IRRADIATION OF THORIUM BASED FUELS AT RESEARCH REACTORS OF TROMBAY

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Abstract

Utilization of large thorium reserve of our country for power production had been a prime goal from the inception of our three stage nuclear power programme. This long term policy is well reflected in irradiation programme of Thoria and Thorium rods at research reactors Cirus and Dhruva, located at Trombay. The first batch of Thorium / Thoria rods were loaded in Cirus reactor on August 28, 1960, immediately after attaining first criticality on July 10, 1960 which reflects the priority the Thorium utilization programme received from our planners. The programme was aimed for gaining sufficient experience with Thorium fuel cycle by the time our first stage of nuclear power programme attained maturity. In Cirus, these rods are irradiated in the annular gap, called J-rod annulus, between the two graphite reflectors around the pile. About 200 Thorium / Thoria rods had been irradiated in Cirus before the reactor was shut down permanently on 31st December 2010. A few Thoria assemblies have also been irradiated in fuel positions of Dhruva reactor. Apart from these, a few fuel assemblies made of ThO₂, PuO₂ and UO₂ were irradiated in Pressurized Water Loop of Cirus to study fuel and clad behavior under high temperature and pressure, before their induction in power programme. Presently, fuel cluster with AHWR type Th-Pu and Th-LEU MOX fuel pins are being irradiated in a regular fuel position of Dhruva for obtaining vital information and experience related to Thorium based MOX fuel cycle.

This paper highlights the experience gained in irradiation of Thoria / Thorium rods, expected yield of U²³³, contamination level of U²³² and handling of these assemblies at research reactors Cirus and Dhruva.

Key Words: J-rods, Thoria, Pressurised Water Loop, AHWR fuel pin, MOX,

1. Introduction: Research Reactors are one of the most essential nuclear facilities in the design and development of necessary infrastructure for nuclear power programme of any country. They provide a large volume of neutron source for carrying out neutron beam research, production of radioisotopes for application in the field of medicine, agriculture, food preservation and industry, irradiation testing of reactor material and nuclear fuel, neutron radiography and neutron activation analysis.

2. Brief Description of Cirus and Dhruva Reactors: Cirus is a 40 MW (thermal) Uranium metal fuelled, heavy water moderated, light water cooled research reactor commissioned in July 1960. Sea water is used as secondary coolant and sea is ultimate heat sink. The maximum thermal neutron flux of Cirus reactor is 6.6 x 10¹³ n/cm²-sec. Cirus reactor was operated successfully for nearly five decades with very good availability and capacity factors. The reactor was under refurbishment during 1997 – 2003 and after successful completion of various refurbishing activities, it resumed operation from October 2003. After fifty years of service, Cirus reactor was permanently shut down on 31st December 2010.

Dhruva is a 100 MW reactor with a maximum thermal neutron flux level of 1.8x10¹⁴ n/cm²-sec. It uses natural uranium metallic fuel with aluminium cladding and heavy water as coolant, moderator and reflector. The vertical coolant channels are semi-permanent structural component and replaceable. This provision has given the requisite flexibility to alter the core configuration in case of any requirements. The use of metallic U fuel enables the reactor to provide adequate excess reactivity for various irradiation programmes. With this excess reactivity and provision for interchangeability of fuel and other in-pile positions, utilization of reactor has been enhanced by providing additional isotope tray rods, irradiation of thorium and Thoria based MOX fuels for
generating experiences for thorium fuel cycle, irradiation of structural materials like seamless zircaloy samples etc.

3.0 Thoria Rods of Cirus: A typical thoria rod assembly is about 23’ long and it consists of two sections. The bottom section is about 12’ 3” long, 1.6” OD and it contains the thorium dioxide pallets in a finned aluminum tube. The top section, known as shielding section is 10’ 5” long and is made of carbon steel and Aluminum rods. Irradiation of Thoria rods generates heat which is removed by the normal structural air cooling; about 14000 CFM, supplied to the J-rod annulus from east and west air intakes. Neutron flux in J-rod annulus is 1X10^{13} n/cm^2/sec. J-rod annulus (2.5” wide) is formed by the annular gap between two concentric rings of nuclear grade graphite blocks around reactor vessel. Cirus had design provision to irradiate 100 air cooled assemblies in J-rod annulus. Metallic thorium assemblies were irradiated up to 15000 reactor MWD and thorium oxide assemblies (containing about 132 thorium oxide pallets; each pellet has 1” OD and 1” length) were irradiated up to 25000 reactor MWD, before their removal from the reactor. During their stay in pile, J-rods were monitored for their power output as reflected by increase in air temperature. An alarm is generated when the limiting condition (65° C) of air temperature is reached. After schedule irradiation of the J-rods, they are taken out of pile and stored in Dry Storage Block (DSB) in reactor hall. DSB has ventilation and the rods are cooled by air. After about 5-6 years of cooling and storage at DSB, the rods are shipped for reprocessing. The initial lots of J-rods were shipped to Kalpakkam for providing fuel to KAMINI reactor. Subsequent shipments were done to Fuel Reprocessing Division (FRD), Trombay.

3.1 Handling of Irradiated J-rods and Their Shipment for Reprocessing: Handling of irradiated J-rods is a dose intensive job and it requires elaborate preparations and precautions. Radiation field on an irradiated rod depends on its burn-up and cooling period. Normally after 25000 RMWD irradiation and 6-7 years of cooling, the radiation field on contact of the rod comes down to ~ 15 R/hr. However, radiation field at the working area remains about 200-300 mR/hr. To assess the dose budget and to carry out proper planning and preparations, radiation mapping of an irradiated rod is done before taking of the cutting operations to remove fuel portion from the shielding portion. Shipment of cut J-rods involves Cirus Operations, Dhruva Operations and Fuel Reprocessing Division (FRD). A schedule for carrying out this activity is made in consultation with the three agencies. 30-T shipping flask from Plutonium Plant (FRD) is received in Dhruva SFSB trucking area for loading of cut fuel sections. Although loading of cut fuel sections in 30-T shipping flask can be done in Cirus Rod Cutting Building (RCB) also, it had been found easy to carry out these operations in Dhruva SFSB due to more space for movement of vehicle / trolley. Also Dhruva SFSB has lower background radiation field which results into lower dose consumption. The irradiated J-rod assembly is cut at the weakest portion called neck of aluminum shielding section and the fuel section is detached. A bunch of fuel sections are then transferred in a lead shielded trolley to Dhruva SFSB trucking area where the fuels are transferred to 30 ton shipping flask manually and further sent to PP for reprocessing.

3.2 Burn-up Levels of Thorium and Thoria Rods Irradiated at Cirus: Thorium metal J-rods were irradiated up to 15000 RMWD while thoria J-rods of density about 7.1 g/cc were irradiated upto 25000 RMWD. The irradiation limits were based on maximum permissible aluminum sheath temperature of ~ 400° F. At these burn up levels, (U^{232} + Pa^{232}) content is estimated to be about 2.2 ppm for Thorium metal rods while about 3.6 ppm for Thoria rods. With this content of (U^{232} + Pa^{232}), the relatively clean U^{233} can be handled like any other fissile fuel with adequate protection against alpha activity [5]. The reactivity load of a single thoria
rod of higher density (total mass ~16.5 kg) was 0.07 mk/rod while that for the lower density rod (total mass ~12 kg) was 0.05 mk/rod.

3.3 Radioactivity in Thoria Rods of Cirus: To plan handling and reprocessing aspects, it is necessary to know residual activity of a Thoria rod after irradiation. The following table gives this information about a typical Thoria rod having 16.5 kg mass and about 26000 MWD of irradiation\textsuperscript{[2]}:

Table -1: Residual specific activity in a rod for different cooling periods

<table>
<thead>
<tr>
<th>Cooling period (years)</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sp. Activity, (mCi/gm)</td>
<td>2.88</td>
<td>1.55</td>
<td>1.10</td>
<td>0.88</td>
<td>0.77</td>
<td>0.70</td>
<td>0.66</td>
</tr>
</tbody>
</table>

The reduction in specific activity is mainly due to decay of Ce\textsuperscript{144} (T\textsubscript{1/2} ~285 days) and Pr\textsuperscript{144} (T\textsubscript{1/2} ~17 minute). Radiation field on a single rod after about 4 year of cooling is estimated to be 15 R/hr at a distance of 1 foot while U\textsuperscript{233} content in a single rod is estimated to be about 20.6 gm which contains about 3.1 ppm of U\textsuperscript{233}\textsuperscript{[3]}.

4.0 Thorium Irradiations in Pressurized Water Loop (PWL) of Cirus: Pressurised water loop (PWL) in Cirus is abigh temperature and high pressurein fuel facility to test experimental fuel assemblies under simulated power reactor conditions. Zircalloy-2 tube of 57.9 mm O.D. and 5.6 mm wall thickness, passing through one of the 4” ID lattice positions of Cirus forms the Test Section through which demineralised light water at 1500 psig pressure and 500°F temperature is circulated by two canned motor pumps in series. The system was commissioned in 1972 and was extensively used for irradiation of test fuel specimen. Total 15 nos. test fuel assemblies had been irradiated in Cirus PWL. Initial test assemblies for irradiation were having Natural UO\textsubscript{2} fuel material. Irradiations were aimed at development of power reactor fuel and evaluation of the effect of variation in design and fabrication parameters on its performance. Subsequently, mixed oxide fuel assemblies with Natural UO\textsubscript{2}-PuO\textsubscript{2} pins were irradiated to gather data and handling experience towards development of MOX fuel for power reactor. One thoria based MOX fuel assembly (called as AC-6), was irradiated in mid-eighties to study and understand the problems associated with thoria based fuel before their introduction in our nuclear programme. Another MOX fuel assembly BC-8 was irradiated in early nineties to prove the design of (UO\textsubscript{2}-PuO\textsubscript{2}) and (ThO\textsubscript{2}-PuO\textsubscript{2}) pins of PHWR at high burn-ups, develop understanding of the fuel behaviour using advanced fuel cycles and bench mark the fuel design code.

4.1 Re-constitution of AC-6 pins and Re-irradiation after Cirus Refurbishment: The test fuel assembly AC-6, which was a 6 pin cluster with 5 pins of 4% PuO\textsubscript{2} - ThO\textsubscript{2} MOX and 1 pin filled with helium; had been irradiated to 22640 RMWD and stored under water for about twenty years after its removal from pile. It was planned to extend irradiation of fuel pins from this assembly for generating additional data for studying behavior of Thoria based fuel in terms of fission gas release, fuel centreline temperature, fuel pellet swelling, fuel grain growth etc at higher fuel burn-ups. Also, it would provide an experience of carrying out re-irradiation of fuel pins. With this objective, a new fuel assembly consisting of 4 no. of fuel pins of AHWR design (Th- 8% Pu MOX) and 2 no. of fuel pins of AC-6 (TAPS-BWR) design was fabricated and loaded in PWL test section. AC-6 fuel pins were subjected to UT, eddy current testing and liquid N\textsubscript{2}-glycol testing, and found suitable for re-irradiation. Subsequently they were assembled with new AHWR design fuel pins. Presence of irradiated fuel pins in the new assembly for irradiation made its handling different from the handling of normal fuel assemblies of PWL. This needed transportation of assembly from PIED to loading bay of Cirus Rod Cutting Building (RCB) using 20-ton shielded cask, handling of assembly underwater in specially fabricated cans and its transport to reactor hall using RCB buggy, lifting of the assembly in fuelling machine and its installation in PWL test section. As the movement of fuel assemblies from RCB to reactor hall was a non-routine work and in the past no irradiated assembly had been loaded in PWL test section, a dummy fuel assembly of identical dimensions was fabricated to carry out trials for validating the handling procedures. After successful trials, the fuel assembly was installed on 15\textsuperscript{th} November 2010 in test section. By the time the reactor
was shut down permanently on 31st December, the assembly received 811.517 RMWD irradiation. Performance of the assembly during its stay in pile had been satisfactory.

5.0 Irradiation of Thorium Oxide Rods in Dhruva: The programme for irradiation of Thoria assembly in Dhruva was taken up just after its criticality. In order to cut down time on the development of Thoria fuel assembly, it was decided to keep design of these assemblies similar to regular fuel assemblies undergoing irradiation in Dhruva. After satisfactory endurance testing of a prototype assembly in Dhruva flow station, a few Thoria assemblies were installed in regular fuel position of reactor and irradiated successfully up to a maximum burn up of 10,000 RMWD. Each Thoria assembly consisted of two parts namely the reusable seal and shield sub-assembly at top and 7-pin fuel assembly at bottom. The fuel sub-assembly in turn consisted of Thoria pellets of ~3m stack height. Each cylindrical pellet was of 12mm diameter and 12-16mm length having an average sintered density of 9.5 g/cc. The irradiation was extremely helpful in generating first hand experiences of thorium based fuel cycle, especially on U-233 production, contamination level of U-232 in U-233, their variation with burn up. The estimated (U\(^{233}\)+Pa\(^{233}\)) content at the burn-up level of 5,000 and 10,000 RMWD was 53 gm and 91 gm while (U\(^{232}\)+Pa\(^{232}\)) content was 12.5 and 24.3 ppm respectively.\[^4\] The variation of assembly power with respect to burn up and the corresponding maximum fuel temperature were estimated to ensure that adequate safety margin was available for extend the burn up of Thoria to 30,000 RMWD at Dhruva. The activity content of Thoria assembly for different irradiation level at reactor and for different post irradiation cooling period was estimated. This proved to be one of the most valuable inputs for designing the reprocessing facility for Thoria fuel at Trombay.

5.1 Irradiation of Th-Pu MOX fuel: A cluster with AHWR type fuel pins were irradiated in the regular fuel position in Dhruva for a burn up of 20,000 MWd/t. The fuel pins contain Th-Pu MOX fuel with Plutonium fraction of ~1%. The arrangement of the cluster was identical to that of regular fuel assembly of Dhruva. It had a fuel cluster pinned to the Aluminium shield. The Aluminium shield and seal plug were the same as used in regular fuel assembly. The cluster design was kept nearly identical to a standard Dhruva fuel assembly with its component like flow tube, top bulge section, bottom split bulge, top split collar being same to that used in Dhruva fuel. The main modification was in the design of spacer grid arrangement. It consisted of a central structural tube to which spacer grid was attached. Six fuel pins were made to surround the central tube. The spacer pads attached to the fuel pins along with the spacer grip provided the inter pin spacing. The 6 fuel pins and the central tube were held at top and bottom by tie plates.

5.2 Fuel Fabrication and Quality Control: The free standing Zr-2 clad was fabricated at NFC by the conventional pilgering route. Spacer pads were attached to the fuel clad by resistance welding of appendages on empty clad tubes. The TH-1% Pu pellets were fabricated by two processes. The first route was conventional powder-pellet route involving cold compaction and high temperature sintering. The second route was by Coated Agglomerates Process route. The pellets were staked in the clad tubes and sealed at either end by TIG welding of end plugs at Helium atmospheres. The relevant QC and inspection procedures adopted in the fabrication of clad, pellets and fuel pin was similar to that adopted for TAPS-BWR-MOX fuel. The fuel pins along with the central tube were assembled together by welding of the of the aluminium structural components to form the cluster at AAAF. The QC and inspection procedures adopted for the regular fuel assembly of Dhruva was adopted for the test cluster.

5.3 Safety Evaluation: A detailed safety analyses was carried out before loading fuel in reactor. The main points of analyses were:

- Reactor Physic Analysis: The fuel cluster power was evaluated to be ~500 kW at rated power in the designated location at reactor. The peak linear heat rating was estimated to be 40kW/m. The reactivity load due to the fresh AHWR fuel assembly was calculated to be 2.04 mk and was found to be increasing slowly due to accumulation of fission products. At the end of irradiation, the reactivity load increased to 2.5mk.\[^7\]
• Fuel Performance Analysis: This was carried out for the steady state operation using modified GAPCON - THERMAL code, incorporating thermo-physical data of Thoria based fuel. The result of the important fuel design limiting factors is given in the table below[6]:

<table>
<thead>
<tr>
<th>Sr. No</th>
<th>Parameter</th>
<th>Linear Power Rating</th>
<th>Design Limit</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>35</td>
<td>40</td>
</tr>
<tr>
<td>1</td>
<td>Peak fuel centre line temperature</td>
<td>°C</td>
<td>1016.7</td>
</tr>
<tr>
<td>2</td>
<td>Peak cladding outer temperature</td>
<td>°C</td>
<td>110.4</td>
</tr>
<tr>
<td>3</td>
<td>Peak internal pin pressure</td>
<td>bar</td>
<td>5.4</td>
</tr>
<tr>
<td>4</td>
<td>Effective clad strain, after 20GWd/</td>
<td>(%)</td>
<td>0.015</td>
</tr>
</tbody>
</table>

The fuel performance analyses had also been carried out taking into effect of uncertainties in fuel design variables, deviation in the fabrication process parameters, changes in material properties and variation in reactor operating conditions.

• Thermal Hydraulic sub-channel analysis: The analysis was carried out for steady-state operation and under transient conditions of flow coast down for minimum permissible coolant flow of 375lpm. The clad surface temperature and MCHFR was found to be well within safe limits. There was no boiling of coolant predicted to occur as the clad temperature was estimated to be lower than coolant saturation temperature.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Steady State</th>
<th>Flow coast Down</th>
</tr>
</thead>
<tbody>
<tr>
<td>Clad surface Temp, °C</td>
<td>122.9</td>
<td>140.7</td>
</tr>
<tr>
<td>Coolant surface temp. at</td>
<td>150.2</td>
<td>148.6</td>
</tr>
<tr>
<td>location of max. clad temp, °C</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCHFR</td>
<td>4.35</td>
<td>3.889</td>
</tr>
<tr>
<td>Max. coolant exit temperature °C</td>
<td>69.1</td>
<td>75.5</td>
</tr>
</tbody>
</table>

5.4 Experimental Evaluations: As the design of the AHWR fuel pin spacer grid was different from that of regular fuel assembly and the pins were proposed to be kept free in the initially proposed design, it was decided to carry out endurance testing of the assembly at Flow Test Facility (FTF) of Dhruva before loading into reactor. One dummy assembly identical to actual test AHWR fuel assembly except that the six Zircaloy clad fuel pins had SS rods in place of TH-Pu MOX fuel pellets was fabricated at AFFF, Tarapur. During flow testing at FTF, the parameters like flow, pressure drop across the assembly were found to be identical to that obtained during flow test of a regular fuel assembly under similar conditions. Vibration signature of AHWR dummy assembly, one fresh standard fuel assembly and one Aluminium dummy assembly were noted periodically by RED at various flows. The flow test was carried out for ~500 hour to have reasonable assessment of fretting at spacer locations. After that, the dummy assembly was dismantled and cut for inspection. Based on the vibration analyses and observation on the sub-assembly after dismantling and cutting, the design of the AHWR fuel cluster was modified. The fuel pins and central rod were connected to the tie plates at either end.

5.5 Monitoring During Irradiation: The AHWR fuel assembly was irradiated for 3.25 years. During this period, flow through the assembly, its outlet temperature, burn up level and power developed by the rod was monitored continuously. Its vibration level was also monitored periodically through special procedures to ensure its healthiness. The performance of the assembly was normal and was in close agreement with the predicted values.

5.6 Post Irradiation Handling: The assembly was removed after completion of target burn of 20,000MWd/t. Safety aspects during handling in Fuelling Machine was evaluated before its installation in pile. As assembly power was increasing with burn up, its decay heat at the end of irradiation after 2 hour of cooling was evaluated and found to be 5.3kW as compared to 13.5kW for Dhruva fuel assembly. Based on this, the maximum allowed dry handling time of 5 minutes being followed for Dhruva fuel was considered safe for AHWR fuel also. As the specific heat of MOX fuel is less than metallic Uranium fuel, the pin central temperature and cladding temp during dry handling was re-evaluated and found to be around 500°C. This was
considered to be acceptable for both oxide fuel and Zr-clad. The shielding adequacy of FM considering both neutron and gamma source was evaluated and found to be adequate for handling the AHWR fuel. The contribution of spontaneous fission neutron strength in PU-240 was also considered while evaluating shielding adequacy. The AHWR fuel safely removed from reactor and transferred to SFSB for post irradiation cooling.

The fuel cluster portion will be sent to PIED for detailed inspection in hot cell of PIED.

6.0 Irradiation of Th-LEU MOX fuel: After successful completion of irradiation of TH-PU MOX fuel in Dhruva, another cluster with containing AHWR type Th-LEU MOX fuel was installed in regular fuel position of Dhruva for a similar target burn up of 20,000MWd/t. The UO$_2$ content in Thoria MOX fuel is 11 wt% and the LEU is 8.94% enriched. The arrangement of test assembly, its fabrication methods, QC and inspection procedure was identical to that of the first assembly. The proposal had similarly undergone safety evaluation and regulatory review before it was loaded in reactor. The test irradiation will be helpful in studying thoria based fuel behaviour in terms of fission gas release, fuel centreline temperature, fuel pellet swelling, fuel grain growth etc. It will also provide vital information and experience related to Thorium fuel cycle aspects.

7.0 Future Plans: Refurbishment of Dhruva is being planned for safety upgrades and life extension. The IPL facility of 150mm diameter and of 1.2 MW capacity will be commissioned at central position of Dhruva during that time. The facility will be used for testing of advanced Thoria based fuels under simulated power reactor conditions.

8.0 Concluding Remarks: The irradiation programme of Thorium / Thoria and Thoria based MOX fuel assemblies from very inception of our atomic energy programme has given deep insight into Thoria based fuel. The irradiation programme yielded into a firsthand knowledge about fuel fabrication, quality assurance, out-of-pile testing and handling aspect of fresh and irradiated assembly to the designer, fabricator and operators. The programme of early irradiation of Thoria rods in Cirus resulted into fuel supply for KAMINI reactor. Subsequent irradiations of Thoria based MOX fuels in Cirus PWL and Dhruva has helped in study of thoria based fuel behaviour in terms of fission gas release, fuel centreline temperature, fuel pellet swelling, fuel grain growth etc on completion of its PIE at hot cell in PIED. The irradiation programme has provided feed material for carrying out reprocessing studies on Thoria based MOX fuel. The major goal of fuel development programme is to develop an analytical approach for predicting the behaviour of power reactor fuels and to compare results of the in-pile and out-of-pile test with prediction. These irradiations have helped in pursuing that goal.

9.0 References:

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