Performance of Water Cooled Nuclear Power Reactor Fuels in India – Defects, Failures and their Mitigation

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Abstract

Water cooled and moderated nuclear power reactors account for more than 95% of the operating reactors in the world today. Light water reactors (LWRs) consisting of pressurized water reactor (PWR), their Russian counterpart namely VVER and boiling water reactor (BWR) will continue to dominate the nuclear power market. Pressurized heavy water reactor (PHWR), also known as CANDU, is the backbone of the nuclear power program in India. Updates on LWR and PHWR fuel performance are being periodically published by IAEA, OECD-NEA and the World Nuclear Association (WNA), highlighting fuel failure rate and the mitigation of fuel defects and failures. These reports clearly indicate that there has been significant improvement in in – pile fuel performance over the years and the present focus is to achieve zero fuel failure in high burn up and high performance fuels. The present paper summarizes the status of PHWR and LWR fuel performance in India, highlighting the manufacturing and the related quality control and inspection steps that are being followed at the PHWR fuel fabrication plant in order to achieve zero manufacturing defect which could contribute to achieving zero in – pile failure rate in operating and upcoming PHWR units in India.

1. Introduction

Nuclear power reactor and related nuclear fuel cycle technology have reached a state of industrial maturity in India. Presently, 21 reactors are in operation with installed capacity of 5780 MWe and 6 reactors with total capacity of 4300 MWe are in advanced stage of construction [1]. The nuclear power program in India was launched in 1969 with the construction and commissioning of 2 boiling light water reactors (BWR) of 200 MWe at Tarapur Atomic Power Station (TAPS 1 & 2), in collaboration with General Electric (GE), USA. The reactors are still in operation with de- rated capacity of 160 MWe. Pressurized heavy water cooled and moderated reactor (PHWR), a pressure tube type, horizontal reactor system, is the backbone of the nuclear power program in India. The construction of the first two PHWR 220 MWe units in the country was initiated at the Rajasthan Atomic Power Station (RAPS) in the mid 1960s as a collaborative venture between Atomic Energy of Canada Limited (AECL) and Nuclear Power Corporation of India (NPCIL). RAPS 1 was jointly commissioned in 1972 but RAPS 2 was delayed and commissioned in 1981, mainly with indigenous effort of NPCIL. Thereafter, NPCIL has progressively standardized and constructed 14 units of PHWR 220 and 2 units of PHWR 540 MWe Currently, 16 PHWR 220 MWe and 2 PHWR 540 units are operating at plant load factor above 80%. India's first venture on large (≥ 700 MWe) nuclear power plant and pressurized light water reactor (PWR) started with the construction of 2 units of VVER 1000 MWe, the Russian acronym of PWR, at Kudankulum (KKNP) in collaboration with Rosatom, Russia. KKNP1 attained first criticality in 2013 and was synchronized with India's southern grid in October 2013. KKNP1 is currently under first re-fuelling shut down.

The six reactors under construction include 4 units of PHWR 700 MWe, the second VVER 1000 at Kudankulum (KKNP2) and a prototype fast breeder reactor of 500 MWe (PFBR 500) at Kalpakkam. India is planning to construct several Gen III^{+} light water reactor (LWR) parks, with total installed power of ~ 40,000 MWe in collaboration with Rosatom, Russia (PWR – VVER 1000), Areva, France (PWR – EPR 1650 MWe), General Electric, USA (BWR – ESBWR1350) and Westinghouse, USA (PWR – AP1000). The LWR vendors have guaranteed life time supply of zirconium alloy clad, low enriched uranium (LEU) oxide fuel (< 5% U^{235}) The short term target is to install 14,600 MWe by 2020 but the long term goal is to generate 25% electricity from nuclear plants by the year 2050, through PHWRs, LWRs and FBRs [2, 3].

NPCIL is in charge of design, construction, commissioning, operation and maintenance of water cooled nuclear power plants in India, including the light water reactors (LWRs) from abroad. The Nuclear Fuel Complex (NFC) at Hyderabad is responsible for manufacturing of UO_2 fuel pellets, zirconium alloy clad UO_2 fuel elements /rods and fuel bundles / assemblies for the 18 operating PHWRs and all upcoming PHWR 700 units and the 2 boiling water reactors (BWRs) and all zirconium alloy core components for these reactors. The research and development support, including post – irradiation examination (PIE) of fuel and core structural component is provided by the Bhabha Atomic Research Centre (BARC) at Mumbai.

Nuclear power plants are expensive to build but cheap to operate. Hence, there is a strong economic incentive to operate the plants with high load factor. The continuing public acceptance and growth of nuclear power will also depend on reliable radiological safety and economics of nuclear electricity. Natural uranium, containing U^{235} and U^{238} is the basic raw material for LWR and PHWR fuels. These isotopes mainly emit α particles, are mildly radioactive with very long half life and pose minimum hazard from external radiation. On the other hand, the fission penetrating Y and neutron radiations, in addition to α and β particles. The fuel cladding tube made of zirconium alloy acts as primary containment or the first barrier to fission products and actinides and is also responsible for transferring the fission heat energy from the fuel to the coolant. A defect or breach in the cladding tube will cause fuel failure and lead to release of highly radioactive and health hazardous FP and actinide fuel loss into the primary heat transport system (PHTS), causing rise in the radiation fields and hazards for the plant personnel. Hence, the integrity of fuel cladding tube should remain intact. In other words, the first barrier should be robust from the point of view of safety and economics. Efforts are underway to design, manufacture and operate water cooled power reactor fuels with higher burn up and zero defect and zero failure in the hostile reactor environment of high temperature, high pressure, high fast neutron radiation level, chemical corrosion, vibration and physical stresses.

The present paper summarizes the author's experience at NFC and BARC respectively in manufacturing and PIE of water cooled nuclear power reactor fuels in India, highlighting the fabrication steps and the inspection and quality control procedures that have been upgraded at NFC to have zero fuel manufacturing defects, paving the way for zero fuel failures in reactor.

2. Fuel Assemblies (FAs) for PHWR 220, PHWR 540/700, BWR160 & VVER 1000 in India

Table 1 summarizes the major features of UO_2 pellets, zirconium alloy clad fuel pins and fuel assemblies for operating water cooled nuclear power reactors in India. The UO_2 fuel pellets are stacked into fuel columns and loaded in zirconium alloy cladding tubes which are plugged and seal welded at the ends to encapsulate the fuel. The fuel rods are then clustered in circular, square and hexagonal arrays for PHWR, BWRs and VVERs fuel assemblies (FA) respectively.

The FA and fuel rod of PHWRs are also referred to as fuel bundle (FB) and fuel element (FE) respectively. The structural components of FBs for PHWRs are made of zircaloy 4 and consist of thin walled cladding tubes, end caps, spacer pads, bearing pads and end plates. Zircaloy4 bearing and spacer pads maintain the inter element and element to coolant channel gap and are resistance-welded on to the fuel cladding outer surface. The inner surface of the cladding tubes are coated with a thin layer (~ 5 micron) of graphite. The zircaloy 4 clad UO_2 FEs are arranged in several concentric rings around a central axis. In the FB of PHWR 220, there are 19 FEs of the same diameter of ~ 15 mm and length ~ 500 mm length. These are assembled in concentric circular configuration with 12 FEs in the outer rin, 6 FEs in the intermediate ring and one FE in the centre. Likewise, the FBs of PHWR 540 MWe /PHWR 700 MWe comprise of 37 FEs that are arranged in circular configuration (1 + 6 + 12 + 18). The 19 and 37- element FBs contain ~ 15 kg and ~ 22 kg natural uranium oxide pellets respectively. Figure 1 summarizes the major features of the FBs for PHWR 220 and PHWR 540/700. The FBs are loaded into horizontal pressure tubes which penetrate the length of the calandria vessel. Twelve FBs are loaded into each pressure tube and the number of pressure tubes are 306 and 392 for typical PHWR 220 and PHWR 540/700 units in India. The FBs are loaded and unloaded on - power almost on a daily basis with two fuelling machines located on both ends of the calandria. In PHWRs, the thin – walled zircaloy 4 cladding collapses onto the UO_2 fuel pellets due to the pressurized (~ 10 MPa) heavy water coolant.

LWRs, consisting of PWRs (including VVERs) and BWRs, are vertical pressure vessel type of reactors. The FAs for LWRs have a grid structure with bottom and top nozzles and numerous grid support pieces that firmly hold fuel rods in their precise lattice position, minimizing the risk of vibration-induced abrasion and facilitating the flow of coolant water around the fuel rod. In a PWR fuel assembly the fuel rods are mostly arranged in a 14 X 14 to 17 X 17 square lattice. The PWR fuel assembly is typically 4-5 meters in length, ~ 20 cm across and contains around 500 kg of UO_2 fuel. The FAs of VVERs are similar to that of PWRs in terms of grid structure, height and UO_2 content

Table 1. Major features of fuel pellet, pin & assembly of nuclear power reactors in India

BASIS OF COMPARISON	PHWR 220	PHWR 540/700	BWR 160	VVER 1000
Fuel Pellet : UO_2 in all cases				
U^{235} Enrichment (%)	0.7	0.7	1.6, 2.1 & 2.66	1.6, 2.4, 3.3, 3.7 & 4.1
Pellet shape: Cylindrical	solid with double dish & chamfer	solid with double dish & chamfer	solid with double dish & chamfer	No chamfer, no Dishing, 1.2 mm central hole
Pellet Diameter (mm) Pellet Height (mm) Pellet Density (g/cm3) & % Theoretical Density (TD)	14.4 solid ~20 ≥ 10.60 ≥ 96	12.18 ~15 ≥ 10.60 ≥ 96	12.4 ~15 ~ 10.36 ~ 94	7.60 ~ 10 10.4 – 10.70 ~ 95- 96
Fuel Pin : Zirconium Alloy				
Cladding Material	Zircaloy 4	Zircaloy 4	Zircaloy 2	Zr-1 % Nb
Outer Diameter (mm) Wall thickness (mm) Length (mm)	15.27 0.38 493	13.08 0.38 493	14.86 0.89 3893	9.1 ~ 0.65 3840 (fuel:3530)
End Cap Welding	Resistance	Resistance	TIG	Resistance
Fuel Assembly(FA) &Core				
Fuel Rods/Elements per FA No of FAs in core	19	37	36	311 163
Weight of UO_2 per FA (kg)	~ 15	~ 22	~ 140	~ 490
Total UO_2 Full core (tons) Weight of UO_2 refueling (tons)	60 30-45	111 80-90	40 10-15	~ 80 25-27

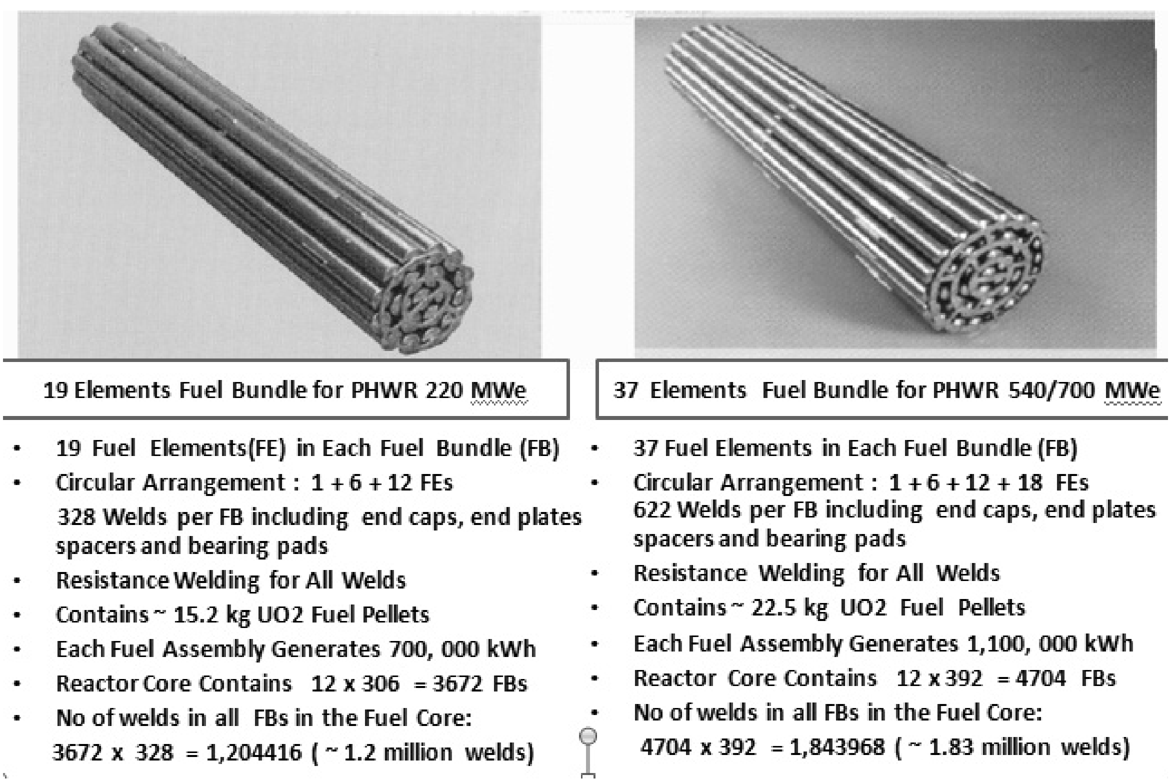


Figure 1. Inter - comparison of fuel bundles for PHWR 220 MWe & PHWR 540/700 MWe

but the fuel rods are assembled in a hexagonal configuration. Figure 2 shows the FA of the VVER 1000 at Kudankulum nuclear power plant (KKNP) and highlights the main feature of the reactor core [4]. The KKNP core contains 163 FAs arranged in hexagonal geometry. Each fuel assembly consists of 311 fuel pins, 18 guide tubes for placing burnable absorber cluster, one guide tube for keeping in core instrumentation detectors and a slotted central tube for structural support. The fuel pins and tubes are held by a framework of 15 hexahedral spacer grids and a supporting tail grid. Four types of FAs having five different U^{235} enrichments viz. 1.6%, 2.4%, 3.3%, 3.7% and 4.1% are used in the core. The initial core load consists of 54 FAs of average enrichment 1.6%, 67 FAs of 2.40% and 42 FAs of 3.62%. The design cycle duration is about 300 effective full power days (EFPDs) and the average life time of an FA in core would be 3. 37 EFPDs operating cycles. The average burn - up of the FAs under steady cycles is 43,000 MWd/tU and the maximum fuel pin burn- up will be 58,000 MWd/tU.

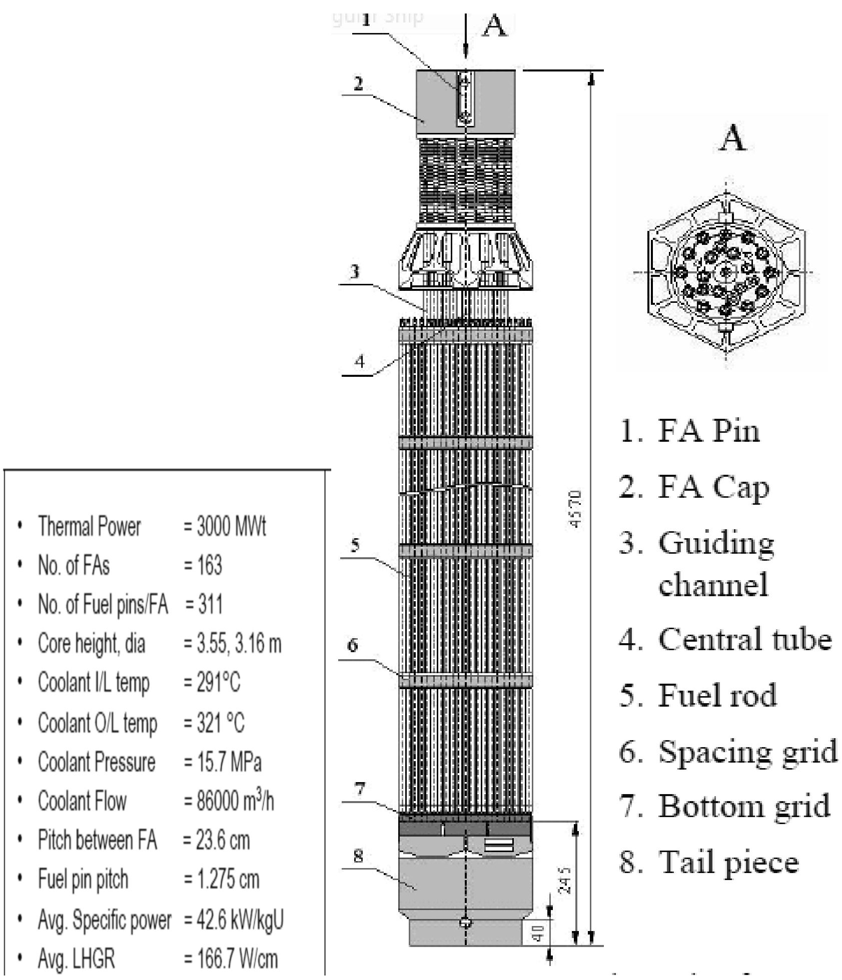
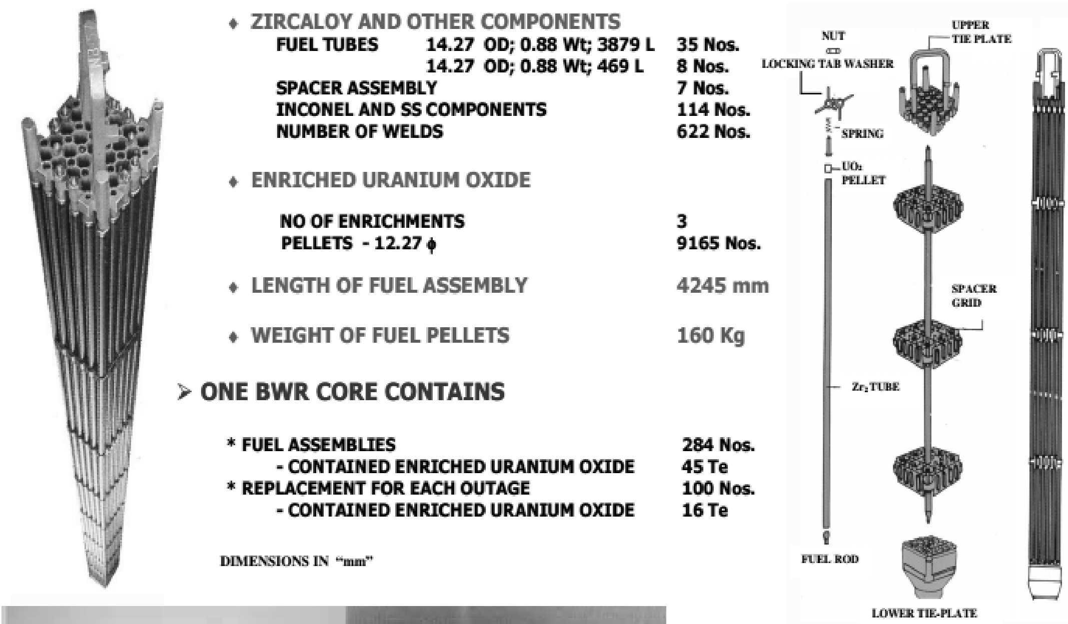


Figure 2. Fuel assembly and reactor core characteristics of VVER 1000 units at Kudankulum[4]

The FAs of BWRs also have grid structure and square lattice where the fuel rods are arranged in 6x6 to 10x10 configuration. A BWR fuel module has 4 FAs and a cruciform shaped control blade. Each FA is enclosed in a zirconium alloy sheath or channel box which directs the flow of coolant water through the assembly and is isolated from its neighbors by a water-filled zone in which the cruciform control rod blades travel. Figure 3 show typical 6 x 6 fuel assembly for the two BWR 160 MWe units at TAPS 1 and 2 [5]. Zircaloy 2 clad UO_2 pellets with U^{235} enrichments of 2.66%, 2.1% and 1.6% has been the reference fuel for the two reactors. However, as part of the program on plutonium recycling in water – cooled reactors some 12 number of 7 x 7 mixed uranium plutonium oxide (MOX) containing up to 5% PuO_2 have been successfully irradiated in TAPS 1&2 in 4 re-fuelling cycles to burn up levels of ~ 20,000 MWd/tU without any failure [6, 7].

LWRs are required to be shut down periodically for refueling outage when 25 – 35% of FAs in the core are required to be replaced with fresh fuel. For ensuring high (> 90%) plant load factor, there is a need to have high fuel burn up, longer operational cycles (18-24 months) and shorter outages of 4 to 6 weeks. The average fuel burn up for Generation III$^+$ PWRs, VVERs and BWRs are in the range of 40,000 to 55,000 MWd/tU.

PHWRs operate on on-power fuelling scheme and old fuel is replaced by new fuel on a near-daily basis, when typically, 4 or 8 fresh bundles are added/ discharged. The core never has a large excess reactivity, except after first commissioning. For PHWRs, the aim is to have the maximum fuel burn-up and the longest possible fuel dwell time in the core permissible from reactor physics stand point. The average burn up of PHWR fuel is ~ 7000 MWd/tU.



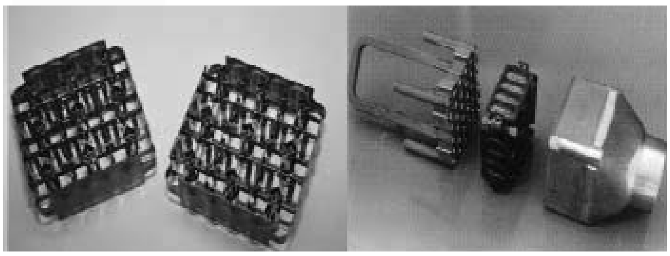


Figure 3. 6x6 fuel assembly with spacers and top and bottom sections for the two BWR 160 MWe units at Tarapur , India (TAPS 1 & 2)

3. Performance of LWR and PHWR Fuels

IAEA and OECD have published review reports [8, 9] on performance of water cooled reactor fuel. These reports provide information on the techniques of detecting leaking FAs and fuel rods, summarizes the statistical data on fuel failures, discuss their root causes and recommend failure mitigation measures.

The leaking fuel rods in LWR core is detected by I^{131} activity, several iodine isotopes concentration and uranium concentration in the primary coolant and in some case radioactive noble gas release in the off-gas system or to the environment. In PHWR too, leaky fuel bundles are detected by monitoring I^{131} in the primary coolant circuit. The location of the failed bundle in the PHWR core is performed on – power by delayed neutron (DN) monitoring of coolant samples from individual fuel channels or by DN monitoring of the reactor feeder pipes [10]. DN monitoring of individual channels on-power is usually done when the I^{131} in coolant exceeds the prefixed values. However, currently, fully automated monitoring of DN signals from individual channels is done on a regular weekly or fortnightly schedule.

LWRs can continue to operate with leaking fuel rods if the specified limits are not reached. In such case the identification and removal is usually postponed until the next planned outage. However, if the radiation level in PHTS becomes too high because of leak or failure of the fuel, the reactor has to be shut down before the scheduled fuel outage and the failed fuel assemblies have to be removed and replaced, causing economic penalty. Since PHWR re- fuelling is carried out on – power, there is no need for shutdown to remove the leaking FBs. Further, because natural UO_2 FBs for PHWRs are inexpensive as compared to FAs of LWR, reconstitution of FBs with defected FEs is not done. The FAs of LWRs are expensive. Hence, there is provision for reconstitution and re-use of failed FAs after removing the leaky rod/rods.

The fuel performance have significantly improved over the years and in several countries zero fuel failure have been maintained in several reactors. The average fuel failure rate of water cooled reactor today is in the range of 10^{-5}. The specific fuel failure rates in PWRs, VVERs, BWRs and PHWRs are respectively 13. 8, 15. 1, 4. 4 and 0.35 per 1000 fuel assemblies discharged [8, 9].

The causes of primary defects in LWR and PHWR fuels could be classified under three main categories, namely,

i) design related defects,

ii) manufacturing defects and

iii) reactor operational defects.

Debris induced fretting is the most dominant mechanism for fuel failures in all types of water cooled reactors. Wherever possible, debris filters are being introduced. For the FAs of PWR and VVER, grid to rod fretting is also a common failure. In BWRs, corrosion by itself or in combination with crud deposits is a major root cause of failure. In PHWR, flaws in the fuel elements induced during fabrication, like faulty end-plug welds and porous "piping" defects in the bar material used for end plugs, can cause coolant ingress in fuel elements leading to failure due to hydriding of zircaloy cladding.

The performance of BWR and PHWR fuels in India has significantly improved over the years though the fuel defect rate is still higher when compared to the BWRs in USA, Europe and Japan and the CANDU - PHWRs in Canada and Republic of Korea (ROK). Figure 4 shows the uninterrupted days of operation of more than a year of representative PHWR 220 MWe (RAPS 2-6, KAPS 1-2, KGS 1-2), PHWR 540 (TAPS 3) and BWR 160 (TAPS 2) units in India during the last few years till May 2015 [11]. So far, more than 3000 fuel assemblies of the two BWR 160 have been irradiated in 47 fuel cycles and in recent years there has been several cycles with zero failure.

Likewise, in the 16 operating PHWR 220 MWe units some 6,00,000 fuel bundles of the 19 - fuel element type have been successfully irradiated to average discharge burn up of ~ 7000 MWD/tU. In addition, in the two PHWR 540 units, more than 30,000 fuel bundles of the 37 fuel element type have been irradiated with significantly low failure rate [12]. In addition, fifty one 19 – element UO_2 fuel bundles containing slightly enriched uranium (0.9% U^{235}) were loaded in 14 channels of MAPS 2 and successfully irradiated to burn up of 25,000 MWd/TeU [13].

Further, fifty 19 - element mixed uranium plutonium oxide (MOX) fuel bundles containing MOX fuel pellets (0.4% PuO_2) in the inner 7 fuel elements were successfully irradiated to 20,000 MWD/TeU in KAPS 1. In addition, some 220 ThO_2 bundles of the 19-element type were successfully utilized for neutron flux flattening of initial cores of KAPS 1 & 2, KGS 1 & 2 and RAPS 2, 3 & 4 and irradiated to 13,000 MWD/Te U^{233} at maximum linear heat rating of 50 kW/m and maximum bundle power of 408 kW [13].

4. Water Cooled Reactor Fuel Fabrication in India – QC & Inspection

The feed materials for fabricating PHWR 220, PHWR 540 /700 & BWR 160 fuels at NFC are:

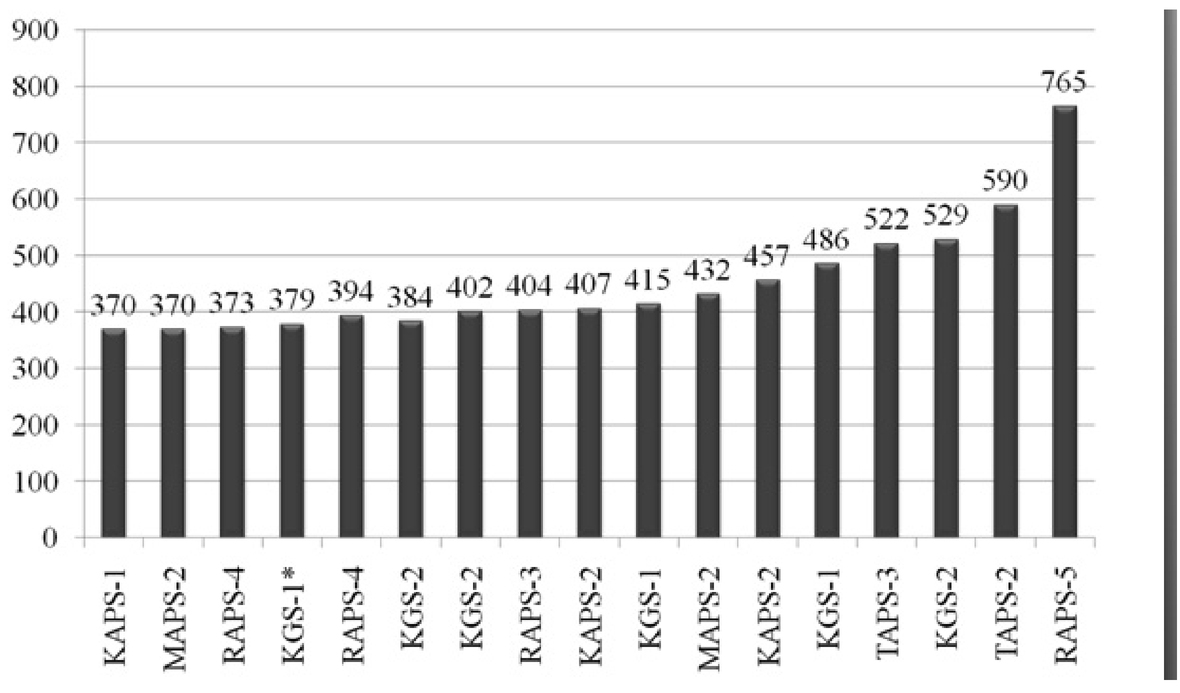


Figure 4. Uninterrupted days of operation of representative PHWR 220 (RAPS 5), PHWR 540 (TAPS 3) & BWR 160 (TAPS 2) units in India in last few years till May 2015

i) imported low enriched uranium (maximum enrichment 2.66%) for the two BWRs; earlier enriched UF6 was imported from USA, France and China but since the last 13 years, enriched UO_2 pellets are imported from JSC TVEL, Russia;

ii) natural uranium ore concentrate (UOC) from uranium mines and mills in India, operated by the Uranium Corporation of India Limited (UCIL);

iii) imported natural UOC from Areva, France and Kazatomprom, Kazakhstan and imported natural UO_2 pellets from JSC TVEL, Russia; soon, Cameco Corp., Canada will supply UOC iv) zircon (zirconium silicate) supplied by Indian Rare Earths Limited (IREL) for manufacturing Hf – free Zr sponge.

Figure 4 shows the process flow sheet followed at Nuclear Fuel Complex (NFC), Hyderabad for manufacturing 19-element and 37- element fuel bundles respectively for PHWR 220 and PHWR 540 units [5, 14]. The classical cold-pelletization followed by high temperature reductive sintering in hydrogen atmosphere (cracked ammonia), at around 1700 C in continuous sintering furnace, is

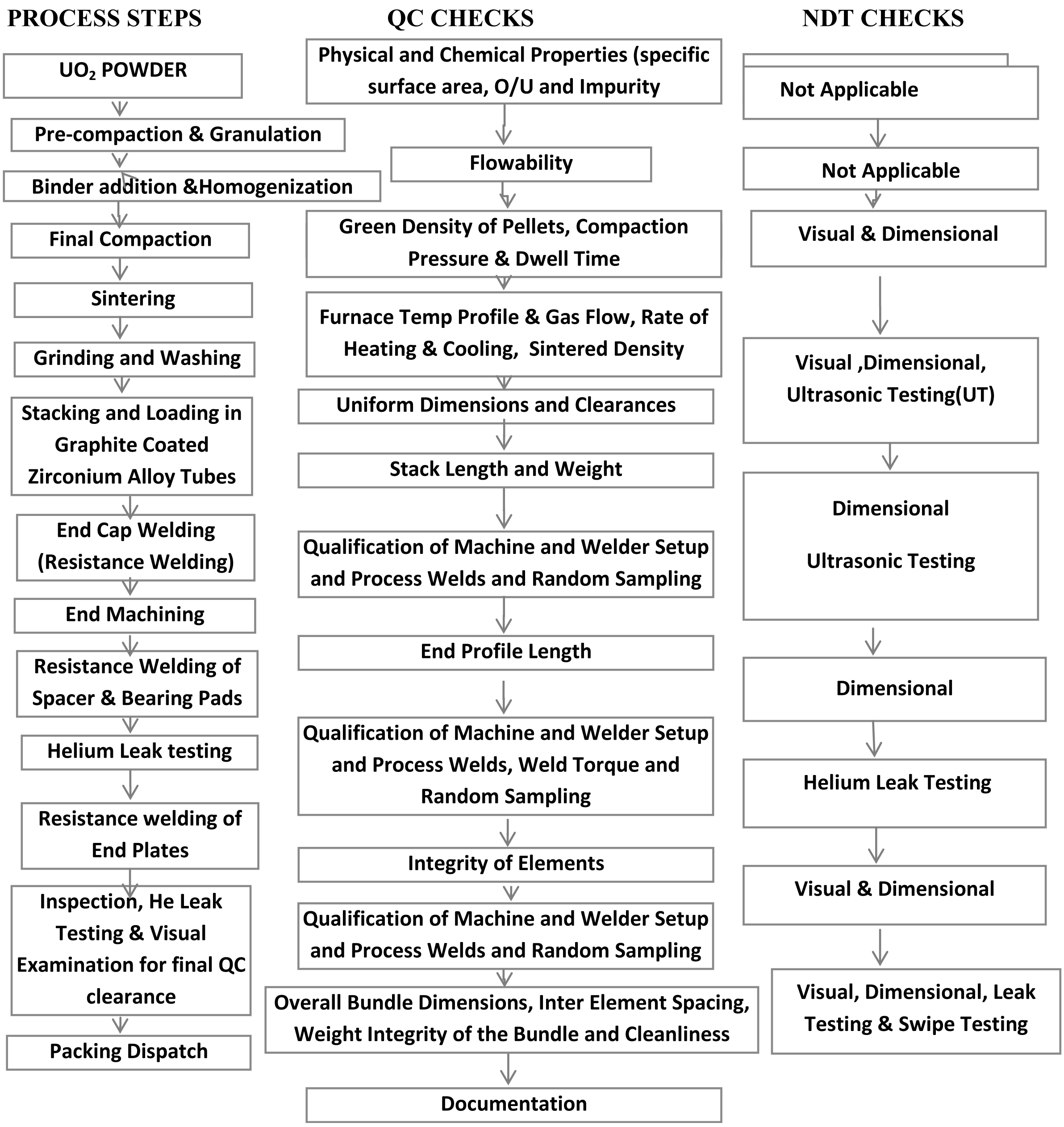


Figure 4. Process flow sheet , QC and NDT steps, followed in NFC for fabrication of zircaloy 4 clad, natural UO_2 fuel bundles for PHWRs 220 and PHWR 540/700 in India

followed for manufacturing high density UO_2 pellets. The powder compaction die – punch set is designed to obtain cylindrical pellets with double dishing and chamfer. The dishing will provide space for accommodating released fission gas and volumetric expansion of fuel pellet during operation. Chamfering will not only minimize pellet chipping but also reduce cladding strain at pellet – pellet interface and avoid 'bamboo ridging' of cladding. The UO_2 pellets should be of high density (≥ 96% TD) and have large grain size (≥ 25 micron) and controlled oxygen to metal ratio in order to have minimum fission gas release during operation. The sintered pellets are subjected to wet - centreless grinding followed by drying in oven before loading in zircaloy 4 cladding tube. The hydrogen content of the fuel pellet should be < 1 ppm in order to ensure no internal hydriding of zircaloy 4 cladding during reactor operation.

The zircaloy 2 and zircaloy 4 components for the FAs for BWRs and FBs for PHWRs respectively are manufactured in NFC, using zircon sand as feed material. Zr is an anisotropic metal with a HCP structure. The major challenge is to keep the Hf impurity to < 50 ppm, get porosity – free homogenous alloy ingots and tailored texture in all zirconium alloy cladding tubes, bars and sheets components. The major alloying elements for zircaloy 2 and zircaloy 4 are ~ 1.5% Sn, ~ 2000 ppm of Fe, Cr and Ni collectively. Zircaloy 4 has no nickel and the Fe and Cr alloying elements are adjusted accordingly. The zirconium alloys are subjected to double or triple melting in high vacuum consumable arc furnace and cast in water cooled copper crucible to obtain the zircaloy ingots. The details of zirconium technology in India are summarized elsewhere [15].

Table 2 summarizes the major steps in fabricating zircaloy cladding tubes and the quality control (QC) checks at each step. The parameters for thermo-mechanical treatments of the zircaloy ingots during hot extrusion, forging and rolling are optimized on the basis of deformation maps. The solid ingots are subjected to hot and cold rolling for sheet products, swaging for bars and multistage pilgering – annealing cycles for cladding tubes. The microstructure, texture and mechanical properties of the finished products are evaluated and the intermediate and final products are subjected to non – destructive evaluation by ultrasonic and eddy current testing.

5. Mitigation of Fuel Failures in India – towards Zero Manufacturing Defect

During the last three and half decades, a significant number of irradiated zircaloy 2 clad BWR 160 MWe fuel rods containing enriched UO_2 pellets

Table 2. Major fabrication steps and QC checks in manufacturing cladding tubes at NFC.

FABRICATION STAGE	QUALITY CONTROL CHECKS
Machined billet inspection	Visual & Dimensions, Ultrasonic Testing for Defects
Hot Extrusion & Annealing	Soaking Temperature & Time, Dimensions, Annealing Temperature & Time, Hardness, Microstructure & Grain size Ultrasonic Testing (UT) for Defects
For Cladding Tubes Multipass cold pilgering of blank to final size with intermediate vacuum annealings followed by : cold working and annealing for BWR tubes, cold working and stress relieving for PHWR tubes	Visual & Dimensions, Annealing Temperature & Time, Tensile properties and Burst properties
Grinding of outer surface and sand blasting of inner surface of cladding tubes	Visual & Dimensions, UT for defects & Wall thickness, Hydride Orientation, Metallography for Grain Size, Corrosion test & Corrosion Product analysis
Tubes cut to finished length	Visual and Dimensions , Bow
Cleaning & Shipment	Cleanliness & Packing

from TAPS 1 & 2, zircaloy 2 / zircaloy 4 clad fuel bundles containing natural UO_2 pellets from PHWR 220 stations and zircaloy 4 clad experimental pins from the pressurized water loop (PWL) of CIRUS research reactor at BARC, have been subjected to post irradiation examination (PIE) at the hot cells of BARC as shown in Figure 5 [15, 16]. Table 3 gives a list of fuel subjected to PIE at hot cells of BARC [17, 18].

Based on the results of PIE, several modifications have been made in the design, manufacturing, quality control and inspection of BWR and PHWR fuel pellet, rods, elements and fuel bundles by collaborative efforts between NPCIL, BARC and NFC.

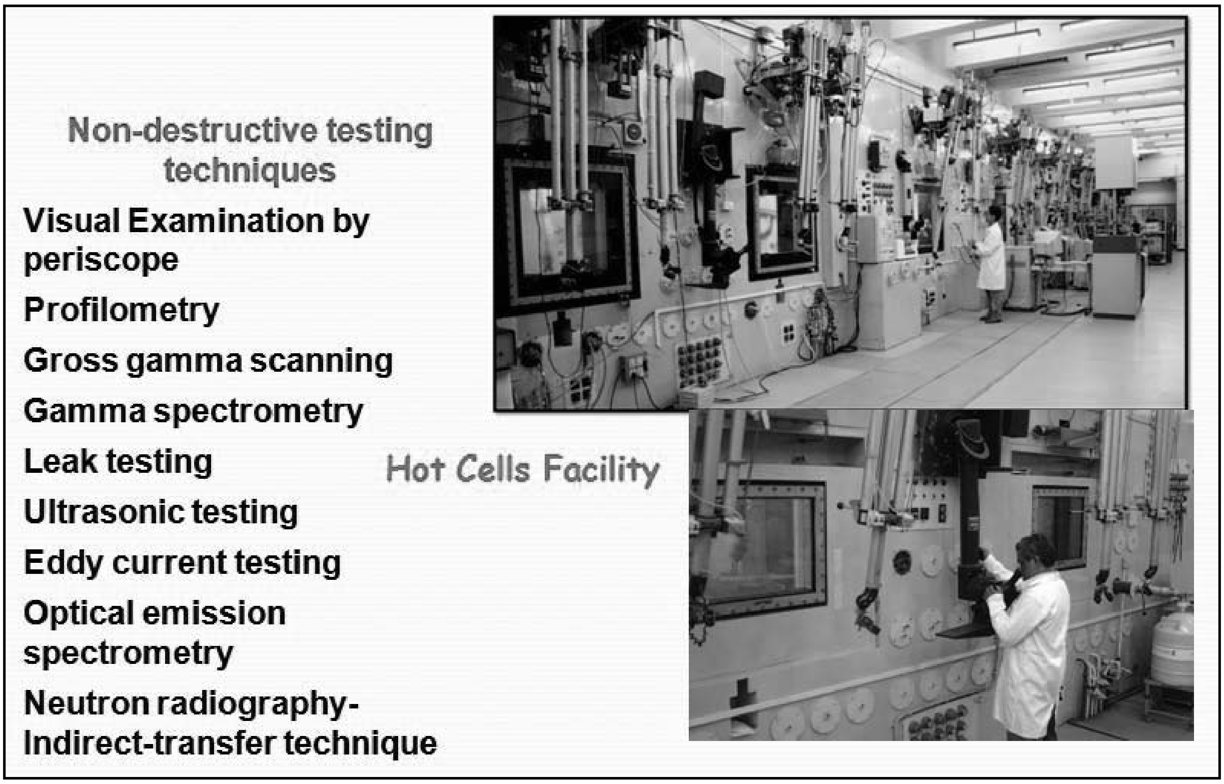


Figure 5. Hot cell facility at BARC, Mumbai for PIE of BWR , PHWR & experimental fuels

Table 3. List of BWR fuel rods, PHWR fuel bundles and experimental oxide and mixed oxide fuel pins subjected to post irradiation examination (PIE) at BARC, Mumbai

Sl No	Fuel Rods, Fuel Bundles & Test Pins	Nuclear Power Reactors	CIRUS
1	Fuel Rods (18 numbers)	BWR-160 MWe (TAPS 1 & 2)	
2	19 - Element Bundles (14 numbers)	PHWR 220 units RAPS, MAPS, NAPS, KAPS etc.	
3	37 – Element Bundles (2 numbers)	PHWR 540 (TAPS 3 & 4)	
4	9 Experimental UO_2 fuel pins		CIRUS
5	Experimental Pins : UO_2-1.5 % PuO_2 (1) UO_2- 4 PuO_2 (15)		CIRUS
6	2 ThO_2 Bundles for flux flattening	From PHWR 220 (KAPS 2)	
7	Experimental Pins : ThO_2 (2 pins) ThO_2 - 4 % PuO_2 (5 pins) & ThO_2 -6.75 % PuO_2 (2 pins)		CIRUS

5.1. Mitigation of PCI-SCC Failure of Zircaloy 2 Clad Enriched UO_2 Fuel in BWR units in India

The failure modes observed during the PIE of 18 fuel rods from the two BWR units at TAPS 1 and 2 in the 1970s were "fretting corrosion" at spacer grid site, internal clad hydriding and pellet cladding interaction (PCI)/stress corrosion cracking (SCC) [19]. Significant nodular corrosion were observed on the zircaloy 2 cladding surface but this did not cause any leaks or failure of the cladding tubes [20]. The nodular corrosion was mitigated in subsequent TAPS 1 and 2 fuel assemblies by using fully annealed zircaloy 2 cladding in place of stress relieved cladding.

The elements which failed due to PCI-SCC had been located adjacent to the control rod. Detailed metallographic examination of these elements revealed that they had experienced severe power ramp during control rod withdrawal which caused a significant rise in the fuel centre temperature, leading to PCI/SCC failure. Analysis of ultrasonic and eddy current testing data on 4000 fuel rods of failed fuel assemblies, collected at the pool side of TAPS 1 and 2 also revealed that nearly 70% of fuel failures occurred in fuel rods which were located adjacent to the control rod.

The PCI-SCC failure was averted in subsequent fuel rods by: i) helium pre-pressurization in the range of 0.25 MPa, in the annular space between fuel pellet stack and cladding internal surface, ii) using cylindrical and chamfered UO_2 fuel pellets with height to diameter ratio in the range of 0.8 to 1.2 and with flat top and bottom surfaces and iii) using cladding tube of higher wall thickness and iv) by proper conditioning during start up after shut down. Special purpose welding machines were developed at NFC for tungsten inert gas (TIG) welding of end caps under high helium pressure.

5.2. Zircaloy 4 clad Natural UO_2 Fuel for PHWR 220 & PHWR 540 units in India

The PHWR fuel element failure observed during PIE at BARC hot cells can be broadly classified as follows:

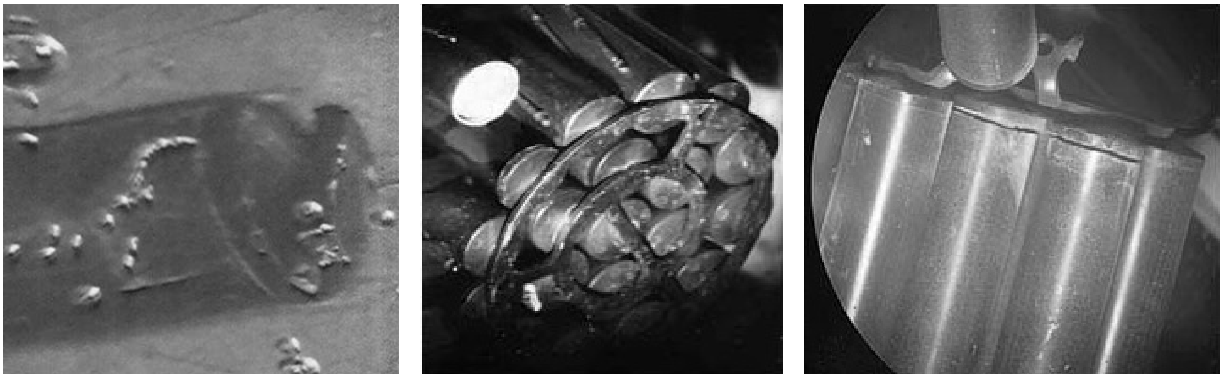
i) PCI-SCC failure due to combined effect of corrosive fission products like iodine and stress at pellet cladding interface due to collapsible cladding during power ramp;
ii) Delayed hydride cracking near end plug weld due to diffusion of hydrogen to areas of high stress and low temperature and precipitation of zirconium hydride;
iii) Manufacturing flaw arising out of micro - porosity in end plug material and also improper resistance welding of end plugs causing lack of fusion leading to failure;
iv) Debris (in the primary coolant circuit) induced fretting failure of fuel cladding;

Most of the above failures are also common for LWR fuel.

5.2.1. Mitigation of PCI-SCC failure in PHWR fuel

PHWR fuel element consist of very high density (96-98% TD) natural UO_2 pellets that are stacked and encapsulated inside thin and collapsible zircaloy 4 cladding tubes. The fuel elements are designed to operate at linear heat rating of 57. 5 kW/m and burn up of 15,000 MWD/tU. The high density UO_2 fuel pellets ensures high uranium content in the fuel and also provide support to the collapsible clad because of the external coolant pressure. In the early 1990s a large number of fuel bundles was found to fail at the PHWR 220 unit at Narora Atomic Power Station (NAPS) by PCI-SCC following power ramp. These fuel bundles were manufactured using zircaloy cladding tubes without any internal coating of graphite. Thereafter, based on the Canadian report on CAN-LUB fuel for mitigating PCI-SCC problem in PHWRs [21], all subsequent zircaloy cladding tubes for PHWR 220 and PHWR 540 units were coated with ~ 5 micron thick layer of graphite. Slurries of deflocculated graphite in isopropyl alcohol is dip-coated on zircaloy cladding tubes. The graphite coated tubes are first dried at ~ 120 C to remove traces of alcohol and then baked at 350 C for 3 hours under a vacuum level of 10^{-4} m bar in order to have an adherent graphite coating and remove all hydrogenous material in the coating. Some 5 mm at both ends of the cladding tubes are kept uncoated in order to facilitate resistance welding of the end caps at a later stage. The hydrogen content of zircaloy cladding tube before and after graphite coating should be below the specified limit of 1 ppm in order to mitigate the problem of internal hydriding during in-pile operation.

During power ramp, the graphite acts as a lubricant and reduces the frictional stress between UO_2 pellet column and the collapsible zircaloy cladding tube, arising out of the higher coefficient of expansion of UO_2 compared to zircaloy. The clad strain at pellet interface is also minimized by chamfer, double dishing and low height to di-



End cap failures — End cap separation

(a) Nitrogen gas bubbles from leaks in end plugs

(b) Dislodging of cladding tube from the end cap weld

(c) cracks in end cap region of several fuel elements

Figure 6. PHWR fuel failure due to leaks in end plugs and resistance welding defects

ameter ratio (0.8-1.2) in UO_2 pellets. The double dishing also provide additional space for accommodating fuel swelling in case of high burn up fuel. The graphite also acts as a diffusion barrier and prevents corroding fission products like iodine, tellurium etc to attack the zircaloy cladding.

In addition to graphite coating and modified pellet shape, the fuelling sequence and adjuster rod insertion are also optimized to reduce the peak power and avoid power transient on starting up following a shut down. Thus, the PCI-SCC failure is mitigated by judicious combination of graphite coating on inner surface of cladding tube, modified pellet shape (chamfer, double dishing and low height to diameter ratio) and keeping the peak power and power transient below the specified limits.

5.2.2. Mitigation of fuel end cap failure in PHWR fuel:

Post irradiation examination (PIE) of 19 element PHWR 220 fuel bundles, both wire wrap and split spacer types, have revealed several instances of leaks in end caps and failures due to end cap cracking or separation as shown in Figures 6 and 7. The end cap weld region could be one of the major sources of defects because of:

i) micro porosity in the rods from which the end plugs were made and
ii) lack of fusion because of improper resistance welding.

Liquid nitrogen-alcohol leak test on fuel pins could locate the leaks as entrapped liquid nitrogen bubble out through the leak on expansion in the warmer alcohol bath as shown in Figure 6a. The lack of fusion during resistance welding has led to circumferential split of the end plug near the weld during in - pile operation and eventual dislodging of the cladding tube from the end cap region as shown in Figure 6 (b) and 6(c). The signals obtained from a sound weld and a defective weld are shown in Figure 7. Presently, the process control of resistance welding operation is carried out by destructive metallographic evaluation of setup welds and the non destructive testing (NDT) based on pulse-echo immersion type ultrasonic testing (UT). A typical system uses 0.2 mm spot focused probe of 30 MHz. A motorized system provides both rotary and linear movement of the fuel element thereby subjecting the entire weld area. Defects are characterized based on the signal shape and amplitude. Calibration is carried out using a natural standard of leaky fuel element and fuel element with 0.04 mm deep, 600 ‘V’ notch on ID and OD of the fuel tube. In addition to UT of set up welds at the beginning of encapsulation campaign, it is essential to carry out 100% inspection of all end cap welds by immersion ultrasonic testing in order to intercept any lack of fusion or other welding defects that could cause fuel failure.

Figure 7 shows typical lack of fusion failure during resistance welding of end plug and the corresponding ultrasonic signal. Ultrasonic signal from a good end plug weld is also shown in Figure 7 for the sake on comparison of an accepted and a rejected weld. Thus 100% inspection of end plug welds by UT will pave the way for zero welding defect of end plugs.

5.2.3. Dislodging of bearing / spacer pads leading to debris – fretting failure:

Presently at NFC, curved bearing pads and spacer pads are resistance – welded on the zircaloy 4

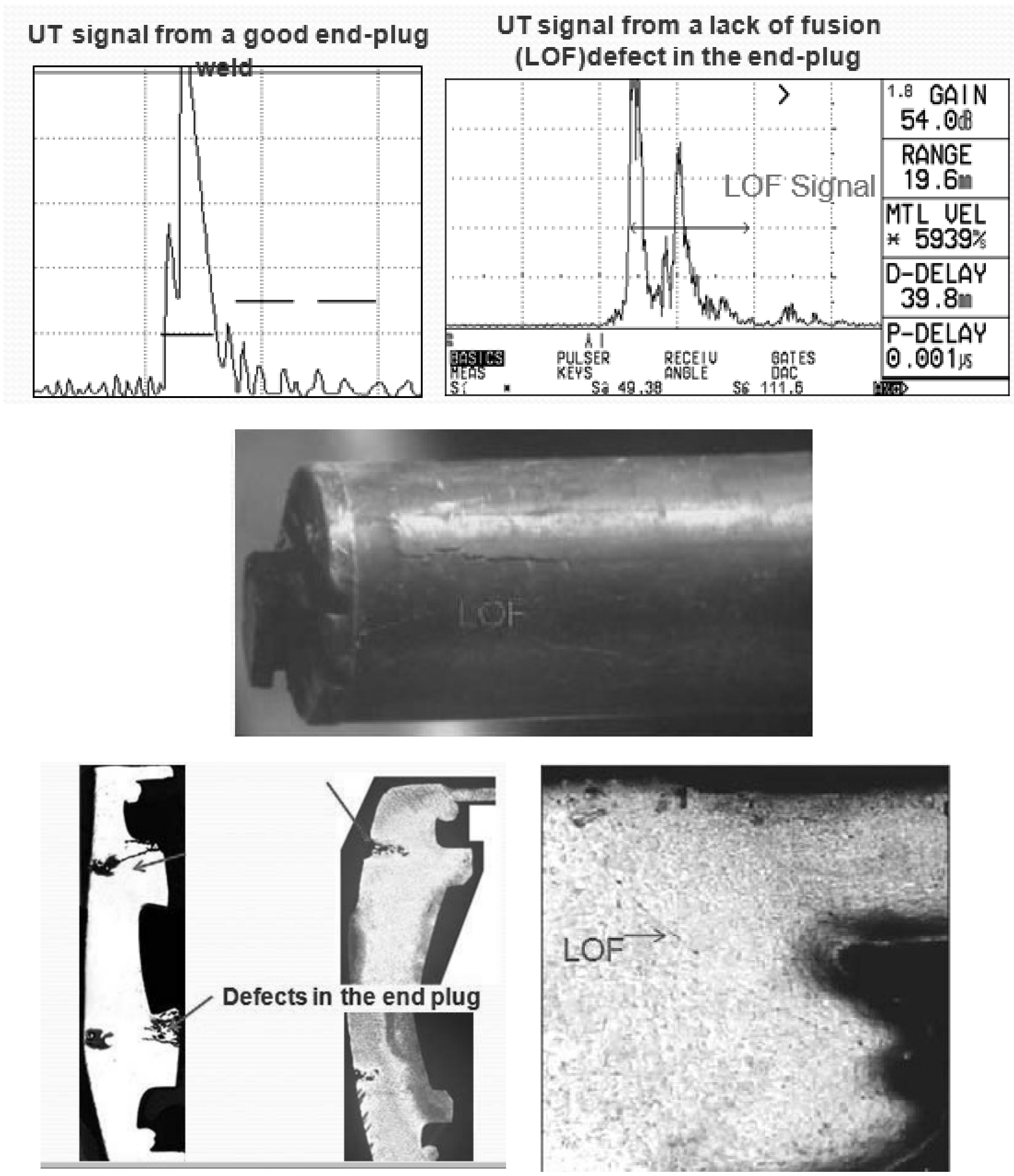


Figure 7. Lack of fusion (LOF) defect and ultrasonic (UT) signal for the same in by brazing resistance - welded end plug welds of irradiated PHWR fuel element .

cladding tubes before they are sent for graphite coating on the inner surface, followed by UO_2 fuel column loading and encapsulation by resistance welding of end caps.

For each piece of bearing and spacer pad appendages, resistance welding is done only at two spots on both ends of the appendages. In most other countries manufacturing PHWR fuel elements, this operation is carried out by brazing the entire mating surface of the bearing and spacer pads with the cladding tube surface with Zr- Be alloy. This ensures adequate bond strength of the appendages. In India, resistance welding is being used to avoid the toxicity associated with beryllium during brazing. During PIE of PHWR fuel bundles, several fuel bundles were found where bearing and spacer pads were found to have pin holes or dislodged [16, 18]. Some of the dislodged pads or other debris in the primary coolant often gets lodged between adjacent fuel element and cause debris induced fretting failure as shown in Figure 8. To mitigate these defects, innovative resistance welding machines are being developed at NFC to ensure that instead of spot welding at two ends

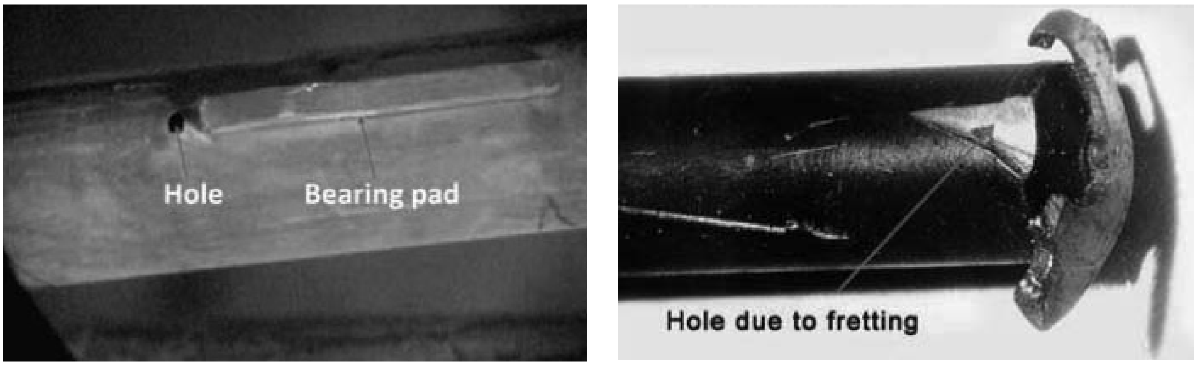


Figure 8. Defects due to improper resistance welding of spacer and bearing pads.

of the spacer and bearing pads, the entire mating surface of the fuel cladding and the spacer and bearing pad appendages are resistance welded.

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