

# BARC

## NEWS LETTER

### ATBR - A THORIUM BREEDER REACTOR CONCEPT FOR EARLY INDUCTION OF THORIUM WITH NO FEED ENRICHMENT

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#### Introduction

Fission energy is released by splitting of a heavy nuclide into two or more lighter ones. Impinging a target nuclide like  $^{235}\text{U}$  with a neutron, one can induce the fission reaction. The fission process releases two or more neutrons that could be used to induce fission reaction in at least one more  $^{235}\text{U}$  nuclide. Repetition of this process establishes the fission chain reaction. If it is controlled such that the neutron population remains constant with time, one can obtain the fission nuclear power on a continuous basis. This is the process that is exploited in nuclear reactors.

Soon after the discovery of fission process, it was recognized that uranium and thorium are the potential nuclear materials available in nature, which hold the key to fission nuclear power. However, there are some features that clearly distinguish these two elements with respect to their exploitation potential. Natural uranium has mainly two isotopes  $^{235}\text{U}$  and  $^{238}\text{U}$  with abundance of 0.72% and 99.274% (balance being  $^{234}\text{U}$ ).

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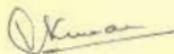
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## EDITORIAL

*India has limited uranium reserve and vast resources of thorium. The present day power reactors use mostly uranium, since thorium has no intrinsic fissile content. ( $^{232}\text{Th}$ - $^{233}\text{U}$ ) fuel cycle is superior to the U-Pu fuel cycle. Availability of  $^{233}\text{U}$  is a prerequisite to start a thermal reactor programme using thorium exclusively.*

*The feature article in the present issue describes a conceptual design of a 600 MWe reactor which is a prebreeder, i.e., efficient  $^{235}\text{U}$  to  $^{233}\text{U}$  converter and eventually a breeder or self-sustaining reactor system with ( $^{232}\text{Th}$  -  $^{233}\text{U}$ ) fuel.  $\text{D}_2\text{O}$  moderator and boiling  $\text{H}_2\text{O}$  coolant flowing in vertical pressure tubes are considered. The seed is 5% enriched  $\text{UO}_2$ . The unique feature of the reactor concept is that it considers no feed enrichment for the thorium of blanket zone.  $\text{ThO}_2$  rods are irradiated in the same reactor for one fuel cycle duration and hence contain some  $^{233}\text{U}$  bred insitu, before they are integrated with fresh seed fuel. The reactivity and power shape are intrinsically maintained with minimum external control maneuvers. Uranium and thorium rods achieve a discharge burnup of 32,000 MWD/T. The reactor is found to have superior safety, economic and operational characteristics. Fuel reprocessing is not needed in prebreeder phase, and is halved in the closed fuel cycle phase.*

*Providing effective healthcare to the teeming population has always been one of the prime concerns of the successive governments in India. Radioisotopes produced in BARC have played a vital role in the national health programme in the treatment of various types of cancers and hyperthyroidism. Isotopes are also widely used in industry, medicine, hydrology, agriculture, research, healthcare and water resources development and management. How the radioisotopes produced in BARC have been successfully used in various applications is the central theme of the second article in this issue.*



Thorium has mainly one isotope,  $^{232}\text{Th}$ . All these isotopes are capable of undergoing fission reaction. However, the fission process in the isotopes of even mass number occurs predominantly with neutrons of energy higher than about 1 MeV, and for neutrons of energy lower than this threshold energy, the fission process is highly improbable.  $^{235}\text{U}$  is the only isotope available in nature for which the fission process can occur with neutron of any energy. Isotopes such as  $^{235}\text{U}$  are termed as fissile isotopes.

The isotopes  $^{238}\text{U}$  and  $^{232}\text{Th}$ , on capture of a neutron and subsequent two  $\beta$  decays, get converted respectively into isotopes  $^{239}\text{Pu}$  and  $^{233}\text{U}$ . These isotopes are fissile ones like  $^{235}\text{U}$ . These are man-made fissile isotopes. Isotopes like  $^{238}\text{U}$  and  $^{232}\text{Th}$  are called fertile isotopes as they have the potential of getting converted into fissile ones.

For fissile isotopes, the probability of fission, normally measured in terms of a quantity called fission cross section, increases significantly by nearly two to three orders of magnitude at low neutron energy. Reactors employing mainly slow neutrons for fission reaction are called thermal reactors since the energy of the neutron is comparable to the thermal or temperature state (kT) of the ambient medium. These reactors use some light nuclide like hydrogen, deuterium or carbon to slow down the fission neutron born in MeV range to thermal energy range of a few eV. The peak of neutron population is typically around 0.025-0.05 eV. These light nuclides are called moderators.

When no moderator is used, the neutron energy will remain high and close to the fission neutron energy. However, there is no possibility of establishing fission chain reaction with natural uranium or thorium with fast fission neutrons. This is because, even though all nuclides are fissionable, the neutrons would undergo predominantly inelastic scattering collisions with the very U/Th isotopes and the neutron energy would reduce soon to energies below the fission threshold energy. One has to necessarily use a sizeable amount of some fissile isotope in order to establish fission chain reaction with

neutrons of fast energy spectrum. Such reactors are called fast reactors.

A large variety of thermal and fast power reactor designs have been evolved in the past. Though uranium and thorium have comparable gross energy potential for generation of fission nuclear power, the present day nuclear power reactors use mainly uranium. This is because thorium has no natural fissile content. The need for external feeding of  $^{235}\text{U}$  or Pu in thorium adds to the fuel cycle cost, thus making it commercially less attractive, at least till  $^{226}\text{U}$  is available. However, it is imperative to produce significant amount of the man-made fissile isotopes:  $^{239}\text{Pu}$  or  $^{233}\text{U}$ , prior to the exhaustion of the small natural content of  $^{235}\text{U}$ , so as to be able to extract nuclear energy from the entire fertile column of uranium and thorium.

It is possible to utilize the excess neutrons born in a fission reaction for prospective capture in fertile isotopes. When more fissile atoms are produced than what are consumed, the reactor is said to be a breeder. This is achieved rather easily in a fast reactor using  $^{239}\text{Pu}$  as fuel since relatively more number of neutrons are born per fission. Nearly 3 neutrons are born per fast fission in  $^{239}\text{Pu}$ , as against 2.4 per fission in  $^{235}\text{U}$  with thermal neutrons. It must be mentioned here that the fissile isotopes also have a possibility of fissionless capture, i.e. a neutron may be simply absorbed and no fission may occur. In this case, not only there will be no fission energy, but there would be no fission neutrons either. To account for this fissionless capture probability in fissile atoms, another quantity called eta ( $\eta$ ) is defined which is the number of fission neutrons born per absorption of a neutron in a given fuel nuclide ( $\eta = \nu\sigma_f / \sigma_a$ ). Here  $\sigma_f$  is fission cross section,  $\sigma_a$  is absorption cross section and  $\nu$  is number of neutrons born per fission. When  $\eta$  is significantly larger than two, one can use one neutron for maintaining fission chain reaction and possibly slightly more than one for capture in fertile atoms and the rest will be lost in parasitic capture in structural materials and through net leakage from the system. Fast reactors in which the fissile atoms grow with time are called fast

breeder reactors (FBRs). The fissile atom production takes place in thermal reactors too. But since  $\eta$  is close to 2 for  $^{235}\text{U}$ , the ratio of net fissile atoms produced to consumed ones is less than unity and hence these reactors are called converters and not breeders. A thermal breeder is perhaps feasible only with the other man-made isotope  $^{233}\text{U}$ . It has the least fraction of fissionless capture.  $^{233}\text{U}$  has  $\eta$  value of about 2.3 or more, nearly in the entire energy range, while for  $^{235}\text{U}$  and  $^{239}\text{Pu}$ ,  $\eta$  dips below 2 at intermediate energies where significant resonance capture (fissionless) occurs. A light water breeder reactor (LWBR) was demonstrated at Shipping Port, Pennsylvania, U.S.A. This reactor employed ( $^{232}\text{Th}$ ,  $^{233}\text{U}$ ) fuel and operated during 1977-82. A net breeding ratio of 1.013 was reported.

A breeder reactor is very attractive since it is capable of generating the fissile atoms not only for its own consumption, perhaps for even additional ones. However, it is seen that one needs stockpiling of the man-made isotope  $^{239}\text{Pu}$  or  $^{233}\text{U}$ , before embarking on deployment of breeder reactors in large scale. In this context, it is worth examining the fissile material production capability of different thermal power reactors that are currently operational.

Deuterium in the form of heavy water is one of the best moderating material because it has very low absorption cross section for thermal neutrons. Natural uranium can be used as fuel in pressurized heavy water reactors (PHWRs). The relative rate of fertile captures in  $^{238}\text{U}$  is high in PHWRs since  $^{235}\text{U}$  content is low. PHWRs are therefore very good thermal converters. In PHWRs, the mean residence time of fuel is less than a year. The average discharge burnup of fuel is around 7,000 MWD/T. PHWRs have limited core excess reactivity during the equilibrium phase and hence they need on-line refueling of practically one fuel channel per day.

Hydrogen of light water is very efficient in slowing down the neutron from fast to thermal energy in least number of collisions. However, the thermal neutron capture cross section for hydrogen is 500 times larger than that of deuterium. Hence, while using light water as moderator,

one must use enriched uranium to compete with the absorption in water. The order of enrichment is 3 to 4% in light water reactors (LWRs). The fissile conversion rate in LWRs is lower. However, the fuel resides for a longer duration of three to six years. Discharge burnup of the order of 30,000 to 45,000 MWD/T is achieved in LWRs. The core excess reactivity is fairly large and hence LWRs need refuelling only once in a year or so. The core excess reactivity is sought to be controlled by soluble boron, burnable poison mixed with fuel or other type of control rods.

For the same gross power generation, the mass of natural uranium used in PHWRs will be nearly 6 to 8 times the mass of enriched uranium fuel used in LWRs. The gross Pu output is more than double in PHWRs. However, the specific content of Pu in discharged fuel is about 0.3% in PHWRs in comparison to about 1% Pu in fuel discharged from LWRs. The LWR fuel contains, in addition, about 1% of unburnt  $^{235}\text{U}$ , which is about 0.2% in PHWRs.

It must be stated here that the Pu discharged from thermal reactors contain fissile as well as fertile isotopes. There are four major isotopes, viz.  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ ,  $^{241}\text{Pu}$  and  $^{242}\text{Pu}$ . Of these  $^{239}\text{Pu}$  and  $^{241}\text{Pu}$  are fissile. The fissile content is typically about 70%. Even in these fissile isotopes, nearly 30% of thermal absorption leads to fissionless capture. The isotope  $^{240}\text{Pu}$  has very large absorption cross section for thermal neutrons. These factors make the Pu to be less attractive fuel in comparison to  $^{235}\text{U}$  or  $^{233}\text{U}$  for thermal reactors. Pu is a good fuel for fast reactors in view of its breeding potential mentioned above.

From the foregoing discussions, the following points can be noted.

- The present day power reactors use mainly uranium as fuel. Hence, only plutonium is produced in these reactors. Thorium as fuel is not introduced in any significant measure so far in the world.
- $\text{D}_2\text{O}$  moderator of PHWRs is useful for good neutron economy. Use of natural uranium limits the core

excess reactivity, but leads to higher Pu production. Though total Pu content in discharged fuel is large, its specific content is low, and a large mass needs to be reprocessed to recover this Pu.

- LWRs have to use enriched fuel. Large excess reactivity is compensated by wasteful captures in some absorbers like boron. The Pu production here is less than half that of PHWRs. But the specific content is three times that of PHWRs. In comparison to natural uranium fuel, nearly  $1/6^{\text{th}}$  or  $1/8^{\text{th}}$  of the mass of fuel is required to be handled both in core and out of core.
- The Pu produced from thermal reactors is a better fuel for FBRs, in view of its breeding potential.

It is seen that there are some good as well as some undesirable features in the above two types of thermal reactors. By combining the merits and eschewing the demerits, it must be possible to evolve a better reactor concept. Induction of thorium is kept as one of the major objectives in the new reactor design presented below.

### Physics Considerations To Induct Thorium

The nuclear characteristic of thorium is to be clearly understood if one desires its induction in thermal reactors in large proportions.  $^{232}\text{Th}$  has nearly three times the absorption cross section for thermal neutrons in comparison to  $^{238}\text{U}$ . It has lower resonance captures. Hence a well thermalized neutron spectrum is suitable for enhancing the captures in fertile thorium. It is interesting to observe the irradiation behavior of thoria ( $\text{ThO}_2$ ) rods or  $\text{UO}_2$  rods kept under the same environment. A constant gross flux level of  $10^{14}$  n/cm<sup>2</sup>/sec was assumed to be incident on these rods for a period of 1500 days.  $\text{D}_2\text{O}$  moderator was considered. Fig.1 gives the total uranium or total plutonium formed respectively from thorium or uranium fuel and the fissile contents thereof in units of g/kg. It is seen that for the thoria rod, the asymptotic uranium concentration is about 16 g/kg. Fissile content remains above 90% for almost 1000 days. In case of natural or enriched uranium the asymptotic plutonium production is barely 6 g/kg. The fissile fraction

rapidly decreases. In fact, the absolute fissile content remains saturated below 4 g/kg after 800 days for natural or enriched uranium. These characteristics are due to relatively high absorption cross section of  $^{232}\text{Th}$  and low one for  $^{233}\text{U}$  in comparison to those of  $^{238}\text{U}$  and  $\text{Pu}$  isotopes respectively. It is interesting to note that  $^{233}\text{U}$  achieves an asymptotic stable concentration that would stay for even some years. This study shows that irradiation of  $\text{ThO}_2$  rods is far superior to that of  $\text{UO}_2$  rods, since one is able to produce higher amounts of a better fissile material for the same neutron fluence.

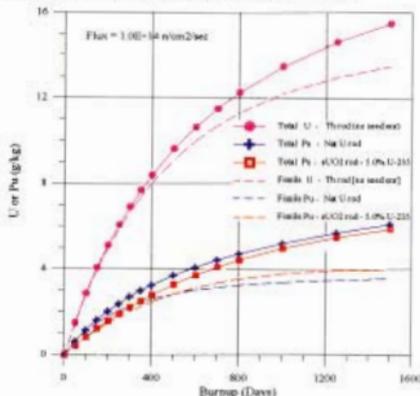


Fig. 1 Production of U from Th or Pu from U as a function of burnup

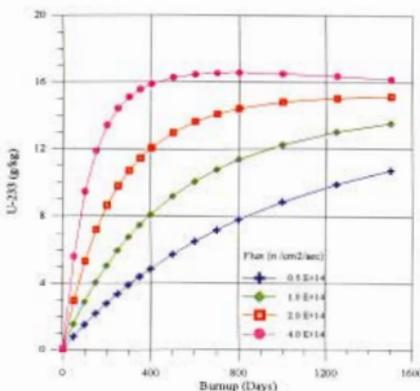


Fig. 2 Production of U-233 with burnup for pure thorium cluster at various flux levels

Fig.2 shows the formation of  $^{233}\text{U}$  when a ring of thorium rods are placed in  $\text{D}_2\text{O}$  and are subjected to irradiation at different flux levels of  $0.5$  to  $4 \times 10^{14}$   $\text{n/cm}^2/\text{sec}$ . It is seen that a flux level of about  $2 \times 10^{14}$   $\text{n/cm}^2/\text{sec}$  is optimal for a fairly rapid production of  $^{233}\text{U}$ .

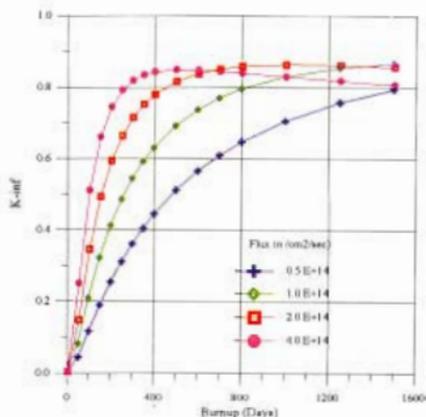


Fig.3 Variation of  $K_{\infty}$  with burnup for pure thorium cluster at various flux levels

Fig.3 gives the reactivity behaviour of these thorium rods. The  $K_{\infty}$  curve closely follows the  $^{233}\text{U}$  variation. The initial sharp increase of reactivity of thorium rods and asymptotic zero slope suggests that one can use it to judiciously balance the fall in reactivity of conventional fuel rods. By such a scheme one can achieve a nearly flat reactivity characteristic for a fairly long duration, in a prospective core design.

If a batch mode of refueling is adopted, the core excess reactivity can be tuned to be nearly zero for full power operation during the entire fuel cycle. In this way, the good features of low excess reactivity in a PHWR and the annual batch mode of refueling practiced in LWRs can both be combined.

Use of some enriched seed zone is necessary to be able to accommodate significant number of fertile thorium rods. However, light water moderator cannot be considered because LWRs are too compact and the thermal flux

level is typically of the order of  $10^{13}$  n/cm<sup>2</sup>/sec, which is too low for a rapid <sup>233</sup>U production. D<sub>2</sub>O moderator is advantageous since it offers large space where the thermal neutrons get accumulated, without the risk of they being absorbed by the moderator itself. High thermal flux of the order of  $10^{14}$  n/cm<sup>2</sup>/sec is possible with D<sub>2</sub>O moderator. Use of enriched fuel and D<sub>2</sub>O moderator will result in a large core excess reactivity. However, instead of wastefully absorbing the neutrons in some control materials, if one can fruitfully use them for fertile captures in as many thorium rods as possible, one would achieve a core design with the best neutron utilization.

A segregated seed and blanket type fuel assembly design is preferable. Within a fuel assembly, a ring cluster form of arrangement is useful since it avoids the corners or edges of a typical square or hexagonal type fuel assembly design which would demand a 2-D or even a 3-D enrichment distribution. In order to maximize the flux incident on thorium rods, they should be placed in the outermost ring of a fuel assembly. When thorium rods without any feed enrichment are placed along with some seed fuel rods, it is observed that the thorium rods generate only 1/1000<sup>th</sup> of the power produced in seed rods, since there is scope for only fast fission at zero burnup. Eventually, when adequate <sup>233</sup>U is formed, they would generate power comparable to the seed fuel rods. This poses serious problem for thermal hydraulic design. To alleviate this problem, it is suggested that a full batch load of thorium clusters should be irradiated, in the same reactor, for one fuel cycle duration of about an year. The thorium rods would behave essentially as a kind of absorber rods during this period. In one year, they would accumulate nearly 60% of the asymptotic <sup>233</sup>U content. Fresh enriched fuel rods in ring cluster form are then assumed to be inserted into the above pre-irradiated thorium rods also in ring cluster form. The power mismatch between the seed and irradiated thorium rods is then found to be acceptably low. These integrated (fresh seed + irradiated ThO<sub>2</sub>) fuel assemblies can then be irradiated for several fuel cycles before being discharged together from the reactor.

For freezing the geometry of the fuel assembly and of the core, some additional considerations should be spelt out.

D<sub>2</sub>O moderator necessitates a pressure tube type design. Vertical core design is preferred to avoid the irradiation creep and sagging problem of horizontal pressure tubes. A hexagonal fuel assembly lattice configuration is chosen for the core. The fuel rods are assumed to have an active length of 360cm. The fuel rod diameter is chosen to be around 1 cm. Boiling H<sub>2</sub>O coolant is preferable since the pressure can be lower in comparison to pressurized D<sub>2</sub>O coolant. This offers the advantage of lower pressure tube thickness for a given tube diameter and better thermodynamic efficiency. D<sub>2</sub>O leakage and accompanied tritium release are also avoided. However, the design should ensure negative coolant void reactivity. It is well known that when the D<sub>2</sub>O coolant of a CANDU type PHWR is replaced by boiling H<sub>2</sub>O, the reactivity due to Loss of Coolant Accident (LOCA) becomes much more positive because of high absorption by H<sub>2</sub>O. This problem has been overcome by choosing a fairly large size fuel cluster. A 127 rod fuel cluster, instead of the standard 19 or 37 rod fuel cluster of PHWRs, is seen to give a zero or negative void reactivity. It was noted that the change in resonance escape probability due to coolant voiding is sufficiently negative and hence nullifies the positive contributions from the other factors. For boiling H<sub>2</sub>O coolant, this phenomenon is more marked than for D<sub>2</sub>O coolant. This is due to the fact that boiling H<sub>2</sub>O not only acts as coolant but makes significant contribution to moderating of neutrons as well.

A large size fuel cluster also offers the advantage of loading a large number of thorium rods in the outermost fuel ring facing directly the thermal neutrons incident from the moderator. In the 127 rods cluster, there will be 91 seed fuel rods distributed in six rings. It was observed that there is a very large power peaking of about 1.6 among the seed fuel rings themselves. Even by using differential enrichment, the peaking factor could not be reduced. On the contrary, there was significant loss of reactivity even after using much higher average

enrichment. The solution to this problem was obtained by replacing the central 37 rods with some pure moderator block like BeO. There would then be just two rings of seed fuel rods. There are 24 rods in the inner ring and 30 rods in the outer one. The power peaking can be easily reduced by choosing the enrichment of the inner ring to be 20% more than that of the outer one. Enriched  $UO_2$  ( $eUO_2$ ) has been considered as seed in the present design. One can consider other types of seeds as well. As a means of increasing the power share of the  $ThO_2$  rods, we chose the diameter of thorium rods to be ~25% more than that of  $UO_2$  rods. The number of thorium rods was reduced from 36 to 30. This has helped in increasing the thorium volume fraction and maintaining the pin-to-pin gap between  $ThO_2$  rods. Each fuel cluster would therefore contain 54  $eUO_2$  rods and 30  $ThO_2$  rods.

The reactor power is chosen as 600 MWe. This power is internationally regarded as optimal with respect to safety and economy. It is not too small for quantum addition of each unit. When the scenarios involving beyond design basis accident are to be considered, the risk for common public can be shown to be acceptably low in comparison to larger size power reactors. For a typical thermodynamic efficiency of 32%, the design thermal power becomes 1875 MWt. Using a typical average linear heat rating of about 170 w/cm and an active core height of 360 cm, the number of fuel assemblies with 84 fuel rods per assembly is found to be ~360.

### ATBR Core Design

The finalized core and fuel assembly design parameters are described in Table-I. Fig.4 gives the cross sectional view of the (54  $eUO_2$  + 30  $ThO_2$ ) rods fuel cluster. A five batch-refueling scheme is assumed, giving 72 fuel assemblies per batch. Apart from the 360 fuel assemblies, the core is designed to accommodate at least one batch size of the (30 natural  $ThO_2$  rods) clusters. Additional 19 natural  $ThO_2$  clusters are also accommodated in the core to cater to xenon over-ride and other control maneuvers. The core thus contains 360 (54 $eUO_2$ +30 $ThO_2$  rods) fuel assemblies + 91 (30 natural

$ThO_2$  rods) clusters. The locations of the thorium clusters are fixed and are assumed to be uniformly distributed at twice the lattice pitch of 64 cm. They span almost the entire core except the peripheral one or two layers of fuel assemblies. Fig.5 presents the optimized loading pattern for equilibrium core. It shows a curious star-like shape. Loading of the fifth cycle assemblies at the core peripheral layer enables low neutron leakage.

The reactivity, relative power and burnup accumulation in the three fuel rings of the ATBR fuel cluster were studied in detail with the lattice burnup code CLUB and 69 group WIMS cross section library.

Interesting features were observed. It was seen that in comparison to an all  $eUO_2$  fuel of average enrichment, ATBR fuel cluster exhibits a relatively flat reactivity characteristic.  $K_{\infty}$  is much lower at zero burnup, but is significantly higher near discharge burnup. The  $K_{\infty}$  of mixed ATBR fuel cluster is increased when the fluence in thorium rods is enhanced. The relative power peak of 1.28 occurs in the middle  $UO_2$  ring at zero burnup, and shifts to the thorium ring later. At 35 GWd/T, the thorium ring relative power is 1.25. For the assembly average burnup of 32 GWd/T, the thorium rods achieve about 35 GWd/T, while the  $eUO_2$  rods achieve a burnup of about 31 GWd/T. It is worth noting that the burnups in the three fuel rings get more or less evened out at the time of discharge. In fact, thorium rods with no external feed enrichment achieve somewhat higher burnup.

The core calculations are done by a new code TRISUL (an acronym derived from Thorium Reactor Investigation with Segregated Uranium Loading) developed for the purpose. It uses finite difference diffusion theory method with hexagonal right prismatic meshes. TRISUL is a coupled neutronics cum thermal hydraulics code. Three dimensional void or steam fraction distributions are calculated with a thermal hydraulics model that is similar to the one used for Tarapur BWR with square fuel assembly geometry.

Table I

## ATBR CORE AND FUEL DESIGN PARAMETERS

|   |                      |   |                 |                  |
|---|----------------------|---|-----------------|------------------|
| Reactor Power   | (MWe)                | 600 (1875 MWt)  |                 |                  |
| Total core flow   | (tonnes/hr)          | 27 X 10 <sup>6</sup>  |                 |                  |
| Average heat rating   | (w/cm)               | 172   |                 |                  |
| No. of rods in a fuel cluster                                 |                      | 84 (54 seed rods + 30 ThO <sub>2</sub> fertile rods)  |                 |                  |
| No. of rods in a natural thoria cluster                       |                      | 30 ThO <sub>2</sub>   |                 |                  |
| No. of (U-Th) fuel clusters in the core                       |                      | 360   |                 |                  |
| No. of natural thoria clusters in the core                    |                      | 72 (fixed) + 19 (moveable) = 91   |                 |                  |
| No. of fuel assemblies per batch                              |                      | 72  |                 |                  |
|   | Ring →               | Inner   | Middle          | Outer            |
| No. of fuel rods  |                      | 24  | 30              | 30               |
| Pitch circle dia  | (cm)                 | 10.4  | 13.0            | 15.8             |
| Fuel Material   |                      | UO <sub>2</sub>   | UO <sub>2</sub> | ThO <sub>2</sub> |
| Feed enrichment   | (%)                  | 5.4   | 4.5             | insitu           |
| Fuel density (smeared with airgap)                            | (g/cm <sup>3</sup> ) | 10  | 10              | 9                |
| Clad Material   |                      | Zr-Nb(1%)   | Zr-Nb(1%)       | Zr-Nb(1%)        |
| Clad ID/OD  | (cm)                 | 1.00/1.14   | 1.00/1.14       | 1.26/1.40        |
| Assembly Lattice Pitch (hexagonal)                            | (cm)                 | 32  |                 |                  |
| Average Fuel Temperature                                      |                      | 600°C   |                 |                  |
| Saturated Coolant Temp. (Boiling H <sub>2</sub> O – 1015 psi) |                      | 286°C   |                 |                  |
| Coolant Inlet Subcooling                                      | (Kcal/kg)            | 7 to 20   |                 |                  |
| Central Moderator Block                                       |                      | BeO   |                 |                  |
| ID/OD of Central Moderator Block                              | (cm)                 | 1.0/9.0 (inclusive of Zr-liner)   |                 |                  |
| Pressure Tube (PT) Zr-Nb (2.5%) ID/OD                         | (cm)                 | 17.6/18.7   |                 |                  |
| Calandria Tube (CT) Zr-2 ID/OD                                | (cm)                 | 20.4/20.7   |                 |                  |
| Moderator Material & Temperature                              |                      | D <sub>2</sub> O - 80°C   |                 |                  |
| Radial D <sub>2</sub> O reflector thickness                   | (cm)                 | 60 to 70  |                 |                  |
| Axial D <sub>2</sub> O reflector thickness                    | (cm)                 | 60  |                 |                  |
| Calandria Tank Size   |                      | 8.4 m dia X 4.8 m height  |                 |                  |
| Control System  |                      |   |                 |                  |
| Fast shutdown + Shutdown hold                                 |                      | 1) Injection of liquid poison in dry tubes<br>2) Stainless steel rings/rods in interchannel space<br>3) Moderator dump during refueling outages |                 |                  |
| Xe over-ride  |                      | Movable ThO <sub>2</sub> clusters (19 nos.)   |                 |                  |

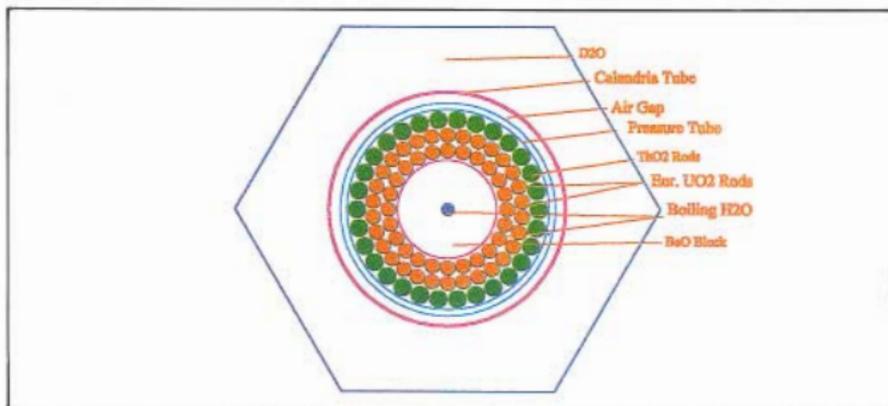


Fig. 4. ATBR - (54UO<sub>2</sub> + 30ThO<sub>2</sub>) rods fuel cluster

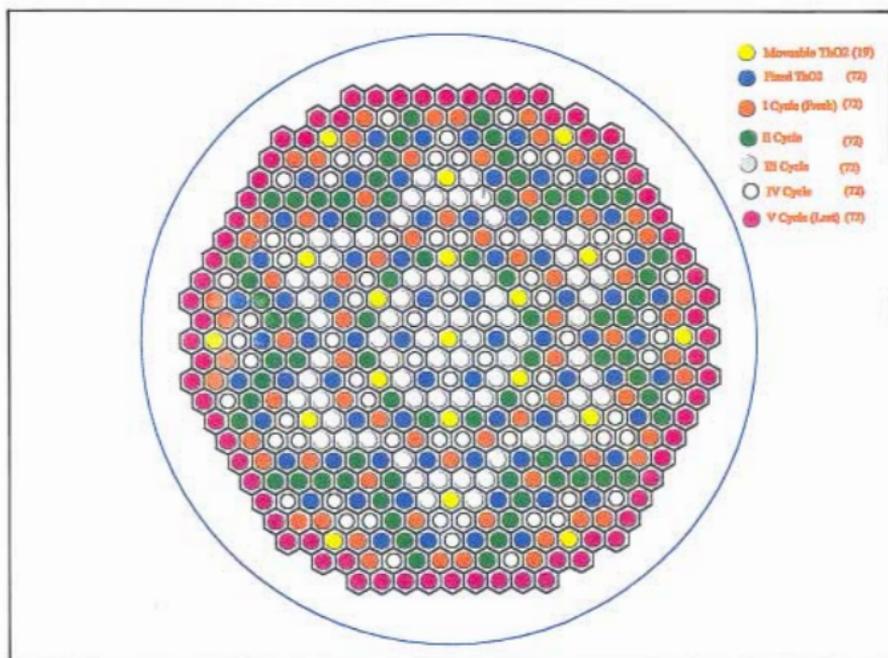


Fig. 5. ATBR core - 360 (54UO<sub>2</sub> + 30ThO<sub>2</sub>) fuel assemblies + 91 (30 natural ThO<sub>2</sub>) clusters

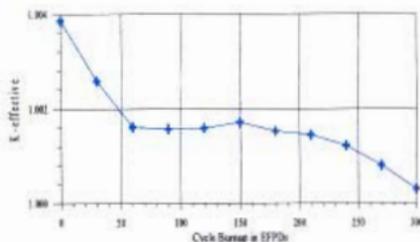


Fig.6 ATBR core K-eff vs cycle burnup

Fig.6 gives the variation of  $K_{eff}$  with cycle burnup. All 91 ThO<sub>2</sub> clusters are fully in during the entire fuel cycle. The overall spread in  $K_{eff}$  is only  $\pm 2$  mk for a cycle length of 300 effective full power days (efpds).

Fig.7 gives the radial, axial and overall peaking factor variation with cycle burnup. It is remarkable to note that the <sup>235</sup>U burning and <sup>235</sup>U production are judiciously balanced so that the  $K_{eff}$  and power distributions are intrinsically maintained within a narrow band without any external maneuvers of reactivity control mechanism like soluble boron, burnable poison or control rods.

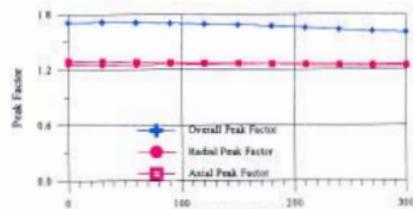


Fig.7 ATBR Core - Variation of Peak Factors with Cycle Length

The core excess reactivity in cold xenon free state is less than 5% throughout the fuel cycle. The control requirement for hot shut down and shutdown hold are seen to be less than 2.5% and 6% respectively, including a shutdown margin of about 1%. The xenon override reactivity for restart after shutdown following a prolonged operation at full power is about 20 mk for ATBR as against 110 mk for PHWR. The 19 moveable thorium clusters can meet this if they are all kept fully IN during

normal full power operation. The equilibrium core was seen to exhibit a negative or near zero void reactivity coefficient.

Table-II gives the summary of fuel contents in the fresh charge of fuel assemblies and the discharged ones for a batch size of 72 assemblies and for a cycle length of 300 efpds. The fresh fuels contain thorium rods irradiated for one cycle duration. They contain <sup>235</sup>U bred insitu. The discharge burnup is around 32 GWD/T. The discharge fuel contains 10R kg of <sup>235</sup>U and 132 kg of total U in thorium rods, 72 kg of total Pu with 54 kg of fissile Pu in uranium rods. The unburnt <sup>235</sup>U is 139 kg.

In order to assess the fuel economy of the ATBR concept, a comparison is made with the serial VVER-1000 MWe reactor that is proposed to be built in India with Russian collaboration. A quantum electricity generation of 10GWe for 30 years with 300 efpds operation per year has been considered. Table-III gives the initial <sup>235</sup>U feed requirement, equivalent mined uranium, and the contents of discharged fuel for the above quantum of electric power production.

Table II

### FUEL CONTENTS IN FRESH AND DISCHARGED FUELS

#### Equilibrium Core - 72 Assemblies Per Batch 5 Batch Fueling - Cycle length - 300 efpds

| Parameter  | Fuel Contents in 72 Assemblies |        |
|--|--------------------------------|--------|
|  | BOL                            | EOL    |
| Weight of Thorium                                  | kg 7609.5*                     | 7343.2 |
| Weight of <sup>235</sup> U in Thorium              | kg 56.4                        | 108.5  |
| Weight of <sup>235</sup> U+ <sup>235</sup> U in Th | kg 56.6                        | 111.7  |
| Weight of U-tot in Thorium                         | kg 59.1                        | 132.1  |
| Weight of U-tot in Uranium                         | kg 9689.7                      | 9242.6 |
| Weight of <sup>235</sup> U in Uranium              | kg 474.8                       | 139.3  |
| Weight of Total Pu in U                            | kg 0                           | 72.4   |
| Weight of Fissile Pu in U                          | kg 0                           | 54.1   |
| Average Discharge burnup                           | MWD/T ---                      | 31600  |

BOL - Beginning of life EOL - End of life

\*Thorium is irradiated for one cycle duration

**Table III**  
**COMPARISON OF FUEL REQUIREMENTS –**  
**10 GWe FOR 30 YEARS - 300 EFPD PER YEAR**  
**ATBR VS VVER**

| Parameter  |                            | ATBR  | VVER  |
|--|----------------------------|-------|-------|
| Reactor Power                                      | Mwe                        | 600   | 1000  |
|  | MWt                        | 1875  | 3000  |
| Average Feed enrichment                            | (%)                        | 4.9   | 4.4   |
| Fuel per year                                      | U in eUO <sub>2</sub> (T)  | 9.69  | 22.2  |
|  | Th in ThO <sub>2</sub> (T) | 7.67  | ---   |
| Fissile per year (300 efpds) <sup>235</sup> U (kg) |                            | 474.8 | 976.8 |
| Fuel for 10 GWe-30 years                           | eUO <sub>2</sub> (T)       | 4845  | 6660  |
|  | ThO <sub>2</sub> (T)       | 3835  |       |
| Equivalent mined uranium                           | (T)                        | 45540 | 55940 |
| Fissile input for 10 GWe-30 years                  | (T)                        | 237.4 | 293   |
| <b>Output Materials</b>                            |                            |       |       |
| Total - Pu   | (T)                        | 36.2  | 68.4  |
| Fissile Pu   | (T)                        | 27.0  | 47.9  |
| Unburnt <sup>235</sup> U                           | (T)                        | 69.7  | 78.2  |
| <sup>233</sup> U                                   | (T)                        | 54.3  | --    |
| Total U in thorium                                 | (T)                        | 66.0  | --    |
| Total Output                                       | (T)                        | 171.9 | 146.6 |
| Ratio of output to input                           |                            | 0.72  | 0.50  |

The ATBR design considers the loading of enriched uranium and thorium in the ratio 50:50 by weight in each cycle (including additional 19 ThO<sub>2</sub> clusters). ATBR requires 27% less enriched fuel, 19% less equivalent mined uranium in comparison to a VVER-1000 MWe reactor design. It produces 47% less Pu. There is a sizeable production of <sup>233</sup>U. The ratio of gross seed output to input is 0.72 for ATBR and 0.5 for VVER. The <sup>233</sup>U production in the additional 19 ThO<sub>2</sub> clusters has not been included in the output of ATBR. It is also possible to load some ThO<sub>2</sub> clusters near core-reflector interface to breed more <sup>233</sup>U. This table demonstrates that the ATBR fuel cycle shall be far more economical and is capable of using wider base of nuclear materials, viz. uranium and thorium.

### Summary

The salient features of the ATBR core can be summarized as follows:

- At full power operation, there is no need for external reactivity control mechanisms.

- The  $K_{eff}$  and the power distribution are intrinsically maintained constant.
- Xenon induced power oscillations are unlikely since thorium rods would curb any local power or flux surge.
- The xenon override reactivity is about 20mk for full power operation. This can be provided by withdrawal of 19 moveable thorium clusters.
- The ATBR core is inherently safe since the most common transients involving reactivity excursions like rod ejection, LOCA, cold water addition etc are either absent or far less severe for this reactor.
- There is a sizeable production of <sup>233</sup>U which is intrinsically proliferation resistant due to formation of the isotope <sup>233</sup>U and high gamma emitting daughter products thereof.
- There is no need for fuel reprocessing, if eUO<sub>2</sub> is available. Even in the closed fuel cycle options, the reprocessing load would be nearly halved, since 50% of the core can continue to use fresh ThO<sub>2</sub> in its natural form.
- Other types of seed zones employing either <sup>233</sup>U in natural uranium/thorium or Pu in natural uranium/thorium are possible. Of these, the option of <sup>233</sup>U in thorium has the potential of being developed into a thermal breeder.
- The ATBR concept has an overall better economic, operational and safety characteristics in comparison to any power reactor design that is currently operational.
- The present work is a theoretical study with the cross section data and calculation tools available with the author. Some uncertainties in the calculated results are admittedly present. Notwithstanding the above, it is claimed that the proposed ATBR has indefatigable design features which are convincingly superior to those of the power reactor designs prevalent today. It is mandatory to perform some physics experiments to refine the design parameters delineated in this paper. The emphasis is laid more on the design philosophy rather than on the design parameters themselves.

# INDIA'S ISOTOPE PROGRAMME IN SERVICE OF MAN AND SOCIETY

## Introduction

BARC has pioneered the production and application of radioisotopes in India for the past four decades.



Fig.1 Isotope Processing Laboratories at BARC

The isotope programme in BARC involved reactor production of over 100 isotopes, their processing into various physical and chemical formulations for use in medicine, industry, hydrology, agriculture and research, as well as development of applications in healthcare, industrial quality control and processing, and in water resources development and management. The success of this programme over the years can be gauged from the following :

- Nearly 150 hospitals are practising nuclear medicine largely with radiopharmaceuticals supplied by the Board of Radiation and Isotope Technology (BRIT).
- About 500 radioimmunoassay (RIA) laboratories are manned exclusively by BARC-trained personnel, using RIA kits developed by BARC and supplied by BRIT.
- Nearly 1 million isotope-based patient investigations per year are done in hospitals all over in India.
- Cobalt-60 radiation sources for the 150 teletherapy units for cancer treatment are provided from BARC/BRIT.
- India has been practising radiation sterilisation of medical products for 25 years. BARC has been

recognised as a centre of excellence in this field by the International Atomic Energy Agency. India operates three plants: Isomed, BARC, Mumbai; Rashmi in Kidwai Memorial Institute of Oncology, Bangalore; and SARC in Sriram Institute of Industrial Research, Delhi. The fourth one RAVI at the Defence Laboratory, Jodhpur, which was also designed and built under BARC supervision, is a multi-purpose irradiation facility including radiation sterilisation.

- BARC has carried out more than 350 isotope tracer and sealed source investigations for industrial troubleshooting and process control. BARC's expertise in this field was provided to almost all countries in the Asia and Pacific region including China and South Korea, and also to Jordan.
- Application of radioisotopes for industrial NDT has a wide base in India with 500 organisations/industrial plants having the requisite facilities out of which 350 offer commercial service.
- The demand for BARC's expertise for gamma scanning of industrial process columns, particularly in the petrochemical industry, is ever increasing.

Some examples of success in specific cases of isotope application are given below:

*Treatment of thyroid disorders :* Radioactive iodine (I-131) supplied by BARC and BRIT provides effective and inexpensive treatment of thyroid disorders like hyperthyroidism and thyroid cancer. A large number of medical centres in India offer this service, of which 20 centres have facilities to treat thyroid cancer.

*Terminal cancer treatment :* BARC has made available inexpensive radioactive phosphorus (P-32) as phosphate to effectively reduce bone pain in terminal cancer patients. In addition to BARC's own Radiation Medicine Centre, 20 other centres carry out this procedure on a regular basis.

*Development of Cs-137 brachytherapy sources* : BARC has developed technology for incorporating Cs-137 in borosilicate glass in non-leachable form and for preparation of sources for brachytherapy of cancer.

*Location of leaks in the 140 km Virangam-Koyali underground oil pipeline* : Using a novel "in-pig" instrumentation technique, BARC was able to detect leaks at the joints of the pipeline using isotope tracers.

*Isotope tracer application in Kudremukh Iron Ore Project* : When the Lakya dam site was under construction as part of the Kudremukh Iron Ore Project in Karnataka, the detection of a large fissure in the cut-off trench brought the whole project to a stop as the extent of the fissure had to be established from the point of view of safety of the future dam.

Radioactive isotope tracer was used to examine the fissure after the underground system was brought to a steady state by continuous water injection. The tracer data conclusively proved that (i) the Lakya river was not connected to the fissure, and (ii) the fissure was localised and was not part of any extended fault.

*Isotope investigation in Salal Hydroelectric Project* : During the construction of a tail race tunnel through a dolomitic hill for the Salal Hydroelectric Project in Jammu, several seepages were encountered. The 2.5 km long tunnel was to carry tail waters from the power station back to the river channel.

By environmental isotope investigation, it was established that the seeping waters were 10 to 15 years old rainwaters trapped in the dolomite and were released whenever a sheer zone was pierced during tunneling.

*Isotope tracer application in Jhamarkotra Mines, Udaipur, Rajasthan* : As the mining in the open cast rock phosphate mine at Jhamarkotra reached the groundwater table, it needed to be dewatered. Intense dewatering at the rate of about 1000m<sup>3</sup>/h resulted in only a marginal fall in the water level leading to suspicion of water ingress from reservoirs such as Jhamri dam, Uday Sagar lake, Bhaghdara reservoir and Jaisamand lake which were all within 9 km from the mine.

The conclusions from the isotope investigation were : (i) the reservoir waters were not connected to the minewaters, and (ii) the pumped mine water appeared to be recycled to the groundwater due to poor drainage.

*The water ingress problem at the Naval Dry Dock, Visakhapatnam, solved using isotope tracer* : Radioactive tracer study using gold-198 to locate the entry of ingress of water into the Matsya dry dock in the Naval Dockyard complex at Visakhapatnam clearly showed the possibility of ingress entry in a narrow band of 3 m width. The ingress entry was then physically located and plugged.

*Cauvery delta water problem* : To augment the supply of groundwater for irrigation in the Cauvery delta, artificial recharge of the local groundwater bodies was considered for implementation.

An environmental isotope and geochemistry investigation clearly showed that the shallow and deep groundwater zones were not interconnected except at isolated places where the clay layer between the two zones was either absent or very thin.



Fig. 2 Radiotracer study on sediment transport on sea bed

*Drinking water problem on the Orissa Coast* : Isotopes were used to find out the reason why groundwater supply in thousands of hand pumps in coastal Orissa turned brackish.

Salinity in groundwater was due to the ancient transgressions of the sea and subsequent entrapment of the seawater in the deltaic sediments. A fresh water

body was located at a depth of 100 m which received modern recharge and which could be developed as a potential source for fresh water.

**Sediment transport off Karwar Coast :** Radioisotope tracer investigations were carried out to determine the direction of bedload transport off Karwar Coast as part of development of the 3<sup>rd</sup> Naval base (Project Sea Bird). The disposal of silt dredged from the proposed turning bay (Binge Bay) at a distance of 4 km south-west of Anjadiv island was found suitable, as the sediment transport at that point was away from the coast.

**Gamma scanning of vacuum distillation column at Haldia Refinery :** The vacuum distillation column at Haldia was designed to produce vacuum gas oil, light oil, heavy oil and short residue. After revamping a few years back, the column produced only spindle oil and nothing else. By carrying out 'on-line' gamma scanning at different zones of the vacuum distillation column, the defects in the column were identified and rectified, resulting in substantial improvement in the column performance.



Fig. 3 ILU-6 Electron Beam Accelerator in Trombay. This is being shifted to BRIT, Vashi, to make it easily accessible to the industry.

**NDT of concrete structures :** Isotope radiography was effectively used to assess the health of reinforcement bars in an old telephone exchange building in Mumbai when it was planned to build an additional floor.

**Radiation cross-linked PE 'O' rings for drum closures :** Radiation cross-linked PE 'O' rings are inexpensive substitutes for fluoro-polymer rings as gaskets in drum closures for high temperature operations. High quality PE 'O' rings produced in millions by the Indian industry are sent to BARC for radiation cross-linking by electron beam using the 2MeV 20 kW ILU-6 Electron Beam Machine.

**Colour enhancement of diamonds and gemstones :** India is a world leader in the export of precious and semi-precious stones.

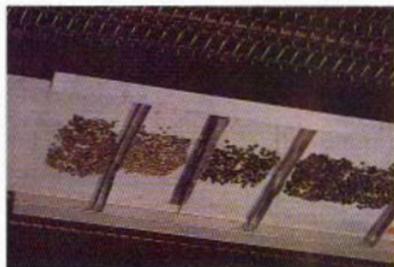


Fig. 4 Diamonds being irradiated under electron beam for colour enhancement

BARC has standardised the procedure for imparting blue, blue-green or green colour to diamonds by electron beam irradiation. The service is now offered on a commercial basis.

#### Conclusion

Radioisotopes produced in BARC are used in the development of the country's healthcare, industry and water resources. The Indian medical community, industrial establishment and the hydrology professionals are very keen to take an ever increasing advantage of isotope technology developed by BARC.

## SSSF AT TARAPUR COMMISSIONED

On July 26, 1999, Solid Storage Surveillance Facility (SSSF), Tarapur, was inaugurated by Dr Anil Kakodkar, Director, BARC, by emplacing a storage unit called overpack into the designated location of SSSF. The overpack consists of two stainless steel canisters each containing about 100 kg of vitrified solid waste produced in the nearby Waste Immobilisation Plant. The storage unit was brought in a 17 tonne lead-shielded cask loaded on a low-bed trailer. The operation of lowering of the storage unit was done using indigenously designed and fabricated remotely operated material handling gadgets and equipments.



*Dr Anil Kakodkar, Director, BARC, inaugurating SSSF, Tarapur, by emplacing an overpack in the thermal vault.*

In his inaugural address, Dr Kakodkar lauded the efforts of the staff of Waste Management Division in the commissioning of SSSF, Tarapur. He termed this as a

shining example of the team work of scientists and engineers involved in design, construction and commissioning of this unique facility. Briefing the members of the Press after the inauguration, Dr Kakodkar mentioned it as a mile-stone in the history of Indian Atomic Energy Programme. He added that there were very few countries in the world who had so far set up such storage facility which is an important step prior to disposal of high level radioactive waste in the deep geological repository.



*Dr Anil Kakodkar, Director, BARC, briefing the Press after the inauguration of SSSF, Tarapur*

SSSF consists of an underground hydraulic vault which houses two inner thermal vaults. The thermal vaults are designed to store about 1700 overpacks for a period of 20-30 years before these are transported to final deep geological repository. Each overpack is designed to hold 1.2 million curie of radioactivity dissipating about 3 kW of heat due to radioactive decay. The removal of decay heat is achieved by natural air convection induced by a 100 m tall stack. This is an inherently self-regulating system capable of taking care of changes in the heat load.

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## CONTRIBUTIONS FOR KARGIL

An amount of Rs. 34.34 lakhs has been collected as contribution to the PM's Relief Fund from the officers and staff of BARC stationed at Trombay and Tarapur, for relief and rehabilitation of the armed forces and their families affected by the Kargil conflict at the border.

## A NEW PROCESS DEVELOPED FOR PURIFICATION OF AMMONIUM DI-URANATE CAKE

Ammonium diuranate cake (ADUC) is produced as a by-product of the thorium extraction plant at Orissa Sand Complex (OSCOM) of Indian Rare Earths Ltd. (IREL), Orissa. The ADUC recovered in the process, however, does not meet nuclear specifications on account of its higher thorium and rare earth contents. The problem of developing a modified process flowsheet to purify this ADUC to the nuclear grade was referred to Uranium Extraction Division (UED), Materials Group, BARC. Laboratory scale studies carried out at the Quality Control Section of UED showed that, with a few minor changes in the solvent extraction process being followed at OSCOM, it would be possible to attain the requisite purity of the ADUC. Scientists of UED subsequently demonstrated the reproducibility of the process on a bench scale at OSCOM and also provided training for the improved analytical technique for thorium.

Based on the encouraging results of the demonstration, a MOU has been signed between BARC and IREL for transfer of technology for the purification of ADUC to the nuclear grade, acceptable for the production of uranium metal. The noteworthy merit of this modified flowsheet is that it can easily be implemented, without any major change, in respect of the process being presently followed at OSCOM. Plant scale operations will commence at OSCOM to purify the entire stock of ADUC in the near future. For the execution of this collaborative

work, IREL will provide the infrastructure and human resources support at OSCOM, Orissa. Scientists from UED, BARC, will participate during the commissioning and successful implementation of the process.

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## SAFETY ASSESSMENT MODEL FOR RADIOACTIVE WASTE DISPOSAL FACILITIES

Environmental Assessment Division of BARC has developed a probabilistic safety assessment model for assessing the performance of near surface disposal facilities for low-level radioactive waste. Two modes of disposal such as single dump and multiple dump are considered in the model. The model can generate the radioactivity release rate into the geosphere, the radionuclide concentration in ground water, the annual effective dose to a member of the critical group through drinking water pathway and the risk to the critical group due to waste disposal practice. The maximum annual effective dose to a member of the critical group for the reference level parameters works out to be  $1.2 \times 10^{-2}$  mSv. The geometric mean of annual effective dose to a member of the public generated through the uncertainty analysis is about 3 times higher than that of the reference level. However, the most probable annual effective dose to a member of the critical group obtained through the uncertainty analysis ( $2.8 \times 10^{-4}$  mSv) is much lower than the dose obtained for the reference level. (*Environmental Modelling & Software*, 14, (1999) 447-460)

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*Editorial Management : T.C. Balan; Computer graphics & layout : P.A.S. Warriyar.*

*BARC Newsletter is also available at URL:<http://www.barc.ernet.in>*

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