Thorium utilization for sustainable supply of nuclear energy

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Abstract. A brief summary of the Indian research and development activities related to Thorium utilization for sustainable supply of nuclear energy is provided in this paper.

1. Introduction

For long time energy security and large scale utilization of Thorium resources in the country, India has a rather unique three stage nuclear power programme. The Pressurized Heavy Water Reactor (PHWR) based on natural Uranium and heavy water belongs to the first stage. The Fast Breeder Reactor (FBR) which will use the ^{239}Pu produced from the first stage and breed ^{233}U from natural ^{232}Th, constitutes the second stage. During this stage ^{232}Th will be used in the blanket. The third stage reactor system will essentially use the ^{232}Th and ^{233}U bred from the second stage FBR. Further, during this stage ^{232}Th will be used in the core. This is essentially the roadmap of India for sustainable supply of nuclear energy [1]. A schematic of the Indian three stage nuclear power programme is shown in figure 1.

Further, India is committed to closed fuel cycle, to use optimally the resources in the country and also to reduce the amount of long lived nuclear waste to be stored in the repository. At present, there are 20 reactors operational generating a total power close to 4800 MWe. Two Light Water Reactors (LWR) of capacity 1000 MWe each, are at an advanced stage of completion. Plans are also underway to build 700 MWe reactors. India is also building a prototype fast breeder reactor of capacity 500 MWe. Soon India will embark on building the first large size ^{232}Th- ^{233}U (^{239}Pu) based Advanced Heavy Water Reactor (AHWR) of 300 MWe [2]. To sum up, India has an ambitious plan of crossing 20000 MWe from nuclear energy by 2020.

2. Thorium Utilisation for sustainable supply of nuclear energy

2.1 Thorium as a fuel for reactors

While it is estimated that the Uranium reserves in the country could be around 60,000 metric tonnes, the amount of Thorium available from the shallow sands alone is expected to be close to 1000,000 metric tonnes. Unlike Uranium which has a small amount of fissile isotope ^{235}U (0.7%) , the element Th does not have any fissile isotope. Hence to use Th as a reactor fuel, it is necessary to produce first ^{233}U from neutron capture of ^{232}Th, followed by decay of ^{233}Th. In the Indian programme, it is proposed to convert ^{232}Th to ^{233}U during the second stage using the FBRs. The Thorium as a fuel has several

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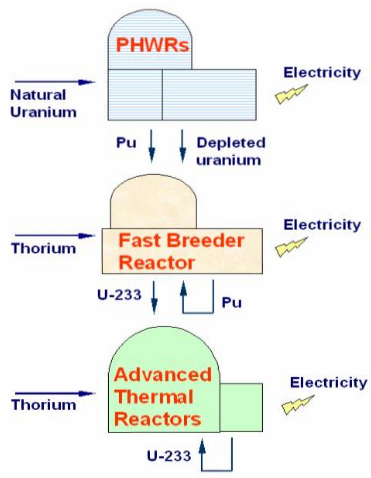

Figure 1. Schematic of the three stage nuclear power programme of India

advantages when compared to Uranium. Uranium and Thorium oxides used as the fuel have different melting points. Thorium oxide has a higher melting point (3300 ° C) when compared to Uranium oxide (2865 ° C) and has a better thermal conductivity. Further, it has a lower fission gas release and good radiation resistance along with dimensional stability. Thorium oxide also exhibits better chemical stability [3].

2.2 Superior nuclear properties of ^{232} Th and ^{233}U

The thermal neutron absorption cross section for ^{232}Th is 7.4 barns and this value is significantly higher than that for ^{238}U (2.7 barns) (in practice one may have to use cross sections over a range of energies). The minor actinide production is also significantly less when using ^{232}Th/^{233}U as a fuel. However, in processing ^{233}U from neutron irradiation of ^{232}Th, one has to worry about the presence of ^{232}U (produced through the (n,2n) reaction on ^{233}U). The problem with ^{232}U is that it has a nucleus emitting high energy gammas in the decay chain. The average number of fission neutrons produced

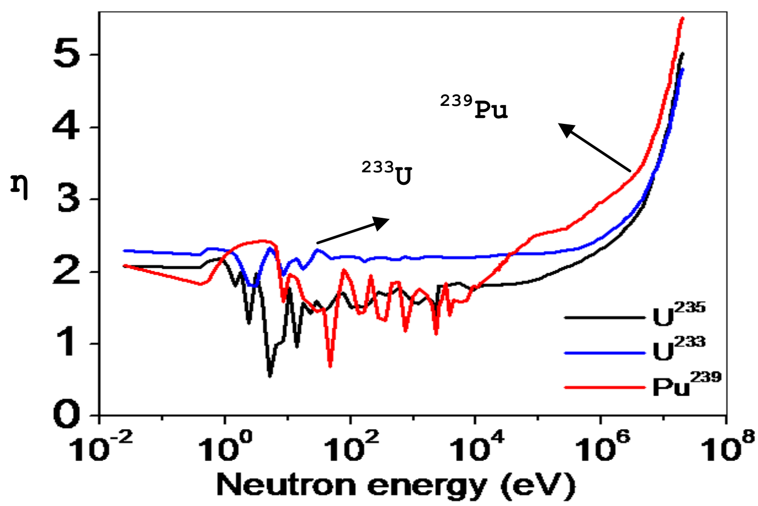

Figure 2. Fission neutrons availability for breeder reactors. The η values for ^{233,235} U and ^{239} Pu are respectively 2.28, 2.04 and 1.94 in the thermal neutron energy spectrum

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per neutron absorbed in the fuel, η is an important quantity in determining the breeding potential from a fertile to a fissile nucleus. The energy variation of this parameter is shown in figure 2 as a function of neutron energy. The value of η decides the fissile material breeding doubling time. The η value should be greater than two, for practical breeding purpose. The η value of ^{239}Pu is below 2 for thermal and keV neutrons and more than 2 at higher neutron energies. The corresponding values for ^{235}U are slightly more than 2 at thermal, lower than 2 at keV region and higher than 2 for MeV neutrons. Interestingly the η values of ^{233}U are uniformly larger than 2, almost in the entire range of the neutron spectrum. In essence the breeding of U -233 from thorium is more efficient than the corresponding breeding of Pu-239 from uranium. In this context, it is of interest to know the nuclei involved in the Th - U fuel cycle. The nuclei of interest connected to the Th-U fuel cycle are shown in figure 3.

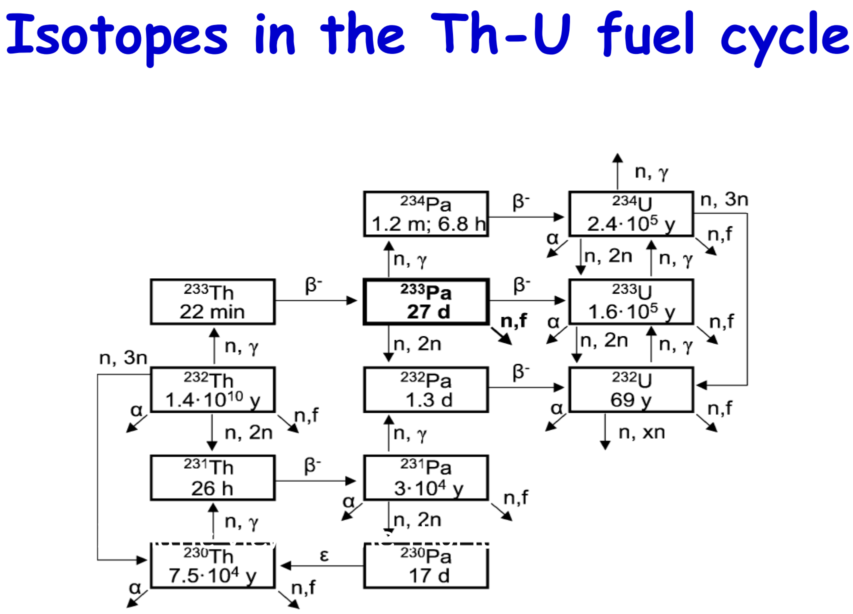

Figure 3. Isotopes in the Th – U fuel cycle

While the required data are available for some of the nuclei , the data situation is far from satisfactory for most of the nuclei in terms of availability over a range of energies and with the accuracy needed. A programme is underway in India to measure the cross sections of interest using the hybrid surrogate ratio method particularly suitable for radioactive targets. Using this method, the ^{233}Pa(n,f) cross sections have been reported for neutron energies above 10 MeV[4]. At lower energies, data are available using the radioactive ^{233}Pa as target . Measurement and evaluation of data for Th-U fuel cycle nuclei are crucial for use of Th as an alternate fuel. Data are also required for some of the minor actinides like Np, Am and Cm and long lived fission products like ^{129}I, ^{135}Cs, ^{107}Pd and ^{93}Zr.

2.3 Evolution of Thorium fuel cycle and related developments in India

As Thorium is going to be the main fuel in the 3rd stage of the Indian power programme and taking into account the sparse data available in the literature for Th-U fuel cycle, India has a comprehensive plan of R & D to take care of all aspects of Th- U fuel cycle. The thoria bundles made in India have been irradiated in some of the existing reactors, to obtain valuable data regarding minor actinides production, burn up and post irradiation handling issues. The thoria bundles have been used in the existing reactors. They have been tested for bundle power upto 400 kW and the corresponding burn up value upto 10 GWd/t. The U isotopes produced after irradiation of natural Th are (approximately) - ^{232}U,0.05%; ^{233}U, 88.8% ; ^{234}U,9.9%:^{235}U,1%, ^{236}U,0.09% ;^{238}U,0.1%. It may be noted that unlike the U-Pu fuel cycle, the elements like Am and Cm are not produced at any significant level. Even ^{237}Np is produced at roughly 1% of what is produced in the U-Pu cycle. These values are in good agreement with the theoretical estimates. It may be noted that the activity level of reprocessed thoria is significantly higher when compared to mined thoria due to the presence of ^{228} Th. After a cooling period of about 20 years, the activity level of reprocessed thoria can be brought down to that of mined thoria. While storing thoria, one has to make suitable provision of ventilation for large release of thoron from the fuel.

India operates a research reactor based on ^{233}U as fuel for many years and this facility(KAMINI) is mainly used for neutron radiography programme. As a part of the 3rd stage power programme, India is developing an Advanced Heavy Water Reactor (AHWR) [4] of capacity 300 MWe. The AHWR is a vertical pressure tube type, boiling light water cooled and heavy water moderated reactor using Th-U MOX and Pu-Th MOX fuel.The salient features of this reactor are : 65% power from Th; Several types of fuel combinations like enriched U, Pu - U, Pu - Th and U-233 and Th MOX; Several passive features- three days grace period without electrical power and with no radiation fall out; Passive shut down system to take care of insider threat scenario; Easily replaceable coolant channels; 100 years design life. The AHWR is accepted by IAEA for study under the INPRO programme.

2.4 Use of accelerator in nuclear power programme

Some years ago, Rubbia and others [5 - 8] proposed the Accelerator Driven Sub-critical System (ADSS) for energy production, fissile material breeding and long lived nuclear waste transmutation. India has a programme to develop sub-critical systems and couple to the external source of neutrons with a view to augment the thorium utilization programme [9]. There are various schemes for production of copious amount of non - fission neutrons from various nuclear reactions. The two frontline candidates are: Photo-neutron and spallation reactions. Using the photo-neutron route it is possible to obtain a neutron source strength of 10^{16} to 10^{17} per second. However, following the spallation route it is possible to achieve a neutron flux of 10^{18} for the same beam power. For 1 GeV protons impinging on a Pb/Bi target, one can get about 30 neutrons /proton and this translates into energy cost of about 30 MeV / neutron. Overall, the spallation route appears to be the best option for production of external non-fission neutrons. Relevant nuclear data of spallation products (toxic nuclei like ^{7}Be, rare earth alpha emitters etc), structural materials(like steel, spallation target window material, etc) are also required for this ADSS programme. Both charged particle and neutron induced reactions take place and the related data are required over a range of energies.

A particularly interesting application (of spallation based ADSS) from the Indian point of view is that of breeding U-233 in an ADSS [10]. The U -233 thus bred may continue to reside in the same ADSS for burning and power production or it may be used in high converting critical reactors. In both the cases there is a net energy gain if we assume that conversion efficiencies are not too low – 30 to 40 % (heat to electrical) and 40 to 50 % (mains to beam power). Besides solid fuelled reactors, there is interest in molten salt reactors (MSR) – both critical and accelerator driven sub-critical –due to their low fissile inventory and excellent neutron economy. With this in view several concepts have been proposed for coupling a sub-critical U core to the external spallation neutrons. In the molten salt ADSS driven system with a 10 MW proton beam fuelled only with Th (without any fissile component) can accumulate enough U-233 to be brought to its full power of 200 MW(e) in about 5 years time. With some U added in the initial stage, the same ADSS can be operated at full power from day one. Once the U – 233 reaches its maximum design concentration, the system will become a net breeder of U -233 which can be used for starting other reactors. In the Th burner concept, a PHWR ADSS, is operated in a once through Th cycle. One starts with a fuel of natural U and Th. During the operation of the reactor, U-233 will be generated and this adds reactivity to the system. The additional reactivity can be compensated by replacing some U by Th. As a result, the amount of Th used in the system gets increased progressively while the U inventory decreased. Ultimately a situation will be reached where the core will consist of only Th and no natural U. Such a sub-critical system would have several advantages – not requiring enrichment or reprocessing at any stage and a high burnup)about 10%) of the Th fuel and also act as a breeder of U – 233. However, it has a low k_{eff} close to 0.9 and gain of about 20 and hence the driver accelerator power required for this system is rather high (estimated to be 30 MW (1000 MeV, 30 mA protons)) for a 200 MWe ADSS. Building accelerators of this type will be a formidable task. The power in ADSS is inversely proportional to the value of sub-criticality and directly proportional to the neutron source strength. If the k_{eff} value is increased to say a value close to 0.95 – 0.98, it is possible to reduce the power of the accelerator to a more reasonable value of 15 MW. If U – 233 is recycled into the system it is possible to achieve this higher value of the k_{eff} . Considerable reduction in accelerator power (almost five fold) is possible in following the concept of one way coupled booster – reactor concept [11].In this case the inner fast core with external spallation source located at the centre boosts the neutron source strength. These neutrons leak into the outer thermal core (PHWR/AHWR) where they undergo further multiplication. The cascade of multiplication gives very high energy gain. Due to the absorber lining and the gap between the fast and the thermal cores, very few neutrons return to the fast core. Hence it is one way coupled between the fast and the thermal cores(shown in figure 4). Finally it is estimated that the beam power required for operating a 750 MW(t) system based on the one way reactor concept,

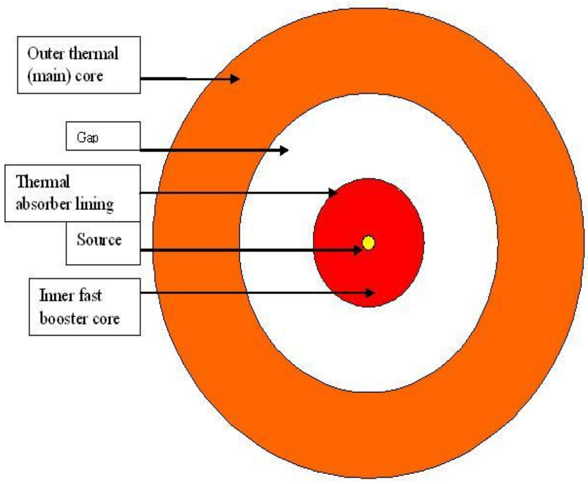

Figure 4. Schematic of the one way coupling concept

is of the order of 1 to 2 MW (1000 MeV, 1-2 mA, protons) and it is clearly feasible to build accelerators of this specification. In this system, the fast core will also have the long lived trans uranic waste and the same will be burnt. The thermal core will consist of ^{232}Th and ^{233}U fuel. However, the systems based on one – way coupled concept outlined above can not be looked upon as accelerator breeders or Th burners operating in once through cycle. They will require recycling of the fuel and will have a negligible contribution to breeding due to the spallation neutron source.

There are a number of technical challenges both in the accelerator and the reactor segments. In India, as a part of this programme, the Department of Atomic Energy has embarked on building a prototype injector for the ADSS. The injector will consist of a high current ion source, 3 MeV RFQ followed by DTL. The low energy high intensity proton accelerator (LEHIPA) which can serve as an injector, is designed to achieve an energy of 20 MeV and deliver proton currents of the order 30 mA. Work is also underway to design the LBE spallation target. Computer simulation of the spallation process and the design of a sub-critical core coupled to the external neutron source is an ongoing activity. In the first phase of the ADSS programme, it is also proposed to couple 14 MeV neutrons from a (d,t) source to the sub-critical core containing U and water to study the neutron multiplication and other related issues. Some details about the ADSS related activities in India are summarized in Ref. 9.

3. Summary
With a focus on large scale utilization of thorium reserves in the country for energy sustainability, India is actively pursuing programmes on different fronts to maximize the energy potential of nuclear fuel material through use of closed fuel cycle and Thorium as a fuel. Both reactor and accelerator developments required for these programmes are underway. The development of fast breeder reactor is a key component in realizing the high level of electricity generation in India, needed for meeting its large demands. The FBR stage is crucial for breeding ^{233}U from ^{232}Th required for launching the 3rd stage of Indian nuclear power programme. The development of Thorium based energy systems is a high priority in India. The research and development activity for achieving an ADSS will be pursued for sustainable nuclear power programme. India would like to invite and participate in related international research and development programmes like spallation target, fuel cycle options, nuclear data and high current accelerators in general.

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