IMPACT OF LMFBR OPERATING EXPERIENCE ON PFBR DESIGN

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Abstract

PFBR is a 500 MWe, sodium cooled, pool type, fast breeder reactor currently under detailed design. It is essential to reduce the capital cost of PFBR in order to make it competitive with thermal reactors. Operating experience of LMFBRs provides a vital input towards simplification of the design, improving its reliability, enhancing safety and achieving overall cost reduction. This paper includes a summary of LMFBR operating experience and details the design features of PFBR as influenced by operating experience of LMFBRs.

1. INTRODUCTION

India has limited uranium and abundant thorium resources. For better utilisation of uranium and to use the available thorium, Fast Breeder Reactor programme is very essential. A 40 MWt / 13 MWe Fast Breeder Test Reactor (FBTR) is in operation at Kalpakkam, since 1985, and has attained its rated power level of 10.5 MWt in Dec 1993 with small Mark I core. As a logical follow-up to FBTR, a 500 MWe Prototype Fast Breeder Reactor (PFBR) is currently under design & development at Indira Gandhi Centre for Atomic Research (IGCAR). Fast reactors like BN-600, SPX-1 and Monju have capital cost significantly higher than PWRs. FBRs, for commercial deployment, after reaching maturity through large scale construction, would be required to have matching unit energy cost with PWRs. Even though incidents of minor to severe nature have occurred in LMFBRs, their operating experience provides a very important input for design simplification, improving reliability, enhancing safety and achieving cost reduction. This paper describes in brief the summary of LMFBR operating experience and details the design features of PFBR as influenced by LMFBR operating experience.

2. SUMMARY OF LMFBR OPERATING EXPERIENCE

- Considerable experience (~ 200 reactor-years) has been gained in the design and operation of sodium systems. Corrosion of structural materials in reactor grade sodium is negligible and purification of sodium for oxygen control by cold trapping is very satisfactory. It has been possible to maintain the purity of sodium in a stable manner.
- Performance of stainless steel of type 304 and 316 for sodium components is excellent. Failure of welds have occurred in the stabilised grade 321 in PFR [1] & Phenix [2] and 15 Mo3 ferritic steel in SNR-300 & SPX-1 [2]. Performance of elevated temperature components on the whole is satisfactory indicating that failure mode of creep-fatigue can be well taken care of in design.

- Very high burnup (192 GWd/t) has been achieved for mixed oxide fuel compared to the initial target value of about 62 GWd/t [3] and there has been very few fuel pin failures. This gives scope for significant decrease in fuel cycle cost. The smaller number of fuel pin failures has led to very clean sodium circuits which has also contributed to low radiological impact.
- The performance of sodium pumps has been very good. The only incident of concern has been oil leaks with a major incident occurring in PFR [4]. Other problems like excessive vibrations, seizure, malfunctioning of speed control etc have been understood.
- IHX performance except for Phenix is excellent. Failure of Phenix secondary sodium outlet header is well understood [5].
- Successful operation of steam generator (SG) holds the key to achievement of high capacity factors. SG requires the high quality during manufacture and a sensitive leak detection system for sodium-water reaction detection and mitigation. PFR incident of failure of 40 tubes of superheater has led to redefinition of design basis leak for SG from the earlier considered incident of double ended rupture of one tube [6].
- Fuel handling incidents have led to interventions & outages in some reactors including FBTR where bypassing of interlocks was done. These incidents call for under sodium scanning before every fuel handling.
- Radiation dose to operating personnel and radioactivity releases to the environment are significantly less compared to PWRs [7]. It is thus possible to define a low target person-Sivert for LMFBR and to reduce the shielding in controlled access areas to enable cost reduction.
- In-service inspection (ISI) is an important means to assess structural integrity and needs attention in design in particular, for main vessel & SG.
- Reactivity incidents have occurred in some reactors and in spite of the best efforts, fully validated explanation has not been possible. It is worthwhile to have a design with less potential for such incidents.
- Fuel meltdown incident has occurred in Fermi due to blockage of subassembly at inlet and at EBR I due to inward bowing of core subassemblies.
- Incidents of sodium leak show need for greater care in auxiliary sodium circuits to minimise failure by thermal striping. There is a need to develop sodium resistant concrete to minimise damages to structures in case of sodium leaks.
- High capacity factors are achievable in LMFBR with sound designs as experienced in EBR II, BOR-60, Phenix (initial years) and BN-600.
- The capital cost of FBR is about 1.5 to 2.5 times that of thermal reactors and significant cost reduction is essential for its successful deployment. Cost reduction measures adopted in LMFBR include elimination of ex-vessel sodium storage, decrease in number & size of components of heat transport system, compact layouts, increasing operating temperature, increasing plant life and increasing fuel burnup.

3. FBTR OPERATING EXPERIENCE

Details of operating experience and incidents that have occurred in FBTR are covered in a companion paper.

4. PFBR DESIGN FEATURES

4.1 Main options

4.1.1 Reactor Power

The successful operation of 500 MWe thermal power plants in India has enabled to fix PFBR reactor power as 500 MWe. Large sized FBR have not indicated any technological problems because of reactor size. Specific capital cost is lower for 500 MWe than for a lower power, say 250 MWe. The design and development efforts needed for 500 MWe and 250 MWe plants are comparable. Pressurised Heavy Water Reactors (PHWR) of 500 MWe are under construction in India. Constructability of 500 MWe PFBR components has been assessed and adequate industrial capability exists within the country.

4.1.2 Fuel

A proven fuel cycle is very essential for PFBR. Though mixed carbide fuel has been used for FBTR due to non availability of enriched uranium, risks associated with carbide fuel fabrication, higher cost coupled with limited burnup potential & limited experience on reprocessing of the fuel have led to adoption of mixed oxide (MOX) fuel. This fuel has shown excellent performance with respect to burnup, has well proven reprocessing technology and has also been used in most of the large sized FBR.

4.1.3 Loop vs Pool concept

Better safety features of the pool concept due to the high thermal inertia of the large mass of sodium in the pool, containment of all radioactivity in a single vessel with no nozzles leading to high integrity of the primary circuit, reliable decay heat removal by independent dedicated sodium loops and satisfactory performance of pool type power reactors abroad have led to adoption of pool type concept for PFBR. The shortcomings of the pool concept as regard to large size of components of reactor assembly, complex thermo-hydraulics of hot and cold pool, interdependence of primary circuit component's construction and maintenance are well recognized and have been looked into.

4.1.4 Operating Temperatures

In order to reduce the unit energy cost, it is essential to adopt a superheated steam cycle. Operating experience of elevated temperature components in FBR indicates that creep-fatigue damage can be well taken care of in the design. Thermo-hydraulic analysis needs to be detailed to have complete knowledge of thermal loading.

Plant temperatures have been arrived at based on structural analysis of hot leg components, in particular Control Plug, limiting clad hot spot temperature to 973 K (700 $^{\circ}$ C), steam generator material as T91 and optimisation studies on heat exchangers (IHX & SG) costs and sodium pumping cost. The temperatures of 820 K (547° C) at hot pool, 670 K (397° C) at cold pool, 628 K (355° C) at IHX inlet, 798 K (525° C) at IHX outlet and steam conditions of 16.7 MPa / 763 K (490° C) at turbine inlet have been chosen. The improved cycle efficiency with higher temperature difference across HXs result in reduction in unit energy cost.

4.1.5 Structural Materials

4.1.5.1 Clad and Wrapper

20 % CW D9 material which has shown excellent performance with oxide fuel has been selected for clad tubes and wrapper. Irradiation of indigenously produced material is planned in FBTR. Wrapper in Cr-Mo grade is also envisaged for future cores of PFBR.

4.1.5.2 Material for Hot leg and Cold leg components in sodium circuits

SS 321 has been rejected due to unsatisfactory performance in FBRs and thermal power stations. SS 347 is expected to behave similar to SS 321 at elevated temperatures. SS 316 LN, which has good high temperature characteristics and provides freedom from sensitisation in as welded state - an important aspect to avoid risk of IGSCC, in the coastal site selected, has been chosen for hot leg components such as inner vessel, control plug, IHX and hot leg of secondary sodium piping. For the cold leg components and secondary sodium piping, SS 304 LN material is found to be adequate. However, use of SS 316 LN for cold leg components and piping would be given consideration where risk of mixup of materials exists. Choice of a single grade also reduces the R&D efforts required.

4.1.5.3 Material for Steam generators

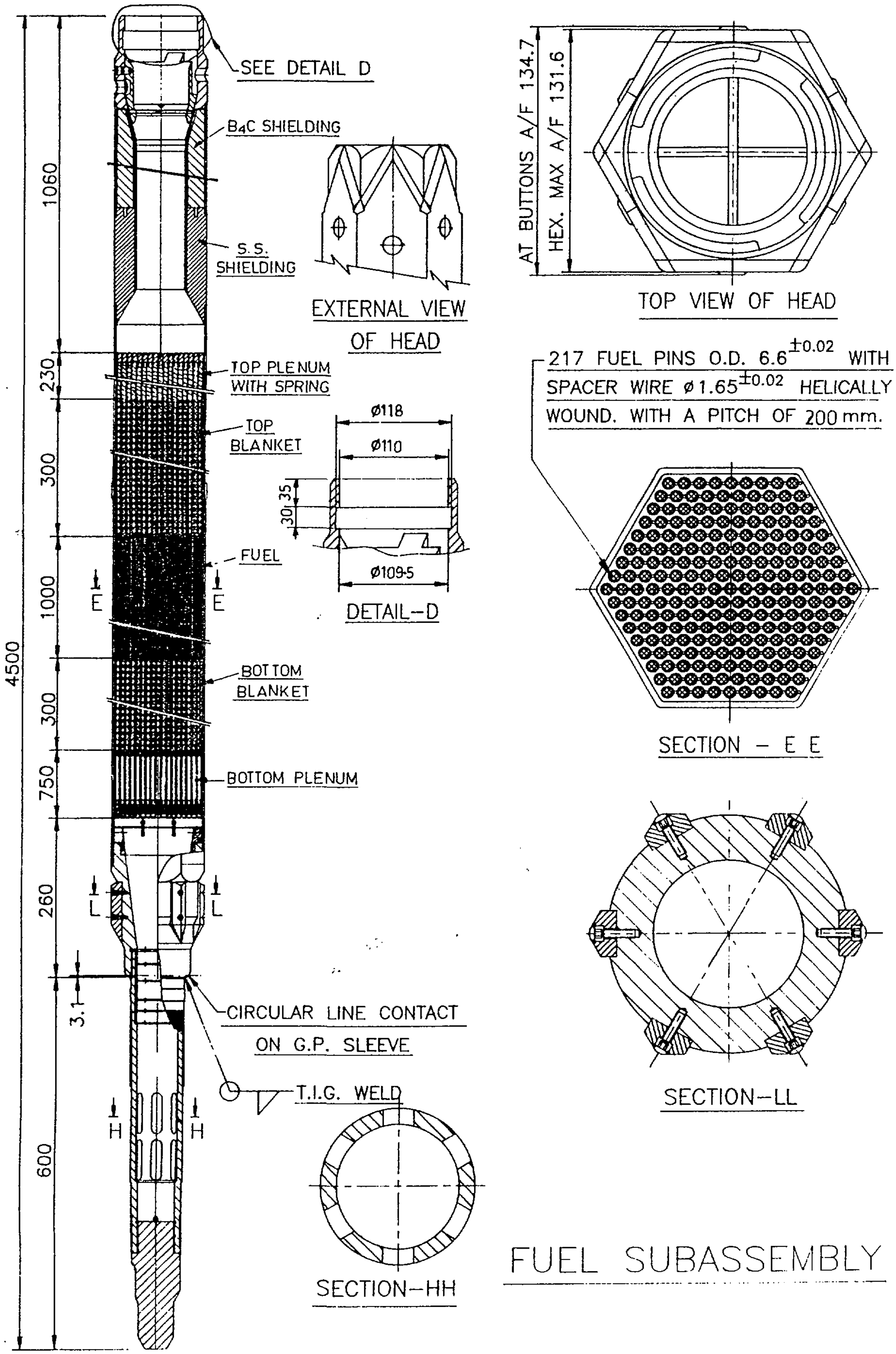
Modified 9 Cr-1Mo (T91) has been chosen for steam generators because of its satisfactory strength at high temperature, freedom from stress corrosion cracking (problem with stainless steels both for chloride and caustic environment) and risk of decarburisaiton (problem with 2.25 Cr-1Mo).

4.1.6 Number of Turbogenerator(TG) sets

Operating experience from nearly 15 TG sets of 500 MWe capacity, currently in operation in India, is excellent. A single TG set has been selected instead of two from considerations of reduction in capital cost and improved capacity factor arising from reduction in outages due to maintenance.

4.2 Core Design

The active core consists of 181 fuel subassemblies with two enrichment zones, of which 85 with ~ 21% PuO_2 content are in the inner enrichment zone and 96 with ~ 28% PuO_2 content are in the outer enrichment zone. Each fuel subassembly consists of 217 helium bonded pins of 6.6 mm outside diameter. Each pin has 1000 mm column of MOX, 300 mm each of upper and lower depleted UO_2 blanket columns and lower fission gas plenum (fig. 1).



SEE DETAIL D
B4C SHIELDING
S.S. SHIELDING
TOP PLENUM WITH SPRING
TOP BLANKET
FUEL
BOTTOM BLANKET
BOTTOM PLENUM
CIRCULAR LINE CONTACT ON G.P. SLEEVE
T.I.G. WELD
1060
230
300
1000
300
750
260
3.1
600
4500
E
L
H
EXTERNAL VIEW OF HEAD
TOP VIEW OF HEAD
AT BUTTONS A/F 134.7
HEX. MAX A/F 131.6
217 FUEL PINS O.D. 6.6±0.02 WITH SPACER WIRE ø1.65±0.02 HELICALLY WOUND. WITH A PITCH OF 200 mm.
ø118
ø110
ø109.5
35
30
DETAIL-D
SECTION - E E
SECTION-LL
SECTION-HH
FUEL SUBASSEMBLY

Fig—1

The fuel pellets are of annular type which enables faster rise to full power. Absence of fuel redistribution & restructuring because of high density helps in attaining high burnup.

There are 3 rows of radial blanket subassemblies and 12 absorber rods arranged in two rings with 9 constituting the Control & safety rods(CSR) and 3 constituting the Diverse safety rods (DSR). Boron carbide with 63% / 50% enriched B_{10} for CSR/DSR respectively is chosen as the absorber material. The control & safety rods are of vented type and this type of design has performed well in FBTR and in other reactors.

Total blockage of SA due to external debris is a low probable event and is taken care by the arrangement of radial inlet of coolant and multiple holes inlet in grid plate sleeves / multiple slots inlet in the SA foot. Total blockage of fuel and blanket SA at outlet is ruled out by provision of an adaptor, which ensures an alternate path for coolant flow. This consists of a annular cylindrical piece with slots provided for sodium flow and this is screwed to the inside of the top portion of the SA. Two sets of 3 nos. of holes are provided on the SA hexcan outside the adaptor. In case of a total blockage of the flow path at the top of the SA, these holes along with the slots in the adaptor provides alternate flow path for sodium. The design objective is to avoid sodium boiling. During normal operation, a small flow of about 0.2 % of the flow through a SA leaks through the above holes.

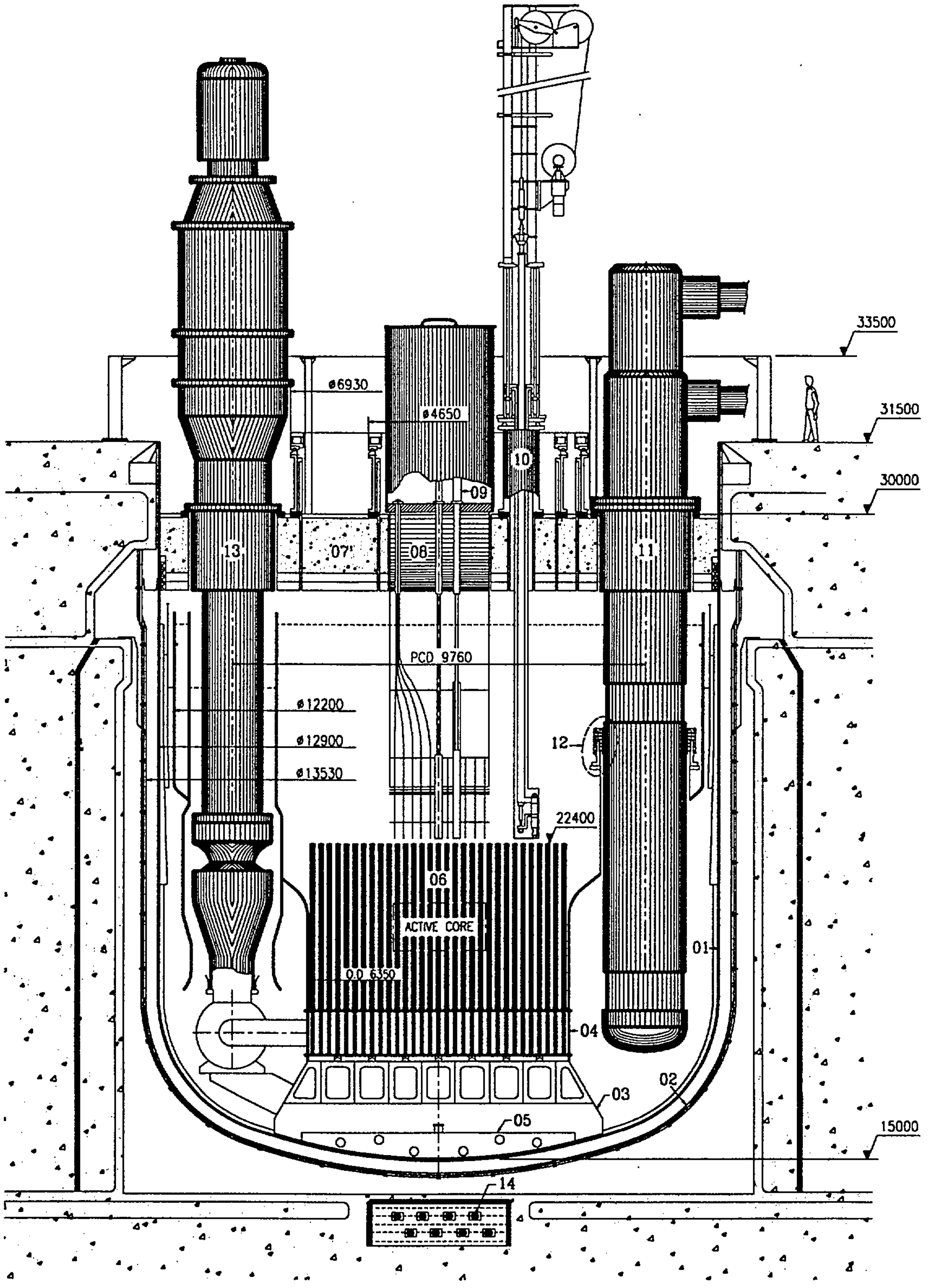
Simple naturally restrained core concept has been adopted which gives negative power coefficient and which also avoids inward radial movement of core subassemblies due to bowing. Use of a separate core barrel has been avoided.

4.3 Reactor Assembly

The reactor assembly consists of main vessel, safety vessel, core support structure, grid plate, inner vessel, roof slab, rotatable plugs and control plug (fig. 2). The main vessel (diameter 12.9 m) contains the entire primary sodium circuit including the 1100 t of primary sodium. The main vessel is cooled by cold sodium to enhance its structural reliability. The main vessel cooling arrangement has been checked for flow induced vibration behavior. The safety vessel follows the shape of the main vessel with a 300 mm nominal gap. The inner vessel separates the hot and cold pools of sodium. Argon gas seal for IHX-Inner vessel sealing has been avoided to minimise the chances of reactivity addition and a mechanical seal design with piston rings has been selected (fig. 3). The seal assembly uses two piston rings and is provided as an integral part of the IHX. It has a face to face contact with a flange integral with IHX standpipe in inner vessel. Compression springs are used to apply the required force in order to minimise leakage of sodium between the flange faces. Hydraulic experiments are planned to verify the quantity of sodium leaking across the seal assembly.

A single grid plate is used to support the core and shielding subassemblies and a fully bolted construction has been adopted. The grid plate has four inlet pipes with a pair of nozzles connected to each of the two primary pumps.

The top shield includes roof slab and two rotatable plugs. Warm roof concept is adopted for top shield to minimise sodium deposition in the annular gaps. The roof slab is a box type structure filled with concrete as the shielding material. It supports the main vessel, primary sodium pumps, IHX, and direct reactor heat exchangers (DHX) of the decay heat removal system. Use of liquid metal seals has been avoided in order to reduce the rotatable support



LEGEND

01. MAIN VESSEL
02. SAFETY VESSEL
03. CORE SUPPORT STRUCTURE
04. GRID PLATE
05. CORE CATCHER
06. CORE
07. TOP SHIELD
08. CONTROL PLUG
09. CONTROL & SAFETY ROD DRIVE MECHANISM
10. IN-VESSEL TRANSFER MACHINE
11. INTERMEDIATE HEAT EXCHANGER
12. IHX MECHANICAL SEAL
13. PRIMARY PUMP & DRIVE
14. NEUTRON DETECTORS

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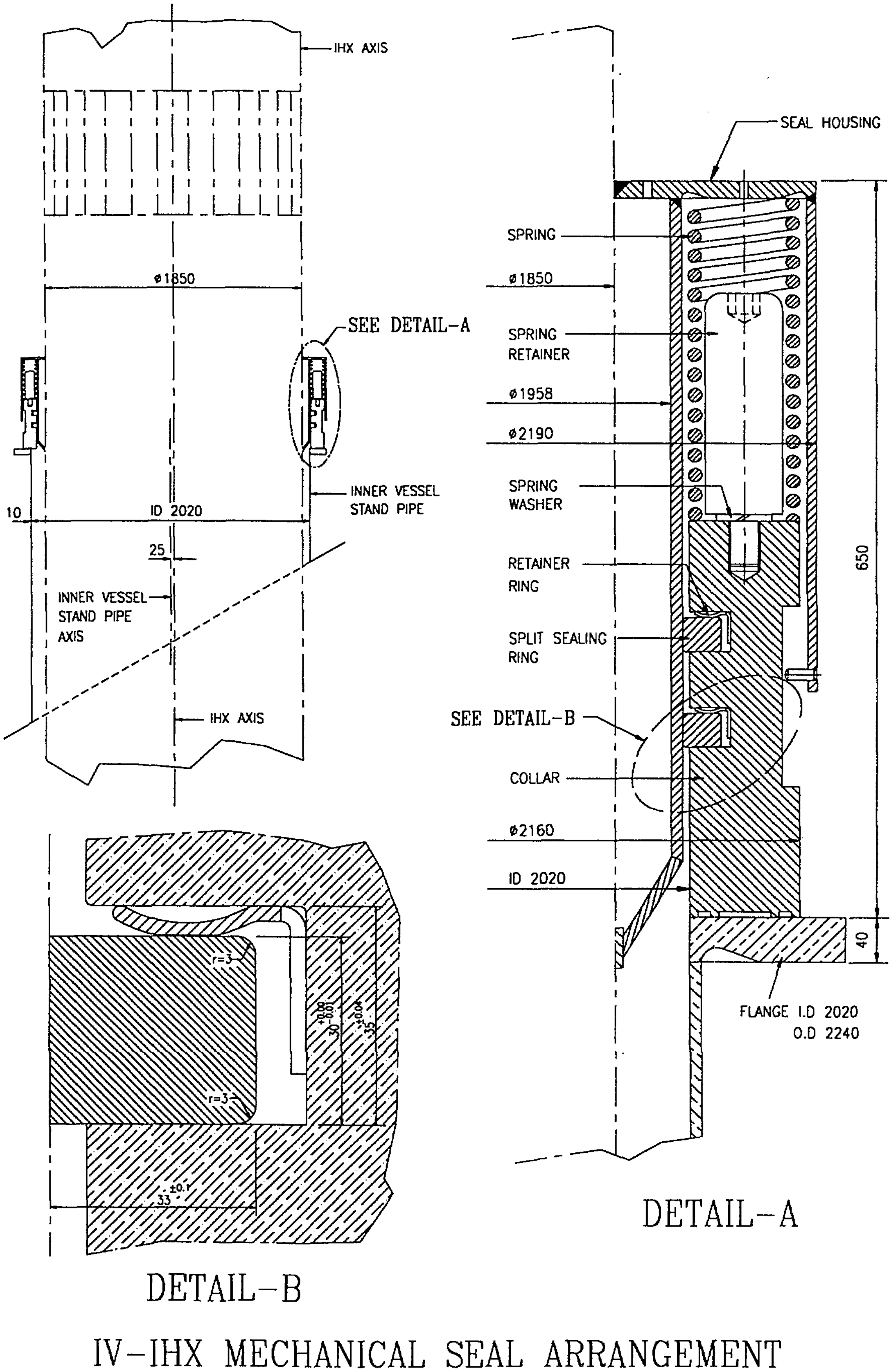
PFBR REACTOR ASSEMBLY

FIG.2



IV-IHX MECHANICAL SEAL ARRANGEMENT

FIG-3

arrangement width (and hence main vessel diameter) and elastomer seals are used to seal the argon cover gas.

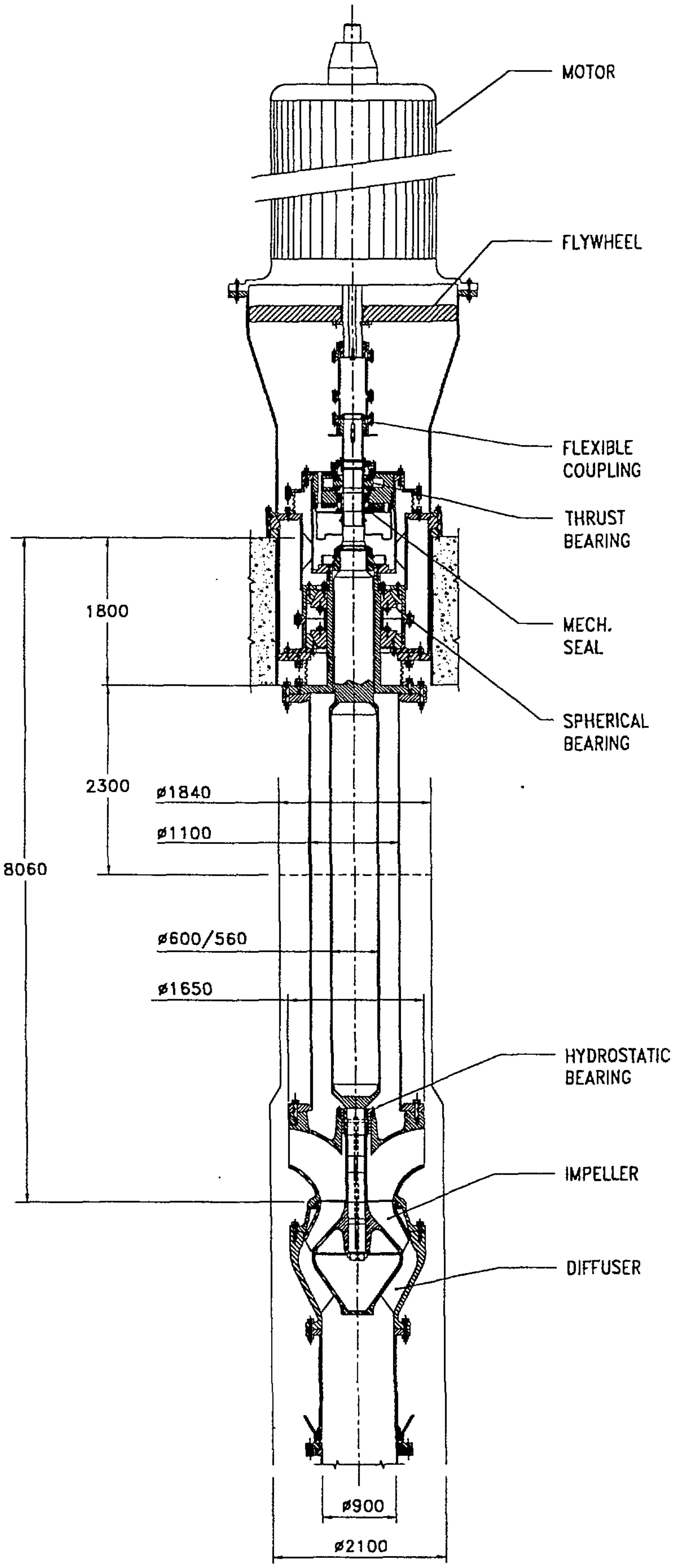
The Control plug supports the 12 absorber drive mechanisms, sleeves which house thermocouples for measurement of outlet temperature of each fuel subassembly and three selector valves with sodium sampling from each fuel SA for failed fuel location. Use of bellows has been avoided for CSRDM to extend the life of the mechanisms and to enhance reactor availability, as bellows failure has been responsible for replacement of CRDMs in reactors using this concept. V-ring seals are used between the stationary sheath and mobile assembly of CSRDM. The core thermocouples are located at a fixed distance of 90 mm from the top of the SA during reactor operation and no Core Cover Plate Mechanism (CCPM) is provided as in FBTR. Thermohydraulic analysis indicates that the thermocouples are immersed in their respective streams at all power levels thereby ensuring adequacy of temperature measurement.

Though the Total Instantaneous Blockage (TIB) of a single SA is categorised as Beyond Design Basis Event (BDBE), an internal core catcher is provided below the core support structure. This is designed for retention of core debris arising out of meltdown of 7 SA based on the SCARABEE tests which have indicated melt propogation at the most to the neighboring six SA.

4.4 Sodium circuits & Components

Detailed optimisation studies on number of loops/components led to the choice of 2-loop concept. Due to adoption of design improvements, the increase in size of the components when the number of loops is decreased is not large and is within the industrial capacity. The reduction in the number of components helps to reduce the capital cost, construction time and the outage time due to generic design failure/inspection/repair of components. Hence the capacity factor of the reactor is expected to be marginally higher for the case with lesser number of loops. Reduction in number of loops also reduces the space required for layout of secondary sodium system components. Hence, 2 loop arrangement has been chosen with two primary pumps and 2 secondary pumps. 2 IHX per loop has been selected based on the economics and is in line with other pool type reactors built so far. The number of SG/loop is based on optimisation analysis of capital cost and outage cost in case of a leak, with due consideration to construction schedule while permitting (N-1) SG modular operation and 4 SG/loop has been chosen.

The Primary pump is a top suction, single stage, centrifugal pump without non-return valve (NRV)(fig. 4), powered by a 3600 kW motor with speed variation of 20-100% of nominal speed. It delivers a flow of 4.13 m^3/s at a head of 75 mlc at an operating speed of 680 rpm. A squirrel cage induction motor fed from current source inverter is selected. A pony motor is provided to run the pump at 20% speed. It is not envisaged to operate the reactor with only one pump in operation. Further, analysis indicates that the flow through the core is adequate in case of one pipe rupture (category 4 event) even without NRV. Hence, NRV is eliminated giving the advantage of increased submergence (for a given main vessel height) thereby permitting higher pump operating speed. Elimination of NRV also increases reliability of the path for decay heat removal. A margin of 1.24 is specified on NPSH ($NPSH_A / NPSH_{3\%}$), which ensures absence of cavitation erosion and gives a pump life equal



PRIMARY SODIUM PUMP

FIG. 4

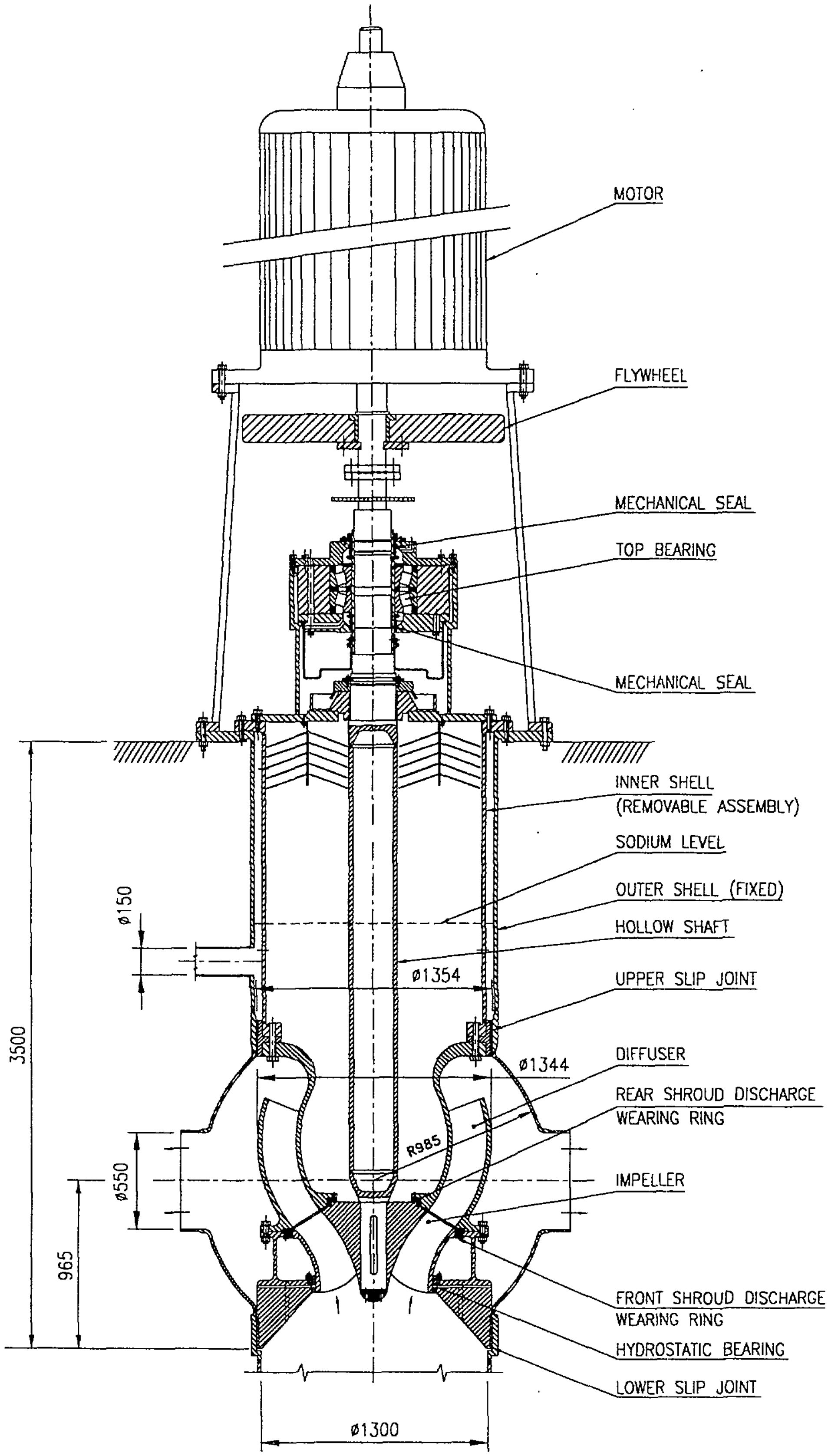
to that of the reactor. The hydrostatic bearing journal is keyed to the shaft and is also provided with a spacer preventing danger of its slippage due to thermal shocks. It has not been possible to avoid use of oil for lubrication of seals and bearings. Hence, efforts are made to avoid entry of oil into sodium. Any possibility of oil leak into the primary circuit is avoided by appropriate surveillance methods as well as by provision of an oil catch pot of sufficient size to accommodate the entire oil capacity. The pump is supported on a spherical seat arrangement to accommodate the differential thermal expansion. Full-scale hydraulic testing of the prototype pump is being done.

The secondary pump is of centrifugal type, mixed flow design delivering a flow of 3.34 m^3/s at a head of 65 mlc at an operating speed of 960 rpm (fig. 5) and is located in the cold leg at a lower elevation with respect to SG. Locating the pump in the cold leg of the secondary circuit is more economical with lower piping costs. The normal cover gas pressure in the pump tank is 0.3 MPa(g). Any danger of flooding of the secondary pump is prevented by a suitably designed piston ring seal (in the upper slip joint) separating the high pressure pump discharge from the relatively low pressure cover gas space. This seal will be experimentally tested to validate its design.

The IHX is a vertical, counter current flow, shell and tube heat exchanger (fig. 6). Each IHX has 3000 straight tubes (19 mm OD x 0.8 mm WT) with primary sodium on shell side and secondary sodium on the tube side. The tubes are arranged in circumferential pitch. A variable flow distribution is provided inside the IHX tubes with a higher flow on the outer rows to improve the thermohydraulic behaviour of the tube bundle. A mixing device is also provided at the secondary outlet to reduce the temperature differences between inner and outer shell of the secondary outlet header. Absence of flow induced vibration of tube bundle and the drain pipe in the downcomer have been verified by theoretical analysis.

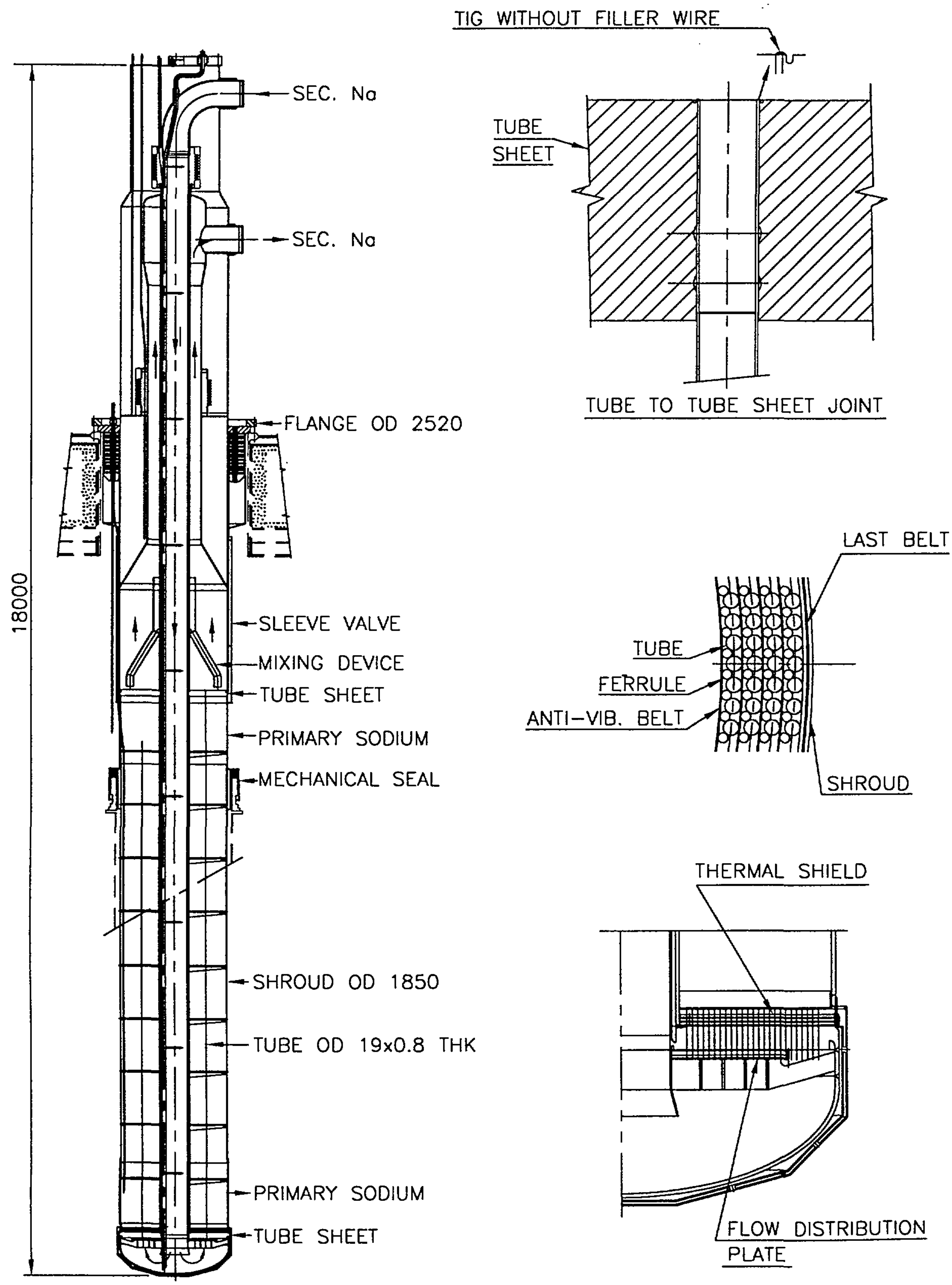
Steam reheat with integrated once-through design for the steam generators has been selected. This has been favoured over sodium reheat as the savings in SG cost, reduction in construction time and ease of design & operation outweigh the marginal advantage in efficiency associated with sodium reheat.

The SG selected is a vertical countercurrent, shell and tube heat exchanger with sodium on the shell side (fig. 7). No cover gas is provided in the SG and a surge tank is provided on the upstream side of the SG. This arrangement is less costly. Experience of multiple SG / loop without cover gas in other reactors is also good. Straight tube design with an expansion bend in each tube located in the bottom portion of the SG above the sodium outlet nozzle has been selected to take care of differential expansion between shell and tubes as well as amongst tubes. Sodium enters the SG through a single inlet nozzle, flows upwards in the annular region & top inlet plenum before entering the tube bundle. A flow distribution device is located in the annular region to bring uniformity in tube bundle flow. Sodium leaving the SG exits through the bottom outlet plenum and a single outlet nozzle. An orifice is provided at the water inlet of each tube of SG from stability consideration. The tubes are supported at various locations by formed type tube bundle support arrangements. Tube to tubesheet joint is of internal bore weld type with raised spigot to enhance reliability (crevice free and radiographable) of this critical weld joint. Long seamless tubes are used in order to reduce the number of tube to tube welds. The inspections proposed for each joint include dye penetrant testing, radiography using anode (microfocus) X-ray and helium leak testing. It is also



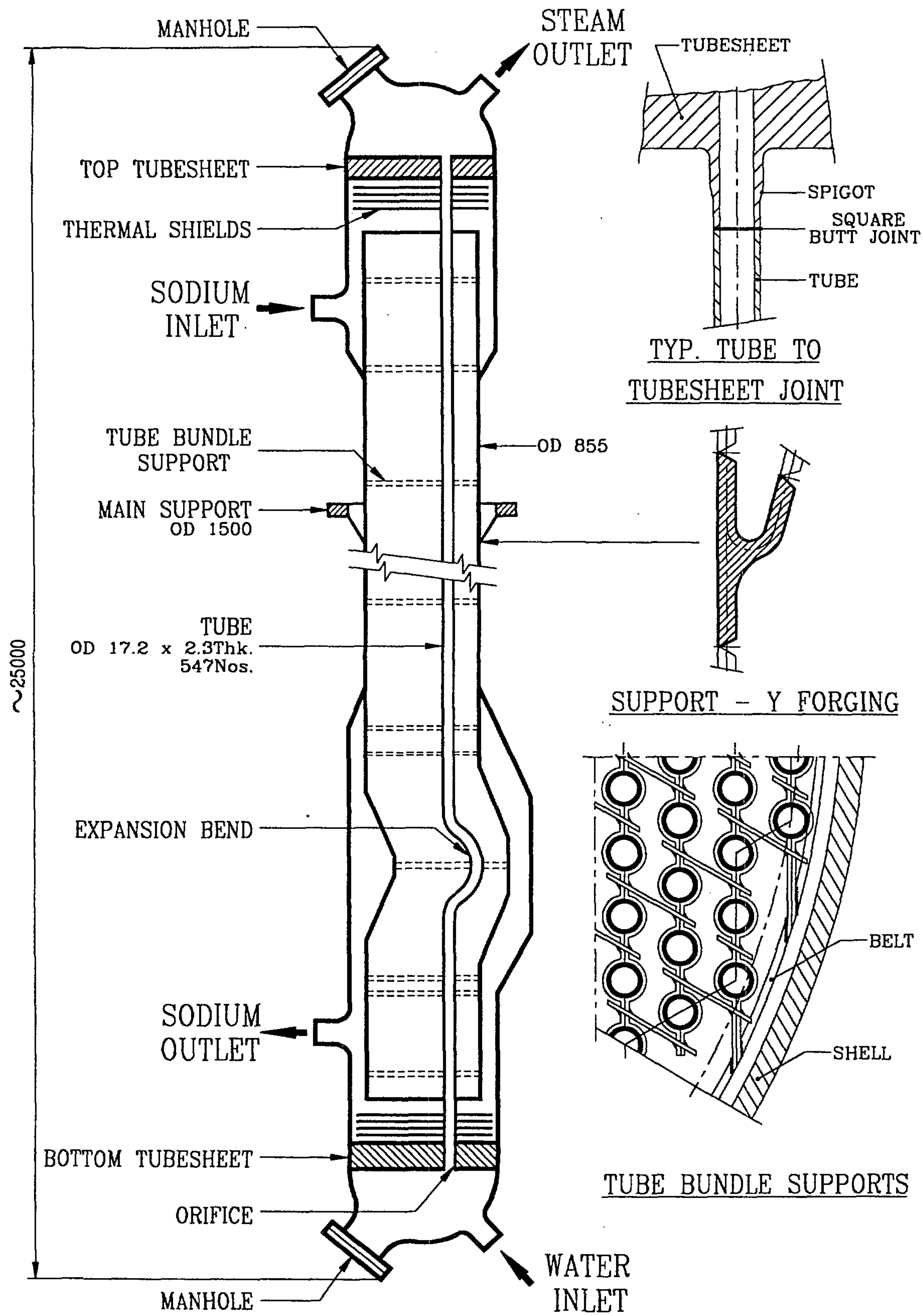
SECONDARY SODIUM PUMP

FIG. 5



INTERMEDIATE HEAT EXCHANGER

FIG. 6



STEAM GENERATOR

FIG. 7

envisaged to postweld heat treat individual joints to avoid risk of stress corrosion cracking associated with hard welds in Cr-Mo steel. Top & bottom tubesheets are protected by thermal shields. Sodium inlet and outlet shell junctions are in the form of pullouts. Manhole is provided on water-steam dished heads to permit access for in-service inspection of tubes and to carry out tube plugging, if required. The design basis accident for SG takes into account the effect of large leaks. Actuation of both the rupture discs located at the inlet and outlet of SG governs the number of failed tubes for leak analysis. Design basis leak is taken as instantaneous double ended guillotine rupture of 3 tubes at the top location of the SG. The reaction products are discharged to the secondary sodium storage tank.

Thermal striping has been the cause of some of the sodium leaks in auxiliary circuits. Mixing devices has been provided to ensure that sodium streams mix with temperature differences lower than the established safe limits.

Oil systems have been avoided wherever possible to minimise the risk of oil fires. The quick actuating valves on the water steam side and sodium side are pneumatic driven. Cooling is by nitrogen for primary cold trap and by air for secondary cold trap. For austenitic stainless steel piping, leak before break concept is used where leak monitoring provision exists. To minimise sodium fires, all the sodium pipes within the reactor containment building are double walled. The dump valves are duplicated in sodium circuits to enhance reliability of dumping, in case of a sodium leak.

It is envisaged to have one primary pump, one secondary pump and a SG as spare.

Operation with one secondary loop at maximum power of 50 % is also planned in case of non availability of one loop.

4.5 Core components handling

In-vessel handling is carried out using two rotatable plugs and an offset (fixed) arm type fuel handling machine (IVTM). An ultrasonic scanner is provided in order to check projection of any SA/absorber rods above the top of the core before starting in-vessel transfer operation. Additionally, strict administrative control on interlocks is to be provided. An Inclined fuel transfer machine (IFTM) is used to transfer the subassemblies from the main vessel to outside.

Ex-vessel sodium storage for removal of decay heat of SA has been avoided and the SA are stored in in-vessel storage locations within the main vessel. The spent fuel subassemblies are stored inside the main vessel for a period of 8 months till the decay power reduces to less than 5 kW and are then shifted to spent fuel storage bay (SFSB). SFSB is a water filled double concrete walled tank.

4.6 Decay heat removal

In case off-site power is available, the decay heat is removed through normal heat transport path of secondary sodium and water/steam circuits. Additionally, an independent safety grade passive direct reactor cooling system consisting of 4 independent circuits of 6 MWt nominal capacity each has been provided. Each of these circuits comprises of one sodium to sodium heat exchanger dipped in reactor hot pool, one sodium to air heat exchanger, associated piping and tanks. Except for the dampers provided on the air side, this system is entirely passive. A slope of 3.5 % is provided for the finned tubes of sodium-air heat exchanger in decay heat removal circuit in order to avoid gas locking.

4.7 Instrumentation and Control (I & C)

I & C, though not having significant impact on capital cost, needs detailed consideration in design as it demands considerable efforts in execution and it has a strong bearing on reactor availability. 2 loop design selected helps in reduction of sodium process instrumentation. The list of trip parameters is based on analysis. In principle, reactor should be shutdown under all design basis events using two independent trip parameters. Reactor shutdown is based on Lowering of rods (LOR) or by SCRAM. Two chromel-alumel thermocouples are provided at the outlet of each fuel SA and are used for SCRAM. Global Delayed neutron detectors (DND) and gaseous fission product detection are used for detection of failed fuel. Only global DND is used for SCRAM. 3 Failed fuel identification modules (FFIM) are provided for locating the SA with failed fuel pins. A bypass electromagnetic flowmeter is provided at the outlet of each primary pump discharge and the flow signal is used for SCRAM.

4.8 In-Service Inspection (ISI)

In-service inspection and monitoring is based on the requirements of ASME section XI, Div 3. For the main vessel, in addition to ASME requirements of continuous monitoring, ultrasonic examination is planned to be carried out through the main vessel - safety vessel interspace (300 mm nominal gap). A periscope is provided for visual examination of reactor internals. Eddy current inspection is under development for the SG tubes. SG tube size and expansion bend design takes into account this inspection requirement. Ultrasonic examination is planned for the dissimilar joints of the roof slab - main vessel and SG transition joint. The subject of ISI for other reactor components important to safety is under study. For the safety related reactor assembly components, which are non-inspectable, an additional factor of safety in design is envisaged.

4.9 Reactor Containment Building

Though the whole core accident is categorised as BDBE, a containment is provided based on the design condition of mechanical energy release of 100 MJ in case of core melt down accident. It has been checked that the main vessel and top shield can withstand this accident. The amount of sodium that is ejected into the containment building does not exceed 1000 kg and preliminary analysis indicate a pressure rise of ~ 10 kPa resulting from the sodium spray fire inside the containment. Aircraft crash is not a design basis event for the containment as the site selected meets the screening distance value of the regulatory code.

4.10 Radiation Protection

The siting, design of the plant and the operating procedures are intended to ensure that the radiation exposure to plant personnel and to the public resulting from the plant operation are controlled so as to comply with the dose limits prescribed by Atomic Energy Regulatory Board (AERB). Adequate shielding is provided wherever required to meet the pescribed dose limits. The targeted collective dose for the plant is 0.5 person-Sv/a (50 man-rem/a). For the general public, the exposure is limited to 0.1 mSv/a (1/10th of admissible dose is apportioned for PFBR).

5.0 SUMMARY

Systematic efforts have been made to take care of the operating experience from LMFBR into the design of 500 MWe PFBR.

FBTR operating experience has improved the confidence level in the design and operation of core, sodium systems, control rod drive mechanisms, fuel handling machines, steam water system and SG leak detection system.

Well proven mixed oxide fuel is chosen as the reference fuel. Pool type concept has been adopted. The plant operating temperatures have been arrived at based on detailed structural analysis of the hot leg components and result in reduced unit energy cost. SS 304 LN / 316 LN is used for sodium systems while modified 9 Cr - 1 Mo is used for SG. 2 loop concept with 2 Primary pumps, 4 IHX and 4 SG per loop has been selected to reduce capital costs and to improve capacity factor. Reactivity incidents have occurred in some reactors and in spite of the best efforts, fully validated explanation has not been possible. Argon gas seal for IHX-Inner vessel sealing has been eliminated and a seal design with piston rings has been selected. Improvement of thermal hydraulics of IHX and provision of a mixing device at secondary outlet have been made. Steam generator design selected takes into consideration the lessons learnt from other operating steam generators and it is expected to realise a more reliable SG.

The design features selected for PFBR are expected to yield an economic, safe and reliable design with improved capacity factor.

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