FUEL DEVELOPMENT & MANUFACTURING PROGRAMME IN INDIA AND ADVANCED FUEL DESIGNS

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ABSTRACT

The emphasis on self reliance in all areas of nuclear fuel cycle technology is the objective of Department of Atomic Energy, India. To achieve this aim, various organisations are working in close co-ordination. This paper contains a brief summary of the work carried out in India on PHWR fuel technology.

INTRODUCTION

The Department of Atomic Energy is responsible for the development and application of Atomic Energy in India. Nuclear Power is a major part of this. It has been our policy from the beginning to develop indigenous capability in the entire nuclear fuel-cycle --- prospecting & mining through fabrication and reactor irradiation to reprocessing and waste treatment.

The power plants in India are designed, constructed & operated by Nuclear Power Corporation (NPC). Presently 8 units of 220 MWe PHWRs are in operation and 4 more are under construction. These units use 19 element fuel bundles. The construction of two 500 MWe PHWRs is being taken up and these reactors will use 37 element fuel bundles. The fuel bundle currently in use are split spacer graphite coated bundles.

Nuclear Fuel Complex (NFC) at Hyderabad manufactures fuel assemblies for use in power reactors. Bhabha Atomic Research Centre (BARC) at Bombay is the research centre for development of thermal reactor fuel cycle. BARC has facilities for qualification testing of fuel and fuel handling system, in-pile loops for irradiation testing and Post Irradiation Examination (PIE) facilities besides a good analytical development capability. The hot cells at Indira Gandhi Centre for Atomic Research (IGCAR) is used for the PIE eventhough IGCAR is mainly working on fast reactor development.

This paper summarises the activities of all these organisations in thermal reactor fuel cycle.

FUEL DESIGN AND PERFORMANCE

The nineteen element wire wrap CANADIAN fuel design was used in 1971 for the RAPS Unit-1, the first PHWR built in India.

In the original RAPS fuel design, the inter-element gaps and bundle to coolant tube gap were maintained by helical wires and bearing pads wrapped and spot welded around the elements. In order to avoid the possible fretting damage caused by these wires to the neighbouring element sheath surface, skewed split spacers and short length bearing pads replaced the helical wire design. This design change was primarily prompted by information available from Canada on the fretting concern. The structural redesign of spacing arrangement is thus essentially same as adopted in Canada. However, the difference exists in the fabrication related design details of these spacers and bearing pads.

In the PHWR fuel manufactured in India, the spacer pads and bearing pads are attached to the sheath by resistance spot welding technique. This was found to be safer, simpler and cost effective technology as compared to the beryllium brazing .

In order to overcome the fuel failures induced due to power ramps or stress corrosion cracking of zircaloy, and to increase the potential of the fuel element to resist the power ramp without failures, the development work on graphite coating of the inside surface of the sheath was taken up. Prototype fuel elements fabricated were test irradiated with encouraging results. This graphite coating technique has been adopted in fuel manufacturing process since 1989 at NFC.

In addition to this change, based on the reactor operating experience, further evolutions in design and fuel management have taken place. The performance of fuel has been steadily improving. The experience gained with 19 element fuel bundle has led to the designs of 37 element fuel bundle for 500 MWe reactor and 22 element fuel bundle design for use in 220 MWe reactors and for use in advanced MOX fuel cycles.

THORIUM OXIDE FUEL

India's three stage nuclear power programme has been drawn with an aim to use the large thorium resources available in the country in the second and subsequent stage of the power programme, which envisages fast breeder reactors and heavy water reactors using advanced fuel cycles. Fuel cycles to use Thorium, Thorium along with U-Pu MOX and Thorium with U-233 bearing fuels have been worked out in this context for PHWR's. Some of these cycles like Once Through Thorium cycle (OTT) envisages use of thorium in PHWR's for extended burnups. There has been a need therefore, to gain experience in irradiation of Thorium in power reactors.

The PHWR produces full power under conditions of equilibrium fuelling by having two burnup zones, the inner zone having a higher discharge burnup than the outer zone. This causes the power distribution to be flat in the centre. But when the reactor is started up, at the beginning of its life, all the fuel is new and there is a power peaking at the centre of the core. In many of the earlier reactors, this was tackled by loading depleted uranium bundles. The depleted uranium was obtained from reprocessing plant. The studies have shown that it is possible to use ThO_2 bundles for initial flux flattening without any adverse impact on shutdown system worths.

Four ThO_2 fuel bundles were loaded in MAPS-1 reactor to study their reactor worthiness. These bundles were in the reactor for 280 FPD. Based on this experience, ThO_2 bundles were used in KAPP-1 & KAPP-2 for initial flux flattening. The ThO_2 bundles have seen a maximum power 315 KW and burnup of 4000 MWd/Te.

Till now 74 ThO_2 fuel bundles have been loaded in the reactors and have a zero defect rate.

PLUTONIUM UTILISATION

The trend towards increase in fuel discharge burnup has been mainly guided by the desire to shrink the spent fuel inventory and to improve the utilisation of uranium resources. The increase in discharge burnup results in a reduction in fuelling rate and hence the load on the fuelling machine.

The best way of utilising plutonium will be in Fast Breeder Reactors. The evaluation of plutonium recycling strategies indicate that the reduction in worth of plutonium in FBRs after one recycle in PHWRs is very small. This is consistent with the studies in LWRs. After analysing different combinations of enrichments of various elements of nineteen element cluster, a configuration with central 7 rods having 0.4% W/O PuO_2 in natural UO_2 and outer rods containing natural UO_2 was selected. This cluster is capable of giving an average discharge burnup of 10300 MWd/Te. The studies on fuel management pertaining to the change over from natural UO_2 core to MOX core has been completed.

The process flow sheet, specifications and quality control documents were evolved. A six pin cluster is undergoing irradiation in Pressurised Water Loop in CIRUS reactor. This cluster has seen a burnup of around 8000 MWd/Te. This irradiation will provide us confidence in our design & fabrication procedure.

22-ELEMENT FUEL BUNDLE DESIGN

A fuel design to generate 15% higher power and having almost the same overall bundle diameter, so as to suit the existing coolant channel diameter has been developed. The design is compatible with the existing fuelling machines and fuel transfer system.

The 22 element fuel bundle design has 14 elements of 13.04 mm diameter in the outer ring, 7 elements in intermediate ring and the central element are of 15.2 mm diameter. This design makes the radial distribution of power amongst fuel elements more uniform. Thus it is possible to obtain a considerably higher bundle power while operating within the permissible linear heat generation rate (LHGR) in the maximum rated element locations in the fuel bundle or provide for additional operating margins.

The prototype 22 element bundles manufactured were subjected to a series of out-of-reactor tests to evaluate the performance with regard to the pressure drop, strength to withstand loads by fuelling machines under normal and off normal conditions, endurance test etc. Fuelling operations were demonstrated by fuelling machine compatibility test conducted on fuelling machine test loop. Two channel load of 22 element fuel bundles were irradiated successfully in MAPS reactor core. About 560 number of 22 element fuel bundles were fabricated further to evaluate production process and are undergoing irradiation.

DEVELOPMENT OF FUEL

The fuel design & manufacture is supported by R&D infrastructure at BARC.

The out-of-pile test loops simulating the reactor operating conditions of temperature, pressure and flow have been extensively used in the qualification of fuel and fuelling machine. The qualification tests include pressure drop test, strength test, wear test and fuelling machine compatibility test. The low temperature facilities have been used to study the flow characteristics of fuel at low Reynolds Number and for studying vibration behaviour. These facilities have been used extensively for qualifying special operating procedures.

An in-pile loop with a heat removal capacity of 400 KW is being used for irradiation testing of fuel for the past two decades. Two more in-pile loops are being added to Dhruva reactor.

In addition to testing, many technologies developed have become part of regular fuel production. The prominent development missions completed include SGMP route for pellet production, double dished chamfered pellets and low temperature sintering of UO_2. The work is in progress for ultrasonic testing of end plug to tube weld. The hot cells at BARC has been used in for the examination of fuel bundles from MAPS.

In addition to these development works, a strong analytical capability in the areas of physics design & core follow up, fuel design, performance evaluation & safety analysis has been developed.

FUEL FABRICATION

The fabrication of 19 element wire-wrap fuel bundle for half the initial charge of RAPS-1 reactor during early 70s is the first major step in the fabrication of nuclear fuel. The fabrication capability has grown to a level that the existing plant is able to meet the requirements of all the operating plants today.

In order to meet the fuel requirements of reactors under construction/planned, it was decided to establish a new 600 tpy plant and this will be implemented in phases.

The civil construction work of this project is in an advanced stage. Some of the equipment have already been received at site and some of the equipment are in the process of erection.

As Nuclear Fuel Complex is an integrated facility where Zirconium Sponge, Zircaloy tubes and other core components are also produced, a new Zirconium Fabrication Plant for clad tube fabrication is also under erection.

Presently at NFC, the Ammonium Diuranate route is followed to obtain the sinterable UO_2 powder. In the new plant, a spray drier is being added for drying ADU in preference to turbo drier presently used. This is expected to yield better flowable powder which will be helpful in overcoming certain problems down the line operations.
The slurry extraction developed in-house for extraction of uranium from the dissolution product practiced in the existing plant has been found performing very satisfactorily. The same has been retained for the new project also.

Continuous precipitation of ADU was practised over a long period. Later, based on the work carried out at BARC wherein it was observed that if the precipitation was carried out under nearly equilibrium condition, the precipitated ADU was more consistent & reproducible. This also resulted in comparatively better filterability and flowable dried ADU. The same process has been adopted in the new plants.

Considerable emphasis has been laid in the new plants on close control of parameters in powder production which is expected to further improve the consistency of powder produced. A distributed digital control system is envisaged to monitor and control various critical parameter from a centralised control room. Also emphasis is put on automation and mechanization to improve productivity and reliability.

A full line of assembly in the new plant will be dedicated to 37 element bundle fabrication. A few such bundles fabricated are undergoing type testing at BARC, Bombay.

There has been continuous attempt to improve and upgrade the processes & achieve self sufficiency in special purpose equipment at NFC. The process of resistance welding the spacer and bearing pads developed at NFC will also be presented in the following sessions. Indigenously developed equipment such as high temperature sintering furnace, end cap welding machines, resistance welding machines for spacers & bearing pads which were described during the third CANDU conference have now been time tested.

CONCLUSIONS

The development of indigenous capability in nuclear fuel cycle has been achieved to a large extent. The performance of fuel in our reactors have improved considerably by incorporating improvements in design, fuel manufacturing techniques & reactor operation so as to minimise power shock to the fuel.

The efforts in the design & development of special purpose indigenous fuel fabrication equipments have paid dividends in terms of improved productivity.

Once the projects at NFC are commissioned, NFC will be able to takeup PHWR fuel fabrication including 37 element bundle. Obviously, the new projects will have excess capacity and will be in a position to undertake any fabrication work. Similarly, the new Zirconium Fabrication Plant will have excess capacity for the fabrication of clad tubes to undertake any job work.