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CONCEPTUAL DESIGN OF PFBR CORE

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Abstract

The design options selected for the core of the 500 MWe Prototype Fast Breeder Reactor are presented. PFBR has a conventional mixed oxide fuel core of homogeneous type with two enrichment zones for power flattening and with radial and axial blankets to make the reactor self-sustaining in fissile material. Pin diameter has been selected for minimisation of fissile inventory. Considerations for the choice of number of pins per subassembly, integrated versus separate axial blankets, and other pin and subassembly parameters are discussed. As the core size is moderate, no special schemes for reducing the maximum positive sodium voiding coefficient is envisaged. Two independent, diverse, fast acting shutdown systems working in fail-safe mode are selected. The number of absorber rods has been minimised by choosing a layout for maximum antishadow effect. Nine control and safety rods are distributed in two rows for power flattening by differential insertion. Three Diverse Safety Rods, are also provided which are normally fully withdrawn. The optimisation of layout of radial and axial shielding and adequacy of flux at detector location are also discussed.

1.0 CORE DESCRIPTION

The basis for Prototype Fast Breeder Reactor (PFBR) core design is safe operation, economic power generation and self sufficient fissile material production. With these requirements, the core design is arrived at by numerous iterations between thermal, mechanical and neutronic analyses. Under Indian conditions, the initial fuel material is derived from plutonium and depleted uranium obtained from spent Pressurised Heavy Water Reactor (PHWR) fuel.

Mixed oxide fuel has been selected for PFBR on account of extensive worldwide experience available with this fuel, proven capability for safe operation to high burnup, ease of fabrication and handling, and proven economic reprocessing. The low breeding ratio and long doubling times for this fuel is not considered a disadvantage in the initial stage on account of the large amount of initial plutonium becoming availiable from the PHWR programme.

A conventional homogeneous type of layout for the core and blankets has been selected. Radial heterogeneous configuration, inspite of advantages of higher breeding ratio and reduced sodium void coefficient, was not considered due to increase in Pu enrichment, higher fissile inventory, larger overall core size, reduction in Doppler coefficient, increased thermal striping protection requirements for above core structures, and possible difficulty in achieving optimum neutronic coupling between core zones without extensive experimental studies. However, the use of axial heterogeneity is retained as an option for the future for some improvement in breeding ratio, reduction in sodium void coefficient and lower displacement damage fluxes.

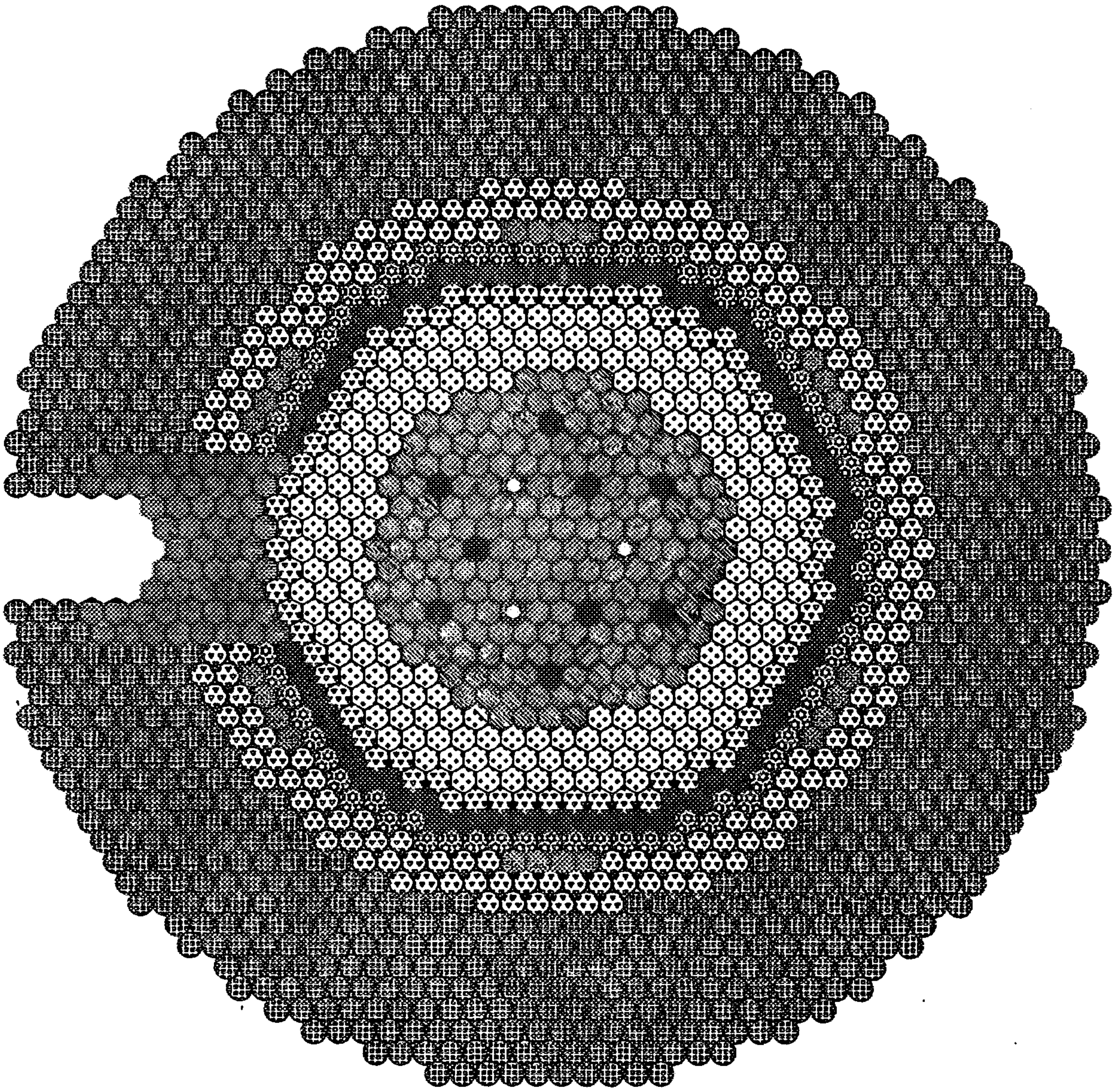
The active core consists of 181 fuel subassemblies of which, 85 are in the inner enrichment zone with 21% PuO_2 content and 96 are in the outer enrichment zone with 28% PuO_2 content. There are three rows of radial blanket subassemblies followed by one row of steel reflector and one row of intermediate shielding boron carbide subassemblies outside of which are the internal fuel storage locations and then the radial bulk shielding subassemblies of steel and boron carbide. There are twelve absorber rods arranged in two rings of which nine constitute the primary Control and Safety Rod (CSR) system and three constitute the secondary Diverse Safety Rod (DSR) system. Fig.1 gives the cross section of the core layout and Table 1 summarises the main parameters of the core.

Each fuel subassembly consists of 217 helium bonded fuel pins of 6.6 mm outer diameter using 20% cold worked D9 alloy cladding separated by helically wound spacer wires giving a pitch ratio of 1.25. Each pin has a 100 cm column of mixed oxide fuel, 30 cm each of upper and lower depleted UO_2 blanket columns and upper and lower fission gas plena. The axial blankets are integrated within the fuel pin. Fig.2 gives the section of the fuel pin and subassembly.

The core restraint against thermal and swelling induced outward movement of subassemblies is provided by spacer pads on the fuel, blanket and reflector subassembly sheaths, at axial position at the middle of the top blanket. This concept has the advantage of moderate interactive forces between subassemblies and permits "flowering" of the fuel subassemblies to some extent to contribute to the negative power coefficient.

2.0 FUEL PIN DIAMETER AND MAXIMUM LINEAR PIN POWER

The choice of fuel pin diameter significantly affects the fissile inventory, fuel cycle cost and the breeding gain. Parameteric studies made for PFBR indicate that the optimum pin diameter from fuel inventory doubling time as well as fuel cycle



SYMBOL	TYPE OF SUBASSEMBLY	No.	MASS PER SUBASSY. IN Kg
▨	FUEL (INNER)	85	245
▧	FUEL (OUTER)	96	245
●	PRIMARY CONTROL ROD	9	200
○	SECONDARY CONTROL ROD	3	200
⊙	BLANKET	186	320
⊛	STEEL REFLECTOR (INNER)	72	355
●	B_4C SHIELDING (INNER)	69	185
⊛	STORAGE LOCATION	75	245
▤	RESERVE STORAGE LOCATION	24	355
▤	ENRICHED BORON SHIELDING	56	185
⊙	STEEL SHIELDING (OUTER)	180	330
⊞	B_4C SHIELDING (OUTER)	903	265

FIG. 1. PFBR core configuration.

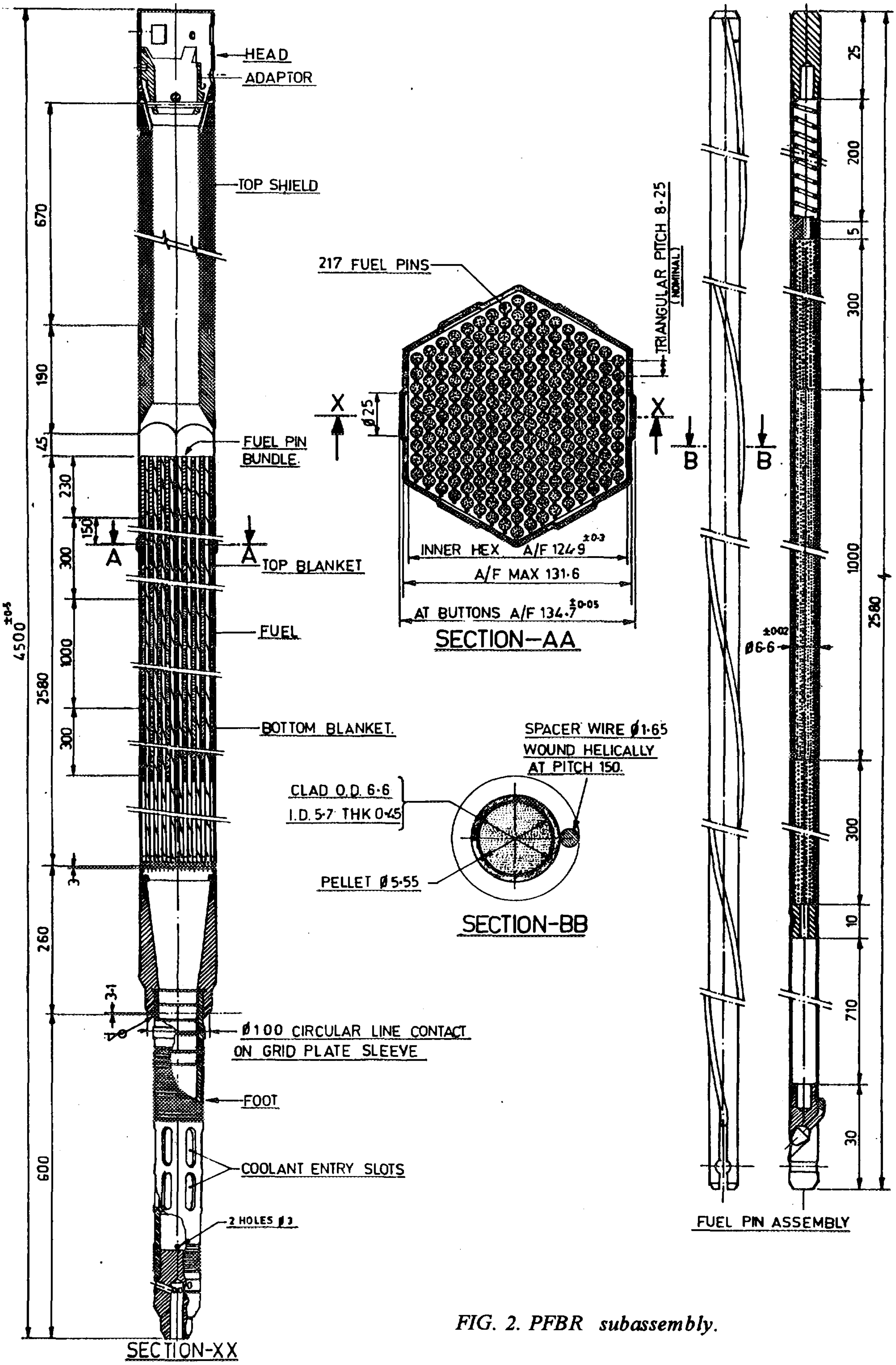


FIG. 2. PFBR subassembly.

TABLE 1. CORE DESIGN PARAMETERS

Core thermal power	1250 MWt
Maximum linear pin power	450 W/cm
Fuel pin outer diameter	6.6 mm
Active core height	100 cm
Equivalent core diameter	199 cm
No. of pins per fuel subassembly	217
Fuel smeared density	82.5% TD
Maximum core Δ T	180^{o}C
Maximum core Δ P	55 m Na
Subassembly pitch	135 mm
Sheath thickness/Subassembly size	3.2/131.3 mm
Fuel pins clad thickness	0.45 mm
Axial blanket thickness (each)	30 cm
Core volumetric fractions	
Fuel/Gap/Sodium/Steel	33/2/41/24 %
Maximum neutron flux	$8x10^{15}$ n/cm^{2}-s
Breeding ratio	1.07
Effective delayed neutron fraction	340 pcm

cost considerations for mixed oxide fuel is 8 to 9 mm depending on the out-of-pile cycle time. However, based on initial plutonium availability, it is decided to restrict the inpile plutonium inventory to two tonnes and consequently the pin diameter is fixed as 6.6 mm, which also has the advantage of smaller core size and lower sodium void coefficient. However, the choice of smaller pin diameter and consequent lower fuel volume fraction has the disadvantages of reduced breeding ratio, reduced core conversion ratio (and hence increased CSR worth for burnup compensation and increased power swing with burnup), reduced Doppler coefficient, reduced fuel residence time and increased fuel cycle cost.

Irradiation testing and basic fuel properties measurements for PFBR fuel composition have not yet been made, but with available data and appropriate uncertainty margins, a maximum linear pin power of 450 W/cm is conservatively fixed.

3.0 SUBASSEMBLY SIZE

Selection of the number of pins per fuel subassembly depends on several factors. A safety consideration is that an individual fuel subassembly must remain sufficiently subcritical under immersion in water. Further, high mechanical loading due to weight and high thermal loading due to decay heat becomes increasingly difficult to be taken into account in the design of the fuel subassembly handling and transport mechanisms. Having fewer number of pins per subassembly favours increased core average coolant outlet temperature and increased average discharge burnup. However, these advantages must be balanced against the disadvantages of having to load/unload larger number of subassemblies at each shutdown, increased number of locations in the grid plate and increased number of core monitoring positions (for outlet temperature and failed fuel). Taking into account the above considerations 217 fuel pins per subassembly has been selected leading to a subassembly size of 131.3 mm (outside distance across flats). For this choice the fuel handling equipment are designed for a maximum extraction load of 15000 N and decay power of 5 kW per subassembly.

4.0 ENRICHMENT ZONING

Power flattening by enrichment zoning permits reduction in fissile inventory for given power output. For cores of PFBR size, two enrichment zones are adequate to achieve the desirable power flattening. Parameteric study made varying zone volume ratio V_1/V_2 from 40/60 to 80/20 showed that for the fresh core the power form factor varied little for V_1/V_2 from 50/50 to 70/30 (less than 2%) with the optimum at 60/40. However, it was found that the stability of the form factor with burnup was better for values of V_1/V_2 less than 50/50. The present selection of V_1/V_2 is 47/53.

5.0 FLOW ZONING

Subassembly outlet temperature flattening by flow zoning reduces thermal striping and increases the mixed mean outlet temperature which improves efficiency. Based on parametric studies the number of flow zones recommended are seven for the fuel, four for the blanket and one each for the absorber, reflector, inner shielding and storage subassemblies.

6.0 ACTIVE CORE HEIGHT

The advantages of small core heights are increased fuel volume fraction for same coolant pressure drop, reduced subassembly length, easier fuel fabrication and reduced sodium void coefficient, but the disadvantages are increased fissile inventory, larger number of pins and larger core radius. The penalty in Pu inventory was calculated as a function of H/D ratio. A limit of 10% was placed on fissile inventory penalty compared to optimum H/D resulting in a lower limit of 0.4 for H/D ratio. Present selection of 100 cm core height has a H/D ratio of 0.5.

Variation of core height from 80 cm to 120 cm was studied by appropriate thermal-hydraulic-neutronic computations. For fixed pin diameter, core pressure drop and core temperature rise, increase of core height decreases fuel volume fraction affecting neutron economy, but at the same time improves H/D ratio leading to reduced neutron leakage and better neutron economy. These two opposing consequences result in optimum core height from fissile inventory and breeding gain considerations to lie between 90 and 105 cm. The study of fuel cycle cost as a function of core height was difficult to make as fabrication cost data as a function of pin length is uncertain but it appeared that the fuel cycle cost was relatively insensitive to the core height around 100 cm.

Further study was made after selection of subassembly size keeping coolant pressure drop and subassembly pitch constant. In this study, other parameters such as pin diameter, clad thickness, pitch ratio, inter-subassembly gap and so on were allowed to vary. On account of increased fuel volume fraction the 80 cm core compared to the 100 cm showed marked improvement in breeding ratio and doubling time at the cost of 6% increase in inpile Pu inventory and nearly 25% increase in the number of fuel subassemblies. The 120 cm core showed 15% reduction in number of subassemblies and 7% reduction in inpile Pu inventory, but marked deterioration in breeding ratio and doubling time. The sodium void coefficient varied in a complex manner with core height on account of simultaneous change of core dimensions and sodium volume fraction. However, the overall change in sodium void coefficient was of the order of only 2%.

7.0 LOCATION OF FISSION GAS PLENUM AND AXIAL BLANKETS

Location of fission gas plenum at the top of the pin is considered desirable in order to prevent the possibility of fission gas passing through the core when there is a fuel pin failure. However, due to higher temperatures a top plenum becomes much longer than a bottom plenum for the same stored gas pressure and mass. Since the relative safety is not much different, for PFBR, the major fission gas storage is in a bottom plenum (71 cm) with a small amount of storage in a top plenum with spring (20 cm).

Based on fuel residence time, buildup of plutonium, coolant pressure drop and fuel fabrication considerations, an axial blanket thickness of 30 cm on each side has been selected. the effect of having axial blanket pins separate from the fuel pins was studied as it leads to lower fuel pin fabrication costs due to shorter lengths and fewer number of blanket pellets. It was found that on account of choice of mainly bottom plenum storage, the separation of the top axial blanket does not lead to much breeding penalty, whereas separation of the bottom axial blanket does lead to considerable reduction in breeding gain. Since overall subassembly length is smaller and subassembly fabrication is simplified, the choice of having both blankets integrated in the fuel pin has been made.

8.0 RADIAL BLANKET THICKNESS

Study of the variation of blanket breeding ratio with radial blanket thickness indicated 90% saturation at a thickness of 40 cm. Consequently, the number of radial blanket rows is fixed at three. A study for reduction of the number of blanket rows to two showed that in this case there is 0.04 reduction in breeding gain and the reactor may not be a net producer of plutonium in case cycle losses exceed 1%.

9.0 REACTIVITY COEFFICIENTS

The inlet temperature coefficient and the power coefficient are overall negative as for other fast reactors of this type with the dominant negative component coming from the Doppler effect in U-238. The positive contribution from the coolant expansion coefficient is smaller and delayed compared to the negative contribution from the Doppler effect and fuel expansion.

No special design provisions have been made to increase the Doppler coefficient or reduce the sodium void coefficient which is about 3.5 dollars for the reference core. As compared to the case of the 8 mm or greater pin diameter, the 6.6 mm pin diameter reference core has marginally smaller Doppler coefficient but also substantially smaller positive sodium void coefficient (lower by over a dollar). However, the effect of voiding leads to a greater reduction in Doppler coefficient for the 6.6 mm pin diameter case as compared to the larger pin diameter cases.

As already stated, decreasing the core height to 80 cm from 100 cm caused little reduction in sodium void coefficient but considerably increased the core size and inpile fissile inventory and was hence not considered.

10.0 REFUELLING INTERVAL

This is an important parameter which affects the core characteristics in several ways. Shorter refuelling interval improves average discharge burnup, reduces excess reactivity and control requirements, improves breeding ratio, reduces number of internal fuel storage locations, reduces the out of pile inventory and improves the doubling time. On the other hand, longer refuelling interval improves the reactor availability and reduces the thermal cycling of the reactor (due to shutdown). There is also an upper limit on refuelling interval which comes from the design peak fuel burnup. It is necessary to choose the (fixed) refuelling interval as a simple fraction (1/2, 1/3 etc.) of the residence time of the peak rated subassemblies to prevent gross reduction in the average discharge burnup.

Studies showed that for a peak discharge burnup of 100,000 MWd/t the suitable refuelling interval is 180 full power days. As initially the fuel burnup will be limited to 50,000 MWd/t the refuelling interval is correspondingly reduced to 90 full power days.

Based on the above choice of refuelling interval the excess reactivity and control requirements, the fuel management scheme and the number of invessel fuel storage locations have been arrived at.

11.0 REACTIVITY CONTROL SYSTEM

The core is protected by two independent shutdown systems each capable of bringing the reactor to cold shutdown. The number of rods in each shutdown system is sufficient to ensure cold shutdown by that system even with its most reactive absorber rod unavailable. There are nine CSRs in the first shutdown system and three DSRs in the second shutdown system. The twelve rods of the reactor are distributed in two rows to maximise the antishadow effect and to minimise the required number of rods.

In order to avoid common mode failure and to achieve diversity, the design of both the systems is in such a way that parts related to safety action are entirely different. There is a mechanical coupling between the mobile assembly of the CSR Drive Mechanism and CSR, whereas there is an electromagnetic coupling between the DSR Drive Mechanism and DSR. Similarly, the rods of each shutdown system are of different design. Two independent plant protection logic systems provide trip signals to the corresponding shutdown system and independent and diverse control system for each shutdown system is provided. The drop time of the rods under emergency conditions is less than a second.

For both CSRs and DSRs, enriched boron carbide is chosen as the absorber material on account of high nuclear cross section, ease of fabrication, availability and good experience. Study of the use of Eu_2O_3 and EuB_6 (natural) was made to avoid the use of B-10 enrichment. However, Eu_2O_3 was only as good as natural boron and EuB_6 (natural) was as good as 30% enriched boron. Since, over 50% enriched boron is required the choice of boron carbide was retained.

The nine CSRs are in both the rings to enable power flattening adjustment by differential insertion. All the CSRs have the same B-10 enrichment and their individual worths depend on their position. Speed of movement is sufficiently low (2 mm/s i.e. maximum 1 cent/s) to enable desired control and hence, separate regulating rods of low worth are not provided. The minimum total worth of CSRs is 8000 pcm. Enough additional worth is provided for calculational uncertainty and burnup.

Three DSRs of equal reactivity worth are in the inner ring. The total minimum reactivity worth of these rods is 3000 pcm. Enough additional worth is provided for calculational uncertainty and burnup. DSRs are not used for reactor control and are in fully raised position during reactor operation.

There is 6000 pcm of core excess reactivity (1700 pcm power and temperature defect, 3100 pcm BOEC to EOEC reactivity swing, 600 pcm operating margin and 600 pcm uncertainty margin). The emergency shutdown reactivity available for scram is over ten dollars and can handle postulated incidents like loss of flow, transient over power, loss of regulation, fuel melting and slumping in a few subassemblies and sodium voiding in a few subassemblies. The shutdown margin in the fuel handling state is 5000 pcm which can cater to hypothetical postulated errors like removal of any two absorber rods in shutdown state or replacement of absorber rod by fuel subassembly.

The possibility of use of central absorber rod was studied as it decreased the total number of absorber rods to ten and also had the incidental advantage of lowering central sodium voiding reactivity gain. However, in a prototype reactor the central location is an important position for experimental access purposes and it was not considered advisable to block it with an absorber rod.

3800 pcm of reactivity is required to compensate the reactivity loss on going from BOL to BOEC in three cycles. This is not provided by the absorber rods but by diluent subassemblies in the initial fresh core which are progressively unloaded during the approach to equilibrium.

12.0 IN-VESSEL SHIELDING

The in-vessel shielding consists of mainly the radial shielding and the upper axial shielding. The shield design is essentially governed by the dose rate specified for personnel working close to the secondary sodium pipelines in the steam generator building area. The dose rate criterion presently specified by the Atomic Energy Regulatory Board of India is 20 mSv/y for occupational exposure. The corresponding criterion for shield design has been specified as 1 micro-Sv/h. The allowable secondary sodium activity has been computed for the above dose limits at a distance of 20 cm from the surface of a secondary sodium pipeline and comes to be 3 Bq/cc.

The fuel storage locations were initially after two rows of stainless steel reflector. To reduce fission rate in the storage locations the outer row of stainless steel reflector was replaced by a row of boron carbide shielding subassemblies. Low fission rates in storage locations were required for reducing shielding and ease of cooling during handling of irradiated subaseemblies. This change also helped in reducing the radial shield thickness.

The in-vessel radial shielding is located after the fuel storage locations. The intermediate heat exchangers centre line is located in the sodium pool at a radial distance of 4 m from the core centre. Radial shield optimisation studies were carried out such the secondary sodium activity does not exceed 3 Bq/cc. Initially, optimisation studies were carried out with shield materials like carbon steel, graphite, 2% borated graphite and stainless steel. This led to excessively thick shields and, in turn, to larger vessel diameter. As reducing the vessel diameter leads to significant reduction in costs, in subsequent studies, only stainless steel and boron carbide have been considered. Using boron carbide it has been possible to reduce the radial shield thickness from 15 rows to 9 rows (each row corresponds to a thickness of 13.3 cm of the material). The proposed configuration for the radial shield consists of two rows of stainless steel followed by seven rows of boron carbide. It was also found that the last three rows of boron carbide could be of vibro-compacted boron carbide powder instead of boron carbide pellets.

The proposed design limits for neutron irradiation dose is 1 dpa and for He production is 0.1 appm for components such as core cover plate, control plug and the grid plate. These limits have been used for the design of the upper axial shields which are integrated in the subassembly itself. Design of upper axial shields were initially carried out with stainless steel and borated graphite. The upper axial shields need to satisfy two criteria viz. the neutron fluence seen by the core cover plate and the secondary sodium activity. It was found that the neutron fluence on the core cover plate was low and did not influence axial shield deisgn which is hence based on secondary sodium activity. The proposed upper axial shield now consists of 15 cm of stainless steel followed by 63 cm of boron carbide. No shield has been provided in the lower axial direction. Calculations show that the helium production rates are much smaller than 0.1 appm and the dpa at grid plate is 0.25 dpa.

In vessel transfer post (IVTP) for transfer of irradiated subassemblies has been located in the radial shield. Hence it was necessary to design locally a better shield to take care of the loss of shielding due to the presence of the IVTP. It is proposed to place 5 rows of enriched boron carbide around the IVTP (Fig.1).

13.0 DETECTOR LOCATION FLUX

The flux at the detector location below the reactor vessel during start-up was estimated using the presence of inherent neutrons in fresh and irradiated fuel. It was found that the flux at detector location was too low for use of fission counters or boron detectors. Feasibility of using a neutron guide to enhance flux at detector location was studied. It was found that provision of neutron guide in blanket first row enhances the thermal equivalent flux by a factor of ten only. It was considered feasible to use high sensitivity He-3 counters without the use of neutron guide for the neutron monitoring during routine startups. For the initial core loading and approach to criticality, use of in pile detector and auxiliary neutron source is planned.

14.0 CONCLUSION

The conceptual design of PFBR core has evolved taking into account constraints of initial plutonium inventory, limited possibility of extensive advanced fuel irradiation testing and limited possibility of experimental investigations on core configurations radically different from the conventional homogeneous mixed oxide fuelled type. In accordance with the constraints, a core design has evolved which places high importance on successful past experience, minimisation of plutonium inventory, small core size, low initial capital cost, economic power generation on closed fuel cycle, and just self-sustaining from plutonium breeding considerations.