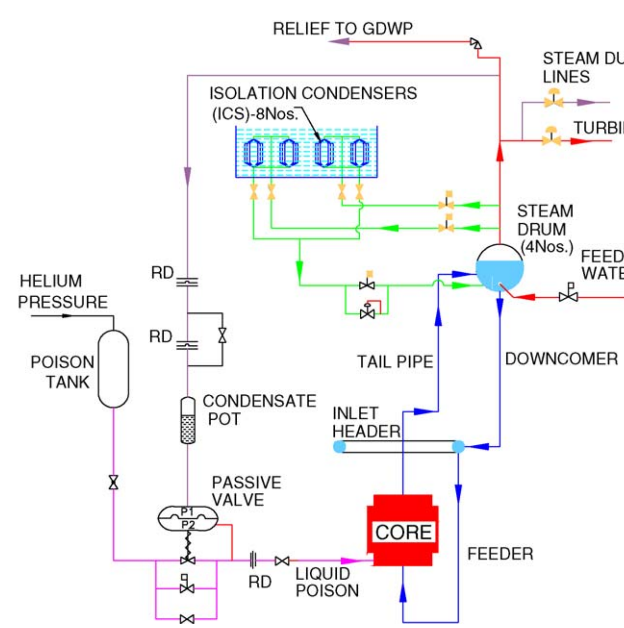

Fig. 11. Passive shutdown device.

7.8. Passive concrete cooling system

This system is designed to protect the concrete structure of the reactor located in the high-temperature zone (V1 volume). The cooling is achieved by circulation of coolant from GDWP in natural circulation mode through cooling pipes located between the concrete structure and an insulation panel. The heat transferred to the insulation panel is transferred to the GDWP water by the cooling pipes, fixed on a corrugated plate on outer surface of the insulation panel. Cooling pipes are placed at an optimum pitch to maintain the concrete temperature below 358 K.

This passive feature has eliminated the requirement of otherwise needed active equipment.

8. Thermal hydraulic analysis

The thermal hydraulic characteristics of a natural circulation reactor depend on the geometry of system, pressure, inlet sub-cooling, feed water temperature, and radial and axial power distribution in the core. The thermal hydraulic design of the AHWR has been carried out to provide adequate stability margin as well as thermal margin. The stability margin is defined as the ratio of successive amplitudes of flow oscillations following a disturbance to the system and the thermal margin is the minimum critical heat flux ratio. Thus, when the stability margin is less than one, the system is stable; if it is more than one it is unstable and if it is one, the system is at the threshold of stability. The MCHFR value has to be much larger than one for having a larger thermal margin for the reactor. The computer codes, TINFLO-S (Nayak et al., 2002) and ARTHA (Chandraker et al., 2002) were used to evaluate these parameters for the reactor. The effect of various parameters affecting the thermal and stability margins has been determined. The thermal hydraulic analyses were carried out to determine the channel flow distribution, exit quality, void fraction in the channels and the MCHFR. Some of the important results of the analyses are given in Table 6. A high value of stability margin is desirable. Several parametric studies were carried out to optimize the geometrical and operating conditions (Kumar et al., 2002). A brief outline of the results obtained is described below.

Table 6
Thermal hydraulic parameters of AHWR

Core fission power	960 MW_t
Core power	920 MW_t
Coolant	Light water
Heated fuel length	3.5 m
Total core flow rate	2237 kg/s
Coolant inlet temperature	533.5 K (nominal)
Coolant outlet temperature	558 K
Feed water temperature	403 K
Average steam quality	18.2%
Steam generation rate	407.6 kg/s
Steam drum pressure	7 MPa
MHT loop height	39 m
Minimum critical heat flux ratio (MCHFR) at 20% overpower	1.67
Maximum channel power	2.6 MW

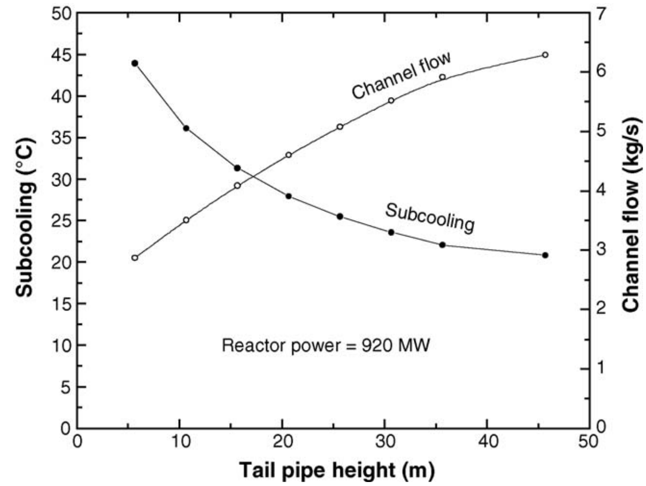

Fig. 12. Effect of tail pipe height on core inlet sub-cooling and channel flow rate.

8.1. Effect of tail pipe height

The variations of core inlet sub-cooling and flow rate with the change in tail pipe height for a high-power (2.6 MW) channel of the reactor are detailed in Fig. 12. It can be seen that with an increase in the tail pipe height the channel flow rate increases and sub-cooling at the core inlet decreases. Fig. 13 gives the variation of stability margin and CHFR with the change in the tail pipe height. Stability margin indicates that the normal operating region is away from the unstable region. Both the stability and the thermal margin increases with increase in the tail pipe height. However, beyond the tail pipe height of 20 m, the increase in the stability margin is only marginal while the thermal margin keeps on improving.

8.2. Effect of tail pipe size

Fig. 14 shows the effect of tail pipe (riser) diameter on the channel flow rate and the sub-cooling for a constant feed water

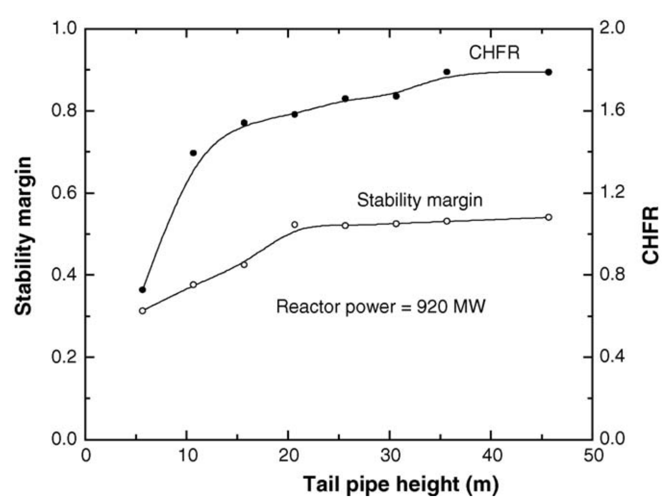

Fig. 13. Effect of tail pipe height on thermal and stability margin.

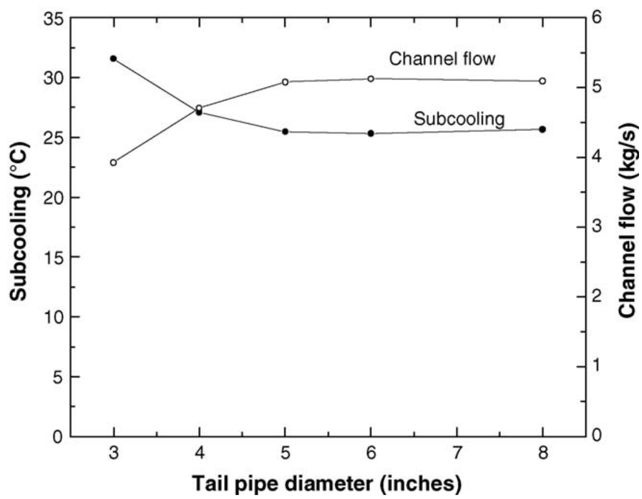

Fig. 14. Effect of tail pipe diameter on core inlet sub-cooling and channel flow rate.

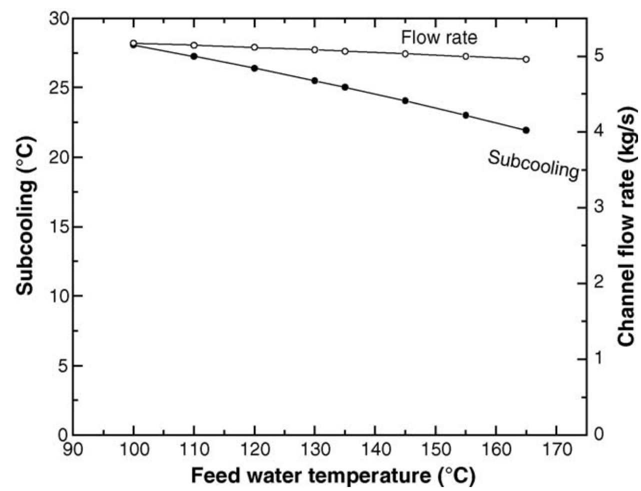

Fig. 16. Effect of feed water temperature on core inlet sub-cooling and channel flow rate.

temperature of 403 K. The channel flow rate initially increases with increase in tail pipe size and saturates beyond 127 mm diameter. The effect of tail pipe size on the stability and the thermal margin is shown in Fig. 15. Initially, with increase in the tail pipe size, the stability margin increases significantly but later on (beyond 127 mm tail pipe size) the increase is only marginal. The thermal margin improves with an increase in the tail pipe size. However, beyond the tail pipe size of 127 mm, the effect of increase in size is not significant.

8.3. *Effect of feed water temperature*

The effect of feed water temperature on core inlet sub-cooling and the channel flow rate is shown in Fig. 16. Both the channel flow rate and the sub-cooling decrease with increase in the feed water temperature. Fig. 17 shows the effect of feed water temperature on the thermal and the stability margin. The stability margin increases with increase in the feed water temperature while the CHFR decreases with increase in the feed water temperature.

8.4. *Stability analyses*

The Ledinegg type of instability occurs when the inlet sub-cooling exceeds 9 K (Fig. 18) for the system pressure of 0.1 MPa and the channel power greater than 315 kW. At a pressure of 1.0 MPa and sub-cooling less than 40 K, this type of instability is completely avoided. Thus, at the operating pressure of 7 MPa, Ledinegg type of instability is not a concern. However, density wave instability occurs even at a pressure of 7.0 MPa and inlet sub-cooling of 25.9 K if the power is less than 49.8% FP (Fig. 19). Thus, controlling the inlet sub-cooling and pressure will avoid both these instabilities.

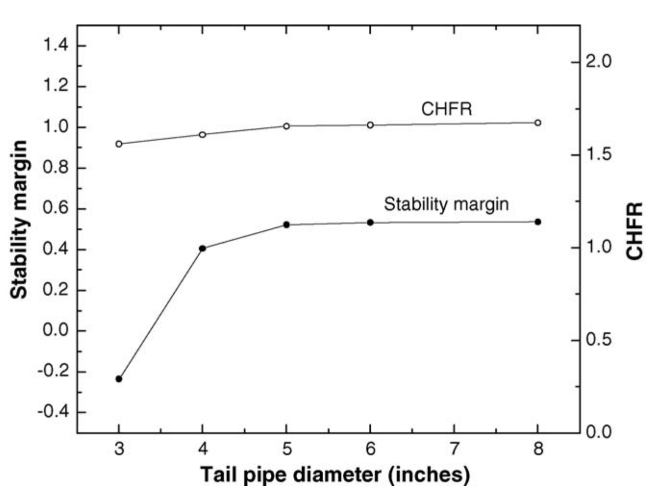

Fig. 15. Effect of tail pipe diameter on CHFR and stability margin.

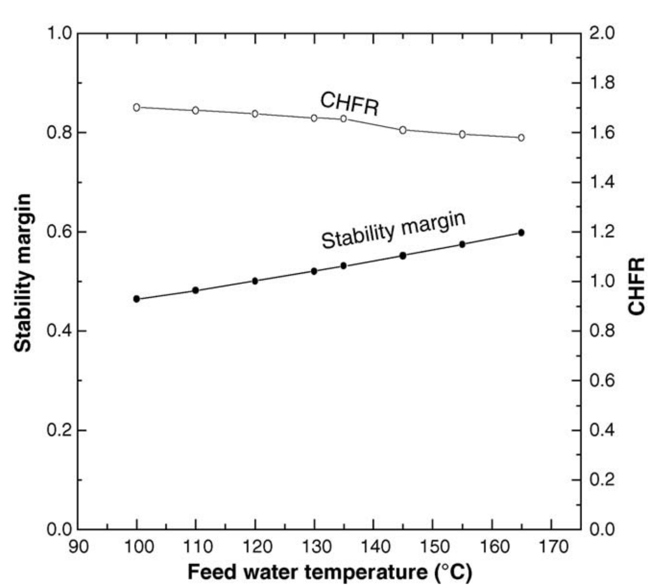

Fig. 17. Effect of feed water temperature on stability margin and CHFR.

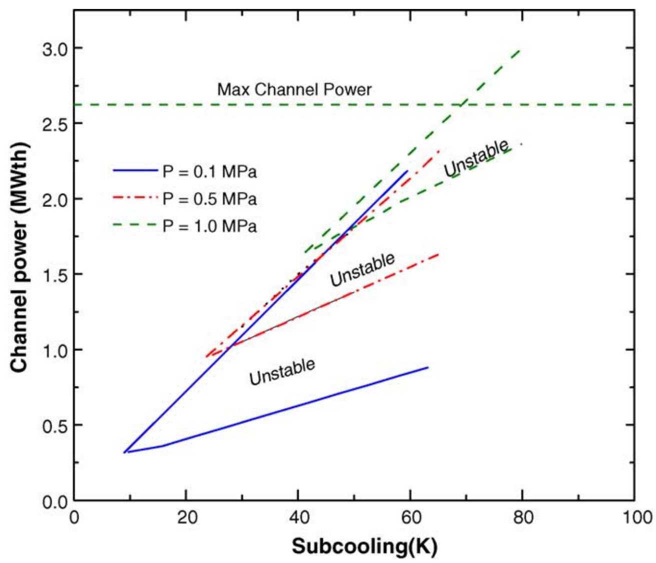

Fig. 18. Ledinegg type instability map for AHWR.

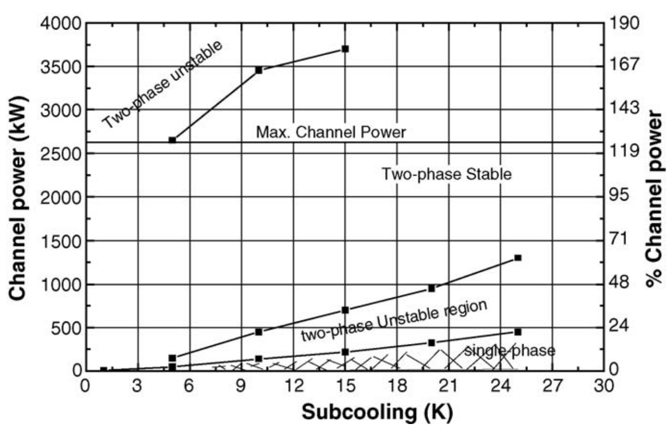

Fig. 19. Stability map for dynamic stability.

9. Experimental programs to demonstrate the inherent and passive features relevant to AHWR and their development status

Several passive features have been adopted for the AHWR for which analytical studies have been carried out. Some of these needs to be validated through experimental programs. A list of the experimental programs is given in Table 7.

A critical facility is being built at Bhabha Atomic Research Centre to validate the physics calculation models used for the AHWR.

An extensive experimental program has been planned and executed to understand the natural circulation characteristics including its stability for the design of the AHWR. This comprises setting up of several experimental facilities. The AHWR natural circulation characteristics during start-up, power raising and accidental conditions have been experimentally simulated in these facilities. In addition to several small facilities, a full size integral test loop (ITL) (Rao et al., 2002) has also been built to simulate the thermal hydraulic characteristics of the AHWR. The facility has the same elevation as that of the AHWR. The facility contains one full size channel of the AHWR, with its associated inlet feeder and tail pipe. The geometry of the feeder pipe and tail pipe of the ITL is retained the same as that of the AHWR, thus it simulates not only the driving buoyancy head, but also the resisting frictional forces which are vital in the simulation of natural circulation. The nominal operating pressure of the ITL is 70 bar and maximum power of operation is about 2 MW, which are closer to the prototypic conditions.

Figs. 20 and 21 show the simulation of decay heat removal behavior of the AHWR in the ITL following a station blackout. Under this condition, due to non-availability of the Class-IV power, the feed pumps are unavailable. The reactor is tripped and the main steam isolation valve (MSIV) closes. The decay heat

Table 7
Experimental program

Main objectives	Enabling technologies	Status of development
Negative void coefficient	Tight lattice pitch	Feasibility demonstrated
	Use of a scatterer cum absorber component within fuel cluster	Physics experiments to be done in the critical facility
Optimum use of passive systems for core heat removal	Natural circulation driven main coolant system	Ongoing and experimental studies planned in the ITL
	Isolation condensers	
	Large passive heat sink within containment	
	Passive valves	R&D in progress
Enhanced safety following LOCA	Passive emergency core cooling system (ECCS)	Planned experiments in the integral test loop
	Fluidic device in ECCS	Ongoing experimental program
	ECCS injection directly into fuel	Ongoing experimental program
	Passive containment isolation	Demonstration planned in a facility under construction
	Core submergence	Passive feature, no R&D required
	One-way rupture disk	R&D planned
	High reliability non-return valve	R&D in progress
Additional features to achieve low core damage frequency	Passive poison injection using steam pressure	Demonstration planned in an experimental facility
Moderator heat removal, stratification in large diameter calandria	Passive moderator cooling	Planned in a scaled model
High-temperature protection of concrete	Passive concrete cooling	Planned experiments in ITL

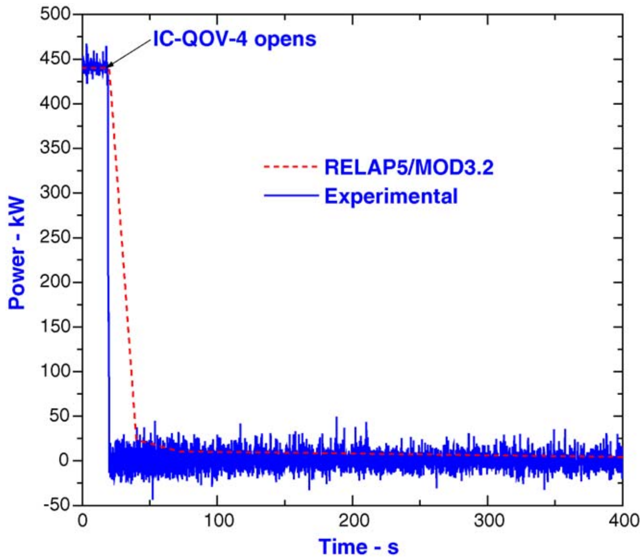

Fig. 20. Simulation of decay heat removal rate of the AHWR in integral test loop following a station blackout.

generated in the core is removed by thermo siphon using the ICs. In the integral test loop, the decay heat generation rate has been simulated by controlling the current flowing through the cluster, heated electrically. From Fig. 21, it can be observed that the MHT pressure continuously falls due to the steam condensation in the ICs. Many other safety experiments are being carried out in this facility to validate the AHWR design concepts.

The critical heat flux (CHF) under natural circulation condition is also a complex phenomena and it has been found that the conventional CHF relationships for forced circulation conditions cannot be applied over the entire range of operation under natural circulation conditions. To study the CHF of the AHWR cluster, several phased experimental program are underway both in BARC and at Indian Institute of Technology, Mumbai. These include conducting experiments in prototypic AHWR clusters

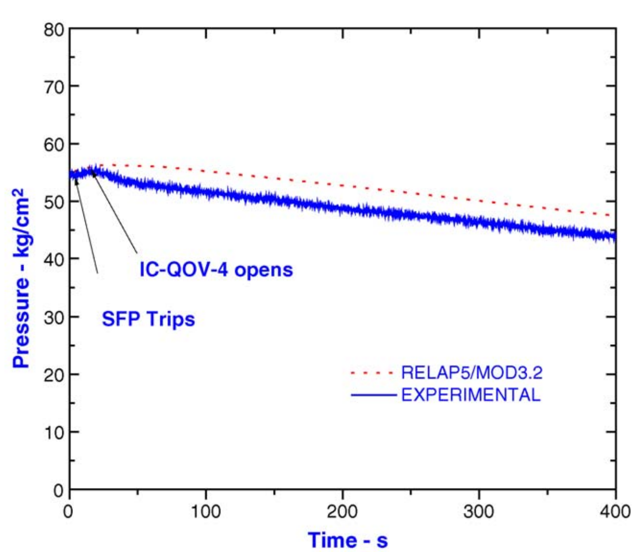

Fig. 21. Variation of steam drum pressure in the integral test loop following a station blackout.

using water as well as Freon as the working fluid and the corresponding nominal operating conditions of the reactor as well as development of mathematical models for critical power prediction based on film flow analysis.

Experiments on condensation of steam in presence of non-condensable gas are being carried out on the passive external condenser tube of AHWR. The purpose of the experiment is to determine the condensation heat transfer coefficient outside the tube in a stagnant steam/air mixture simulating the prototypic conditions. The experiments will be performed on different orientation of the tube with different concentration of non-condensable gases. Further, the tests will be conducted with the natural circulation of water inside the tube with steam/air mixture condensing outside the tube.

10. Safety analyses

The emphasis in the reactor design has been to incorporate passive safety features to the maximum extent, as a part of the defense in depth strategy. The main objective has been to establish a case for elimination of a need for an evacuation planning, following any credible accident scenario in the plant.

A major objective of design of the AHWR has been to provide a capability to withstand a wide range of postulating events without exceeding specified fuel temperature limits, thereby maintaining fuel integrity. The safety analysis of AHWR has identified an exhaustive list of 55 postulated initiated events (PIEs) (Gupta and Lele, 2002). The events considered include a wide range in the following categories:

- small, medium and large break LOCA;
- operational transients involving loss of coolant inventory;
- multiple system failure;
- power transients.

The safety analyses include 10 anticipated transients without scram scenario. The latter include the combination of a frequent event with unavailability of shutdown system. The acceptance criteria for all design basis accidents are given below:

- maximum fuel cladding temperature $\leq$ 1473 K;
- maximum local oxidation of fuel clad $\leq$ 18%;
- maximum fuel temperature anywhere in the core for any transient $\leq$ melting point of ThO_2;
- mass of Zr converted into $ZrO_2 \leq 1\%$ of the total mass of the cladding;
- radiation level at the plant boundary $\leq$ applicable levels for emergency planning.

The analyses indicate that in none of the accident sequences mentioned above the fuel clad temperature exceeds 1073 K.

In the conventional sense, for the purpose of design of the containment, a double-ended guillotine rupture of the 500 mm diameter inlet header has been considered. A clad surface temperature transient for this case is furnished in Fig. 22. This shows the efficiency and adequacy of the designed engineered safety feature, to limit the consequences, well within the acceptance

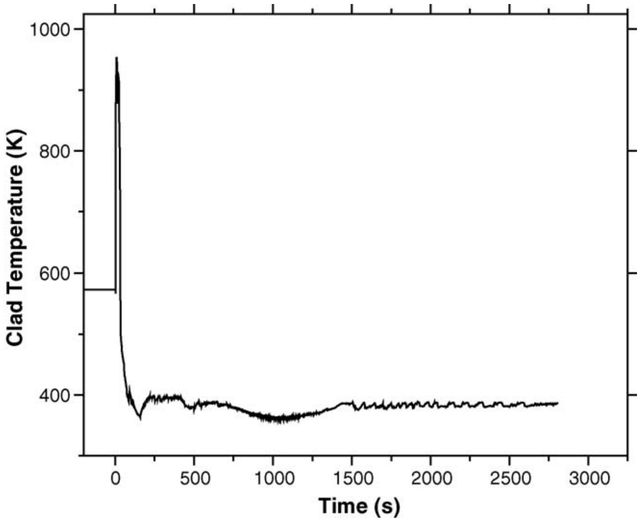

Fig. 22. Clad temperature of the maximum power rated channel for double-ended guillotine break of inlet header.

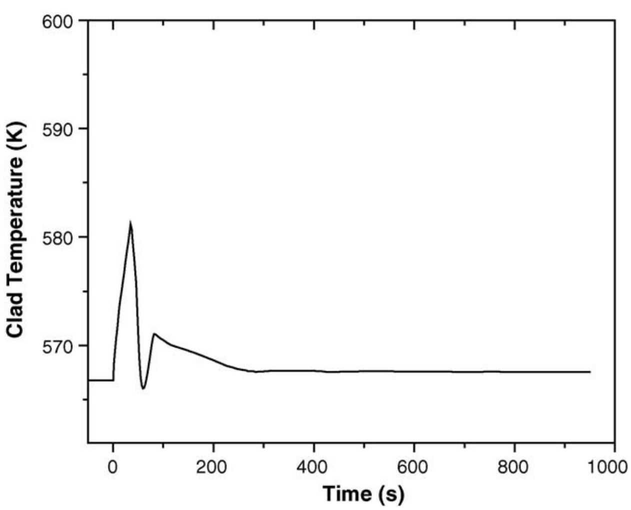

Fig. 23. Clad temperature of the maximum power rated channel under failure of wired shutdown system.

criteria limits. A large number of other accident scenarios conventionally fall within the category of beyond design basis accidents. However, even in these cases, including a case of station blackout together with failures of both the independent fast acting shutdown systems (SDS-1 and SDS-2), it has been demonstrated (Fig. 23) that none of the acceptance criteria for design basis accidents has been violated.

11. Summary

The design of Advanced Heavy Water Reactor incorporates several new features. These include utilization of thorium on a large scale and inclusion of several passive safety features. In addition to development of a system of computational tools to address the issues arising out of these innovations, an extensive experimental programme is under way to validate the design approaches used. At the current stage of design, the safety evaluations carried out so far, indicate that the peak clad temperature remains within acceptable limits for practically the entire range of initiating events and their credible combinations. The reactor is expected to serve as a platform for the development and demonstration of all technologies associated with the large-scale utilization of thorium, and advanced safety systems relevant for water cooled reactors.

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