1.1 REACTOR PHYSICS DESIGN & NUCLEAR DATA

Reactor Physics Design of AHWR

The major design mandates are the production of more than 60% of the total power from Th\(^{233}\)U and incorporation of inherent safety features such as the negative coolant void coefficient. A composite cluster with (Th, \(^{233}\)U) MOX fuel in the inner region and (Th, Pu) MOX fuel in the outer region is designed to meet this desired fuel performance characteristics. Coolant void coefficient of reactivity can be made negative in a slightly under-moderated heavy water system at a lattice pitch of 220 mm. For the design pitch of 245 mm, the coolant void coefficient is significantly positive. Hence, a burnable absorber, dysprosium, is used in a centrally located multi-purpose displacer rod, to obtain the negative coolant void coefficient. Other inherent safety features include negative fuel temperature coefficient, negative power coefficient and low power density. In addition, core excess reactivity is low on account of on-power refuelling.

The basic lattice and composite fuel cluster has been designed for 245 mm pitch. The reactor has two independent, functionally diverse, fast acting shut down systems, namely, shutdown system\#1 consisting of 40 mechanical boron carbide shut off rods and shutdown system\#2 based on liquid poison injection. There are thirteen boron carbide control rods grouped into four regulating rods for fine reactivity control, four absorber rods for xenon override and five shim rods for power setback.

Reactor control and protection systems are designed in a fashion similar to that in currently operating PHWRs except that the reactivity devices occupy lattice locations in the AHWR. A notable feature of AHWR is the xenon override capability to restart reactor anytime following shutdown. Equilibrium core is fuelled to three burn-up zones to achieve a flattened power distribution. Core average discharge burn-up is 24,000 MWd/t.

The cluster design allows use of burnable absorber in the non-fuel multi-purpose displacer rod depending on the fuel type. Use of heavy water as moderator improves neutron economy and relatively soft neutron spectrum imparts flexibility to run it on different fuel-types. During the initial phase plutonium bearing fuel is used to generate \(^{233}\)U required for composite cluster of the equilibrium core. Equilibrium core will be self-sustaining in \(^{233}\)U and runs on composite cluster containing both \(^{233}\)U and plutonium. Two different values of enrichment i.e., 2.5% in the top half and 4.0% in the lower half are used in plutonium bearing fuel pins such that the axial flux peak is shifted away from the high steam quality upper region in the coolant channel. This improves the thermal hydraulic characteristics of fuel channel, with fuel dry-out occurring at a considerably high channel power in comparison to that with a uniform axial enrichment.

Initial core design of the AHWR, approach to equilibrium and refuelling strategy studies along with the development of the flux mapping system, power distribution control system and analysis of various spatial instabilities are being pursued. Physics design to achieve higher fuel burn-ups of about 40,000 MWd/t by lattice pitch reduction and removal of burnable absorber from the fuel cluster are important ongoing activities.

Multi-group data library has been updated for elements of thorium chain and for certain elements specific to AHWR such as burnable absorber dysprosium. Various uncertainties due to thorium data have been identified and experiments are planned in the AHWR critical facility to resolve them. Physics design and preliminary safety analysis for experiments in critical facility have been completed. Core power distribution has been optimized, taking into account the neutronic thermal hydraulic coupling. Studies have been carried out on generation of \(^{233}\)U required for equilibrium core, achievement of self-sustaining uranium cycle, recycling of uranium and its impact on overall fuel cycle including toxicity of spent fuel.

Cross section of the AHWR fuel cluster
Nuclear Data and Needs for Thorium Fuel Cycle Studies

The design of innovative reactor systems needs integral data validation studies to generate accurate nuclear data. The nuclear data from experimental measurements need to be evaluated, for arriving at basic point data libraries in the Evaluated Nuclear Data File (ENDF) format. The condensed nuclear data libraries are then used with Monte-Carlo and deterministic codes for reactor physics evaluations of reactor systems. In the context of thorium-based systems, the nuclear data for thorium fuel cycle studies of the AHWR and the CHTR are of current interest. The nuclear data for ADS systems, for neutron energies beyond 20 MeV are also important. Presently Indian nuclear data activities generically encompass the user-oriented approach starting from data files distributed by the IAEA.

India sponsors a regional IAEA-NDS nuclear data Mirror Site (http://www-nds.indcentre.org.in) for the Asian region.

Nuclear data for Isotopes of Thorium Chain

The new concepts involving thorium systems require detailed basic nuclear data measurements and integral validation studies, as thorium has not received the required attention in the past. The nuclear data of the major and minor isotopes $^{230}$Th, $^{231}$Th, $^{231}$Pa, $^{233}$Pa, $^{232}$U, $^{233}$U and $^{234}$U in the thorium fuel cycle needs to be brought to the present level of quality that exists for the isotopes of the U-Pu fuel cycle. For instance, the self-shielded capture resonance integrals for $^{232}$Th are higher in “jendl3.lib” as compared to those in “endfb6.lib” in the final stage of the IAEA WIMS-D update project, by several tens of percent as compared to 0.1% target accuracy. The design of advanced reactor systems, demands accurate nuclear data, in the resonance region that affect plant safety related feedback coefficients such as Doppler and coolant void reactivity effects as a function of burn up.

Critical Facility (CF)

The AHWR simulations are based on assumptions and modeling approximations, which are sensitive to nuclear data uncertainties especially because of thorium fuel cycle. The evaluation of the lattice characteristics requires experimental validation to freeze the design and obtain regulatory clearance, before fuel fabrication. A multi-purpose Critical Facility (CF) has been designed and is in an advanced stage of construction. The sensitivity calculations illustrate that the critical height of the CF with AHWR representative core increases by 5 cm and 7 cm respectively when the “iaea.lib” and “jendl3.lib” libraries replace “endfb6.lib” WIMS-D library. For natural uranium core this is about 2 cm. The replacement of multigroup data of $^{232}$Th alone in “jendl3.lib” by “endfb6.lib” changes the k-infinity by 10.24 mk, “jendl3.lib” yielding a higher calculated value of k-infinity.
**Compact High Temperature Reactor (CHTR)**

The design of CHTR, which exhibits intermediate neutron spectrum, was strongly influenced by considerations of nuclear data and associated uncertainties during its evolution. The cross sections for several new materials, such as Er, Bi and Ga that were considered for CHTR show large discrepancies in different cross section libraries. It is mandatory to have a negative Doppler feedback effect in the core design. The initial choice of pure $^{233}$U as fuel had been revised due to the calculated positive Doppler reactivity feedback. Further, the calculated Doppler reactivity effect of $^{233}$U has a large uncertainty as the nuclear data of resolved and unresolved resonance region are highly uncertain. As the spectrum covers regions above the thermal range, accurate knowledge of various transport and inelastic cross-sections of various constituents, such as $^{233}$U, $^{232}$Th, Be, Er, Th, which affect the design significantly at high temperatures are required. Experimental work to demonstrate these systems are underway.

**Thorium irradiation in FBTR (Kalpakkam)**

In the 40 MWt Fast Breeder Test Reactor (FBTR), at Kalpakkam, fifty four thorium subassemblies (717 Kg) have been loaded in the 9th ring in the radial blanket after the nickel reflector. Plans are underway to consider loading additional 100 thorium assemblies in 7th and 8th rings as well. The $^{233}$U produced in this reactor will have low content (5 ppm) of $^{232}$U as compared to several hundreds of ppm in other situations such as a normal fast reactor core, ADSS core with thorium and thorium in PHWR. The reason for the expected low ppm of $^{233}$U in FBTR is understood as due to three factors influenced by nuclear data and physics considerations: Nickel reflector brings the neutrons below the threshold of (n, 2n) reaction in $^{232}$Th. The effective $^{231}$Pa (n, $\gamma$) cross section is much lower in a fast spectrum as the capture cross section falls rapidly with increasing energy. Thirdly, the accumulation potential of $^{233}$U produced is more in saturation in a fast spectrum making the ppm content of $^{233}$U in $^{232}$U much smaller.

**PIE studies of irradiated thorium bundles in PHWRs**

Identical loading of thorium bundles was used in KAPP-1 & 2, KAIGA-1 & 2 and RAPS-3 & 4 to attain flux flattening in the initial core. The thorium oxide used is about 400 kg in all the 35 bundles put together in a reactor. The bundles loaded in KAPP-1 & 2, KAIGA-1 & 2 and RAPS-3 & 4 have already been discharged from the core. Samples were obtained from one of the irradiated ThO$_2$ bundles and have been analyzed experimentally by alpha spectrometry for $^{232}$U and by thermal ionization mass spectrometry for $^{233}$U, $^{234}$U, $^{235}$U and $^{236}$U by two different groups in BARC. The previous analyses by two teams in BARC gave a factor of six to eight under-predictions in the production of $^{232}$U. The discrepancy was traced back to the fact that the effective one-group values of cross sections for isotopes of thorium fuel cycle and the use of assumptions in the ORIGEN code are not applicable to the irradiation of thorium in PHWRs. Simulation of the thorium experiment using the new WIMS-D libraries, has been successfully attempted. Sensitivity results of different modeling approaches such as single cell versus super-cell model and treatment of (n, 2n) process (pseudo-fission versus explicit) to prediction of isotopic contents of urania have been obtained. The results are shown in Table.

Generation of integral data by gamma spectrometric analysis of the irradiated thorium fuel is also a part of this activity.

A direct consequence of $^{232}$U concentration in bred $^{233}$U from PHWRs, is its effect on radiation shielding modification in the AHWR critical facility. The outer thickness of the concrete has been increased by nearly 10% to compensate for the additional gamma dose emanating from the fuel clusters based upon experimental results of post irradiation analyses and using the new basic evaluated nuclear data files.

**IAEA-CRP on nuclear data for thorium cycle**

The active participation in the IAEA-Co-ordinated Research Program, on the “Evaluated Nuclear data for Thorium- Uranium Fuel Cycle" enables to share information and to benefit from the developments related to the use of thorium around the world. An experimental benchmark based upon a 30 kW, $^{233}$U fuelled research reactor KAMINI, at Kalpakkam, is under preparation. KAMINI is a low power research reactor designed and built by a joint venture of BARC and Indira Gandhi Centre for Atomic Research (IGCAR). KAMINI is the only reactor in the world operating with the $^{233}$U fuel. Preparation of a benchmark on thorium irradiation experiments and burnup measurements in PHWRs is also underway.
Advanced Heavy Water Reactor

**Use of updated nuclear data for safety analyses and operation of existing reactors**

The Fuel Temperature Coefficient (FTC) of PHWR fuel, calculated by the new 69-group “iaea.lib” library gives significantly different results at higher burn-ups and explains as a preliminary observation, the unexpected power rise that occurred in the KAPS-1 unit. In a PHWR, the FTC which is negative at low burnups becomes less and less negative and even turns positive at some burnup. The precise crossover point in burnup where the FTC becomes positive depends on many parameters such as the temperature range and 19 versus 37-rod cluster. The FTC is due to the combined effect of Doppler effect and fuel re-thermalization effect. Recent calculations of FTC of PHWR lattices, performed independently by several researchers, illustrate the following: The 27 group WIMS1981 library has a crossover point, for FTC at about 12000 MWd/t burnup; at about 9400 MWd/t with the same but 69-group library, at about 6000 MWd/t for a 19 rod cluster with the new “iaea.lib” library and at about 4500 MWd/t for 37 rod cluster of PHWR with the “iaea.lib” library. The crossover point of the FTC is not just the issue but how negative it should be, in order to overcome positive reactivity, that includes the positive xenon kill feedback whenever power transient occurs. The calculated coolant void reactivity using the new “iaea.lib” library is observed to be lower than the earlier results obtained using the 1971 library. The KAPS-1 overpower transient could be explained only with the use of new IAEA multi-group nuclear data libraries.

**International collaborations in experimental programmes**

As multiple fuel cycles (e.g., U-Pu, Th-U), with the option of closing the fuel cycle are envisaged, the nuclear data requirements that are needed to develop the new systems with high burnup are demanding and include the entire range of actinides and fission products for multiple fuels. The neutron time of flight experiments to measure neutron induced reaction cross sections from 1eV to 250 MeV, in CERN (Geneva, Switzerland) and neutron transmission experiments using the 150 MeV electron linac at Pohang in South Korea, would help to reduce the existing uncertainties in simulation studies of new and advanced reactor concepts. The new measurements of nuclear data of these experiments and analyses including covariance error information are of importance. Experimental activation measurements to perform neutron source flux characterization in DHURVA 3001 beam hole has been taken up in order to initiate neutron time.

**XnWlup software for inter-comparison of WIMS-D multigroup cross section**

An upgraded version of the computer program ‘XnWlup’ has been developed in Visual C++ to produce readily, histogram plots of the multi group cross sections of a selected nuclide, as a function of neutron energy. It also provides the comparison of the nuclear data of different nuclides from different libraries.
The upgraded version, Xnwlp3.0 has extra features as sought by the users and includes plots and inter-comparisons of self-shielded resonance integrals at various background dilutions and temperatures.

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- **Nuclear Data for AHWR Analysis**

Nuclear data is one of the major inputs for the physics design of advanced reactor systems utilising thorium, such as the AHWR. The basic experimental data of the available Th-U cycle available requires to be established with the accuracy of current day standards. The requirements of accuracy for the basic data of the individual isotopes for thermal reactor systems, will have to be improved in order to predict, for instance, k-effective within 0.5%, feedback coefficients within 10% and integral reaction rates within 1.0%. The aim of the task is to generate a processed multi-group data-set for AHWR analysis, which would entail benchmarking, data processing, integral testing, front-end and back-end analysis and sensitivity studies of the systems using thorium. There are several ingredients in this task like, assessment of the energy group structure of thorium cycle evaluations, research towards indigenous data processing capabilities and development of experimental capabilities and nuclear data evaluation methodology.
The availability of the basic nuclear data for the thorium cycle has been assessed and the multi group processed library for isotopes has been updated. The thorium-fuelled lattices have been benchmarked to validate the available basic data. Further the nuclear data set with materials and material composites have also been updated for analysis of the advanced reactor systems. Experiments have been formulated to qualify the nuclear data, by validation through post-irradiation examination.

The review of the basic data isotopes of the Th-U cycle such as $^{233}$Th, $^{233}$Th, $^{234}$Th, $^{231}$Pa, $^{232}$Pa, $^{232}$U, $^{233}$U, $^{234}$U etc. existing as of now exhibit large discrepancies. The thorium fuel cycle data in the WIMS data set has been extended with $^{232}$U and $^{231}$Pa. The $^{232}$U content in the irradiated thoria bundles of PHWR has been estimated with the WIMS dataset and WIMSD code system. The ratio of $^{232}$U/$^{231}$U is calculated as ranging from 450 ppm to 590 ppm and is in good agreement with experimental results.

Sensitivity studies done for AHWR with different evaluated data sets like, ENDF-B/VI, JENDL 3.2 and JEF 2.2 show significant differences. For example, the $^{232}$U absorption reaction rates differ by 5% in thermal energy range and about 10-15 % in higher energy ranges.

The void reactivity studies for AHWR lattice also show a scatter of about 25 %. WIMS data set has been updated with isotopes of dysprosium for AHWR analysis.

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