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Multi purpose research reactor

V.K. Raina[a,*], K. Sasidharan[a], Samiran Sengupta[a], Tej Singh[b]

[a] Research Reactor Design & Projects Division, Bhabha Atomic Research Centre, Mumbai 400085, India
[b] Research Reactor Services Division, Bhabha Atomic Research Centre, Mumbai 400085, India

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Abstract

At present Dhruva and Cirus reactors provide the majority of research reactor based facilities to cater to the various needs of a vast pool of researchers in the field of material sciences, physics, chemistry, bio sciences, research & development work for nuclear power plants and production of radio isotopes. With a view to further consolidate and expand the scope of research and development in nuclear and allied sciences, a new 20 MWt multi purpose research reactor is being designed. This paper describes some of the design features and safety aspects of this reactor.

1. Introduction

High flux research reactors play a crucial role in providing unique facilities for basic research in frontier areas of science and for applied research related to development and testing of nuclear fuels and other materials. These reactors also cater to the increasing need of radioisotopes for application in the fields of medicine, agriculture and industry.

Keeping in view the projected requirements of reactor based facilities for basic and applied research as also the projected demand for radioisotopes in the country, construction of a new "multi purpose research reactor" (MPRR) is essential.

The MPRR with its high neutron flux and irradiation volume will provide the appropriate platform for research in reactor fuels, reactor materials, condensed matter research for study of structure and dynamics of materials, stress analysis of engineering components especially reactor materials, dynamic radiography, time of flight refractrometry, small sample investigations for the study of new and novel materials such as ferromagnetic, opto-electric materials, protein crystals, fullerenes, porous-silicon, etc., including high pressure and milli-Kelvin range studies which will greatly help the nation to keep abreast of the developments in the field of material sciences and development of novel materials and alloys.

The MPRR will supplement the isotope production capacity of Dhruva research reactor to meet the projected requirements of various isotopes beyond the year 2010. Installation of a xenon gas loop, for production of I-125, which has a great demand for medical and industrial applications, is also being explored.

The proposed 20 MWt research reactor will have a maximum thermal neutron flux of 5×10^{14} n/cm^2/s. The reactor will be fuelled with Low Enriched Uranium dispersion type fuel and will use light water as coolant and moderator. The reactor core will be surrounded by an annular heavy water tank to achieve a high neutron flux and irradiation volume to maximize the number of irradiation positions available for isotope production and material irradiation. Most of the irradiation positions will be accommodated in the heavy water reflector tank surrounding the core.

2. Physics design

2.1. Design objective

The primary objective of the nuclear design of the MPRR is to obtain high usable thermal and fast neutron (>821 keV) flux levels in experimental/irradiation positions, low nuclear fuel inventory, minimum number of reactivity control devices, optimal fuel thermal hydraulics and high reactivity safety margins. Design also caters to experimental and irradiation reactivity loads as well as operational reactivity requirements, fuel burn-up and xenon over-ride.

* Corresponding author. Tel.: +91 22 25594608; fax: +91 22 25505311.
E-mail address: vkrain@magnum.barc.ernet.in (V.K. Raina).

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A sub-optimal H/^{235}U atom ratio is adopted in the design to ensure that void coefficient of reactivity is sufficiently negative. The design aims to obtain an average discharge ^{235}U fuel burn-up of 35 at% at 20 MWt reactor operations corresponding to a fuel residence time of at least 3 months. Fuel shuffling and/or partial replacements for burn-up reactivity compensation would be effected once in about 4 weeks.

2.2. Core and reflector

The reactor core is constituted by 37 lattice positions laid in a square pitch of 81.7 mm. The nominal equilibrium core consists of 28 standard fuel assemblies and six control fuel assemblies. The central position and two peripheral positions will be used for high neutron flux experiments/irradiations. The nominal ^{235}U mass in the equilibrium core would be about 6.0 kg.

The core is surrounded by a 600 mm thick heavy water reflector. Given the relatively small size of the reactor core for achieving high neutron flux level, the number of experimental/irradiation positions that can be provided in the core is rather limited. Incorporation of a large heavy water reflector tank around the reactor core helps in sustaining high thermal neutron flux levels over a large radial distance where most of the experimental/irradiation positions are easily accommodated. The core configuration along with experimental facilities in the reflector tank is shown in Fig. 1.

Graphite fillers have been used to occupy the space between core periphery and the inner surface of the reflector tank. As presence of water in this region reduces the thermal neutron flux levels in the reflector region, water content in this region will be kept as low as practically achievable.

The MPRR core will be loaded with plate type fuel assemblies. The fuel meat is loaded with Low Enriched Uranium (19.75%, w/w ^{235}U) in the form of U_3Si_2 dispersed in aluminium matrix with a loading density of 3.0 g/cm^3. The meat has a thickness of 0.6 mm and is clad with 0.4 mm thick aluminium alloy. A coolant gap of 2.5 mm is maintained between the fuel plates. Fuel assemblies are of two types:

(i) Standard fuel assembly.
(ii) Control fuel assembly.

A standard fuel assembly consists of 18 fuel plates and two Al-alloy inert plates. The nominal amount of ^{235}U in a plate is 13.85 g. The control fuel assembly is basically a standard assembly with six fuel plates removed to create two adequately sized gaps to accommodate two hafnium (Hf) absorber elements.

2.3. Physics design approach

The core nuclear design calculations have been carried out in three stages as indicated below. All the codes used for these calculations were validated against the IAEA benchmark reactors and the results obtained were found in good agreement with those reported in IAEA-TECDOC-233 (1980) and IAEA-TECDOC-643 (1992).

- Plate cell calculations.
- Assembly level or super-cell calculations.
- Core level or global calculations.

2.3.1. Plate cell calculations

A plate cell consists of fuel meat, clad and associated light water coolant/moderator. Plate cell calculations have been carried out using the transport theory lattice computer code WIMS

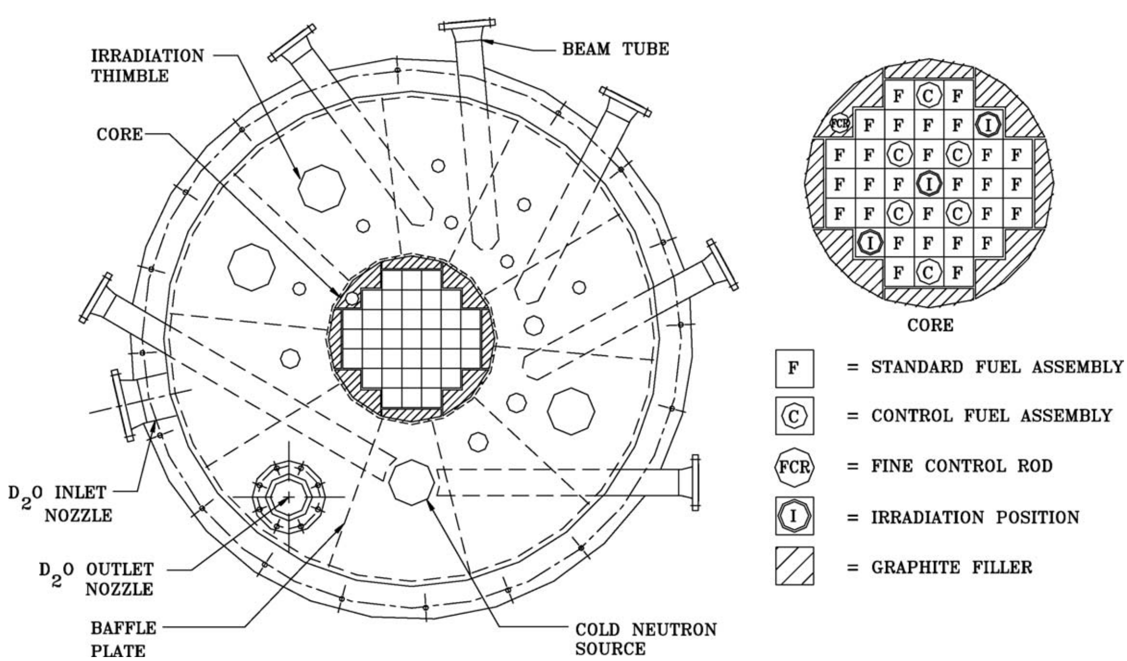

Fig. 1. Core configuration and experimental facilities.

package (Askew et al., 1966) originated from the Winfrith Laboratory. The lattice code version WIMS-D/4 (Hallsall, 1980) has been employed with its own 69 group nuclear data library which consists of 14 fast energy groups above 9.118 keV and 42 thermal groups below 4.0 eV and 13 groups in the energy range of 4.0 eV to 9.118 keV to cover the resonance range. Main transport calculations were done in appropriately chosen 27 neutron energy groups of the standard 69-group WIMS library. The 27 group cross-sections for the plate cell were condensed to three group homogenized lattice data. The three groups are 10–0.821 MeV (fast), 821 keV to 0.625 eV (intermediate) and <0.625 eV (thermal). All operating conditions including fuel burn-up have been considered. Sensitivity analyses have also been done using different cross-section libraries such as ENDF/B-VI.8, JENDL-3.2, IAEA and JEF-2.2 (Leszczynski et al., 2002).

Each square lattice cell (8.17 cm × 8.17 cm) area containing the fuel elements is represented by two material regions. The inner plate cell region (7.02 cm × 6.5 cm) represents the fuel portion of the assembly in the center of the multiplying cell. The peripheral region around the fuel region represents the extra aluminium in the fuel plates, the side plates, inert plates and remaining water.

There are two different types of plate cell geometries: one for a standard fuel element and the other for control fuel element. For other materials appearing in non-fuel region (such as heavy water, light water, aluminium and graphite filler), the cross-sections have been generated by considering them to be one pitch away from the plate cell boundary with light water present between them.

2.3.2. Assembly level calculations

The super-cell group cross-sections have been generated by weighing the corresponding macroscopic cross-sections of inner and outer regions with their volumes and group fluxes in corresponding regions. The group fluxes in different regions of the super-cell have been calculated using the two-dimensional transport theory computer code TWOTRAN (Lathrop and Brinkly, 1970) in *X–Y* geometry with reflective cell boundary conditions.

For assessment of the reactivity worth of a control rod, the group flux distributions have been obtained with and without hafnium blades being present in the control element lattice cell. Corresponding controlled super-cell properties were generated, in a similar way as for other multiplying cells.

2.3.3. Core level calculations

Three-dimensional global simulations of the reactor were carried out using the computer nodal code FINSQR (Singh and Sengupta, 1999). The code is used to solve the multi-group neutron diffusion equations by using finite Fourier transform technique in *XYZ* geometry. The code calculates the core reactivity, core fission power distribution and multi-group fluxes in and around the core.

2.4. Reactivity control and shut down devices

The reactivity is controlled by vertical movement of six fork type control cum shut off rods, with twin blades made of 3.25 mm thick, 60 mm wide and 600 mm long hafnium (Hf) blades. Hafnium thickness is chosen to ensure that the absorber section is black to thermal and epi-thermal neutrons. The reactor power is regulated by vertical movement of a lower worth hafnium/cadmium cylindrical fine control rod. The fine control rod will be located in the graphite filler region around the core. Locations of the six control cum shut off rods and the fine control rod in the core are shown in core map shown in Fig. 1.

The normal scheme of reactor control envisages controlled (auto) adjustment of the position of the fine control rod. Worth of the fine control rod is deliberately chosen to be less than the β_{eff} of the core to restrict the consequences of a loss of regulation scenario.

Six control cum shut off rods are to be moved in a bank for coarse control of reactivity. The total reactivity worth of the control cum shut off rods, which are also required to perform the function of a fast shut down device, is adequate to provide a shut down margin of at least 50 mk. The shut down margin is more than the worth of a fresh standard fuel assembly in a central location plus a reactivity uncertainty of 10 mk. The total reactivity worth of the control cum shut off rods including fine control rod in the equilibrium core is about 145 mk and meets the shut down margin requirement at the beginning of cycle of the equilibrium core.

The reactor core loadings for operation at 20 MWt will be designed to satisfy the following reactivity related safety criteria.

- The sub-criticality at complete shut down state should be at least 50 mk. Complete shut down state is defined as all control rods and regulating rod in fully inserted state in the core.
- At partial shut down state, the reactor should be sub-critical by at least 10 mk. The partial shut down state will typically correspond to the six control cum shut off rods at 65% IN and the fine control rod in fully IN state.
- Core loading changes will be carried out in partial shut down state of the reactor.
- The core loading will be such that any five of the six control cum shut off rods (i.e. with the maximum worth rod considered unavailable) should be able to maintain the reactor sub-critical by at least 10 mk.

2.5. Reactivity effects of power operation

2.5.1. Power and temperature

As the reactor power is raised, temperature in different regions of the core starts increasing and affects the reactivity of the reactor. An assessment has been made at an average fuel burn-up of 25% using the earlier mentioned computer codes. The estimated temperature coefficients of reactivity of fuel and coolant and power coefficient of reactivity (with six control cum shut off rods fully out) are given below:

Fuel: −0.013 mk/°C.
Coolant: −0.006 mk/°C.
Power: −0.03 mk/MW.

2.5.2. Void reactivity

The reactor will have negative void coefficient of reactivity both in fresh and irradiated conditions of the fuel assemblies. Hence, formation of incipient steam voids during operation is not expected to lead to unsafe reactivity transients. The estimation of void coefficient of reactivity has been carried out by generating the lattice parameters at different densities of the coolant at an average burn-up of 25%. The core reactivity is found using FINSQR at different densities. The value of void coefficient of reactivity is estimated to be about (−) 0.7 mk/% of void.

2.5.3. Xenon

Xenon and samarium are two important saturating fission products, which are of concern during the operation of a thermal reactor. The equilibrium xenon load is estimated to be about 30 mk. Being a high thermal neutron flux reactor, the load due to xenon increases rapidly following reactor trip. A provision of about 15 mk reactivity is made to over-ride xenon for a period up to 30 min after reactor trip. Should reactor poison out, restart-up of the reactor will be possible after a period of about 25 h. The equilibrium reactivity load of samarium is about 5 mk.

2.5.4. Fuel burn-up

The core is designed to operate at a thermal power of 20 MW without refueling for a 4 week operating cycle. The average burn-up in equilibrium core at beginning of cycle (BOC) and end of cycle (EOC) are estimated to be about 25 and 30% atom, respectively. The average estimated burn-up coefficient for the equilibrium core is about (−) 0.052 mk/MWd. The reactivity loss per day is estimated to be about 1 mk.

2.5.5. Core excess reactivity in equilibrium core

The formation of core configuration is such that core excess reactivity of 90 mk will be available at beginning of cycle for about 4 weeks of continuous operation of reactor and to cater to xenon, temperature, xenon over-ride, burn-up and isotope/experimental irradiation loads without refueling/shuffling of the fuel elements. The core excess reactivity at end of cycle will be about 65 mk. In equilibrium core, the various reactivity loads are as follows:

	Reactivity load (mk)
Xenon	28.5
Samarium and other FPs	5.5
Power and temperature	1.0
Core burn-up	25.0
Xenon over-ride	15.0
Experiments/irradiations	15.0
Total	90.0

2.6. Neutron flux and power distribution

The maximum unperturbed thermal neutron flux at the central water hole in the core and in the reflector region is estimated to be about 5×10^{14} and 2×10^{14} n/cm^2/s, respectively, at 20 MWt power of the reactor. Mapping of the thermal neutron flux is given in Fig. 2. The fast neutron flux above 821 keV in the central water hole is estimated to be about 1.0×10^{14} n/cm^2/s. The estimated maximum fission power levels in a standard and a control fuel assembly with all control rods at top position are estimated to be about 786 and 564 kW, respectively. The overall peaking factor (including axial and local) has been estimated to be about 1.5. The partially inserted control rods will peak the neutron flux towards the bottom of the core. At the beginning of cycle, the control rods will be at about 50% IN position in the core and the maximum power of standard and control fuel assemblies is estimated to be 760 and 454 kW, respectively, at 20 MWt power of the reactor. In this condition, the maximum overall peaking factor for standard and control fuel assemblies will be 1.8 and 1.94, respectively.

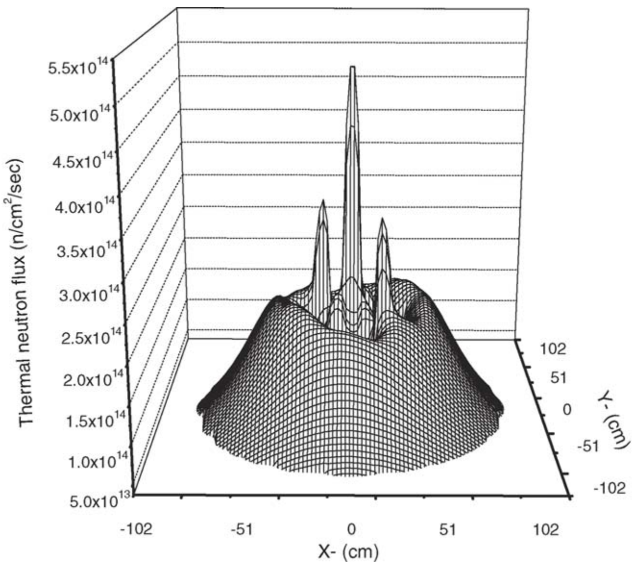

Fig. 2. Thermal flux mapping in the core and heavy water reflector.

3. Fuel

As discussed in the physics design, the reactor will be loaded with Low Enriched Uranium plate type fuel of U_3Si_2 dispersed in aluminium matrix. U_3Si_2 dispersion fuel has been chosen considering high uranium density in fuel meat, good compatibility with aluminium, high thermal conductivity, excellent blister resistance threshold (about 515 °C), stable swelling behaviour under irradiation, high fission gas retaining capability, low release of volatile fission products and better fabricability. The powder metallurgy process is used for the fabrication of U_3Si_2 dispersion fuel (Hegde et al., 1993). Aluminium alloy is chosen as clad material considering its high thermal conductivity, small cross-section for neutron absorption and low cost.

The core will be loaded with two types of fuel assemblies comprising of the standard fuel assemblies and control fuel assemblies.

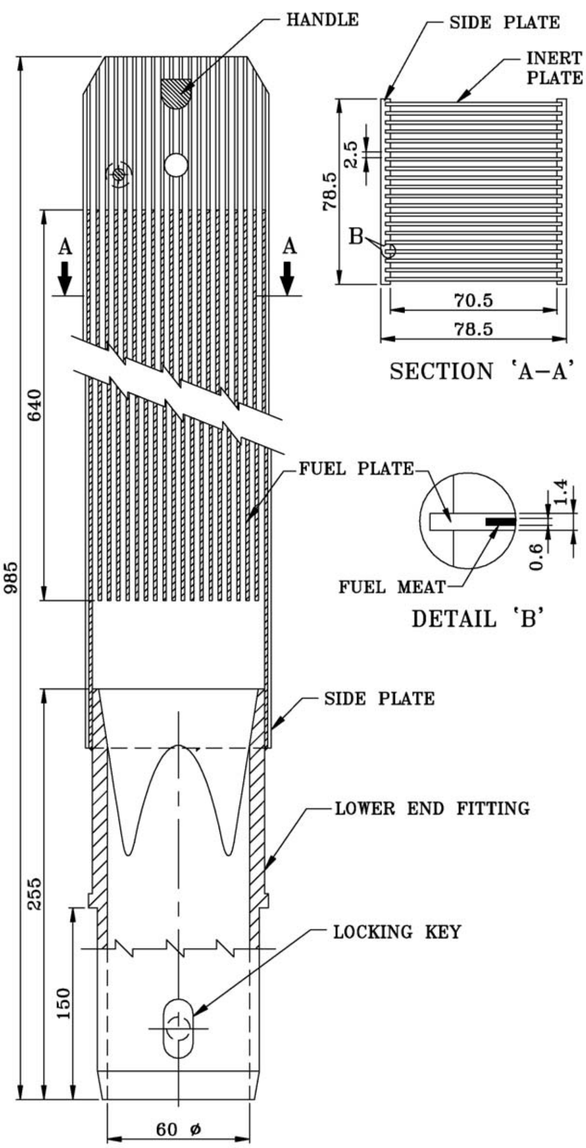

Fig. 3. Standard fuel assembly.

3.1. Standard fuel assembly

The general arrangement of the standard fuel assembly is shown in Fig. 3. Standard fuel assembly consists of 18 fuel bearing plates and two aluminium inert plates swaged into the grooves provided in the side plates. A water gap of 2.5 mm is maintained between fuel plates/inert aluminium plates. Each plate has a meat thickness of 0.6 mm. The fuel meat is clad with 0.4 mm thick nuclear grade aluminium sheet. A fresh standard fuel assembly contains about 250 g of U-235.

The fuel box is welded to a lower end fitting. The lower end fitting provides smooth transition from square cross-section of the fuel box to a cylindrical toe for supporting the fuel assembly on the grid plate. A collar and a key provided in the toe portion ensure proper support and orientation of the fuel assembly. A handle has been provided at the top of the fuel box for remote handling.

3.2. Control fuel assembly

The general arrangement and details of control fuel assembly are given in Fig. 4. Control fuel assembly is basically a standard fuel assembly with six fuel plates removed to create the space to accommodate the twin blade fork type absorber element. The twin blades are placed symmetrically with respect to central axis of the fuel assembly. Each assembly has a special attachment of a guide tube at the top and 7.2 mm wide recess right through for movement of each absorber blade. The recess has been created by providing two aluminium guide plates swaged into the side spacer plates of the fuel box. The gap between fuel bearing plates is 2.5 mm. The same gap is maintained between a fuel bearing plate and the inert plate. The absorber element used for control fuel assembly is made of hafnium. The movement of the absorber blade of the control fuel assembly is controlled by the drive unit provided on the platform at the pool top.

Since the fuel will be operated at high power density, the fuel loading has been restricted to 3 g/cm^3 and fuel meat temperature is limited to 130 °C. These limits have been imposed to avoid swelling of irradiated fuel of high density due to inter-diffusion between U_3Si_2 and aluminium resulting in the formation of $U(AlSi)_3$. The clad temperature is also limited below onset of nucleate boiling limits to prevent reactivity effects due to void formation. Maintaining low clad temperature also reduces oxide formation on the clad surface, which in turn prevents fuel meat temperature increasing to higher values.

In order to ensure the fuel safety, the coolant velocity has been so chosen that for the hottest standard and control fuel assembly, the fuel meat and clad temperatures do not exceed the prescribed limits. The plot of fuel meat, clad and onset of nucleate boiling (ONB) temperatures along the fuel length of the highest rated standard assembly considering nuclear and engineering peaking factors is given in Fig. 5. The variation in fuel clad temperature and onset of nucleate boiling temperature for some of the postulated initiating events are discussed in the section on safety analysis.

4. Plant layout

The overall layout of the reactor complex is shown in Fig. 6. The reactor building which houses the reactor block and the fuel storage block is a rectangular RCC building with internal dimensions of 40 m × 30 m × 20 m height above the ground level. It is a confinement building designed for an internal pressure of 200 mm water gauge and external pressure of 100 mm water gauge. The personnel entry into the reactor building is through an air lock (PAL), which is approached through a service corridor of reactor annex. For vehicle entry into the reactor building a vehicle air lock (VAL) is provided at the rear side of the building. Service building is a two storied RCC building located adjacent to the reactor building and houses electrical substation, water treatment area and other plant auxiliary systems. The plant auxiliary system includes air compressors, refrigerating machines,

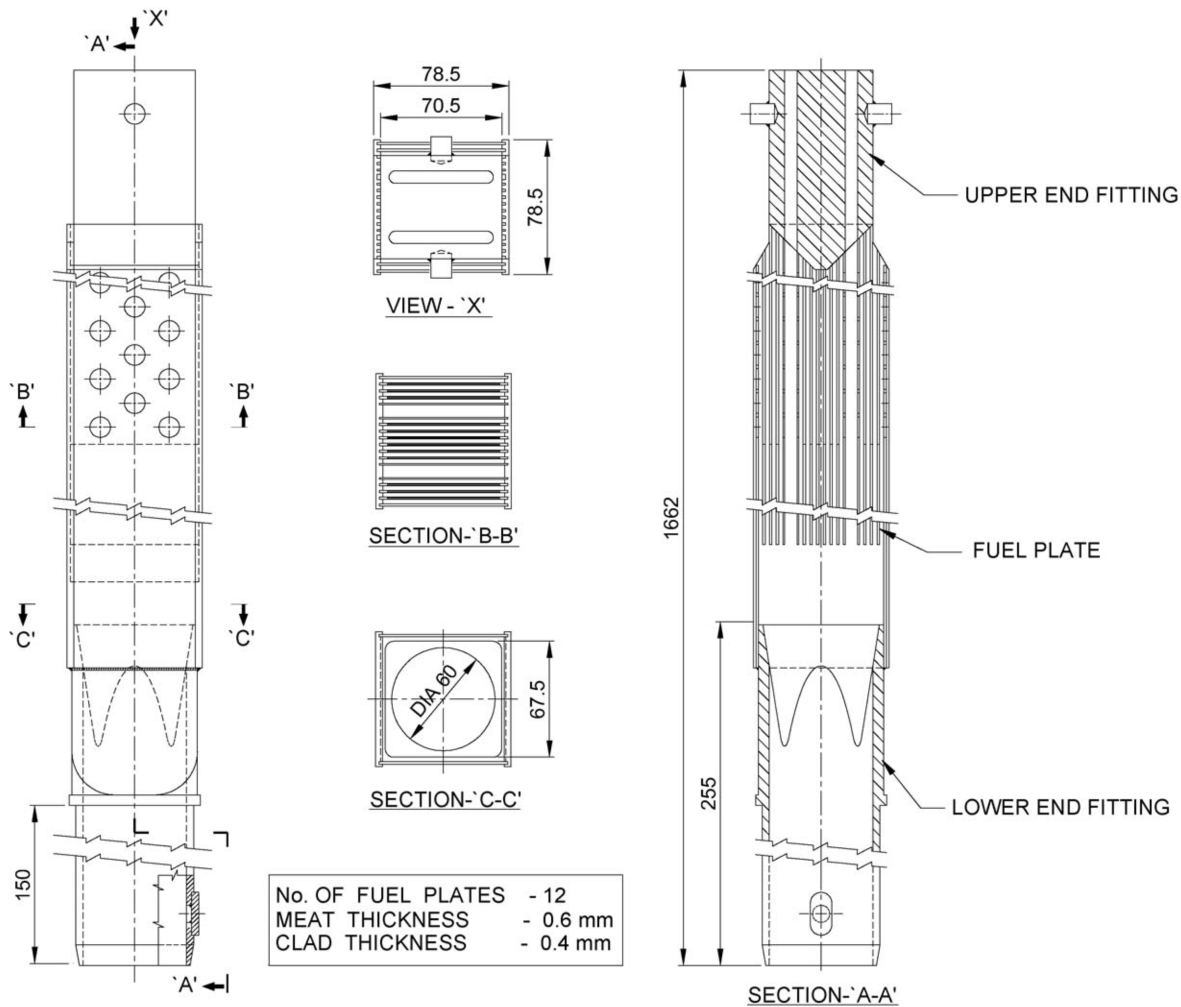

Fig. 4. Control fuel assembly.

air handling and ventilation system. A service corridor provides access to service building from other buildings. The control building is a three storied building which locates the central control room, automatic constant voltage rectifiers (ACVRs), uninterrupted power supply (UPS) sources, battery banks and associated panels. Adjacent to control building emergency diesel generators and motor alternators are provided. A guide tube laboratory (GT-LAB) used for neutron physics experiments using cold neutron source is located at basement elevation adjacent to the reactor building. The filter house is located behind the reactor building and houses the ventilation system equipments and HEPA filter banks. A stack of 100 m height is located behind the filter house for discharging gaseous effluents to atmosphere.

5. Reactor and fuel storage block

The reactor and fuel storage block consists of two water filled bays isolatable by sealing gates with inflatable seals. The reactor bay houses the core, core support structures, beam tubes, piping of primary and reflector cooling systems, neutron sensors, natural circulation valves, etc. Fuel storage bay houses racks for fuel and active assemblies and primary coolant system piping and has provision to lower the fuel transfer shielding flask at a specified location in the pool. The layout drawing of reactor and fuel storage block is shown in Fig. 7. The floor and the inside walls of the reactor bay and fuel storage bay are lined with SS 304L plates to ensure leak tightness of the pool and to facilitate better control of pool water chemistry and ease of decontamination, if required.

The reactor block along with the fuel storage block is housed in the reactor building measuring 40 m × 30 m × 20 m height. The reactor bay/pool is 11.5 m deep with a wall thickness of 2.4 m, which provides requisite biological shielding in radial direction. The pool water provides shielding against radiation in axial as well as in radial direction and also acts as moderator and coolant for the reactor core. The pool water also acts as the heat sink in the event of non-availability of forced cooling.

Fuel storage bay is designed to store all the fuel assemblies that would be discharged from the core in 15 years of full power operation with sufficient sub-criticality margin. The reactor bay and the fuel storage bay are interconnected and are serviced by a common movable service trolley mounted on rails at the pool top to facilitate safe handling of fuel, isotope and other irradiation assemblies.

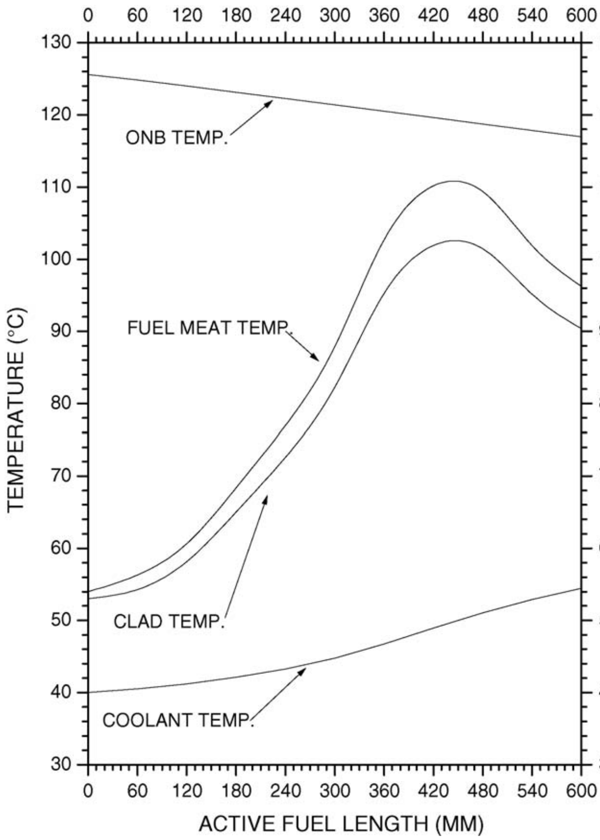

Fig. 5. Temperature distribution across fuel plate of hottest standard assembly.

The reactor core is located 9 m below the water level in the reactor bay. The core is constituted by standard and control fuel assemblies mounted on a grid plate having 37 lattice positions laid on a square pitch of 81.7 mm. An annular reflector tank of aluminium alloy, 635 mm i.d. × 1895 mm o.d. and 1100 mm height surrounds the core to provide 600 mm thick heavy water reflector around it. This tank also accommodates horizontal beam tubes and vertical experimental and irradiation positions as indicated in Fig. 1. The grid plate and the reflector tank are mounted on a support structure, which also acts as the bottom outlet plenum for the primary coolant system. The space between core lattice assemblies and the inner shell of the reflector tank is filled with graphite fillers for neutron economy. The graphite fillers are supported on the grid plate. The arrangement of the core, the reflector tank and the support structure is shown in Fig. 8.

A fixed platform above the core at pool top is provided to serve as support for control rod drives, actuating mechanism for natural circulation valves, vertical tubes for ion chambers, various cables and for ease of handling fuel and irradiation assemblies.

Researcher's area and the reactor pool are sized such that, in the event of a rupture of beam tube, core submergence of minimum 1 m is assured with the pool water reaching equilibrium height between reactor pool and researcher's area.

6. Primary coolant system

Primary coolant system is designed to remove the heat generated in the reactor core, reactor pool and fuel storage pool. The heat picked up by the primary coolant system is transferred

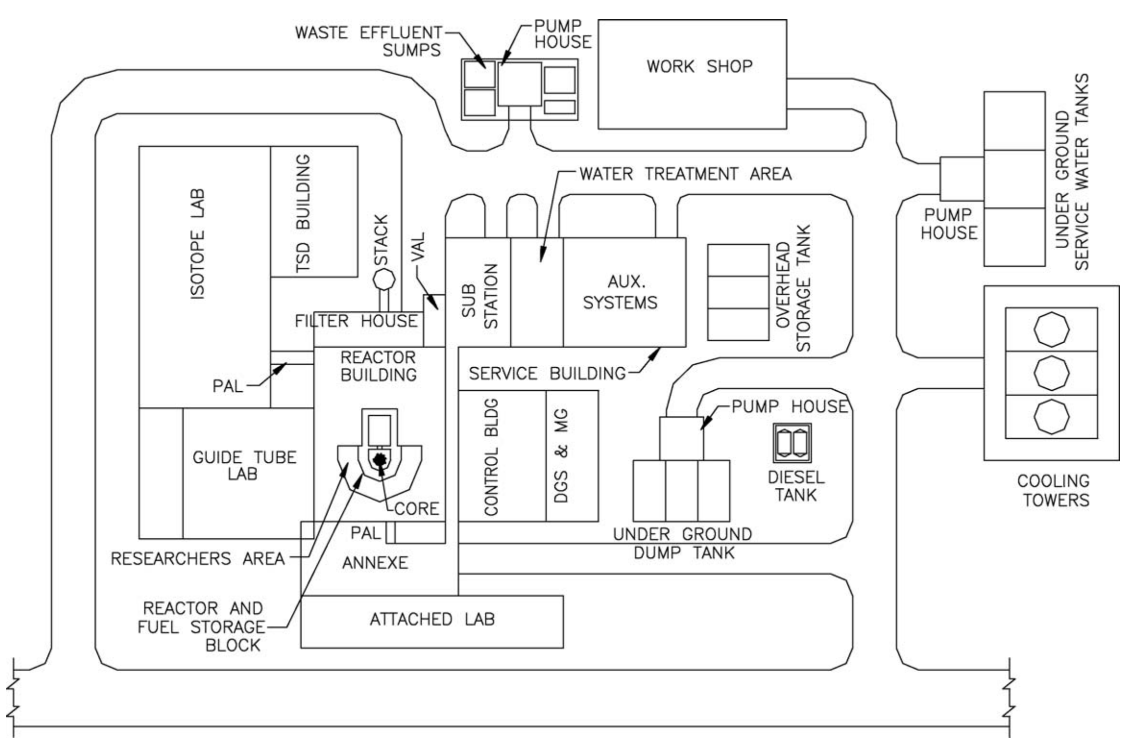

Fig. 6. Overall plant layout.

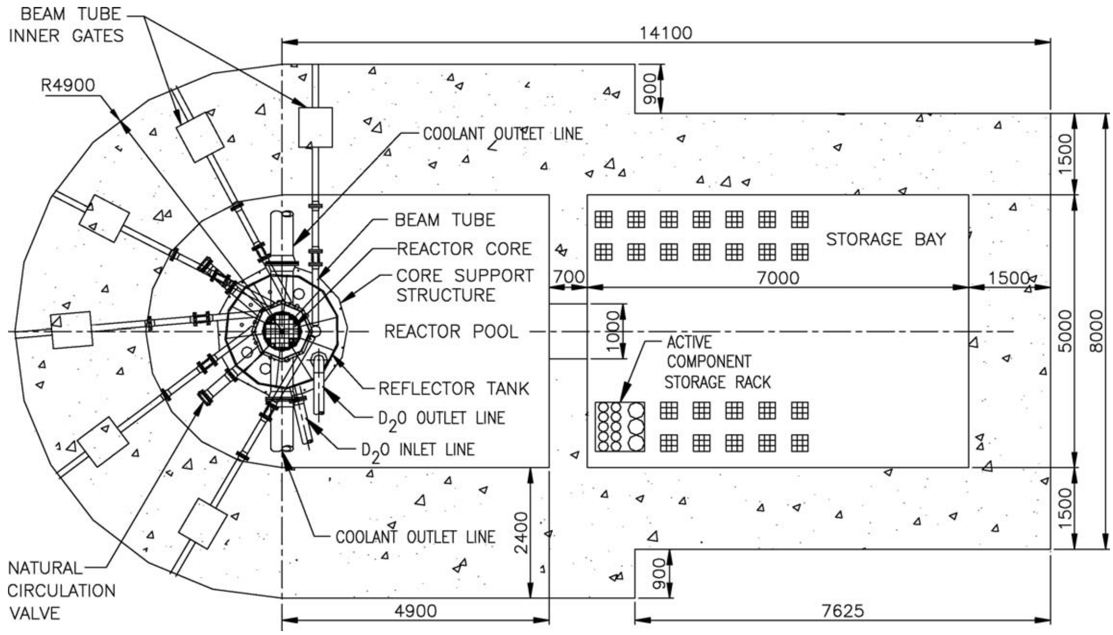

Fig. 7. Reactor and fuel storage block.

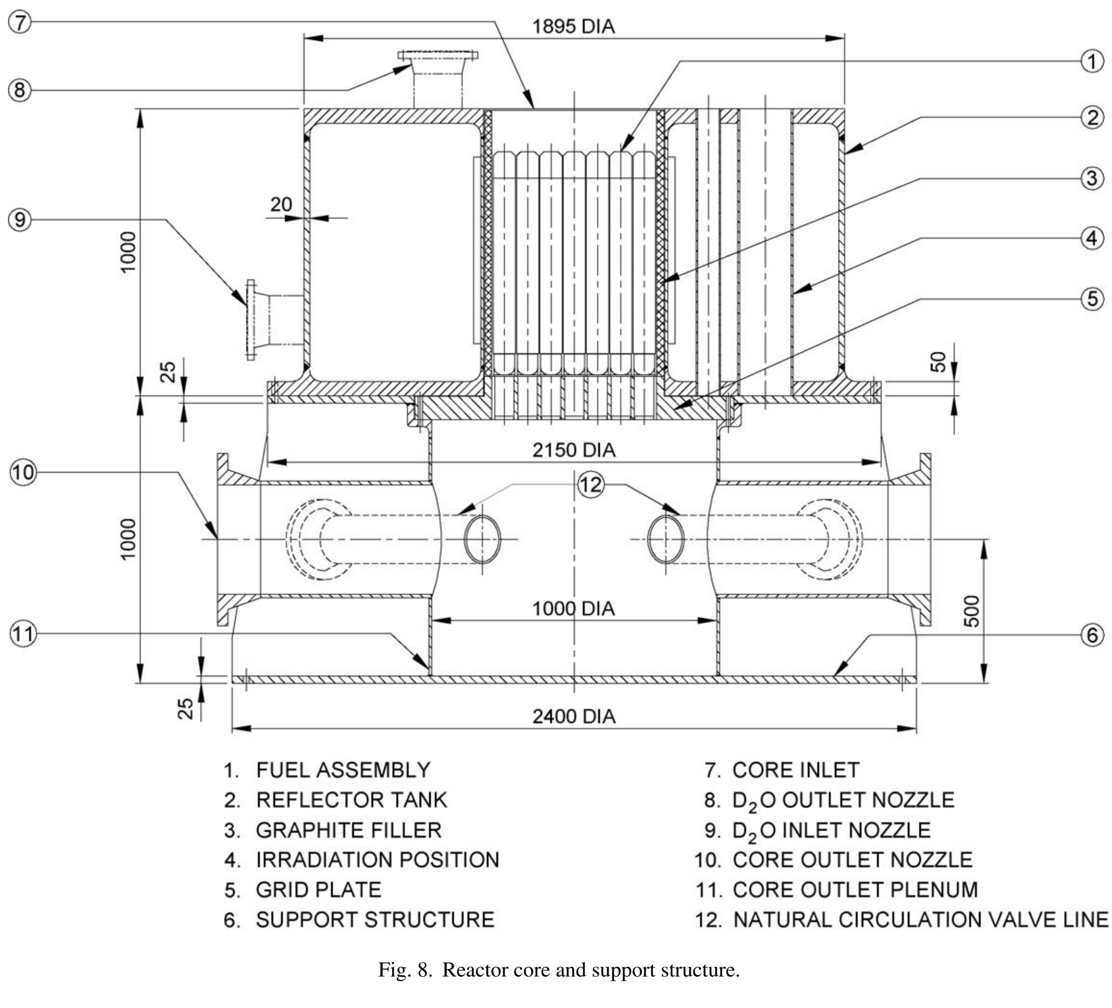

Fig. 8. Reactor core and support structure.

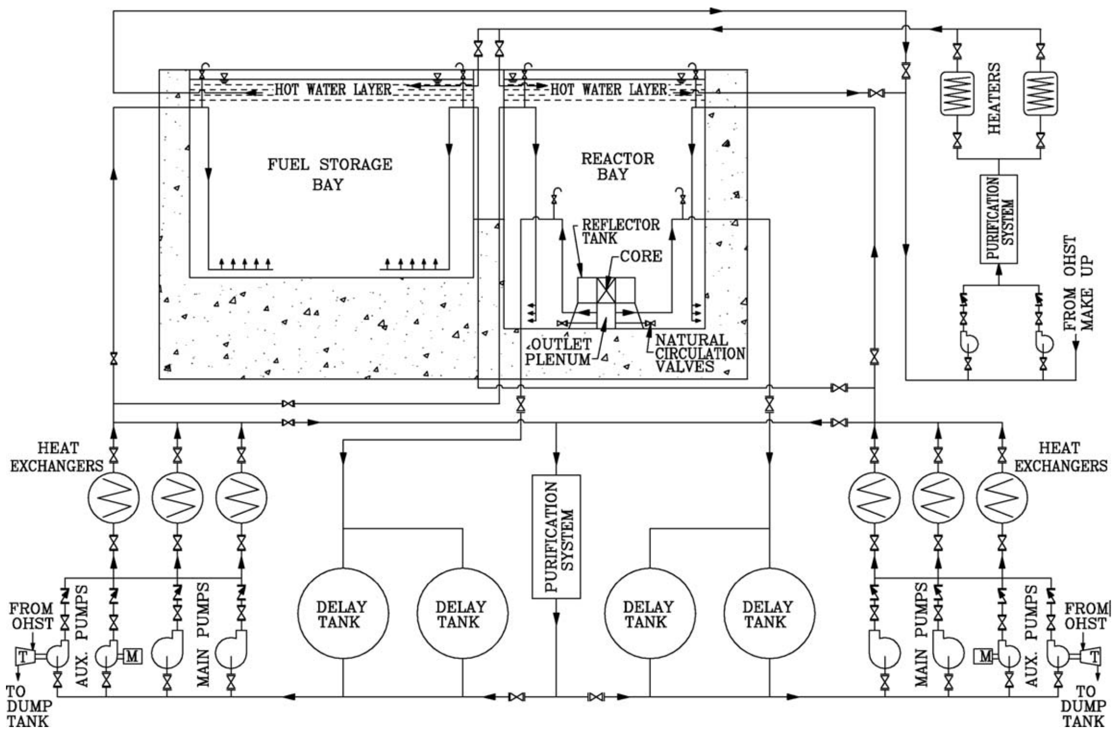

Fig. 9. Primary coolant system.

to the secondary coolant system through a set of heat exchangers. The secondary coolant system, in turn, rejects the heat to atmosphere through a set of cooling towers.

A simplified flow diagram of the primary coolant system is given in Fig. 9. The reactor core is cooled by demineralised water in the reactor pool flowing from top to bottom through the core. The water leaving the core enters the outlet plenum. The outlet plenum is bifurcated into two independent loops, each consisting of a set of delay tanks, recirculation pumps and heat exchangers. Hot water from core outlet after passing through the delay tanks is sent to the heat exchangers by the recirculation pumps where heat is transferred to the secondary coolant. Cold primary coolant water from heat exchanger outlet is fed back to reactor pool through a distributor. A part of the flow taken from the heat exchanger outlet is also sent to the fuel storage pool separately.

Forced coolant flow through the core is maintained from top to bottom with a view to reduce the radiation field and the airborne activity at the pool top. This will also maintain pool water activity and temperature at a reasonably low value.

For reducing the radiation field due to ^{16}N activity sufficiently and to permit access to the coolant system equipment during reactor operation, four delay tanks of 17 m^3 capacity each are provided at the core outlet. These stainless steel tanks are sized to provide a minimum delay of about 1 min. Various configurations were considered inside the delay tank to get uniform flow distribution and maximum time delay. Maximum delay was obtained through the provision of a combination of perforated and solid circular plate at the inlet and outlet of the tanks. The velocity distribution and stream function distribution plots are given in Fig. 10.

Each of the two loops has two main coolant pumps of 167 l/s capacity each. During normal operation all the four pumps will be operating. The pumps are provided with class IV power supply. Appropriately sized flywheel is provided on each main coolant pump to ensure adequate coast down cooling flow during pump trip transients.

As a measure of sufficient redundancy and diversity, two auxiliary pumps of 50 l/s capacity are provided in parallel to the main coolant pumps in each loop for core cooling during outage of main coolant pumps. One pump is provided with uninterrupted ac (class II) power supply and the other one is driven by a hydraulic turbine which runs by the gravity flow of water stored in the overhead storage tank (OHST). Though only one auxiliary pump is adequate to meet the shut down cooling requirements of the reactor, the system is designed to start two auxiliary pumps (one on motor prime mover and the other on turbine prime mover), on failure of main coolant pumps. Should the designated auxiliary pumps fail to start on demand; the standby pump will start automatically after a preset time delay. The overhead storage tank is designed to provide sufficient water inventory to cater to 2 h of auxiliary pump operation, though 1 h operation of the pump is adequate. Beyond 1 h the core cooling can be achieved through natural convection cooling by opening the natural circulation valves provided in the outlet plenum. Natural convection cooling is resorted to during prolonged shut down state of the reactor also. Two flapper type natural circulation valves are provided at the core outlet plenum. Opening of one valve is adequate to achieve natural convection cooling of the reactor.

Siphon break lines are provided on reactor pool and fuel storage pool inlet lines as well as core outlet lines to avoid complete

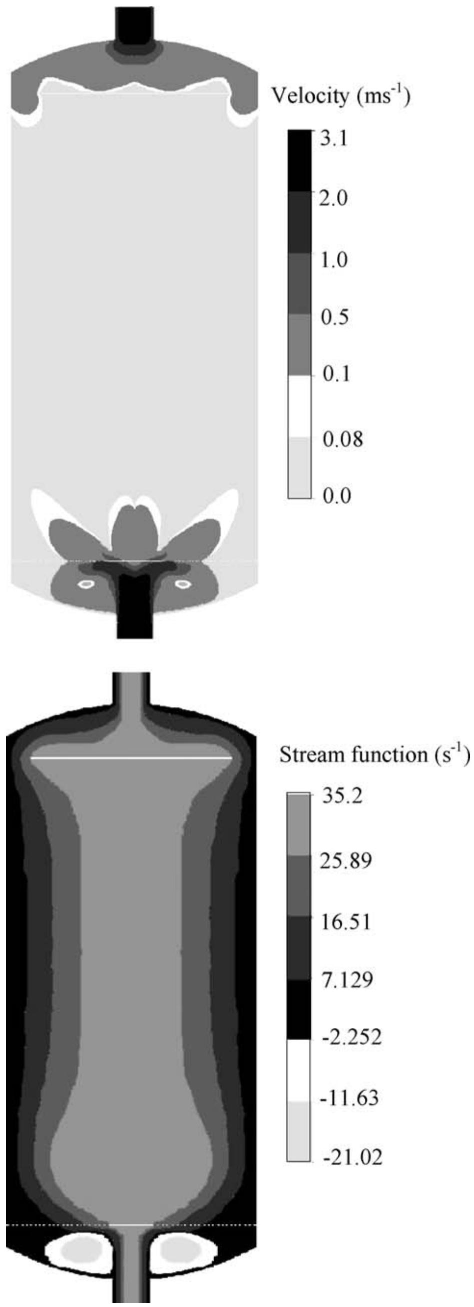

Fig. 10. Velocity and stream function distribution.

pool water drainage in the event of a pipe failure outside the pool.

In order to minimize the radiation level at the pool top due to various radionuclides present in the pool water, particularly ^{24}Na and ^{41}Ar, a hot water layer of low active water having a temperature of about 5 °C above the bulk pool water temperature will be maintained at the top region of the pool. Since the temperature of water at the pool surface is maintained higher, the tendency of active water to rise up to the pool surface is reduced significantly. In order to maintain the hot water layer at the pool surface, water is drawn continuously at a rate of 3.3 l/s from the pool top surface and passed through mixed bed ion exchangers to ensure that water is free from radionuclides. This water is heated by a set of heaters and fed back to the hot water layer at the top of the pool. The hot water layer is maintained during normal operation as well as shut down state of the reactor unless the reactor shut down is planned for a prolonged duration.

A part of the coolant flow from heat exchanger outlet from each loop is combined and sent through a set of filters and ion exchangers to maintain the pool water clarity, chemical quality and activity within specified limits.

During class IV outage or extended shut down state of the reactor secondary coolant system is not available for reactor cooling and the decay heat is dumped to the reactor pool. A pool cooling system is provided to remove the heat from the pool under this situation.

7. Reflector cooling system

Heavy water of high isotopic purity filled in an annular tank surrounds the core to act as the neutron reflector. Heavy water is recirculated continuously through the reflector tank to transport the heat generated in the reflector and its associated structures to the secondary coolant through a heat exchanger. Heat is ultimately discharged to atmosphere through a cooling tower.

The heavy water recirculation system consists of a set of three pumps (two operating), two heat exchangers (one in service), a delay tank and an expansion tank with associated piping and valves. The delay tank of 4 m^3 capacity is provided at the reflector tank outlet to provide about 1 min delay to reduce the radiation field due to ^{16}N activity on system piping and equipment in accessible areas. The expansion tank is riding on the system and helps to maintain heavy water system at the desired pressure absorbs the volumetric changes due to temperature variations and provides continuous inventory status of heavy water in the system.

Helium is used as cover gas for heavy water system. Cover gas is used to maintain the heavy water pressure in the reflector tank and heat exchangers at the desired value, to facilitate filling, draining and drying of the system/equipment in addition to avoiding the ingress of atmospheric air and moisture into the system.

The cover gas system consists of helium storage tank, expansion tank, recombination unit, a cooler and a blower for recirculation of helium. Helium from the expansion tank is drawn through a recombination unit and a cooler by means of blower and discharged back in to the expansion tank.

8. Control and instrumentation system

The control and instrumentation system of the MPRR comprises of neutron detectors and instrumentation channels for source range start-up and power range operation, dedicated microcomputer systems for power regulation and protection, drive mechanisms for control cum shut off rods and intelligent data acquisition and trend monitoring systems. All parameters of the control and instrumentation system are grouped into three major sub-systems, i.e., safety, controls and information, with each sub-system being an independent functional unit.

8.1. Safety system

The system is designed to process and monitor key process variables important to reactor safety and to automatically initiate safety actions in order to maintain reactor parameters within safe limits. The safety control and instrumentation systems initiate no actions during normal undisturbed operation, but take priority over all other control and instrumentation system actions when required.

This system comprises of all control and instrumentation systems connected to the reactor protection system and makes use of various trip parameters, generated from neutronic and process instrumentation signals to initiate protection action. The protection system has three channels, parameter-wise, with local coincidence logic and connected to two diverse chains for protection action. Each chain forms a bank and each bank is capable of independently tripping the reactor. All the trip signal outputs from the local coincidence logic, going to the two banks are segregated in a manner to ensure that each initiating event is covered by two independent parameters.

8.2. Control system

The control system is designed to ensure proper operation of the plant and to provide continuous information regarding the status of the plant variables. It also aids the operator in initiating manual action as and when required. Those functions needing immediate attention are brought to the central control room.

8.3. Reactor regulation and protection

The reactor regulating system (RRS) is configured with an independent dual-processor hot standby microcomputer system. The power regulation algorithm computes the difference between the digitised feedback signal from neutron sensors and the demand power, and the control output is generated, which is fed to a stepper motor drive for precision position control of the fine control rod. The generation of error signal is through a fixed control law, resident in the embedded microcomputer system.

Regulation and protection action is carried out with the help of six control cum shut off rods and one fine control rod. These control cum shut off rods have a higher worth and are used for coarse control from start-up to full power operation, as well as to effect a protection action, by gravity drop, in the event of a parameter exceeding its pre-set value. In addition to the six control cum shut off rods the fine control rod is used for fine adjustments and power trimming. In the event of a protection action, while the control cum shut off rods drop down by gravity, the fine control rod is driven down.

The drive mechanism for the control cum shut off rods consists of an electronic driver unit, a drive motor coupled through gearing arrangement and an electro-magnetic clutch coupled to the control cum shut off rods. On a trip signal, the electromagnetic clutch de-energizes, thereby allowing the control element to be inserted into the reactor core under gravity. The electronic driver unit provides the necessary control outputs for speed and position control.

8.4. Information system

The computer aided process information system provides a global information source. Intelligent information processing and comprehension, enables it to display process conditions and process sequences with high information content for safety related and operational tasks.

The output of the safety system and control system are isolated and connected via a common redundant plant bus, to the information system. The process variables to be monitored are recorded by means of sensors/transducers located in the field, having built-in features of signal conditioning, digitising and communicating with the bus, using a standard communication protocol.

8.5. Alarm annunciation

The system is designed to provide extensive alarm filtering, prioritising and interrogation capabilities to enhance operator recognition of events and plant state. The input to this system emanates from a dedicated plant display system, having an open architecture, which permits extensive display and annunciation enhancement. There is also an advanced annunciation feature, having the provision of special safety system impairment levels and potential operating policy violation alarms, with a display/report capability which details the resultant system unavailability or margin encroachment under the prevailing failure or configuration changes. This also provides a flexible navigation system for selective display and information retrieval as desired by operation and maintenance staff.

9. Confinement and reactor building ventilation system

Confinement system is provided for complete isolation of reactor building from outside atmosphere in the event of high radioactivity in the reactor building. Facility for controlled release of air from the reactor building to the atmosphere through a filtered venting system has also been provided towards lowering of the reactor building pressure to sub-atmospheric value in a controlled way. Air treatment plant (ATP) is provided for supplying fresh filtered and conditioned air to the reactor building.

The confinement system is a low leakage rate system designed to ensure that the off-site whole body and thyroid doses to the members of public even under postulated accident condition do not exceed the dose limits prescribed by International Commission for Radiation Protection and BARC Safety Council.

Once through ventilation system is chosen to avoid the possibility of recirculation of active air and to prevent the build-up of air-borne activity in any area of the building. Active areas, such as beam tubes and associated gate chases, shielded rooms housing equipment of coolant and reflector cooling systems are maintained at lower pressures compared to adjoining areas. The exhaust air from the reactor building is passed through a bank of HEPA filters before it is released to the atmosphere through a stack.

10. Electrical power supply system

Electrical power supply system has been designed to provide the required quality of electrical power to various reactor systems and auxiliaries through grid supply, diesel generator sets and un-interrupted power supply systems. The system provides emergency electric power supply to safety related systems by automatic power transfer, so that power supply to essential loads is restored within the permitted interruption time, in the event of loss of normal power or failure of any single active/passive component.

Adequate care has been taken to provide sufficient operational flexibility, information acquisition and diagnostic features for better reliability and easy maintainability. The electrical system layout ensures that the principle of independence and diversity are met and that interference from power circuits with control cables and control signals are minimized.

11. Safety analysis

Safety analysis of the reactor is in progress to ensure that the overall safety objectives have been met. The analysis is carried out to evaluate the capability built in the facility to accommodate and control disturbances or failures. This involves analysis of response of the reactor to various postulated initiating events (PIEs). Since it is neither practical nor necessary to include each event for a detailed analysis, only those events that produce boundary cases for the design are considered for detailed analysis. Various postulated initiating events have been categorised into anticipated operational occurrences and accident conditions on the basis of their expected frequency of occurrence and potential radiological consequences. Some of the postulated initiating events categorized into each group are given below:

Anticipated operational occurrences

(a) Loss of normal electrical power supply.
(b) Loss of heat sink.
(c) Inadvertent opening of one of the natural circulation valves.
(d) Unbalanced shim control rod positions.
(e) Fuel element clad failure.

Accident conditions

(a) Start-up accident.
(b) Primary coolant pump seizure.
(c) Primary coolant system pipe failure.
(d) Ingress of light water into heavy water reflector.
(e) Ingress of heavy water into primary coolant system.
(f) Fuel channel blockage.
(g) Failure of experimental setup.

The results of the analyses are compared with the design limits and radiological acceptance criteria to ensure their acceptability. Safety analyses will form the basis for the design of systems/items important to safety and for the selection of operating limits and conditions of the reactor.

The event sequence and results of the analysis of a few typical postulated initiating events which result in reduction of core flow, with and without reactor trip, are discussed below.

11.1. Loss of normal electrical power supply

11.1.1. Event sequence

Loss of normal electrical power supply will result in failure of main coolant pumps and secondary coolant pumps. Failure of main coolant pumps will result in continuous reduction of coolant flow through the core depending on the coast down characteristics of primary coolant system and its pump motor assembly. The reactor will trip due to loss of normal power supply. This trip is backed up by reactor trip on low primary coolant flow (initiated at 85% of normal integrated flow of the two loops). On loss of normal electric supply, auxiliary coolant pumps will get signal to start. Normally, one motor driven auxiliary coolant pump from one loop and one turbine driven auxiliary coolant pump from the other loop will start operating. During the flow coast down, as the main coolant pump discharge pressure falls below the pressure head developed by the auxiliary coolant pump, the down stream check valve of the auxiliary coolant pump will open to deliver flow through the system. After 1 h from the reactor trip, when the core decay heat reduces sufficiently, the auxiliary coolant pumps can be stopped and natural convection core cooling is established by opening the natural circulation valves at core outlet.

11.1.2. Analysis and results

An analysis has been carried out using the computer code RELAP5/MOD 3.2 (Fletcher et al., 1995) to ensure that the fuel and clad temperatures are well within the acceptable limits during the above transient. In the analysis first reactor trip on power failure is ignored. Second reactor trip initiated by primary coolant low flow is about 3 s after power failure is considered for the analysis. Allowing a delay of 300 ms for shim rod actuation and another 500 ms for shim rod drop, the reactor is assumed to be tripping 3.8 s after the loss of power supply.

The analysis indicates that the maximum clad temperature of hottest standard fuel assembly and control fuel assembly are 112.5 and 109 °C, respectively. These values are well below the onset of nucleate boiling (ONB) temperatures. The variation of flow, power and temperatures of the hottest standard fuel assembly is shown in Fig. 11. The minimum departure from nucleate boiling ratio (DNBR) values for hottest standard and control fuel assemblies are 2.42 and 2.46, respectively. After 1 h from reactor trip when the auxiliary coolant pumps are stopped, flow reversal takes place. During the flow reversal, clad temperatures for the hottest standard fuel assembly and control fuel assembly increases from 45.2 to 72.5 °C and 44.7 to 71.8 °C. The variation of flow, power and temperatures of the hottest standard fuel assembly is shown in Fig. 12.

11.2. Inadvertent opening of a natural circulation valve

11.2.1. Event sequence

Inadvertent opening of one of the natural circulation valves creates a bypass flow path parallel to the core causing reduction in core flow during normal operation and consequent increase in fuel and clad temperatures. Since gross system flow does not

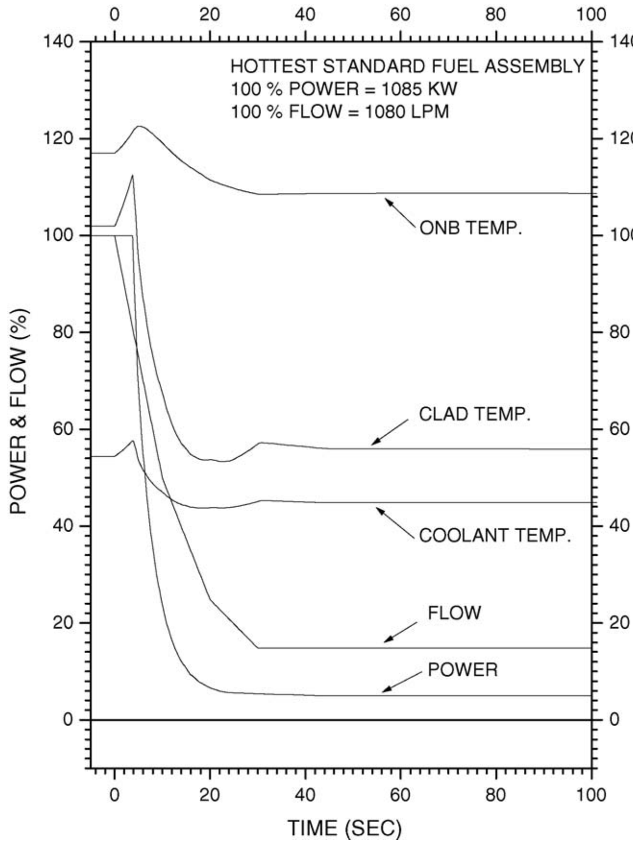

Fig. 11. Variation of power, flow and temperatures on loss of normal power supply.

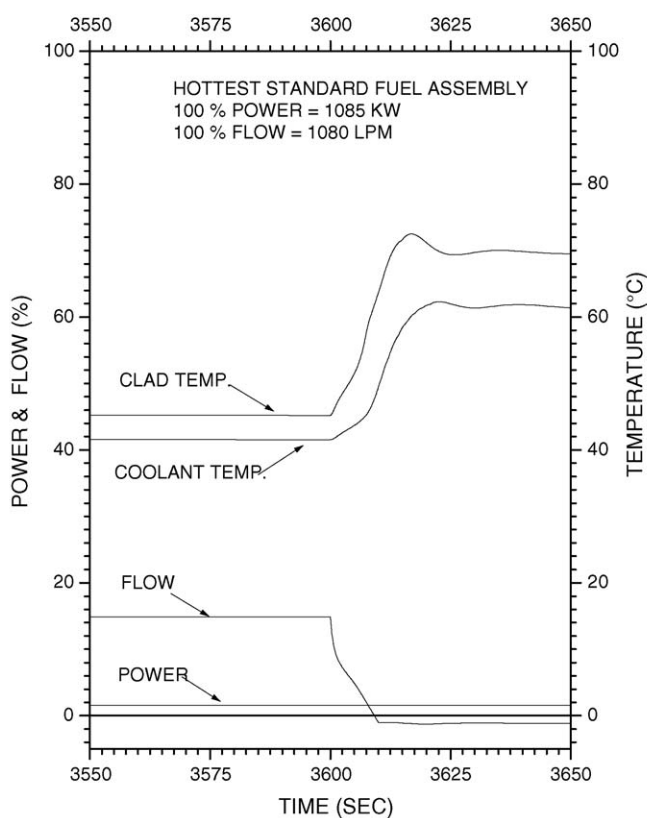

Fig. 12. Variation of power, flow and temperatures during the transition from forced flow to natural circulation flow under reactor shut down condition.

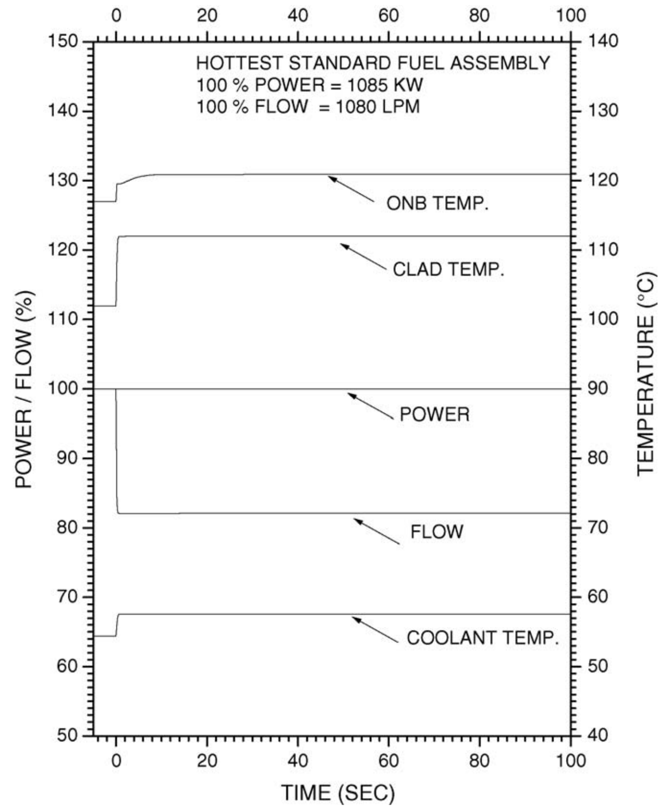

Fig. 13. Change in flow and temperatures during inadvertent opening of one natural circulation valve.

reduce, reactor trip on “low recirculation flow” will not register in this case. The reactor will trip on opening of the closed limit switch of the natural circulation valve. An analysis was, however, carried out to study the consequences in case the reactor fails to trip.

11.2.2. Analysis and results

The analysis has been carried out using the computer code RELAP5/MOD 3.2. Analysis indicates that the flow through the fuel assembly reduces to 82% of normal value. The clad temperature of the hottest standard fuel assembly and control fuel assembly increases from 102 to 112 °C and 98.8 to 108.5 °C, respectively, with reactor operating at full power. No boiling is observed in the core. The variation of flow, power and temperatures of the hottest standard fuel assembly is shown in Fig. 13. The minimum DNBR value for the hottest standard and control fuel assemblies are 2.37 and 2.41, respectively. No adverse consequences are observed.

11.3. PSA for MPRR

Work on development of full scale Level 1 probabilistic safety assessment (PSA) model is underway. This study will enable assessment of safety aspect of the reactor in an integrated manner. The scope of this study is limited to full power operation considering reactor core only as the source of hazard. The objective of this study includes:

(a) Determination of 'core damage frequency'.
(b) System analysis to identify design weaknesses.
(c) Analysis of human action important to safety.
(d) Analysis of common cause phenomenon.

The list of initiating events from the Design Safety Analysis Report forms the basic source of input. Apart from this master logic diagram and the review of the initiating events given in the literature on plants having similar design will form an added input. Since this probabilistic safety assessment study is being done at the design stage of the plant, the reliability data from the generic source will form the main input for this study. However, for crucial components/systems like control and shut off rods, shut down cooling pumps, natural circulation valves, emergency power supply system and primary coolant system piping, etc., data from plant specific source will be used in this analysis.

12. Concluding remarks

The multi purpose research reactor with its high neutron flux and large irradiation volume will provide a versatile facility to meet the future requirements for basic and applied research, material irradiation and production of radioisotopes for application in the fields of medicine, agriculture and industry.

References

Askew, J.R., Fayers, F.J., Kemshell, P.B., 1966. WIMS–D/4, A general description of the lattice code. J. Br. Nucl. Energy Soc. 5, 564–585.

Fletcher C.D., Schultz R.R., et al., 1995. RELAP5/MOD3 Code Manual, Idaho National Engineering Laboratory, NUREG/CR—5535, INEL—95/0174.

Leszczynski, F., Aldama, D.L., Trkov, A., 2002. WIMS-D Library Update-Final report of a Co-ordinated Research Project, IAEA-DOC-DRAFT.

Hallsall, M.J., 1980. Summary of WIMS-D/4 Input Options, AEEW-M.1327.

Lathrop K.D., Brinkly F.W., 1970. Theory and use of the general geometry TWOTRAN Program. LA—4432.

Hegde, P.V., Prasad, G.J., Ganguly, C., 1993. Powder metallurgy route for fabrication of Al-UAl_3, Al-U_3O_8, Al-U_3Si_2 dispersion fuels for Nuclear Research Reactors. Trans. PMAI 20, 27–31.

Singh, K., Sengupta, S.N., 1999. A finite Fourier transform method for three dimensional steady state reactor core calculations. Ann. Nucl. Energy 26, 533–541.