Shaping of nuclear fuel fabrication in India – a journey of self-reliance

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The Indian nuclear programme stresses upon the indigenous development of technologies and self-reliance. The article presents an overview of nuclear fuel fabrication for research and power reactors. In the recent past, stress is on the automation of various manufacturing and inspection processes. This results in consistency and repeatability in quality. A wide variety of nuclear fuels have been used in the research reactors and this provides options for the future. The fabrication of fuel for the research reactors is done at the Atomic Fuels Division, Bhabha Atomic Research Centre, Trombay, and the Nuclear Fuel Complex, Hyderabad has the mandate to supply fuel bundles to various power reactors. This article gives an overview of developments over the years and the present status. It also covers fuel manufacturing for a variety of research and power reactors in the country.

Keywords: Automation, indigenous technologies, nuclear fuels, quality control, self-reliance.

Introduction

THE Department of Atomic Energy (DAE) envisaged the construction of a series of pressurized heavy water reactors (PHWRs) for power production in the first phase of the Indian nuclear power programme (NPP). The Nuclear Fuel Complex (NFC), an industrial establishment of DAE, caters to the production of fuel for power reactors. The selection of the site for NFC at its present location in Hyderabad was decided by Homi Bhabha himself after reviewing other possible sites. NFC was identified for the production of fuel (uranium dioxide), zirconium alloys and other manufacturing facilities required for fuel production. Notwithstanding the initial decision to import the full initial charge, the first half charge of the fuel for the commissioning of RAPS-I was produced in the Bhabha Atomic Research Centre (BARC), Mumbai by early 1972 using imported zircaloy tubes and components. This laid the basis for the indigenous manufacture of nuclear fuel in the country. BARC developed the process for the production of finished UO_2 pellets from magnesium di uranate (MDU) from Uranium Corporation of India Limited (UCIL) and all the zircaloy components required for the reactor programme starting from zircon sand obtained from M/s Indian Rare Earths Ltd, Mumbai.

Though some of the equipment for pellet production, assembly fabrication and tube production were imported, the equipment for the entire chemical processes in uranium and zirconium were indigenously manufactured. The timeline of PHWR and BWR fuel production at NFC is as follows

Reactor-grade ZrO_2	:	April 1971
Reactor-grade Zr sponge	:	June 1971
Nuclear-grade UO_2 powder	:	February 1972
PHWR fuel tubes	:	April 1973
Nineteen element PHWR bundle	:	June 1973
Enriched UO_2 powder	:	September 1973
Calandria tubes	:	May 1974
BWR fuel tubes	:	August 1974
Coolant tube	:	August 1979

The requirement of fuel, reactor components and other commitments of NFC have since been met with complete satisfaction for the power programme implemented by DAE. Though initially finished fuel was being imported for BWRs, NFC developed the manufacturing processes for fuel through the powder–pellet route starting from UF6 of required enrichments.

The journey of nuclear fuel fabrication in India started with the fabrication of fuel for the research reactor CIRUS, and already includes fabrication of the fuel for the prototype fast breeder reactor (PFBR). The following sections give an overview of the various nuclear fuels manufactured for research and power reactors.

Nuclear fuel fabrication for research reactors

The fuel fabrication of the research reactors is done at the Atomic Fuels Division (AFD) of BARC. The facility caters to manufacturing and supply of fuel for a wide variety of research reactors. Some of the important fuels for research reactors that have been successfully manufactured are described below.

Apsara

It was decided to build a reactor of the swimming pool-type early in 1955, named APSARA to enable the scientific

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community to carry out various physics experiments and related research. The basic design of the reactor was completed in July of that year and the entire reactor, including the building was designed and made within one year by a group of homegrown experts. However, highly enriched uranium (HEU)-based fuel elements were imported from the United Kingdom. The reactor gained its first criticality on 4 August 1956, and was the first one in Asia. It was decommissioned in 2010. The fuel was in the form of an alloy with U^{235} enrichment limited to 93% w/w. Overall dimensions of the fuel elements were $73 \times 73 \times 905$ mm and a standard fuel element had 12 fuel plates. Each fuel plate consisted of 0.5 mm thick uranium–aluminium alloy meat clad with 0.5 mm thick aluminium alloy. BARC was responsible for the research and development of a variety of nuclear fuels catering to research reactors, as described later.

CIRUS

After the commissioning of the APSARA reactor, a decision was also made to build a 40 MW_{Th} NRX-type research reactor at Trombay in collaboration with Canada, later named CIRUS, to carry out engineering and physics experiments on the design of future power reactors. The fundamental reason for import of fuel for APSARA was that fuel fabrication facilities did not exist in India then, and the fuel fabrication technology was nascent. This was also realized by Bhabha and therefore in 1956, activities related to fuel development and technology upgrades for self-reliance and self-sufficiency in this area were started under the project 'FAGGOTS' in the then Metallurgy Division of BARC. This was the first major step in the country to increase the capability and expertise for nuclear fuel fabrication. Based on the available technical information, a flow sheet for the fabrication of aluminium-clad uranium metal fuel for CIRUS from metal ingot was developed. The project 'FAGGOTS' was renamed the Atomic Fuels Division (AFD) in 1964. A Uranium Metal Plant (UMP) was also set up for the production of nuclear-grade uranium metal ingots by calciothermic reduction of UF_4. The first uranium ingot was delivered by UMP in January 1959 and a prototype fuel element was fabricated in June 1959. More than half of the initial core for CIRUS was fabricated at AFD and thereafter, subsequent refuelling requirements of the reactor were also met successfully till the reactor was decommissioned in 2010.

Metallic fuel fabrication flow sheet for CIRUS included vacuum induction melting and casting, high stage rolling, beta-heat treatment and machining of uranium rods with subsequent cladding in aluminium-finned tubes, seal welding and assembly. The quality control steps included ultrasonic testing of uranium billets, pressure testing of finned tubes, eddy current testing of fin tubes, machined uranium rods and fuel pins after cladding, radiography of welds and glycol leak testing of fuel pins. Applications of various non-destructive techniques (NDTs) were introduced in India through CIRUS fuel fabrication. Initial loading of the reactor required about 10 tonnes of uranium in the form of 190 fuel elements. Each fuel assembly was about 9.75 m long and consisted of 59 components. The central active portion consisted of a uranium metal rod, 34.5 mm in diameter and 3.1 m long, cladded in 1S aluminium-finned tubes.

Zero energy reactor for lattice investigations and new assemblies (ZERLINA)

For studying various geometrical aspects (lattice parameters) of a reactor fuelled with natural uranium and moderated with heavy water, a zero-energy critical facility named ZERLINA was built in early 1961. The driver fuel for ZERLINA, which was mainly uranium metal rods clad with aluminium, was also fabricated in AFD according to the requirements and needs. The total quantity of uranium utilized in the first core of the reactor was 3.7 tonnes.

Pu reactor for neutronic investigations in multiplying assemblies (PURNIMA-1)

Keeping the fast reactor programme in view, the then management decided to develop plutonium fuel fabrication technology in the country. Therefore, the Plutonium Fuel Facility in the Radiometallurgy Laboratory at BARC was commissioned in 1968. It is worth mentioning that this facility is completely different from the uranium fuel fabrication facility keeping in view the radiotoxicity and criticality hazards associated with plutonium-bearing materials. To start with, this facility was only a laboratory-sized installation and designed with the flexibility for safe handling of different types of plutonium-bearing materials, including alloys, oxides, carbides, etc. The first major plutonium fuel fabrication job executed in this facility was the manufacture of full core requirement of 186 fuel elements for the zero-energy fast reactor PURNIMA. This was India's first critical facility with plutonium-based fuel and attained criticality on 22 May 1972; it was later renamed PURNIMA-1. The fuel elements consisted of a stack of sintered PuO_2 pellets, with an average density of 90% theoretical density (TD), which were tightly held by a spring between a pair of 80 mm long axial molybdenum reflectors in a 0.5 mm thick stainless steel cladding tube (495 mm long and 11 mm outer diameter). The method adopted for the fabrication of the fuel elements comprised of production of ceramic-grade sinterable PuO_2 powders through the Pu(IV) oxalate route, and manufacturing of dense pellets by cold compaction followed by high-temperature sintering. The tube had helium as filler between the fuel pellet stack and clad, and was encapsulated by TIG-welded endplugs. Of the 495 mm fuel pin length, the active fuel column was 180 mm long. The finished fuel elements were

subjected to rigorous dimensional check, X-ray radiography examination, helium-leak detection and checking of surface contamination. These fuel elements were arranged in a triangular array. The reactor after serving its intended purpose was decommissioned in 1973.

Pu reactor for neutronic investigations in multiplying assemblies (PURNIMA-2)

With the impetus on thorium utilization for making the Indian nuclear programme self-reliant, the initial development work on thorium and thoria was started in 1956. At that time there was the requirement of thoria for irradiation in the J-rod annulus of CIRUS for conversion studies. In total around 90 positions were available for the insertion of J-rods. These rods were also fabricated in AFD using low-density thoria pellets (8 g/cm^3) and thereafter stacking these pellets in 1S aluminium finned tube. Thoria pellets of such low density were fabricated by carrying out sintering at 1200°C for 1 h to facilitate dissolution and extract U^{233}. The uranium-233 extracted after irradiation of thoria J-rods in CIRUS was later utilized for studying the core characteristics of the first homogeneous-type reactor in India called PURNIMA-2. In this reactor, U^{233}-based uranyl nitrate solution was used as fuel with a beryllium oxide reflector to make the reactor critical on 10 May 1984. The reactor achieved its criticality with 397 g of U^{233} and was used to carry out many criticality experiments. PURNIMA-2 made significant contributions to the development of technology for U^{233}-fuelled reactors and was decommissioned in 1986.

Dhruva

In the early 1970s, a decision was made to set up a research reactor with higher neutron flux to meet the growing demand for radioisotopes and also for advanced research in basic sciences. Accordingly, a project named R-5 was initiated to design and erect a 100 MW_{Th} research reactor with natural uranium as fuel and heavy water as moderator and coolant. The reactor after attaining criticality on 8 August 1985 was named Dhruva. Under R-5 project, not only were the reactor design and fabrication indigenous, but also the entire fuel fabrication scheme including the selection of materials, feed and quality assurance. The experience and confidence gained in the fabrication of fuel elements for CIRUS for several years were successfully transformed into the development and fabrication of fuel for the Dhruva reactor at AFD.

In contrast to CIRUS where each fuel assembly is a single-pin design, the fuel cluster for Dhruva consists of seven pins to provide a higher surface area. The fuel assembly consists of a fuel cluster, aluminium shield, and seal and shield plug. Each fuel cluster of Dhruva comprises seven natural uranium metal rods each of 12.7/12.0 mm diameter and ~3 m length, canned in 1.00/1.35 mm thick 1S aluminium finned tube and assembled in 57S aluminium alloy tube employing tie plates and spacers. Fuel pins are fixed at one end of the tie plate and are guided through the other tie plate to accommodate axial growth or contraction. In all, there are 129 fuel cluster positions in the reactor. Based on the extensive experimental results, the fuel design selected has fuel pins attached to the bottom tie plate and guided by the top tie plate. In the initial days, the fabrication flow sheet for making Dhruva fuel clusters was more or less on similar lines as that used for CIRUS fuel rods. However, because of the smaller diameter of the uranium rod and high L/D ratio, many process variables like hot rolling parameters and beta heat treatment, etc. were optimized after taking several trials or experiments.

For Dhruva fuel production, in addition to rolling, a copper tube-jacketed extrusion of the beta heat-treated uranium billet is used. The jacketed extrusion route has increased the production rate and higher yield of uranium with a significant reduction in air activity. Modification in the fuel assembly procedure from the vertical position at an elevation of 4 m from the ground to using a horizontal fixture with split bushes and rollers resulted in a ten-fold increase in the production rate. Blistering, a major cause of rejection of fuel pins, has been controlled by eliminating the pickling operation, a major source of moisture, and introducing in its place caustic soda cleaning, with a careful study of the growth rate of oxide over the uranium rod, preheating the cladding tubes and dehumidification of the assembly area.

Fast breeder test reactor

Based upon the experience gained by France on its fast reactor RAPSODIE and also on the upgraded and higher power version of FORTISSIMO, DAE, India decided to enter into collaboration with French Atomic Energy Commission (CEA) France for building an Indian fast breeder test reactor (FBTR). It was also envisaged that FBTR would be similar to the French FORTISSIMO reactor with about 40 MW_{Th} power and a neutron flux of around 3×10^{15} n cm^{-2} s^{-1}. The fuel proposed initially for FBTR was on similar lines as that for FORTISSIMO; however, to avoid the import of enriched uranium, it was planned to use mixed oxide (MOX) fuel with higher plutonium content. Therefore, a joint study was initiated on the characteristics of this type of fuel. However, it was soon realized that MOX fuel with 70% PuO_2 and 30% UO_2 showed excessive swelling characteristics due to fuel coolant chemical interaction and therefore was found to be incompatible. It was also evaluated that the extent of swelling can be lowered by reducing the O/M ratio to 1.80; however, this would result in a drastic reduction in thermal conductivity. Therefore, based on these studies, it was considered that MOX fuel with higher plutonium content was not suitable for FBTR application. During this

time, a study was also carried out to identify alternative fuels like carbide- and nitride-based fuels for FBTR. In those days, mixed carbide and mixed nitride fuels were considered to be advanced fuels since they inherently offered a much greater potential for improved fast reactor performance. Besides higher heavy atom density, lower light atom fraction and higher thermal conductivity, they also had excellent fuel coolant compatibility. Hence, it was decided to use mixed carbide fuel as the driver fuel for FBTR with a composition of 70% PuC and 30% UC. The decision to use mixed carbide as driver fuel for FBTR was challenging because of the following reasons:

(a) Fabrication of mixed carbide fuel involved additional processing steps compared to MOX fuel.
(b) Carbides are highly susceptible to oxidation and hydrolysis and are pyrophoric in the powder form, thereby requiring high-purity inert atmosphere cover to handle inside a glovebox.
(c) Stringent control of carbon stoichiometry is needed to avoid the metallic phase totally and also to maintain the higher carbide phase within the acceptable limit of less than 15 vol%.

However, as always the flip side to these issues is weighted out by the major advantages of using 70% PuC and 30% UC fuel over its uranium-rich system of MOX.

(a) It is well known that plutonium monocarbide has a small range of carbon solubility unlike uranium carbide which is a line compound. Therefore it is relatively easy to prepare 70% PuC–30% UC as a nearly single-phase material.
(b) A large proportion of higher carbide could be accepted in this plutonium-rich fuel because Pu_2C_3 has a lower carbon potential than U_2C_3 and therefore it will have less potential for clad carburization up to 1000 K.
(c) This combination can tolerate more oxygen impurity without gas-phase carburization of the cladding because the partial pressure of carbon monoxide in equilibrium with plutonium monooxycarbide is much lower than with uranium monooxycarbide.

The mixed carbide fuel fabrication facility was commissioned during 1980–83 and after sufficient trials with uranium carbide, regular production of SS 316 (20% cold-worked) clad $(U_{0.3}Pu_{0.7})C$ fuel for FBTR core was started at the Radiometallurgy Division of BARC in 1984. The first core of 23 fuel sub-assemblies (61 fuel pins per sub-assembly containing 50 g fuel pellets per pin) was completed in about a year. Fuel elements with two different compositions were fabricated in RMD, BARC for FBTR and named Mark-I and Mark-II type fuel. The Mark-I fuel composition was $(U_{0.3}Pu_{0.7})C$, while the Mark-II fuel composition was $(U_{0.45}Pu_{0.55})C$. In addition, a few MOX assemblies fabricated at BARC, Tarapur were also delivered to FBTR to understand MOX fuel behaviour and its characteristics under a fast neutron spectrum. This know-how was later transformed into the fabrication of fuel for the PFBR reactor.

Kalpakkam mini reactor (KAMINI)

As discussed in the previous section, the emphasis of policymakers was always on the indigenous development of the plutonium fuel cycle programme. Therefore, efforts were made to develop different types/combinations of fuel. Under this umbrella, Al–Pu alloy was first prepared successfully in the early 1960s by aluminothermic reduction of PuO_2 powder in the presence of excess aluminium and cryolite. It was found that the reduced plutonium immediately alloyed with excess aluminium and the cryolite fluxed Al_2O_3. In the early 1970's, Al-clad Al–Pu (≤18% Pu) plate fuel elements were fabricated using the picture-frame technique followed by roll-bonding process for nuclear physics experiments in the ZERLINA reactor. This know-how and development were later transformed in RMD, BARC for the fabrication of Al-clad Al-20% U^{233} and Al–23% Pu plate fuel elements for the PURNIMA-III critical assembly. The purpose of building the PURNIMA-III critical facility was to study the core of the KAMINI reactor which was conceptualized in those days. The PURNIMA-III reactor achieved its criticality in 1992 and after successful experiments related to different core combinations (with reflectors) for KAMINI, it was disassembled and the fuel was transferred to the KAMINI reactor site at Kalpakkam in 1996. Parallelly, the 30 kW research KAMINI reactor was built to facilitate studies related to neutron radiography, neutron activation analysis and radiation physics research. The KAMINI core required around 600 g of fissile inventory in the form of U^{233}. The core consists of 72 plate-type fuel elements in nine fuel assemblies containing eight plates each. The overall dimensions of the fuel plate were 260 mm × 62 mm × 2 mm and those of the fuel meat were 250 mm × 55 mm × 1 mm. The major steps used in the fabrication of fuel for KAMINI include master alloy preparation, fuel alloy preparation and ingot casting, ingot rolling to the desired size and shape, core preparation and picture-framing, roll-bonding, blister test, core location and trimming to the required size and swaging in the sub-assembly form.

Apsara-U

In recent times, DAE has taken a policy decision of converting the Apsara reactor core from high-enriched uranium (HEU) to low-enriched uranium (LEU). This decision is in line with the present International practice and was recommended by RERTR (reduced enrichment for research and test reactor) programme, which was later converted into GTRI (global threat reduction initiative).

Hence, it was decided to dismantle the old Apsara reactor and build a completely new indigenous swimming pool-type reactor with 2 MW power. The reactor was later named Upgraded Apsara (Apsara-U).

The decision to build Apsara-U also warranted the indigenous development of fuel with LEU. The LEU fuel fabrication for Apsara-U required the setting up of a dedicated facility to meet the initial core requirements as well as the recurring refuelling requirements of the reactor. In addition, completely new technology was required to be developed in-house to make plate-type fuel elements with uranium silicide dispersed in an aluminium matrix. The major challenge in the development of fuel was to increase the heavy-metal loading in the fuel meat by almost five times to compensate for the loss of fissile content due to the use of HEU in place of LEU. The development and delivery of LEU fuel for Apsara-U were accomplished by the Metallic Fuels Division, NFG, BARC, and a complete flow sheet was developed for making the fuel for the Apsara-U reactor. One of the unique features of the flow sheet was the development of a novel method for synthesizing uranium silicide compound by powder metallurgy route using uranium metal powder and silicon powder as starting material. An indigenous effort was initiated for synthesizing uranium silicide by powder metallurgy route. It was also realized that the product synthesized by this method does not require any additional stage for homogenization, had lower impurity pickup, and also had a single phase within the detectable limits of the instrument used for phase characterization.

The fuel plate elements fabricated for Apsara-U were completely different from KAMINI, as they are almost three times longer with very high fuel phase volume and thin cladding. The fuel elements were fabricated using picture-frame technique followed by roll-bonding operation. Multiple passes of rolling were performed to get the desired product and after necessary characterization the fuel elements were assembled to make SFA (standard fuel assembly) and CFA (control fuel assembly). The reactor requires 11 SFAs and 4 CFAs to operate at its rated power. After successfully carrying out the fuel fabrication programme the assemblies were transferred to the reactor site and the first criticality in Apsara-U was achieved on 10 September 2018.

Nuclear fuel fabrication for thermal power reactor

The supply of fuel for the power reactors, reactor core components and assemblies is the mandate of NFC. These requirements have since been met with complete satisfaction for the Indian nuclear power programme implemented by DAE. Though initially finished fuel was being imported for the boiling water reactors (BWRs), NFC developed the manufacturing processes for fuel through powder–pellet route starting from UF_6 of required enrichment. The fuel fabrication for PHWRs was divided into two separate streams, i.e. IAEA safeguarded facility for producing fuel from imported uranium raw material (Block-A) and domestic safeguarded plants for producing fuel from Indian raw materials (Block-B). Following are the key developments in the area of fuel fabrication at NFC.

UO_2 powder production

The flow sheet for UO_2 powder production can be divided into wet and dry processing (Figure 1)[1]. The ore concentrates as received from the mills are usually processed through a series of chemical and metallurgical processes for conversion into nuclear-grade UO_2. The ore concentrates are dissolved in nitric acid and converted to crude uranyl nitrate $UO_2(NO_3)_2$ solution for further purification. This solution is then purified by solvent extraction with a mixture of 30% tributyl phosphate (TBP) as solvent diluted with kerosene. Multi-stage pumper decanter cascade of solvent extractors is used, which can handle up to 20% solids in the input solution. In this process, uranium is selectively extracted into the extract stream while the impurities remain in the raffinate stream. The uranium-rich extract stream is stripped with de-mineralized (DM) water to produce a pure uranyl nitrate solution. This solution is then precipitated with vapour ammonia for the production of ammonium diuranate (ADU) slurry. The slurry obtained

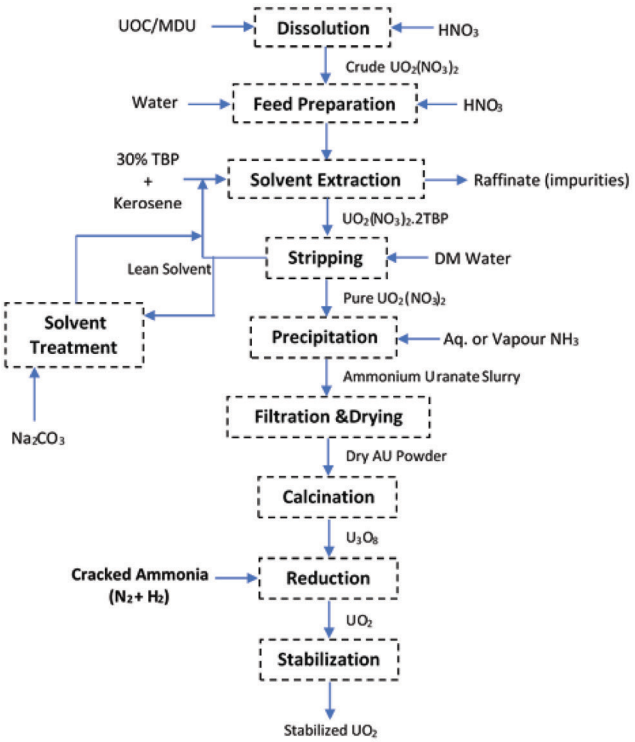

Figure 1. Typical flow sheet for nuclear-grade UO_2 powder production.

after precipitation is filtered and dried. The dried powder is calcined in air at about 600°C. In this step, the ADU powder decomposes into triuranium octaoxide (U_3O_8), which is then reduced in a cracked ammonia atmosphere to obtain nuclear-grade UO_2 powder. The UO_2 powder thus obtained is pyrophoric and oxidizes once it is in contact with air forming U_3O_8. Thus, the UO_2 powder needs to be stabilized under a controlled oxidative atmosphere to form a thin layer of U_3O_8 on the UO_2 kernel. This stabilized UO_2 powder is then sent to the pelletization section for pellet fabrication. Several improvements have been carried out in the powder production to obtain high-purity sinterable powder through a series of hydro-chemical and pyro-processing steps. Some of the major improvements in each step are described below.

Improvements in UO_2 powder processing

Dissolution and solvent extraction: Design and development of a slurry extractor in place of a pulsed column for purification was done. Capacity augmentation through the existing infrastructure was carried out by adopting several process-intensification techniques across all process steps. Positive pressure dissolution, hydrodynamic modifications in slurry extraction, introduction of mechanical coalescers in stripping (back-extraction) and introduction of additional heaters in drying increased the production capacity[2].

High-efficiency, high-capacity scrubbers were designed and developed to effectively scrub NOx from off-gases before releasing it into the atmosphere. Processing of indigenous materials like sodium di-uranate (SDU) posed a significant challenge in dissolution and solvent extraction due to the presence of a high concentration of carbonates, silica, zirconium and dissolved organics. The challenges of high frothing and crud formation in dissolution and inhibited phase separation in solvent extraction were mitigated through the development of a novel dissolution technique using a phosphoric acid complexing agent.

The quality of powder production was greatly improved with the adoption of PLC-based SCADA control system. Automated scrubber slurry transfer system for calcination furnace has led to the elimination of manual scooping, which in turn has resulted in the ease of operation and significant lowering of shop-floor contamination levels, airborne activity, radioactive exposure and man-shifts/month.

Precipitation: Equilibrium precipitation in place of continuous precipitation for improved consistency. The batch-type process was followed to have better control on the final powder characteristics like morphology, etc. With the aim of producing higher sinterable UO_2 powder, the ADU precipitation process was modified using vapour ammonia instead of ammonium hydroxide. Due to this, the process for the preparation of ammonium hydroxide was eliminated and precipitation time was reduced from 22 to 1.5 h, i.e. by 15 times.

Raffinate cake processing: A novel method for calcination of UNRC and converting it to processed uranium cake (PUC) with a reduction of volume by 80% has been developed and adopted. Presently, nearly 50 MT of UNRC is being calcined per month. In Block-A, uranium from PUC is recovered through dissolution, whereas in Block-B an MoU has been signed with M/s IREL for recovery of uranium from PUC.

Drying, calcination and reduction: A high-efficiency turbo drier (stirred bed) was designed and incorporated in place of a manual drier (static bed). Further improvement was made by adopting the spray-drying process which avoids the formation of agglomerates during drying. A rotary tubular furnace for calcination in place of tunnel type was conceived for continuous operation with mechanical seals to avoid powder activity.

Processing of SDU: SDU produced from Tummalapalle through alkali leaching contains strong anionic complexes like carbonates and acid insolubles which hydrolyse the solvent[3,4]. These anionic complexes mainly result in crud formation with the solvent TBP due to the formation of stable complexes of uranium. An innovative, simple and cost-effective technique for processing SDU by introducing a novel reagent in the dissolution has been developed. This has inhibited the formation of all strong stable uranium complexes, thereby preventing the detrimental effect on the solvent during extraction. This technique was adopted at a plant scale and sinterable-grade UO_2 powder has been produced through this route. This is a significant step as larger quantities of this raw material are expected to be supplied by UCIL in the future.

Direct reduction in powder processing: Manufacturing of pellets from UO_2 powder generated from SDU source raw material was observed to have inconsistent and fluctuating sintering density. To solve this issue, a novel method of direct reduction of optimized ADU has been developed and adopted for processing indigenous material in Block-B. This has eliminated the calcination step leading to shortening of the process flow sheet, elimination of U_3O_8 sampling, power savings to the extent of 1000 kWh/MT and saving of manpower. These developments are highly effective and helped to achieve greater than 95–97% sintering acceptance on par with the international benchmark.

Improvements in pellet production

Figure 2 shows a typical flow sheet for UO_2 powder–pellet production. Qualified UO_2 powder lots are processed through the powder–pellet route. The poor flowability of UO_2 powder due to fine particle size requires pre-compaction and granulation operations. The granulated powder is mixed with organic lubricant for reducing die-wall friction

during the final compaction operation. In final compaction, the powder is pressed into green pellets which are sintered in a reducing atmosphere at a high temperature to attain specified high-density pellets with the required O/U ratio. The O/U ratio and sintered density of the pellets affect the thermal conductivity which controls the centre-line temperature and fission gas release. Pellet ends are designed with dishes to accommodate thermal volumetric expansion of the plastic core of the pellets. The pellet ends are chamfered on the edges of the flat pellet surfaces to minimize chipping during loading and subsequent element-handling. Chamfering also reduces sheath strain at the pellet interfaces. Following are the key improvements in dry powder processing and UO_2 pellet production[5].

Pre-compaction: Development and adoption of roll press pre-compaction and granulation is a key process development. Using a comill with the roll press, the pre-compacted powder is granulated to the required size fraction. This reduces the airborne activity drastically due to the high output of the roll press. In this process, the extent of fine powder which has poor sinterability can be controlled effectively[6].

Final compaction: Adoption of admixed powder lubricant in place of liquid die-wall lubricant, adoption of high-performance compaction die made in tungsten carbide, development and adoption of high purity 100% organic powder lubricant (Tristar) for replacement of zinc-sterate are the major process developments incorporated for improving the quality and throughput of green pellet production.

A state-of-the-art rotary powder compaction press was adopted for the final compaction and production of green UO_2 pellets in Block-A. This resulted in an increase in productivity by 2.5 times, yielded more consistent density in green pellets and prevented the generation of contaminated hydraulic oil, unlike earlier hydraulic presses. This rotary press technology has helped in introducing automatic boat charging, which is an integral part of the press. The imported version of the hydraulic press has been successfully indigenized through reverse engineering. Figure 3 shows the UO_2 pellets in both green and sintered form.

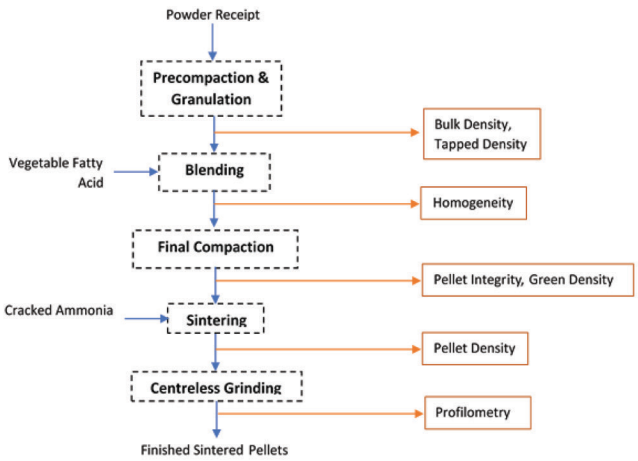

Figure 2. Typical flow sheet for UO_2 powder–pellet production.

Sintering: The pellet ends are chamfered on the edges of the flat pellet surfaces to minimize chipping during loading and subsequent element handling. Chamfering also reduces sheath strain at pellet interfaces. Molybdenum heated, multi heat zone pusher type reducing atmosphere high temperature sintering furnaces are employed for sintering of green pellets. The temperatures in each preheat and high heat zones are controlled through PID controllers. These furnaces are indigenously developed with complete automatic charging and discharging of pellet-boats.

Development and adoption of high-performance dispersion-strengthened moly alloy (MLR) for sintering furnace charge carried have significantly increased the charge carrier life by 2–3 times compared to the conventional TZM alloy used previously.

Towards increasing the production capacity in sintering operation, high-capacity boats were designed for better interaction of cracked gas with the pellets. Capacity per boat was increased by 30% through this design modification. Also, modification in the charging and discharging tunnel of the sintering furnaces was carried out in-house for adopting these high-capacity boats.

Automation in pellet processing: Automation in the pellet fabrication process has eliminated manual handling leading to a reduction in handling defects, spillages and radiation exposure to operating personnel. Automated granule charging into final compaction presses, automated green pellet density measurement system, PLC-based green pellet boat charging system, automatic sintered pellet discharging to centreless grinder through bowl feeder replacement of manual sintered pellet stacking and loading operation with a vision-based robotic system, and automatic guided vehicle for transportation of green pellet boats to sintering furnaces are examples of highly effective self-reliant automatic systems incorporated successfully in the high-capacity production lines.

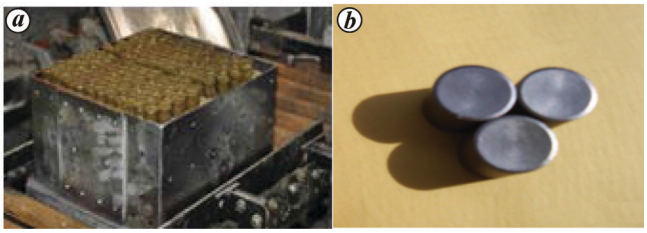

Figure 3. ***a***, Green UO_2 pellets in molybdenum boat. ***b***, Sintered UO_2 pellets.

These innovative developments along with the incorporation of high-productivity automatic equipment in powder production and pellet production have significantly increased the fuel production capacity progressively from the year 2000. The quantum of annual UO_2 powder production was increased by 20–30% by developing oxidative dissolution of rejected sintered UO_2 pellets. Thus, the challenge of recycling sintered UO_2 rejects was solved.

There was a gap between the cumulative fuel annual requirement and the supply of uranium raw material from UCIL. Only after the relaxation of guidelines of the Nuclear Suppliers Group to facilitate international civil nuclear trade[7], could India import natural uranium ore concentrate (UOC) from countries like France, Kazakhstan and Canada, and import UO_2 pellets from Russia. PHWR fuel produced at Block-A is delivered to IAEA-safeguarded PHWRs at Rajasthan, Kakrapara and Narora (total of 10 PHWRs 220 MWe). PHWR fuel fabricated at Block-B is delivered to six nos 220 MWe PHWRs and two nos 540 MWe PHWRs operating at Kaiga & Madras and Tarapore respectively. After the bifurcation of PHWRs and fabrication facilities at NFC. There is no shortage of natural uranium, and hence the operating reactors are run at 85–90% capacity factor.

Manufacturing of zirconium alloy components for fuel bundle

Zirconium-based alloys are used as a material for both fuel element cladding and in-core structural components in water-cooled thermal reactors. Zircaloy-2 is used in the BWR fuel bundle components and Zircaloy-4 is used in the PHWR fuel bundle components. Following is a brief description of the manufacturing of these alloys for fuel.

The impurities in zircon are purified, separation of silica is carried out by fusion with NaOH and removal of sodium silicate after leaching with water followed by solvent extraction using TBP, followed by precipitation by NH_4OH and calcination to produce Hf free zirconium oxide. The high-purity zirconium metal (sponge) is made using Kroll's process. In this process, zirconium tetrachloride obtained from zirconium oxide is reduced using magnesium. Employing the vacuum distillation process, high-purity zirconium metal sponge is made from the raw cake obtained after reduction. For the fabrication of zirconium alloy components required for fuel cladding, zirconium sponge is the starting material. Pure zirconium sponge is blended with alloying elements and made in the form of compacts. These compacts are electron beam-welded to form a consumable electrode for vacuum arc melting. Using multiple vacuum arc melting, the ingots are produced. Hot forging is used for processing the ingots obtained from vacuum arc melting. Hot extrusion is used for the manufacture of tube hollows from the billets obtained after forging. During extrusion of zirconium alloys, copper jacketing is used to prevent oxidation and for lubrication during extrusion.

Sheets required for flat products like endplates, spacers, etc. are manufactured in the following manner. The ingot obtained by multiple melting is extruded into a rectangular section, heat-treated by beta-quenching and surface-machined. These machined rectangular billets are hot- and cold-rolled to sheets. The sheets are annealed in a vacuum in the vertical furnace for obtaining a flat surface. The components are punched from the sheets and strips made using above route. Zircaloy rod material is required for the components like end plug for the fuel bundle. For production of the rods, multiple times melted ingot is extruded to round billets which are beta-quenched. The machined billets are then hot-extruded to a bar shape. The extruded bars are cold-processed by cold swaging. For cladding tubes, hollow billets are obtained after pilot hole drilling and piercing. These hollow billets are beta-quenched and machined. The hollow tubes (known as mother hollows) are obtained by hot extrusion in which a conical die with a mandrel is used. Cold pilgering is used for both diameter and thickness reduction to make thin-walled clad tubes. Thermo-mechanically processed clad tubes are made by pilgering followed by vacuum annealing[8]. Figure 4 shows the flow sheet for the manufacture of zircaloy fuel tubes and components and the role of some of the critical steps in the process flow sheet.

The major challenges in the fabrication of zirconium alloys for fuel assembly and core structurals are given below[9].

- Optimizing the process route for reducing the spread in the mechanical properties like uniaxial tensile and biaxial burst properties (total circumferential elongation, TCE).
- Processing route favourable texture and hydride orientation.

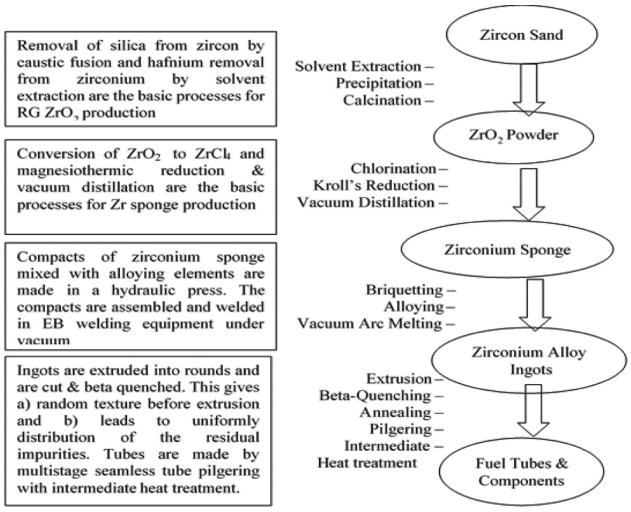

Figure 4. Flow sheet for the manufacture of zircaloy fuel tube and components, and the role of some of the critical steps in the process flow sheet.

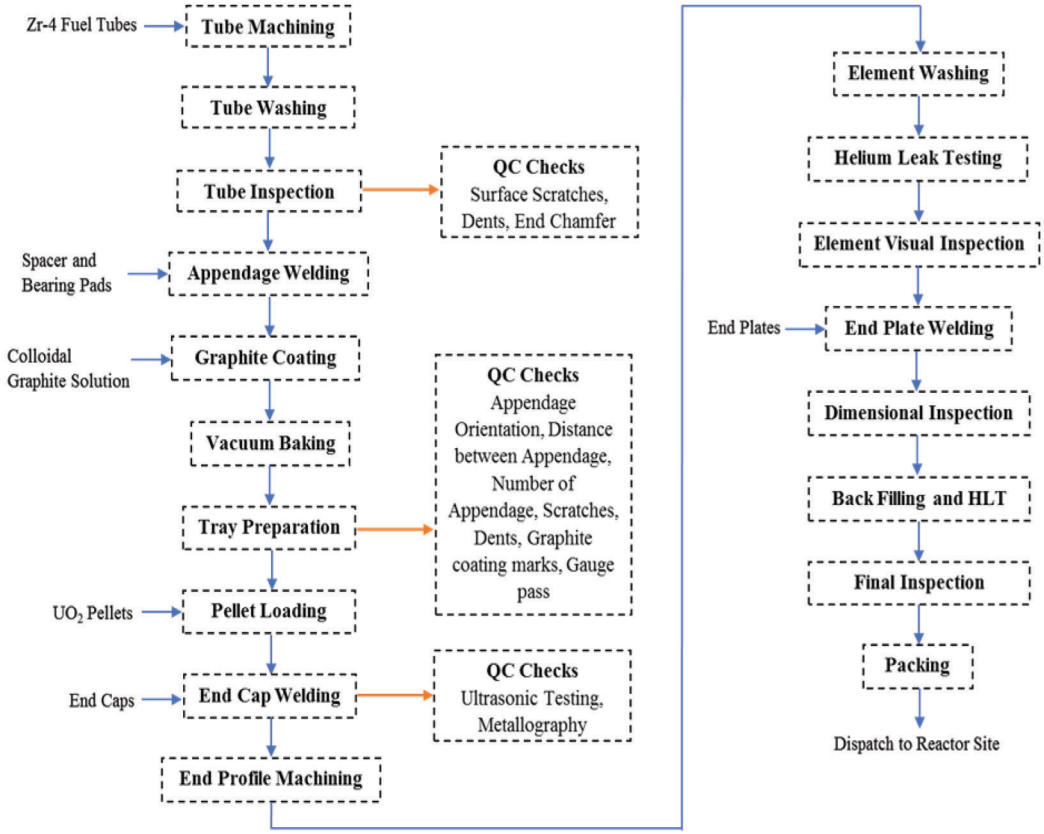

Figure 5. Typical flow sheet for PHWR fuel assembly production.

- Development of automation of visual and dimensional gauging and sorting of fuel tubing.
- Controlling the wall thickness variation and ultrasonic defects in the fuel tubes.
- Detection of fine longitudinal subsurface defects in zircaloy rod material, which could result in helium leak through the end plug of the fuel element.

With extensive studies and trials, these challenges have been overcome. One of the changes made in the process route for fuel tube production is proper preparation of the extruded blank for pilgering. The preparation operation includes machining the ID and OD surfaces of the extruded blank, and controlling the wall thickness variation before cold pilgering to tubes. For obtaining favourable texture in the fuel tube, high wall thickness reduction to the outside diameter (Q-ratio) is maintained.

Fuel assembly manufacturing

Figure 5 shows a typical flow sheet for PHWR fuel assembly production. Zircaloy tubes received from the zirconium fabrication plants are machined to the required length and profiled for welding. The empty fuel tubes in the fuel bundle are denoted as central, inner and outer elements depending on their final placement/position in the fuel bundle[10]. Zircaloy components like spacers and bearing pads are washed and de-greased before welding operations. Spacer pads and bearing pads are welded onto the thin zircaloy fuel tubes by resistance welding or by beryllium brazing method. These appendages are welded at specified positions. Zircaloy fuel tubes loaded with UO_2 pellets are encapsulated with two zircaloy end plugs on both sides using the 'resistance butt welding process'. Helium gas is purged to achieve cover gas requirement in the fuel element, which also provides an inert atmosphere during the welding operation. A high current at low voltage is passed through this joint which results in heat generation depending on the contact resistance. This heat brings the material to a plastic state and metallurgical jointing is achieved. Fuel elements, after end welding, are machined at the ends to get the required end profile and length.

The PHWR fuel bundle fabrication involves handing and joining of a large number of intricate components like spacer pads, bearing pads, end caps, end plates, etc. using resistance welding. The empty tube appendage welding technology developed earlier was automated with simultaneous spot-welding of the spacer pads and bearing pads to the tube through an in-house developed Integrated Spacer and Bearing Pad Welding Unit. The empty tube appendage welding process has been developed and incorporated

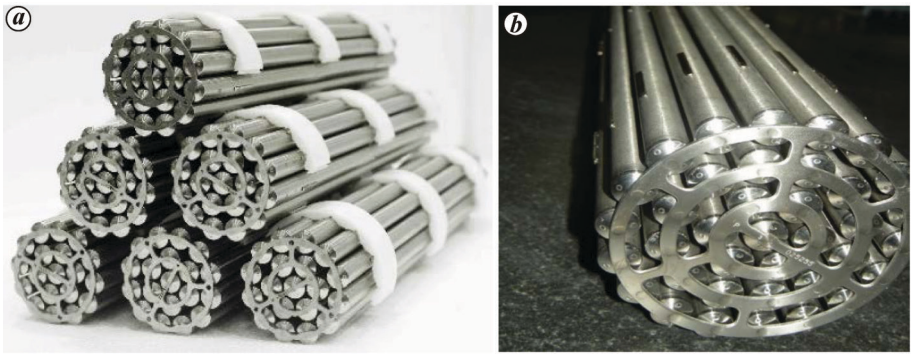

Figure 6. (***a***) Nineteen-element and (***b***) 37-element PHWR fuel bundles.

from 2003. To supply fuel to the 540 MWe PHWRs (TAPS-3&4), a separate line of assembly production was developed in 2004.

Based on the experience during the initial production, improvement of quality and productivity of various products, including nuclear fuel bundles for PHWRs and BWRs, some of the important developments in the first three decades are listed below:

(a) Development of split spacer fuel bundles design from wire-wrap fuel bundles design using a resistance welding process. The split spacer was incorporated into the fuel bundle from 1986 onwards.
(b) Thin-layered graphite coating of zircaloy tube to minimize the chances of PCI and fuel failure was incorporated with in-house equipment from 1990 onwards.
(c) A fully integrated appendage welding system was designed and developed. The machine consists of a fully automated bearing pad and spacer pad welding for all the tube varieties for the fuel bundle (inner, central and outer).
(d) Finalizing design and manufacturing of 37-element fuel bundles for 540 and 700 MWe PHWRs.
(e) Development of manufacturing practices for the production of depleted UO_2 bundles. The depleted UO_2 fuel bundles are required for the performance of flux flattening during the commissioning of PHWRs.
(d) Development of a process for the production of thorium-oxide bundles.
(g) Design and development of automated weld strength testing machine with computerized data acquisition system.

Automation in the assembly manufacturing process

All the imported end-cap welding machines have been retrofitted in-house with advanced automation handling and safety systems. New end-cap welding machines have been developed indigenously, brought into the production line with improved end-cap welding recoveries. The end-cap welding and consequent machining operations have been integrated with the production line, which has improved productivity by 150%. The six-axis robotic end-plate machine installed in fuel bundle fabrication has resulted in complete automation of the end-plate welding operation resulting in a reduction in operator fatigue, manpower savings, improvement in machine availability and consistent bundle recoveries of more than 99.5%. This system was developed for the production of 19-element PHWR fuel bundles and was later successfully deployed for the fabrication of 37-element type bundles. Figure 6 shows typical 19-element and 37-element PHWR fuel bundles.

Fuel manufacturing for BWR

For the manufacturing of BWR fuel assemblies, 4 m long zirconium alloy tubes are end-machined and the bottom plug is welded by TIG welding. The enriched pellets are loaded into the tubes and degassing operation is carried out to remove moisture from the pellets. Top plug welding is carried out for the loaded tubes to form fuel pins. The end plugs are unique for each enrichment for ease of identification. Gamma scanning is done to check the enrichment of the loaded pellets in each fuel pin. The fuel pins are then converted into assemblies by arranging them in a specified fuel matrix with the help of spacer grids and tie plates.

Fuel fabrication for fast reactors

When DAE decided to build a FBTR at Kalpakkam, AFD, BARC took the responsibility of supplying all the core sub-assemblies to the reactor. Accordingly, developmental work was initiated at AFD in 1970, that involved drafting of specifications for materials, freezing flow sheet for thorium-oxide pellet fabrication, standardization of component machining methods and their assembly processes, development of special fixtures and fabrication of prototype assemblies for type-tests. A decision was also taken at that time to set up a facility at NFC that will take up the production of sub-assemblies for the first core and also for the subsequent replacement requirements of FBTR. While

development work was being carried out at AFD, construction activity was started at NFC with the help of detailed engineering provided by the project group of AFD and Civil Engineering Division (CED), NFC. After completion of the erection and commissioning of equipment and machinery, the FBTR facility went into production in 1980.

Core sub-assemblies for FBTR are high-performance subassemblies put to severe operating conditions and hence a high standard of quality, reliability and reproducibility is maintained in every aspect of fabrication. Unlike fuel assemblies of PHWRs and LWRs, FBTR sub-assemblies are component predominant. A fuel sub-assembly of FBTR consists of as many as 511 machined components of 35 different types. All these components are required to be machined and inspected to a high degree of precision[11].

When IGCAR completed the design of core assemblies for the 500 MWe PFBR, NFC initiated developmental work for establishing fabrication flow sheet for various types of assemblies which are much more complicated than that of FBTR. NFC made all efforts and indigenized the manufacture of critical components like SS D-9 fuel tubes, hexcans and other components required for both FBTR and PFBR. NFC has been supplying all the sub-assemblies and components on time[12].

Prototype fast breeder reactor

The fast breeder reactor programme in India was conceptualized to optimize available fuel resources in the country and make the Indian nuclear programme self-reliant. Accordingly, the MOX fuel development programme was also initiated in the Radiometallurgy Division of BARC. For this, MOX fuel assemblies were fabricated initially for BWRs and PHWRs to develop technological capabilities. The MOX assemblies were later loaded in two BWRs at Tarapur in the late 1990s and later in KAPS-1, one of the commercial PHWRs. Based on the technology developed at BARC, an Advanced Fuel Fabrication Facility at BARC (presently known as the Fuel Facility under NRB, BARC) was set up with the mandate for manufacturing MOX fuel pins for PFBRs. The MOX fuel pellets fabrication for PFBR was also challenging due to its small size and also for being annular. Annular pellets are required for accommodating fission gas and also lowering fuel centreline temperature.

Development of metallic fuel for fast breeder reactor

It was perceived that the huge demand and growth rate expected for nuclear power in India can only be met through the use of metallic fuels in fast rectors, which promise a high breeding ratio and low doubling time. Hence R&D related to the development of fast reactor fuels based on metallic fuels, viz. ternary U–Pu–Zr and binary U–Pu alloys was started at BARC. Two types of fuel element designs: (i) sodium-bonded U–Pu (15–19%)–Zr (6–10%) alloy and (ii) mechanically bonded binary U–Pu (Pu ~ 15%) alloy are under consideration.

Quality control in nuclear fuel manufacturing

PHWR fuel

Quality assurance through several quality checks in nuclear fuel fabrication is of utmost importance to ensure fuel performance during reactor operation. Adoption of quality control standards and automated systems has been the backbone of quality assurance activities in the processing and handling of the varied and large number of critical components[13,14].

(a) Zircaloy tube and components: Quality checks include dimensional, visual, chemical, non-destructive (ultrasonic and eddy current tests), mechanical (tensile, burst and hardness tests), metallurgical (grain size and hydride orientation tests), and corrosion (steam autoclave) tests.
(b) Ceramic fuel fabrication: UO_2 powder for physico-chemical properties like O/U ratio, specific surface area, density (bulk and tap), particle size distribution and chemical analysis (enrichment – uranium-235 content, equivalent boron content and moisture content). Sintered and ground pellets for density, chemical analysis, microstructure, visual and dimensional tests.
(c) Assembly operations: Visual inspection, dimensional, metallographic end-plug weld evaluation, cover gas analysis, mechanical shear tests for appendage welds, graphite coating thickness measurement, non-destructive testing like ultrasonic testing (weld evaluation), radiography (BWR plug welds), helium leak testing, gamma scanning, etc. is carried out during various production stages.

BWR fuel

The quality control requirements of BWR fuel assemblies include carrying out destructive and non-destructive tests on various components, fuel pins and finished assemblies. The facility is equipped with state-of-the-art digital radiography system, helium leak detectors, gamma-scanning equipment, film digitizers and a metallography facility for carrying out various tests.

Fast reactor fuel

The quality control tests include welding qualification, dimensional, visual and BDT tests on various assemblies of FBTR and PFBR. Quality control and inspection activities are carried out at NFC, IFSB located at IGCAR, Kalpakkam and various vendor sites.

Automation in inspection

Automated inspection in quality-related aspects is being carried out using various non-destructive and vision-based systems. Automated inspection of PHWR end-cap welds, helium leak testing, visual inspection of radioactive pellets and other components are carried out. The helium leak testing system is integrated with the production line, which results in a smooth flow of the production line without any offline inspection interruptions. The machine vision system for pellet inspection carries out an automated visual inspection. This helps in the reduction of man-hours required for production and reduces external radiation dose to the operating personnel. For visualizing the defect a high-speed line scan camera is used and the pellets are rotated to visualize their entire surface. The system utilizes a library of defects generated over a period to discriminate between defective and acceptable.

An in-house developed automatic ultrasonic testing system checks all end-cap welds in fuel elements. This has improved reliability in the process operations. This system has been presently integrated with the helium leak testing system for improving productivity. Automated inspection line for helium leak testing, weight and gauging for fuel bundles has been indigenously developed and adopted. This has greatly helped in the reduction in radiation exposure, operator fatigue and increased productivity.

Manual inspection of the small-sized intricate spacer and bearing pads has been replaced with high-resolution, vision-based inspection systems. Vision-based inspection system for end caps and end plates has been introduced, which measures ten-dimensional features in less than 10 sec and has increased the inspection rate by more than 30 times.

Summary

Nuclear fuel manufacturing is one of the most challenging technologies due to complexities and stringent design requirements. The operating environment is extremely harsh and safety in operation is of prime importance. Thus, robust manufacturing process of the nuclear fuel meeting the design requirements is extremely crucial. The present state of nuclear fuel manufacturing in India for the research and power reactors has evolved over nearly 70 years and has matured matching the best international standards. Self-reliance in the development of this technology is one of the most important characteristics of the Indian nuclear power programme. The present stress is introduction of automation in the production line and inspection methods. This article gives an overview of the various developments over the years and the present status. It also provides an overview of nuclear fuel manufacturing for the research and power reactors in the country.

1. Pramanik, D., Arora, U. K., Sheela and Hemanth Rao, G. V. S., Fuel pellet materials for PHWRs, and NFC fabrication experience. In International Conference on Asian Nuclear Prospects, Indira Gandhi Centre for Atomic Research, Chennai, 2010.
2. Swamy, J. V. *et al.*, Characterization of ammonium uranates for production of sinterable grade UO_2 powder – microstructure and texture. In Characterization and Quality Control of Nuclear Fuels (CQCNF), 2015.
3. Gupta, R., Kundu, A. C. and Sarangi, A. K., Uranium mining, milling and tailing disposal – best practices. In Proceedings of National Seminar on Environmental and Sociological Implications of Mining in North-East India, Shillong, 2003.
4. Suri, A. K., Innovative process flow-sheet for recovery of uranium from Tummalapalle ore. *BARC Newsl.*, November–December 2010, No. 317.
5. Saibaba, N., Recent technological advances in PHWR fuel manufacturing. In CQCNF, Hyderabad, 2015.
6. Ganguly, C., Ceramic nuclear fuels for power reactors. *Proc. Indian Natl. Sci. Acad.*, 2001, **67**(4, 5), 531–574.
7. Grover, R. B., Opening of international civil nuclear cooperation with India and related developments. *Prog. Nucl. Energy*, 2017, **101**, 160–167.
8. Tewari, R., Srivastava, D., Dey, G. K., Chakravarty, J. K. and Banerjee, S., Microstructural evolution in zirconium-based alloys. *J. Nucl. Mater*., 2008, **383**, 153.
9. Kapoor, K., Ramana Rao, S. V., Lahiri, D. and Sanyal, T., Characterization of microstructure, texture and residual stress in modified Zircaloy-4 PHWR fuel tube material using X-ray diffraction. In Conference on Advances in X-Ray Techniques in Research and Industry, 2006, pp. 117–136.
10. Hemantha Rao, G. V. S., Fabrication, characterization, quality control and non-destructive testing of PHWR fuel. In Proceedings of Workshop on Modelling and Quality Control for Advanced and Innovative Fuel Technologies, Italy, 2005.
11. Baldev Raj, S. L., Mannan, P. R., Vasudeva Rao and Mathew, M. D., Development of fuels and structural materials for fast breeder reactors. *Sadhana*, 2002, **27**, 527.
12. IAEA, Structural materials for liquid metal cooled fast reactor assemblies. IAEA Nuclear Energy Series NO NF-T-4.3, 2012.
13. Kamalesh Kumar, B., Ahmad, A., Goswami, P., Subramanian, K. S., Mistry, R. K. and Prahlad, B., Development of automated ultrasonic testing machine for PHWR endclosure welds. In Characterization and Quality Control of Nuclear Fuels (CQCNF), India, 2012.
14. Phani Babu, C., Subramaniam, K. S., Kamalesh Kumar, B., Yadav, M. K., Subba Rao, Y., Saibaba, N., Automated inspection systems for PHWR fuel at Nuclear Fuel Complex. In 13th International Conference on CANDU Fuels, Kingston, 2016.

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