Computation Tools for Search of Optimal Thorium Fuel Cycle and Their Fuel Management Strategies

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ABSTRACT

India has vast resources of thorium. There is considerable interest in evolving optimal fuel cycle and fuel management strategies for large scale utilisation of thorium. A code system called 'PHANTOM-EXCEL-TRISUL' has been developed for the study of a variety of core designs tailor-made for use of thorium in power reactors. PHANTOM code considers fuel design in ring cluster form. EXCEL code considers a hexagonal lattice structure. Both are lattice burnup codes. They use the 69 group cross section library in WIMS/D-4 format. TRISUL is a core diffusion analysis code. TRISUL incorporates a thermal hydraulic model to determine the axial distribution of steam volume fraction in each fuel channel. A semi-auto-refuelling scheme allows the study and intercomparison of a variety of reloading strategies. A distinguishing feature of the code system, is the possibility to prescribe the lattice parameters of thoria and other mixed fuel clusters as a function of the accumulated neutron fluence in thoria fuel rods. In core simulation assigning the correct thoria rod fluence to each fuel cluster is crucial for determining the correct core characteristic. Since they are not known a priori they are iteratively determined so as to be consistent with the refuelling and fuel management schemes.

1. INTRODUCTION

Thorium does not have any intrinsic fissile content. Use of externally fed seed material like ^{235}U or Pu can allow thorium to be used like uranium. However it would increase the fuel cycle cost significantly. Hence thorium is yet to be used in any appreciable amount for routine power generation. There are proposals to use the dismantled weapon grade uranium or plutonium for introduction of thorium within the constraints of existing PWR or VVER fuel assembly designs [Radkowsky and Galperin, 1998, Galperin and Segev, 2000]. Their proposal is basically an open fuel cycle or once through option. High burnup in thorium is achieved by allowing them to reside in the core for six or nine fuel cycles while the seed zones follow the normal annual batch refuelling scheme. One of the main design objectives of this design is the incineration and disposal of weapon-grade materials.

In Indian context, it is desirable to design systems in which residual fissile content in the discharged fuel is maximised. In a closed fuel cycle option, one can extract much more power when the fuel has a re-use potential. India has thorium reserve six times that of uranium reserve. We therefore look for core designs ideally suited for effective utilisation of thorium. A novel means of inducting thorium in power reactors was conceived by us (Jagannathan et al., 1998). In this concept, loading of seedless thoria rods in the ambience of high thermal neutron flux is suggested. These neutrons are produced by fission in the neighbouring seed fuel zones, get slowed down and remain trapped in islands of D_2O moderator medium with low absorption probability. Such flux traps are created by deliberately replacing the fuel assemblies at regular lattice intervals with D_2O moderator medium. Seedless thoria rods are loaded at these flux traps. When they are irradiated for a duration of one fuel cycle length of one to two years, they acquire sufficient quantity of ^{233}U. Thereafter they can behave like regular fuel rods. Irradiated thoria rods are assumed to be mechanically juxtaposed with fresh seed fuel rods of enriched uranium or other equivalent seeds and irradiated for few more fuel cycles. By optimally choosing the number, size and location of the seed fuel rods and the seedless thoria rods, it was possible to achieve nearly the same discharge burnup from both the types of fuel rods. The residence time for thoria rods is just one additional fuel cycle, i.e., the first fuel cycle in which they accumulate ^{233}U in the same reactor. The weight fraction of seedless thoria loading in the equilibrium core is 50% or more by design. Since breeding of fissile atoms from natural thorium rods is achieved in the same reactor, this reactor concept has been symbolically named as 'A Thorium Breeder Reactor' (ATBR). The large fraction of thorium loading in the ATBR concept has an additional advantage. The high excess reactivity of seed fuel zone is balanced by the low reactivity of thoria rods. With burnup, the seed zone looses reactivity, while thoria rods gain reactivity. They are balanced in such a way that the core excess reactivity intrinsically remains low and flat for a fairy long cycle duration of 300 to 660 days (Jagannathan et al. 2000a). By the choice of optimal refuelling scheme, a near flat power distribution is achieved at the beginning of an equilibrium fuel cycle and becomes inherently flatter as cycle burnup proceeds. There is practically no need for conventional reactivity control in the form of mechanical absorber devices, soluble boron or burnable poison. Near elimination of neutron absorption in control absorbers leads to better utilisation of fuel, and a better fissile conversion ratio.

The other salient core characteristics of ATBR core are given in the Ref. Jagannathan et al. (2001). Extension of the ATBR idea to seed materials like PuO_2 or $^{233}UO_2$ in ThO_2 and their comparison with enriched UO_2 seeded core is presented in Ref. Jagannathan et al. (2000a). Sensitivity of the ATBR core characteristics to different cross section libraries which have been released as part of the IAEA Co-ordinated Research Programme on WIMS library update project (Ganesan and Trkov 1999 & 2000), is presented in Ref. Jagannathan et al. (2000b).

The aim of the present paper is to briefly describe the computation tools used for the study of thorium fuel cycles. The lattice codes PHANTOM and EXCEL are described briefly. In TRISUL code, the special modelling for evaluation of the key information of thoria rod fluence for every fuel cluster is described. Some illustrative results of TRISUL

code are presented. Some discussion is made on the engineering feasibility of the proposed ATBR concept and possible variants thereof.

2. COMPUTATION TOOLS

Here we describe the calculation model for the study of ATBR type cores. The optimised core design parameters are given in references and are not repeated here. ATBR core is designed for a power level of 600 MWe or 1875 MWt. The core typically has 360 (seed + irradiated thoria) fuel clusters. The core considers, in addition, loading of at least one batch size (72, 90 or 120) of natural thoria clusters in each fuel cycle.

2.1 The PHANTOM Code

The ring cluster form of fuel design is analysed by the CLUB module (Krishnani, 1982) of the PHANTOM code system (Jagannathan et al. 1990). It uses a combination of interface current method in the annular ring regions and first flight collision probability method for any heterogeneity within a ring. The basic cross section data is 69 group WIMS library. Recently, the PHANTOM code was made operational with any of the new libraries, based on the latest point data files like JEF-2.2, ENDF/B-6.5 or JENDL-3.2, which became available as part of the IAEA CRP on WIMS library update project (WLUP) (Ganesan and Trkov 1999 & 2000). The code uses the burnup chain as stipulated in each version of the WIMS library, i.e., variable number of explicit fission products and actinides as specified in the library is considered. One pseudo fission product accounts for the absorption of the fission products not considered explicitly. The actinide chain is from ^{232}Th to ^{244}Cm and includes several minor actinides of neutronic importance. Alpha mode of decay of some heavy actinides is also considered. Rhodium-105 and xenon-135 are considered as saturated fission products.

The lattice calculations are done in two major steps. In the first step, the seedless thoria cluster of a single ring is simulated as a function of neutron fluence using specific flux levels within the expected range of flux values in the core. In the second step, the irradiated thoria rods after a given fluence are assumed to be integrated with fresh seed fuel rods and burnup study is carried out for the mixed fuel cluster using an average linear power rating.

The geometrical disposition for seedless thoria cluster is schematically shown in Fig. 1. The problem domain consists of the thoria cluster, boiling H_2O coolant, structural materials of pressure tube and calandria tube, air gap, and the D_2O moderator in one hexagonal lattice pitch. There is a central BeO block. Reflective boundary condition is applied on the problem cell boundary. The burnup study is carried out for constant one group flux values like (0.5, 1.5 and 2.5) $\times 10^{14}$ n/cm^2•s^{-1} up to irradiation time of 1500 days. Homogenised two group lattice parameters of the thoria cluster cell as a function of thoria rod fluence is collected for subsequent core diffusion calculations. We consider two energy groups with energy cut-off at 0.625 eV.

The geometrical disposition for the mixed fuel cluster, referred to hereafter as ATBR fuel cluster, is schematically shown in Fig. 2. The ATBR fuel cell consists of two rings of fresh seed fuel rods surrounded by irradiated thoria rods, boiling H_2O coolant, structural materials of pressure tube and calandria tube, air gap, and the D_2O moderator in one hexagonal lattice pitch. There is a central BeO block of smaller diameter. Reflective boundary condition is applied on the problem cell boundary. The burnup study is carried out parametrically for specific fluence values for thoria rods. This is effected by picking up the starting fuel isotopic densities of thoria rods, from the above calculations for thoria clusters, corresponding to the fluence values of (flux = 0.5, 1.5 or 2.5) $\times 10^{14}$ $n/cm^2 \bullet s^{-1}$ times irradiation time of 300, 500 or 700 days. These irradiation times cover the range of fuel cycle duration that may be achieved, depending on the reload batch size. An average linear heat generation rating of 160 w/cm is considered. The homogenised two group lattice parameters as a function of burnup up to 60 GWD/T is collected for subsequent core diffusion calculations.

2.2 The EXCEL Code

EXCEL code was developed for the burnup study of fuel assembly with hexagonal lattice arrangement, as in Russian VVERs. WIMS library is used as the basic cross section set. It simulates each heterogeneity of the fuel assembly by a series of one dimensional multigroup transport calculation. Interface current method or the first flight collision probability method is used. One-d supercell simulation is done for the entire fuel assembly cell to obtain the multigroup spectrum across the assembly. The homogenised cross section of the fuel pincell and other cells are collapsed to few (typically five) energy groups. The fuel assembly is then treated by two dimensional few group diffusion calculation. Hexagonal finite difference method is used. The homogenised cross section for the entire fuel assembly in few (typically two) groups is obtained for subsequent core diffusion calculations. EXCEL code was validated with the study of VVER reactor benchmark analysis (IAEA, 1995).

Recently, EXCEL code was made compatible with the latest 69 group libraries mentioned above. The burnup model described above for PHANTOM code has also been incorporated in EXCEL code. Test runs are in progress.

It is expected that EXCEL code would become another important design tool for study of thorium fuel cycles. Some preliminary explorative study has been initiated to check the feasibility of ATBR physics ideas in a fast reactor spectrum (Jagannathan, 2001).

Since both EXCEL and PHANTOM codes are written in variable dimensions, it should be possible to adopt larger number of energy groups as well. When the cross section library with more number of energy groups is suitably devised, one may use the above lattice codes for reactor systems with fast or intermediate neutron spectrum with a degree of reliability comparable to that of thermal reactor systems.

2.3 The TRISUL Code

The code TRISUL, as described here, has been specially developed for the ATBR core design computations. TRISUL is an acronym derived from ***T***horium ***R***eactor ***I***nvestigations with ***S***egregated ***U***ranium ***L***oading (Jagannathan, 2000).

TRISUL code solves the multi-group steady state diffusion equations,

$$-\nabla.D_g\nabla\varphi_g + \Sigma_{rg}\varphi_g = \frac{\chi_g}{K_{eff}}\sum_{g'=1}^{G}\nu\Sigma_{fg'}\varphi_{g'} + \sum_{g'=1,g'\neq g}^{G}\Sigma_{s,g'\to g}\varphi_{g'}$$

The boundary conditions are of the type: $D\frac{\partial\varphi}{\partial\vec{n}} + \Gamma\varphi = 0$ for each group.

The symbols in Eq. (1) have standard meaning. Center-mesh finite difference method with one hexagonal mesh or six triangular meshes per hexagonal cell is used.

The main task in TRISUL analysis is to make use of the lattice parameter database, described in Section 2.1 or 2.2 and assign appropriate cross section to every mesh point in the problem domain. We shall describe here the methods used for the burnup study of the ATBR core. As described earlier, the ATBR equilibrium core consists essentially of two types of fuel, viz., natural thoria clusters and seed fuel clusters integrated with irradiated thoria clusters. Fig. 3 shows the typical core layout for equilibrium core studies. The core is described by homogenised fuel regions surrounded on all sides by some reflector regions.

The scoping study of the equilibrium core is normally done with six-fold symmetry. It is possible to use reflective as well as rotational symmetric boundary conditions. In case of reflective symmetry, the symmetry line passes through the centers of the hexagonal assembly and hence half assemblies appear on the symmetry line. Because of half assembles, it is found that there are an unequal number of fuel locations belonging to each fuel batch. In case of rotational symmetry, only integral hexagons are considered. It is noted that with rotational symmetry boundary condition, one can consider the same number of fuel assembly locations for every fuel batch number. This feature is useful in devising a semi-auto-refueling scheme. By arranging the fuel assembly burnups at the end of a fuel cycle in ascending order, slicing them into number of batches of equal number of assemblies and linking the locations of assemblies of the same order number of every batch, the fuel-reshuffling scheme could be semi-automated. The fuel-loading pattern is then described by simply specifying the batch number for each assembly location in the core. As a rule, one can follow a typical (*OUT-IN* + low leakage loading scheme) as a starting guess pattern. The loading pattern can then be improved by observing the calculated radial power distribution and shuffling the fuel assemblies in pair so as to achieve a balanced power distribution by trial and error method.

To perform the few group diffusion calculations, it is necessary to prescribe the lattice parameters for every subregion having a hexagonal or triangular right prismatic shape. The cross sections for the thoria clusters is considered as:

$$\Sigma_{thoria} = \Sigma (\phi_{his}, d, v)$$

ϕ_{his} is the average flux seen by the thoria rods up to the present irradiation time 'd' days, during the first cycle of irradiation. 'v' is the volume fraction of coolant void or steam.

The cross sections for the ATBR mixed fuel cluster is considered as:

$$\Sigma_{ATBR\ fuel} = \Sigma (E, \phi^0_{his}, d^0, v)$$

E is fuel burnup in MWd/tonne, ϕ^0_{his} and d^0 are the average flux history seen by the thoria rods and the time elapsed at the time of their integration with fresh seed fuel rods. d^0 is normally the cycle length considered for the equilibrium core. 'v' is the volume fraction of coolant void or steam.

It is seen that to enable interpolation of cross section using the pre-generated parametric database of lattice constants, one must know the fuel burnup, thoria rod fluence and the void fraction pertaining to a fuel mesh. We will explain the method of obtaining this information. All these parameters depend on the flux and power distribution prevalent in the core which are yet to be determined. It is therefore necessary to devise iterative procedures so that all the above three dimensional (3-D) distributions shall have an internal self-consistency.

a) **Fuel burnup**: While starting the equilibrium core study, the burnup and the initial fluence values for fresh thoria clusters are set to zero. For other ATBR mixed clusters the guess values of the 3-D burnup distribution at the beginning of cycle (BOC) is defined as,

$$E_n(z) = \frac{E_d}{N} \sum_{m=1}^{n-1} P_m \cdot \sin(\pi z/H) \cdot C$$

The axial profile E_n (z) is the same for every fuel assembly belonging to the fuel batch number 'n'. E_d is the discharge burnup in MWd/tonne, N is the total number of fuel batches, 'z' is axial distance from bottom of the active core, H is the core height along with axial reflector saving, C is a normalization constant such that the axially averaged burnup value corresponds to the mean burnup of m-cycled fuels, P_m is a guess value of power fraction in the fuel assemblies which have completed (m-1) fuel cycle operation. P_m can be obtained with a trial core simulation with all P_m initially set to 1.0. E_d is obtained from,

$$E_d = POWER \times DAYS / (N_b \times W)$$

'POWER' is full power of the reactor in MW (thermal) and 'DAYS' is cycle length in days (effective full power days – EFPD), 'N_b' is the number of fuel assemblies in a batch and 'W' is the initial heavy metal weight in one fuel assembly in tonnes.

The above definition of burnup distribution is used only for first core follow-up run. Subsequently the end of cycle (EOC) burnup profile and the user-specified fuel reshuffling scheme are together used to define the 3-D burnup distribution at BOC. ϕ^0_{his} and d^0 are picked up from the thoria rod locations of same order number. At the start of equilibrium core study, the guess values of ϕ^0_{his} and d^0 are taken as some average guess value. For example, for ATBR, they are 1.2×10^{14} n/cm^2.s^{-1} and the cycle length in days respectively. After a core followup run for one fuel cycle, the average flux seen by the each thoria cluster would be available. This data and the number of days of operation are important parameters in determining the lattice parameters for an ATBR fuel cluster. Thus the flux at the locations of thoria clusters are used in assigning the flux history interpolation parameter for the entire lifetime of a ATBR mixed fuel, which moves from first to n^{th} batch locations in successive fuel cycles.

The above procedures allow us to define the 3-D distributions of exposure or burnup, 'E' in MWd/tonne, flux history (in 10^{14} n/cm^2.s^{-1} units) in thoria rods and time of irradiation in days. The void distribution would depend on power distribution, the coolant flow thorough each channel and the inlet enthalpy of the coolant.

The thermal hydraulic model to determine 'v' is explained in Ref. Jagannathan, (2000). Briefly, the model is as follows. The initiation of subcooled boiling is estimated by the model of Levy, (1966). The corresponding flow quality, which is different from the equilibrium quality, is obtained by the model of Zuber and Findlay, (1964). The void-quality correlation is as per Zuber et al. (1967). These models are used in the Tarapur BWR simulator code COMETG and were seen to perform reasonably well (Jagannathan et al., 1984).

After obtaining all the interpolation parameters, the cross sections for a given mesh is obtained by linear interpolation with burnup, Lagrange interpolations for flux history, irradiation time and void. Void distribution is updated after every eight iterations. During void loop, the other power-dependent feedback corrections due to Doppler and xenon are also applied. It is necessary to evaluate the average fuel temperature and average fuel rod power in a mesh, relative to the rated power, in order to apply the above feedback corrections.

The mean fuel temperature in a fuel cluster is determined by assuming a simple linear variation of the form,

$$T_f = 286 + (600 - 286) \cdot P_{rel}$$

where, P_{rel} is the power in a fuel mesh relative to the reference power used in lattice calculations. T_f is 600°C when $P_{rel} = 1.0$, and is equal to the coolant temperature of 286°C when P_{rel}-= 0.0. The fuel temperature 'T_f' is used to evaluate the perturbation to epithermal cross section '$\delta\Sigma_{a1}$'.

$\delta\Sigma_{a1}$ is evaluated as,

$$\delta\Sigma_{a1} = \Sigma_{a1}^{pert} - \Sigma_{a1}^{ref.}$$

Σ_{a1}^{ref} is the epithermal absorption cross section at the nominal fuel temperature of 600°C.

$$\Sigma_{a1}^{pert} = [\nu\Sigma_{f1} + \nu\Sigma_{f2} \,.\, c - k_{\infty}^{pert.} \;\Sigma_{a2} \,.\, c\,] / k_{\infty}^{pert.}$$

where, $$c = \Sigma_{1\rightarrow 2} / (\Sigma_{a2} + \Sigma_{2\rightarrow 1})$$

All cross sections correspond to the reference state. The perturbations are evaluated at zero burnup. The ratio ($\Sigma_{a1}^{pert} / \Sigma_{a1}^{ref}$) or ($1 + \delta\Sigma_{a1}/ \Sigma_{a1}^{ref}$) is used to perturb the cross section at any burnup, void and thoria rod fluence.

To evaluate the xenon perturbation correction, the saturated xenon load for a given power level is converted into equivalent thermal absorption cross section perturbation, or '$\delta\Sigma_{a2}$' by the following formula.

$$\delta\Sigma_{a2} = \Sigma_{a2}^{pert} - \Sigma_{a2}^{ref.}$$

$$\Sigma_{a2}^{pert} = [\nu\Sigma_{f1} \bullet \Sigma_{2\rightarrow 1} + \nu\Sigma_{f2} \bullet \Sigma_{1\rightarrow 2} - k_{\infty}^{pert.} \bullet \Sigma_{a1} \bullet \Sigma_{2\rightarrow 1}] / [k_{\infty}^{pert.} \bullet (\Sigma_{a1} + \Sigma_{1\rightarrow 2}) - \nu\Sigma_{f1}]$$

Σ_{a2}^{ref} is the thermal absorption cross section of the fuel with no xenon. Σ_{a2}^{pert} is the thermal absorption cross section of the fuel with xenon corresponding to rated power level. k_{∞}^{ref} and $k_{\infty}^{pert.}$ are the infinite multiplication factors pertaining to the above two states. The magnitude of '$\delta\Sigma_{a2}$' values as evaluated at the rated mean power level at a given burnup is designated as 'x_1'. The '$\delta\Sigma_{a2}$' values are also evaluated at various power levels at zero burnup. The dependence of '$\delta\Sigma_{a2}$' on the relative power 'P' in a mesh is obtained as,

$$\delta\Sigma_{a2} = [(x_1 \bullet (1 + x_2\,(P)) \bullet P] / [P + x_2\,(P)] - x_1$$

It can be seen that when P = 0, we get $\delta\Sigma_{a2} = - x_1$, and for P = 1.0, $\delta\Sigma_{a2} = 0.0$. This is because the hot database considers saturated xenon corresponding to nominal average power level and no perturbation is required, if the relative mesh power is unity. The above equation is used to evaluate $x_2(P)$ as a function of the relative mesh power 'P'. The x_1 values are evaluated as a function of fuel burnup. The dependence of $x_2(P)$ on relative power level 'P' is assumed to be the same at all burnups.

3 TYPICAL RESULTS OF THE ATBR AMSLYSIS

TRISUL code is used at present for equilibrium core studies. It computes eigenvalue or k_{eff}, 3-D distributions of flux, power, void, fuel burnup and void history, fluence in thoria assemblies etc as a function of fuel cycle burnup. During core follow-up runs, the power level is maintained at full power of 1875 MW thermal. All thoria clusters remain fully *IN*. No external control absorber is present. The power distribution is improved by optimizing the refueling scheme. For a given fuel cluster design, it is

possible to vary the cycle length by changing the fuel batch size. For instance, for the same number of 360 ATBR fuel clusters in the core, a 3, 4 or 5 batch fueling would give a batch size of 120, 90 or 72 assemblies. The larger batch size would give a longer cycle length. Since the fissile conversion in thoria clusters is enhanced by higher fluence, the longer cycle length is found to have an added advantage. By virtue of the ATBR design philosophy, the core excess reactivity was found to remain low and flat for as long as 22 months with a Pu seeded core (Jagannathan et al. 2000a). Long fuel cycle has the merit of allowing sufficient time for fuel reprocessing, prior cooling time and time for re-fabrication. The refueling pattern should however be optimized for each seed fuel type, batch size and cycle length. A few of optimized refueling patterns for enriched UO_2, PuO_2 and $^{233}UO_2$ seeds were documented in the Ref. Jagannathan (2000). They present a variety of attractive floral designs. Fig.4 gives the k_{eff} plot with cycle burnup for Pu-seeded ATBR core with three batch fueling. The k_{eff} values using two data libraries of ENDF/B-6.5 and JEF-2.2 sets differ by about 0.7%, but show very similar trend.

Figs. 5 to 8 give the typical epithermal and thermal flux distributions near the core mid-plane at BOC and at EOC (660 EFPD), for the Pu seeded core. These distributions show absolute fluxes. It is seen that the epithermal flux shows dips at regular intervals. These are locations of natural thoria clusters where the fission neutron production is negligible at the beginning of cycle and remains low during the first cycle of irradiation. However due to the slowing down process, the ambient epithermal flux gets converted into thermal flux. Since the absorption potential of natural thoria clusters is less than that of the seeded regions, there is a net accumulation of thermal flux in these regions. There are, therefore, small mounds seen in the thermal flux distribution at the thoria cluster locations. The thermal flux is also seen to rise in the D_2O reflector region, where epithermal flux has fallen down. The thermal flux profile thus presents the shape of a larger size cake. Figs. 9 and 10 give the 3-D plots of power distribution at BOC and EOC. The power distribution is plotted in the form of block histogram, where the height represents the relative assembly power. Thoria assembly powers are almost zero at BOC and are not visible. However at higher burnups small blocks appear at thoria locations, representing the relative power in them. It is seen that power peaking factor intrinsically decrease with cycle burnup. It is worth mentioning that the gross 3-D peaking factor intrinsically decreased from 1.724 to as low as 1.272, radial peak from 1.286 to 1.052 and axial peak from 1.266 to 1.186 for the Pu seeded core during a cycle length of 660 EFPD.

4. ENGINEERING DESIGN CONSIDERATIONS

In our view, the in situ fissile breeding and burning of thoria rods, as proposed in the ATBR concept, is essential to keep the fuel cycle cost low and comparable to the fuel cost incurred in present day power reactors. It is believed that the attractive physics characteristics of the core, described above, would provide the incentive for overcoming the engineering challenges required in the ATBR design.

Some discussions are perhaps worthwhile to explore the engineering feasibility of ATBR design. ATBR considers pressure tubes diameter of 17.6 cm, arranged in a tight hexagonal lattice pitch of 30 cm. This was necessary to ensure negative void coefficient

in case of LOCA (loss of coolant accident). The design of coolant pipe routing at the inlet and outlet is rendered difficult with the above tight lattice pitch. The pitch may possibly be increased up to 35 cm or so. However, in order to preserve the D_2O moderator volume, one must consider some relatively low absorbing and non-moderating blocks like lead in the moderator region. This can more or less preserve the neutronic characteristic of ATBR core. The overall calandria size would however increase.

The other alternative studied, was a pressure vessel type design with D_2O used as coolant and moderator. The Atucha-1 plant at Argentina considers such pressure vessel design (Corcuera, 1999). Some preliminary calculations gave the following inferences. The engineering problems of fuel cluster design are eased out. Coolant circuit design is simpler. D_2O being very low neutron absorber, compared to boiling H_2O, the thermal neutron utilisation factor increases by 5% and is ~0.99. The limitations come from the total number of fuel channels that can be accommodated in a given size of pressure vessel. D_2O being a poorer moderator requires high V_m/V_f to get a reasonably high k_∞. The fuel cluster size was decreased to (30 seed + 24 irradiated thoria rods) and was accommodated in a coolant tube of 12 cm. A tight lattice pitch would allow more fuel channels to be accommodated in a given pressure vessel. When a tight hexagonal lattice pitch of 18 cm was used, the resonance escape probability became too low (<0.5). Even though high resonance captures lead to better fissile conversion, k_∞ remained too low at all burnups in comparison to the k_∞ of the present ATBR fuel cluster design. At higher pitch, the reactivity is gained, but the slope of k_∞ with burnup was also increased. Depletion of fissile isotopes in seed zone remained high due to high flux level in a tight pitch and high effective cross section at higher pitch. Notwithstanding these inferences, it must be possible to design a pressure vessel system with ATBR like characteristic. It is found that a hexagonal pitch 22-24 cm would give the optimum k_∞ vs burnup characteristic. With the pressure vessel dimensions of Atucha plant, it is estimated that it is possible to design a reactor with a gross power of 300 to 350 MWe only. If pressure vessel diameter can be of the order of ~6m, the reactor can be designed for higher power level. Since this system considers three costly components, viz., pressure vessel, enriched seed and D_2O moderator, the economics of such a system is questionable.

Another alternative to be studied was a design similar to ATBR, but with a fast neutron spectrum. Though our code system in the present state is not capable of studying a fast reactor due to limitations of cross section data, a conceptual proposal for a fast thorium breeder reactor is suggested for other competent groups to explore such a possibility (Jagannathan, 2001). The engineering problems associated with the primary coolant circuit design of the ATBR would not be there in such designs and hence deserve to be studied more rigorously.

5. CONCLUSIONS

India has interest in evolving optimal fuel cycle strategies to maximize the power extraction potential from its vast resources of thorium. In this paper we have given a description of some of the calculation tools developed for this purpose. Lattice burnup codes PHANTOM and EXCEL were developed for the analysis respectively of the Indian

PHWRs and the proposed Russian VVERs. Considerable effort was made to make these codes compatible with the latest better qualified data libraries that have become available from the IAEA sponsored CRP activity of WLUP. Incorporation of generalized burnup chains from ^{232}Th to ^{244}Cm and explicit treatment of nearly 60 fission product nuclides has improved the burnup model in these lattice codes. The diffusion code TRISUL has been specially developed for the equilibrium core studies of new designs like ATBR. Special care has been taken to make several internal self-consistency checks, since there are no similar codes developed elsewhere. TRISUL code can consider variable number of energy groups. It can therefore be used for a fast reactor version of the ATBR type core with as many energy groups as required.

In our view, the reactor design for the best use of thorium may differ from the ones that are considered for uranium fuel, in order to exploit the higher fissile conversion potential of thorium. ATBR design is one of the possibilities. In order to verify the physics design characteristics as calculated by theoretical codes, it is suggested that some calculation benchmark be formulated using core design parameters of ATBR. When the characteristics are verified by other peer groups, it can pave the way for exploring several similar designs and evolve a system that has operational characteristic superior to the existing ones and is also feasible from engineering design considerations. For better utilization of thorium, this R&D effort is considered to be a very essential step.

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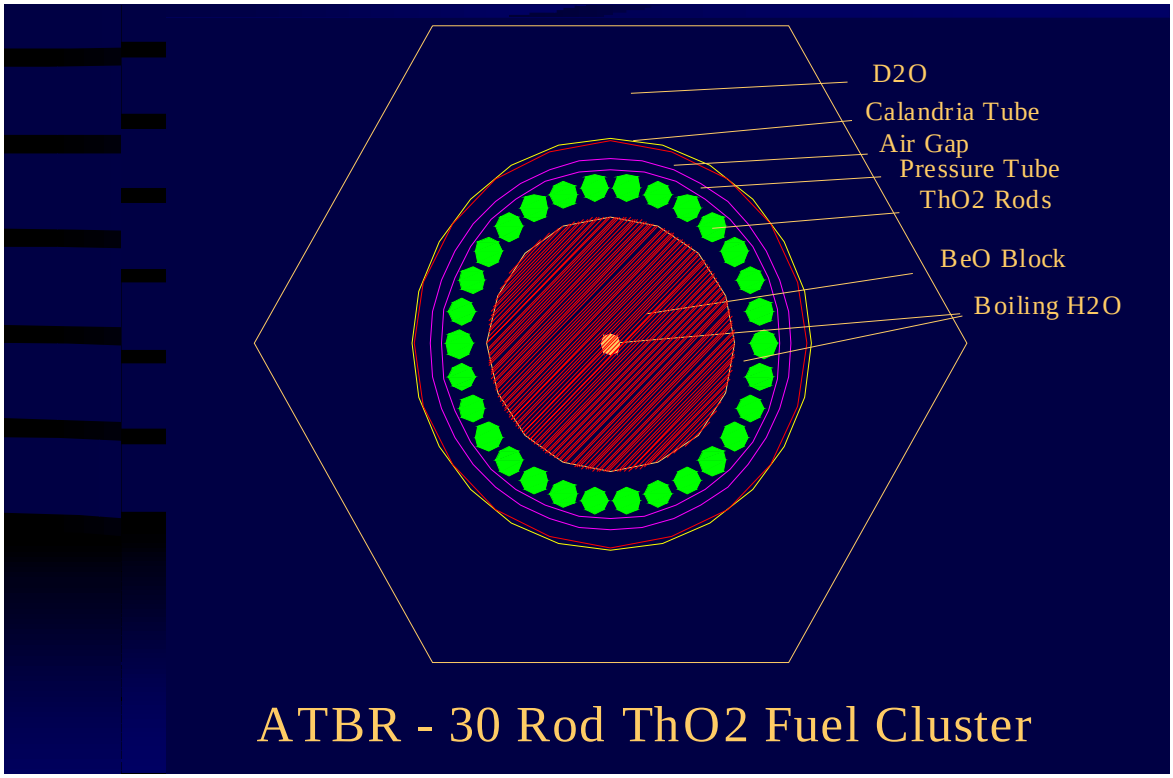

D2O
Calandria Tube
Air Gap
Pressure Tube
ThO2 Rods
BeO Block
Boiling H2O
ATBR - 30 Rod ThO2 Fuel Cluster

Fig.1

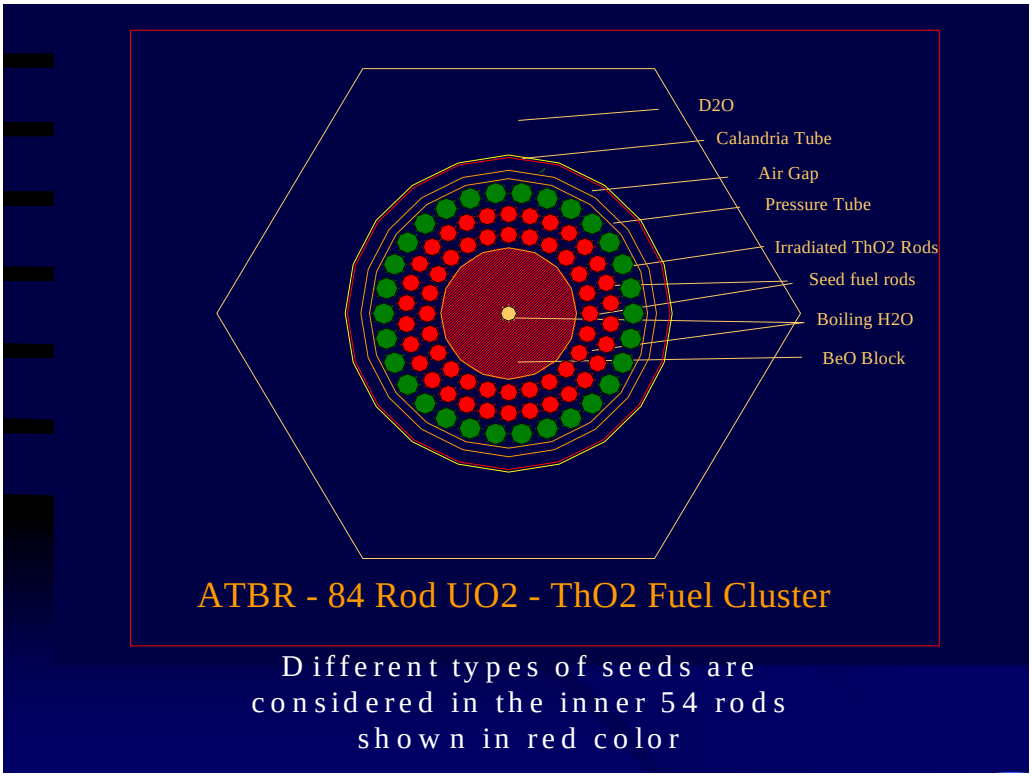

D2O
Calandria Tube
Air Gap
Pressure Tube
Irradiated ThO2 Rods
Seed fuel rods
Boiling H2O
BeO Block
ATBR - 84 Rod UO2 - ThO2 Fuel Cluster
Different types of seeds are considered in the inner 54 rods shown in red color

Fig.2

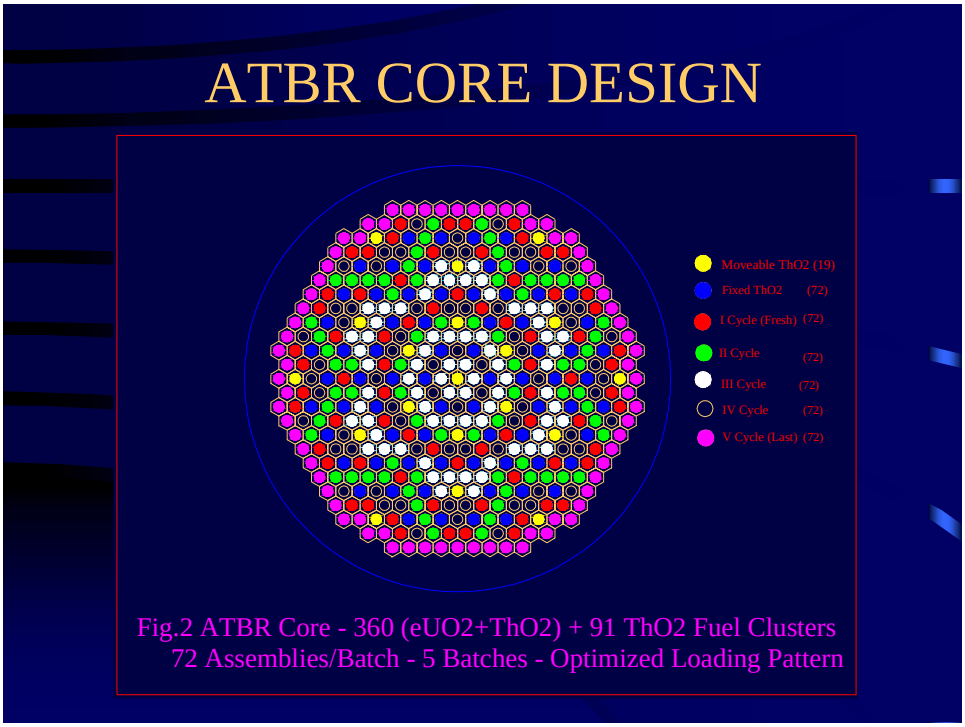
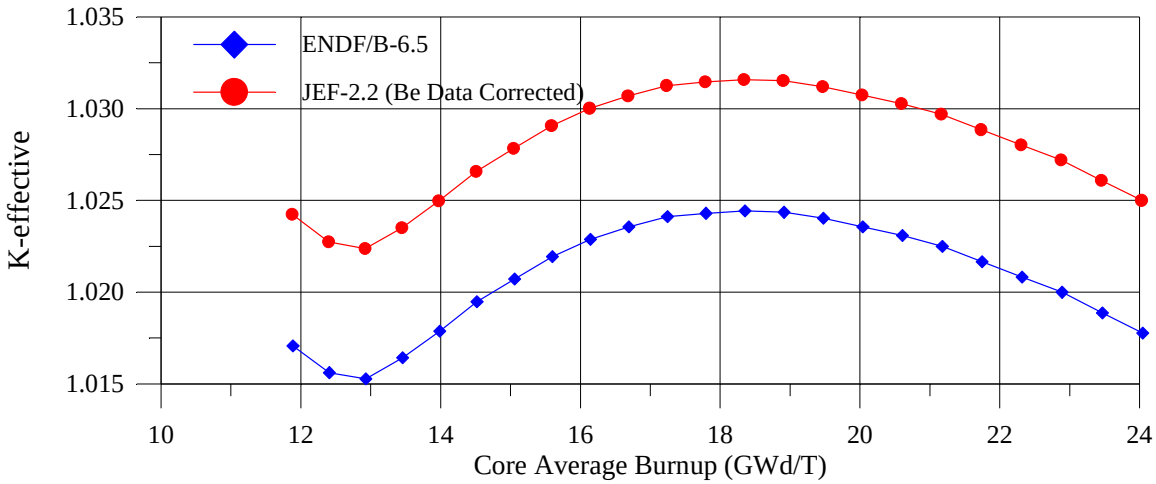


Fig.3



ATBR Equilibrium Core - K-eff vs Cycle Burnup
Pu Seeded Core - Three Batch Fueling
120 Assemblies per Batch - Cycle length = 660 EFPD

Fig.4

ATBR –600 EQUILIBRIUM CORE WITH PuO_2 SEED IN ThO_2 EPITHERMAL FLUX DISTRIBUTION AT CORE MID-PLANE - BOC – 0 FPD

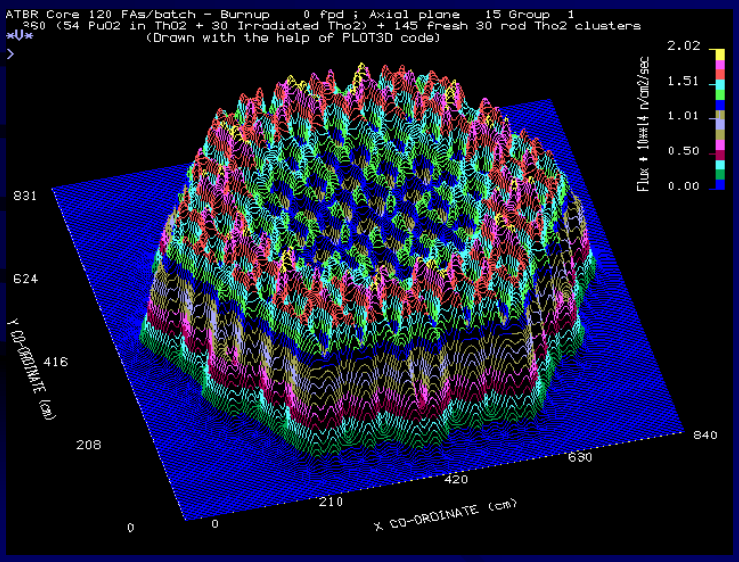


Fig.5

ATBR –600 EQUILIBRIUM CORE WITH PuO_2 SEED IN ThO_2 THERMAL FLUX DISTRIBUTION AT CORE MID-PLANE - BOC – 0 FPD

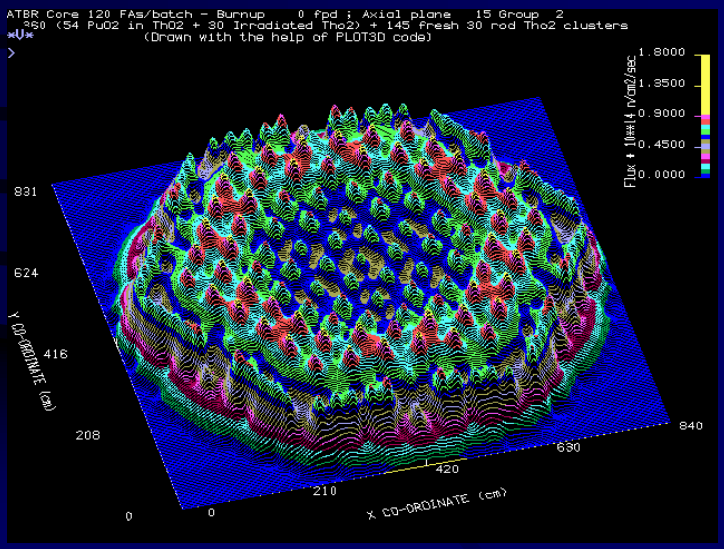


Fig.6

ATBR –600 EQUILIBRIUM CORE WITH PuO_2 SEED IN ThO_2 EPITHERMAL FLUX DISTRIBUTION AT CORE MID-PLANE - EOC – 660 FPPD

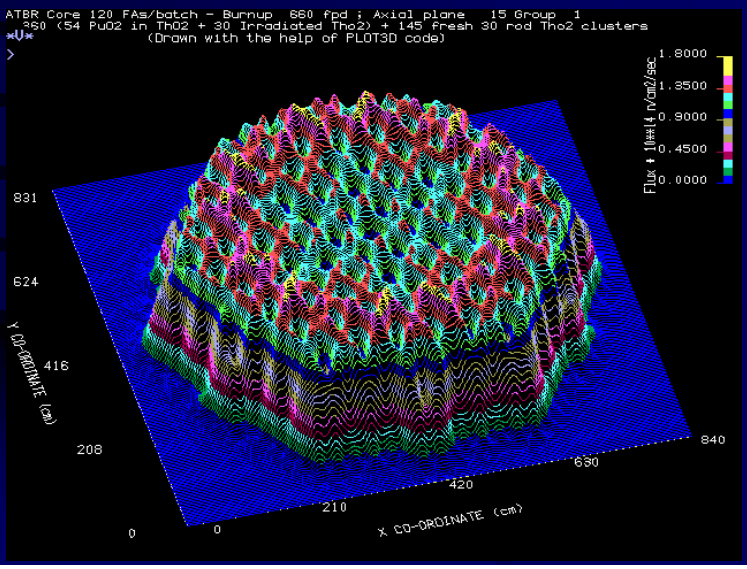


Fig.7

ATBR –600 EQUILIBRIUM CORE WITH PuO_2 SEED IN ThO_2 THERMAL FLUX DISTRIBUTION AT CORE MID-PLANE - EOC – 660 FPPD

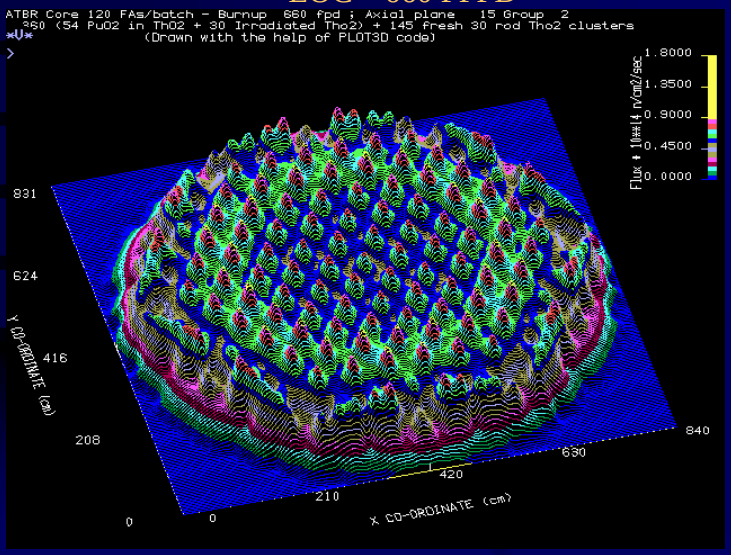


Fig.8

ATBR –600 EQUILIBRIUM CORE WITH PuO_2 SEED IN ThO_2
POWER DISTRIBUTION AT CORE MID-PLANE - BOC – 0 FPD

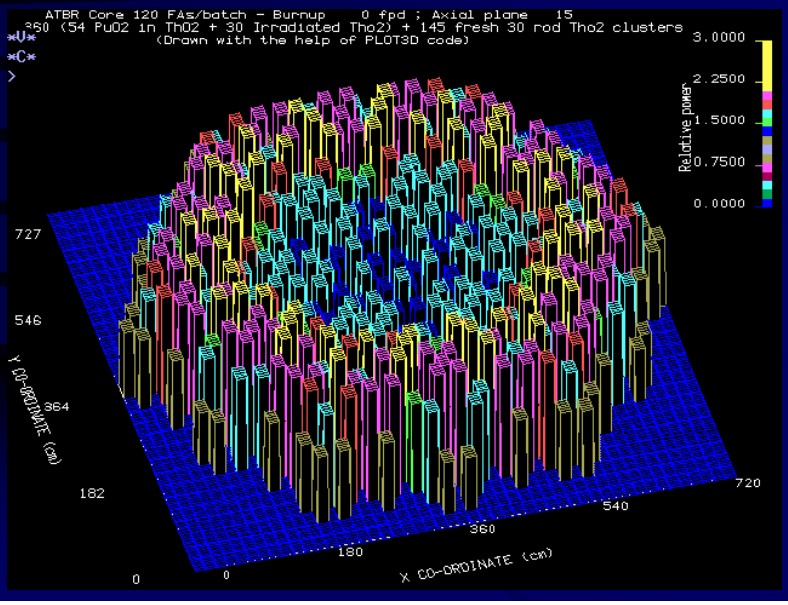


Fig.9

ATBR –600 EQUILIBRIUM CORE WITH PuO_2 SEED IN ThO_2
POWER DISTRIBUTION AT CORE MID-PLANE - EOC – 660 FPD

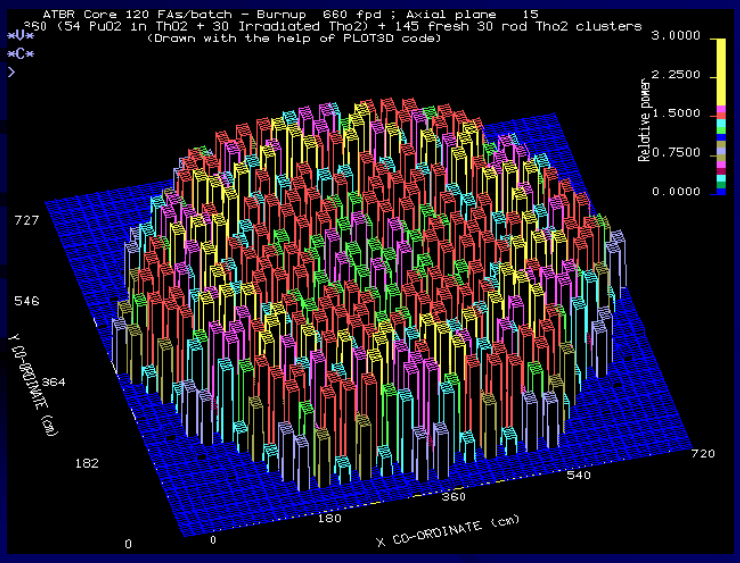


Fig.10