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Performance Assessment of Fuel and Core Structural Materials Irradiated in FBTR

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Abstract

Post-irradiation examination (PIE) is a vital link in the nuclear fuel cycle for providing valuable feedback on the performance and residual life of the fuel and structural materials to designers, fabricators, and reactor operating personnel. The challenging task of setting up of α,β,γ inert atmosphere hot cell facility for PIE of Fast Breeder Test Reactor (FBTR) was accomplished successfully and irradiation performance of the FBTR mixed carbide fuel was assessed stage wise at various burnups starting from 25 GWd/t upto 155 GWd/t. With FBTR being used as a test bed for irradiation experiments on various FBR fuels and structural materials, PIE of various materials subjected to experimental irradiation like the PFBR MOX fuel, FBTR grid plate material have also been carried out to provide valuable feedback to the designers. This paper highlights the (i) results of comprehensive PIE carried on mixed carbide fuel & structural material (ii) control rod performance and (iii) outcome of the examinations on the experimental irradiated subassemblies. .

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Keywords: Post-irradiation examination; carbide fuel; burn-up; performance assessment

1. Introduction

The Fast Breeder Test Reactor (FBTR) at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, uses indigenously developed mixed carbide (U,Pu)C as the driver fuel and 20 % cold worked SS316 as the core structural material. The initial conservative design burn-up and linear heat rating was pegged at 50 GWd/t and 320 W/cm respectively for the FBTR fuel based on out-of-pile experiments, design calculations and limited experience gained by the operation of FBTR [1]. Since the fuel used in FBTR was unique and the fuel behaviour was not established, the continued operation of the reactor and attainment of optimum linear heat rating and high burn-up depended on the feed back regarding the actual performance of the fuel.

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A series of α-tight hot cells [2] with inert nitrogen atmosphere ventilation was established in the Radio Metallurgy Laboratory (RML) to carry out the Post-Irradiation Examination (PIE) of pyrophoric mixed carbide fuel of FBTR. The concrete shielded hot cells designed to handle irradiated material with radioactivity up to 3.7×10^7 GBq have walls of 1200 mm thick made of high density concrete. The hot cells have state-of-art techniques [3] covering a wide spectrum of non-destructive as well as destructive metallurgical examinations of irradiated materials. The non-destructive techniques include visual examination, dimensional measurements, X-ray and neutron radiographic examinations, eddy current testing, leak testing, gamma scanning etc. The destructive techniques include metallography, fission gas extraction and analysis, high temperature tensile tests, micro-hardness measurements, small specimen mechanical testing and electron microscopic studies. The irradiation performance of the FBTR mixed carbide fuel was assessed stage wise at various burnups starting from 25 GWd/t upto 155 GWd/t.

Besides the carbide fuel assembly, the irradiation behaviour of various other core materials like the control rod assembly (B_4C material with stainless steel clad) have also been investigated in the hot cells. With FBTR being used as a test bed for irradiation experiments of candidate fuel and structural materials of fast reactors, PIE has an increasing role to play in characterizing their irradiation behaviour and provide feedback to the designers.

This paper will discuss the performance assessment of the FBTR driver fuel and structural material at different burn-up, and the role of PIE in progressive enhancement of the burn-up to a record high of 165 GWd/t. The role of PIE in performance assessment of materials like B_4C control rod, MOX fuel of PFBR composition and FBTR grid plate material will also be presented.

2. Beginning of life performance assessment of carbide fuel

PIE was initially carried out on seven experimental fuel pins of compositions $(U_{0.3}Pu_{0.7})C$ and $(U_{0.45}Pu_{0.55})C$ irradiated to low burn-ups of 1 to 10 GWd/t (16-100 Effective Full Power Days- EFPD) at a Linear Heat Rating (LHR) of 160 W/cm. The aim of the experimental irradiation was to study the fuel swelling and cracking behaviour during irradiation in the beginning of life. X-radiography of experimental fuel pins with two different fuel compositions indicated clearly the presence of inter-pellet gaps and pellet-clad gap. Progressive increase in fuel stack length was observed with increasing burn-up. The stack length increase in Mark-II fuel pins was found to be lower than that of Mark-I fuel indicating that the lower plutonium content fuel swells at a lower rate. The ceramographic examination of low burn-up fuel (Fig.1) revealed that the swelling and cracking of the carbide fuel within few days of irradiation reduces the fuel-clad gap leading to enhanced gap conductance. This feedback from PIE gave enough confidence to optimize the conditioning period at lower linear power and to increase the linear heat rating of the fuel progressively to 320 W/cm.

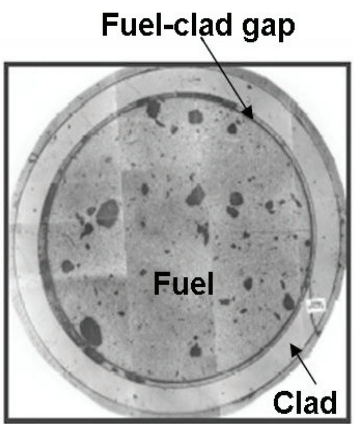


Fig.1. Photomosaic of Carbide fuel pin cross Section after a burnup of 1600 MWd/t

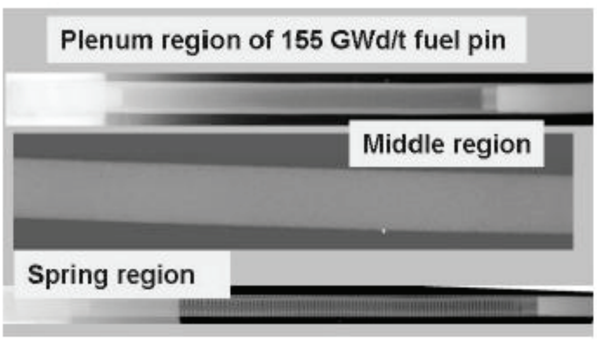


Fig.2. X-Radiographs of fuel pin after 155 GWd/t burn-up

3. PIE of FBTR driver fuel subassemblies

Systematic performance evaluation of fuel and structural material was carried out at various stages of burn-ups of 25, 50, 100 & 155 GWd/t to understand the irradiation behaviour and the FBTR fuel burn-up was progressively increased after considering the implications of extending the burn-up at each stage. The salient results of the PIE are presented in this section.

3.1. Evaluation of fuel performance

3.1.1. Radiographic examination

X-Radiography of fuel pins provides valuable information like increase in pin diameter, increase in fuel stack length, reduction in pellet-to-pellet gap and pellet-to-clad gap and abnormalities like chipping and cracking of fuel pellets. Evaluation of the radiographs of fuel pins after 25 GWd/t and 50 GWd/t in general revealed presence of pellet- to-pellet gaps and pellet-to-clad gaps at the ends of the fuel column. In the case of fuel pins of 100 GWd/t burn-up, the pellet-to-pellet gap and pellet-to-clad gaps appear to have closed at the centre of the fuel column, while pellet-to-pellet gap was still observed in the end of the fuel column. The radiography of 155 GWd/t fuel pins indicated the closure of pellet-to-pellet gap and pellet-to-clad gap almost throughout the length of the fuel column. Fig. 2 shows typical X-radiographs of irradiated fuel pin of 155 GWd/t burn-up.

Neutron radiography of fuel pins [4] at different burn-ups did not indicate any evidence of actinide redistribution. The fuel stack lengths measured from neutron radiographs compared well with that of the measurements from X- radiographs.

3.1.2. Ceramography of fuel-clad cross section

Metallographic examinations of the fuel pin cross sections at various stages of burn-ups gave valuable information on the fuel-clad gap and microstructural evolution of the carbide fuel. Fig. 3 shows the photomosaics of fuel pin cross-sections at the centre of the fuel column after 25, 50 100 & 155 GWd/t burn-ups. Progressive reduction in the fuel-clad gap and radial cracks were observed in 25 & 50 GWd/t burn-up fuel pins. The fuel-clad cross section of 100 GWd/t burn-up fuel pin revealed absence of the gap at the centre of the fuel column with circumferential cracks whereas the cross section at the end of the fuel column indicated presence of minimum fuel-clad gap with radial cracks.

In 155 GWd/t burn-up fuel pins, the fuel-clad gap had closed completely along the entire length of fuel column with circumferential cracks in the centre and end of the fuel region indicating initiation of Fuel Clad Mechanical Interaction (FCMI). The steep increase in the axial stack length of the fuel column beyond 100 GWd/t burn-up measured from X-radiographs is also indicative of the restrained swelling in the radial direction. Exhaustion of porosities was noticed in the outer rim in 155 GWd/t burn-up fuel pin at the centre of the fuel column indicating that the fuel is undergoing hot pressing/creep deformation due to fuel swelling under clad restraint. Evaluating the pellet diameters by image analysis of photomosaics [5], the volumetric free swelling rate of fuel was estimated to be around 1.2% per atom % burn-up at 25 GWd/t and 1% per atom % burn-up at 50 GWd/t.

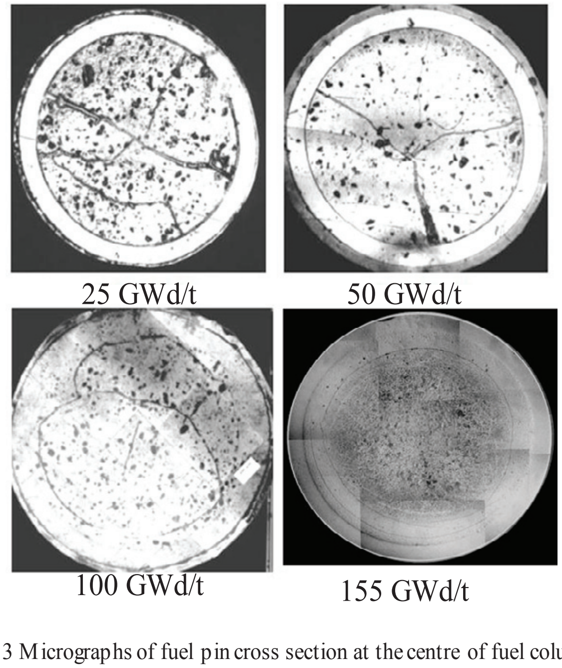


Fig. 3 Micrographs of fuel pin cross section at the centre of fuel column at various burn-up

3.1.3. Fission gas release

Fission gas release on fuel pins after 25 GWd/t burn-up was found to be less than 1 %. In 50 GWd/t burn-up fuel pins, fission gas release was varying from 8 % - 18 %. Fission gas release measurements after 100 GWd/t burn-up indicated that the gas release is in the range of 4 – 14 % [6]. The lower fission gas release in 100 GWd/t burn-up fuel pins is attributed to the reduction in fuel operating temperatures due to closure of fuel-clad gap. Maximum fission gas release estimated on 155 GWd/t burn-up fuel pins was 16 % and the corresponding internal pressure in the fuel pin was measured to be 2.1 MPa.

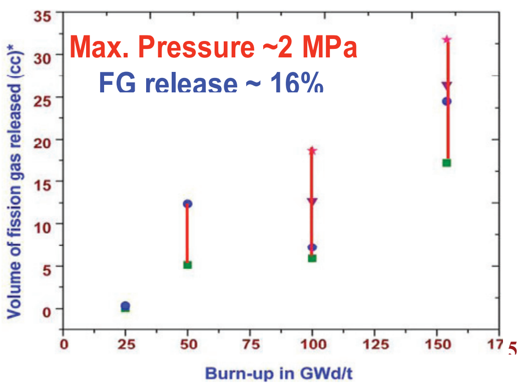


Fig. 4 Volume of fission gas released & plenum pressure as a function of burn-up

The ratio of Xenon to Krypton was estimated to be around 13 indicating predominantly plutonium fission. The helium content in the fuel pins at different burn-ups indicated that most of the helium produced due to alpha decay is released from the fuel matrix to the gas plenum. Fig. 4 shows the volume of fission gases released & maximum plenum pressure as a function of burn-up.

4. Performance assessment of structural materials

Neutron irradiation of SS316 core structural materials induces two types of damaging phenomena (i) changes in dimensions associated with swelling and irradiation creep and (ii) degradation of mechanical properties and embrittlement associated with microstructural evolution [7]. The irradiation induced changes in mechanical properties is a function of neutron fluence (dpa) and the irradiation temperature.

4.1. Metrology of hexagonal wrapper and cladding

The maximum increase in the dimensions of hex wrapper and fuel pin was seen to occur in the region of peak dpa close to the centre of the fuel column. The trends in the changes in corner-to-corner distance (CCD) & flat-flat distance (FFD) of the wrapper and diametral strain of the fuel pin with dpa are shown in Fig. 5. The hex wrapper and the fuel pin did not indicate any increase in its dimensions after lower dpa irradiation corresponding to 25 & 50 GWd/t burn-ups. Beyond about 30 dpa, the dimensions of both the wrapper and fuel pin were seen to progressively increase with increasing dpa. The rate of increase in the dimensions was higher beyond 100 GWd/t burn-up as compared to that at lower burn-ups. The maximum % increase in fuel pin diameter was significantly higher than that of hexagonal wrapper dimensions on account of the higher temperatures of the cladding close to the peak swelling temperature of 20 % cold worked SS316.

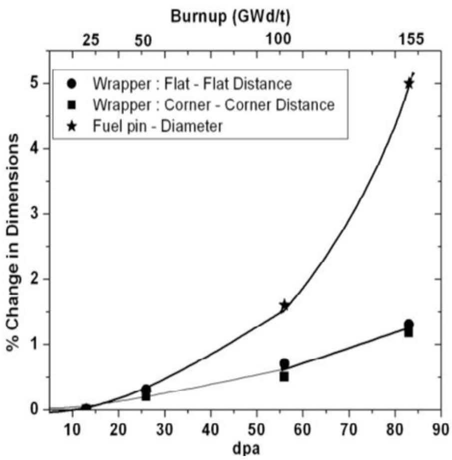


Fig. 5 Variations of hex wrapper and fuel pin dimensions with dpa

Similar trends were also observed in the void swelling estimates of clad and wrapper samples determined from density measurements. Considering a nominal gap of 0.7 mm between fresh fuel subassemblies in the FBTR core, the dilation of 0.65 mm measured in 155GWd/t burn-up wrapper is one of the major limiting factors in increasing the burnup of fuel subassemblies from fuel handling considerations.

4.2. Mechanical properties of cladding and wrapper

The mechanical properties of irradiated SS316 cladding were determined by remote tensile tests carried out on tube specimens sectioned from various locations along the length of the fuel pin corresponding to a combination of dpa (0-83) and irradiation temperature (430-500°C). The tests were conducted at temperatures corresponding to (i) reactor operation conditions (ii) fuel handling operations (180ºC) and (ii) ambient conditions (25ºC). It was seen that the Ultimate Tensile Strength (UTS) of the cladding shows a significant decrease at displacement damages > 60 dpa both in high temperature and room temperature tests (Fig. 6), while the uniform elongation was around 3 - 4.5 % (Fig. 7). Similar trends of decreasing UTS with increasing dpa have been reported in the literature [8].

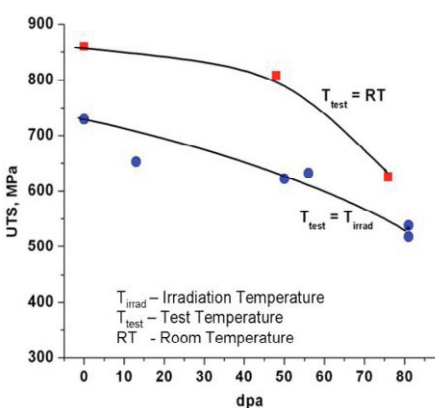


Fig .6. Trends in the UTS of cladding with dpa

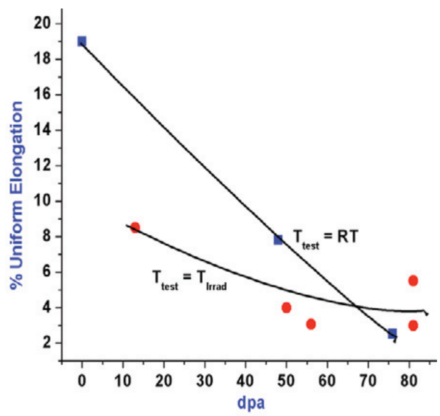


Fig. 7. Trends in Uniform elongation of cladding with dpa

The tensile properties of the hexagonal wrapper were evaluated by shear punch tests involving blanking a 1.0 mm thick and 8.0 mm diameter specimen in a test fixture using a flat cylindrical punch. The load-displacement plot obtained during the punching operation was analyzed and correlated with the conventional tensile test data. Tensile-to-shear punch property correlation was established from standardization experiments on various cold rolled and solution annealed microstructures of SS316.

Shear punch tests carried out on specimens extracted from wrapper indicated that there is an increase in the room temperature yield strength (YS) and UTS with increasing dpa (Fig. 8) and a decrease in the ductility. The tensile properties of wrapper showed a hardening behaviour as its irradiation temperature is around 400-430ºC.

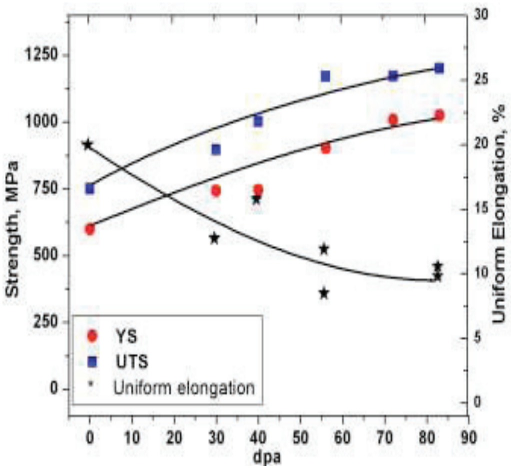
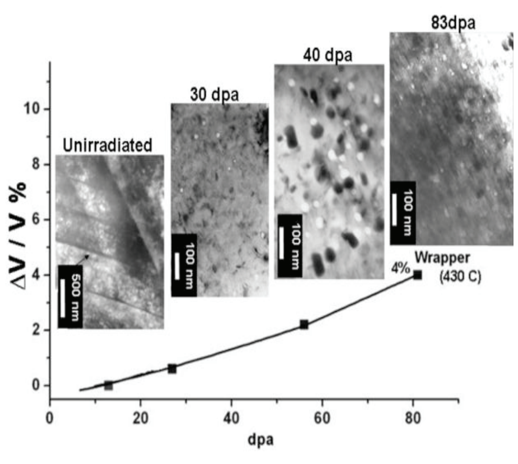


Fig. 8.The trends in the strength and ductility of SS316 wrapper as a function of dpa.

4.3. Microscopic studies of SS316 cladding & wrapper

Optical microstructural examination of the irradiated cladding was carried out to study the microstructure and look for any carburization of the clad inner diameter. The clad microstructure has not revealed any carburization on the inner diameter of the clad tube. Microhardness measurements on the cross-section of the irradiated clad did not indicate any evidence of carburization.

Transmission Electron Microscopy (TEM) studies of hexagonal wrapper showed extensive void formation beyond 40 dpa in addition to precipitation and dislocation loops. Fig. 9 shows the TEM micrographs at different dpa superimposed on the swelling curve. The void density showed a progressive increase with displacement damage. The precipitates were identified to be mainly of nickel and silicon enriched M_6C type of η phase, whereas radiation induced G phase was also observed at 83 dpa. The precipitates were found to be associated with the voids possibly due to the growth of precipitates by diffusion of solutes along with surplus vacancies. The retention of cold worked structure was unambiguously seen after 83 dpa irradiation which suggests that no recrystallisation has taken place and irradiation hardening is more prominent than the softening effects as indicated in tensile properties [9].



In summary, the PIE on 25 GWd/t & 50 GWd/t revealed that the fuel has a lower swelling rate than the expected and swelling was getting accommodated in fuel porosities. Sufficient fuel-clad gap was available to increase the burn-up beyond 50GWd/t without initiation of Fuel Clad Mechanical Interaction (FCMI). The wrapper and cladding strain was also found to be nominal.

Micrographs of fuel at 100 GWd/t indicated the closure of fuel-clad gap and restrained swelling phase at the center of fuel column, while after 155 GWd/t burn-up, the entire fuel column was in the restrained swelling phase. Exhaustion of sinter porosities in the outer rim of the fuel and circumferential cracking throughout the fuel column after 155 GWd/t burn-up indicated increased fuel swelling and FCMI. The fission gas release and the plenum pressure were found to be very nominal in the high burn-up carbide fuel.

There was significant increase in the dimensions of hexagonal wrapper and fuel pins beyond 100GWd/t burnup. The void swelling and the consequent dimensional changes on the structural materials showed a faster rate of increase at 155 GWd/t burn-up as compared to the results obtained at lower burn-ups. The cladding showed a significant loss of strength and ductility beyond 60 dpa. The main limiting factors for further enhancing the burn-up were (i) dilation of wrapper and its impact on the fuel handling operations (ii) FCMI and loss of cladding mechanical properties. The thermo-mechanical analysis of the fuel subassembly based on the PIE results indicated possibility of a marginal increase in the fuel burn-up beyond 155 GWd/t. Presently, the burn-up of a one representative subassembly has reached 165 GWd/t with any fuel failure.

5. PIE of control rod assembly

FBTR has six control rod assemblies for reactor start up, shutdown and control of reactor power. Each control rod consists of nine sintered boron carbide pellets (90% enriched in B^{10} isotope) stacked to a length of 430 mm inside SS316 cladding. The control rod moves axially inside an outer hexagonal sheath made of SS316 during raising and lowering. A few incidents of the dropping of the control rod of FBTR were reported while lifting it to increase the reactor power. This called for a detailed investigation to assess cause of the incident and study the irradiation behaviour of the B_4C absorbing material and SS316 cladding.

To investigate the possibility of interference between the control rod and the outer sheath, precise dimensional measurements were carried out using customized devices designed and fabricated exclusively for this purpose.

Dimensional measurements did not indicate any significant changes in the outside diameters of the control rod or the inner diameters of the satellite tracks of outer sheath. Minor misalignment (Fig.10) of the order of 1.01mm and 0.83mm was observed in the axes of the control rod and the outer sheath which could have led to the interference between them in unfavorable orientations especially during raising of power.

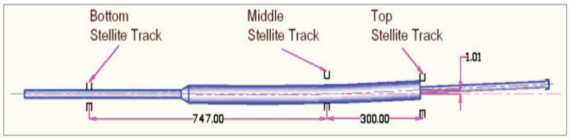


Fig. 10. Axis Misalignment of Control rod

Neutron radiography was carried out using KAMINI reactor to visualize the control rod internals. The radiographs were obtained (Fig. 11) by indirect technique using dysprosium as converter screen at different power levels in two orientations. The neutron radiographs revealed that internals are intact without any blockages that could restrict coolant flow. No gross depletion of B^{10} was observed in the neutron radiographs.

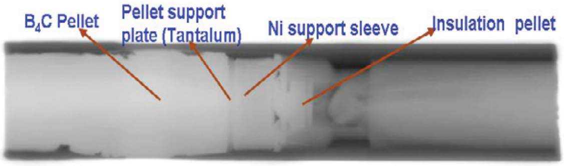


Fig. 11. Neutron Radiographs of the bottom portion of control rod showing support sleeve, insulation pellet and boron carbide pellet

Cracking behavior of boron carbide pellets was assessed by X-ray radiography. Since the existing X-ray radiography port can handle only fuel pins, a separate facility was erected by modifying one of the radiation survey ports for radiography of large diameter control rod. X-ray radiography clearly indicated extensive cracking and fragmentation of the pellets exposed to higher neutron fluence (Fig. 12).

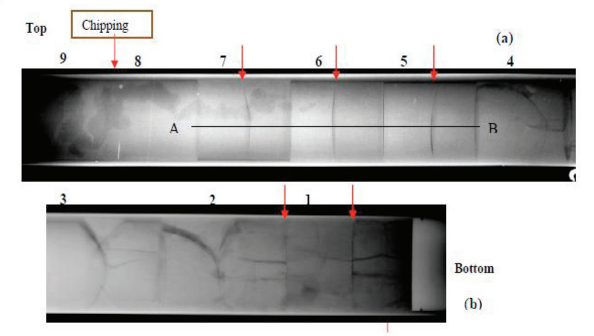


Fig.12. Radiography image indicate the pellet regions of the control rod (a) top portion. (b) bottom portion - The radial cracks at the centre of the pellets are marked by arrows

After cutting the control rod cladding using Nd-YAG laser, considerable difficulty was experienced in the retrieval of pellets due to the presence of sodium and sodium oxide in the pellet-clad gap. The pellets were retrieved using a specially made fixture to push the pellets out of the control rod clad.

Volumetric swelling of the boron carbide pellets measured by density measurements indicated marginal increase in volume of around 2 % in the bottom pellets which has been exposed to higher neutron dose (Fig. 13). Ceramography of B_4C revealed extensive cracking of the bottom pellets (Fig.14) and clad microstructure did not indicate any evidence of chemical interaction between the B_4C pellet and the clad.

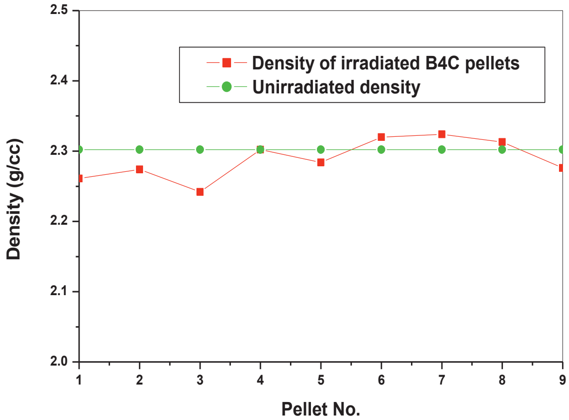
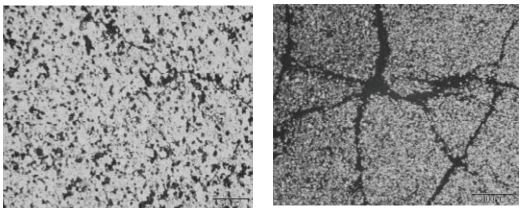


Fig.13: Density variations in the irradiated B_4C pellets



Pellet No. 2 Pellet No. 4

Fig. 14: Photomicrographs of B4C pellet cross-section

Metallographic examination of control rod cladding was carried out to study the interaction of boron carbide pellet and the cladding. The microstructure of the clad cross-section did not indicate any significant interaction between the boron carbide pellet and stainless steel cladding revealing absence of absorber clad chemical interaction as observed by optical microscopy.

Using Laser ablation mass spectrometer, the isotopic ratio of $^{10}B/^{11}B$ was determined. The mass spectrum clearly revealed the presence of ^{7}Li in the irradiated pellets confirming the consumption of ^{10}B. However, ^{10}B depletion in all the pellets was found to be less than 1%.

In summary, PIE of control rod assembly has provided valuable information regarding the dimensional changes, pellet integrity, swelling behavior etc. It has been concluded that the dropping incident encountered could be due to the marginal interference between control rod and outer sheath. Pellet integrity assessment indicated that due to extensive cracking and fragmentation it may not be possible to directly reuse most of the pellets. PIE has indicated that the boron carbide pellets and the structural materials have not reached life limiting conditions.

6. PIE of experimental subassemblies

FBTR is also being used as a test bed for irradiation experiments on various fuels and structural materials of importance to our FBR programme. In such irradiation experiments, hot cell facility is a very crucial link for post-irradiation handling and examination of the irradiated material and to provide valuable outcome and feedback to the designers.

6.1. Residual life assessment of FBTR grid plate

With FBTR in operation for more than 24 years, it was necessary to assess the residual life of the grid plate. Irradiation damage to grid plate is one of the main life limiting factor. Irradiation experiments were carried out to calculate the dpa rate on the grid plate of FBTR and to evaluate the changes in the mechanical properties of the grid plate material.

An experimental subassembly consisting of pre-fabricated miniature tensile and disc specimens of FBTR grid plate material (SS316) was irradiated in the 4^{th} ring of FBTR for duration of 58.18 effective full power days (EPFD). The displacement damage of the irradiated specimens ranged from 1.08 - 2.57 dpa. The irradiation capsule was dismantled in the hot cell facility and remote tensile tests at the irradiation temperature of 350°C, 400°C and 25°C were performed using the miniaturized tensile specimens. Special wedge type gripping systems were designed and fabricated for holding thin specimens during the high temperature test.

The stress strain curves obtained by analyzing the load-displacement plot recorded during the test are shown in Fig 15. It is seen that annealed SS316 material undergoes an increase in YS and UTS (radiation hardening) with respect to the unirradiated values for all the irradiated conditions. The increase in YS from virgin condition is considerably higher than that of UTS increase. The hardening is accompanied by a decrease in uniform elongation to about 20-30% for the various irradiated samples. The 2.57 dpa specimen exhibiting a uniform elongation of above 20% at test temperatures of 28 C, 350°C & 400°C indicates retention of adequate residual ductility in SS 316 irradiated to this displacement damage.

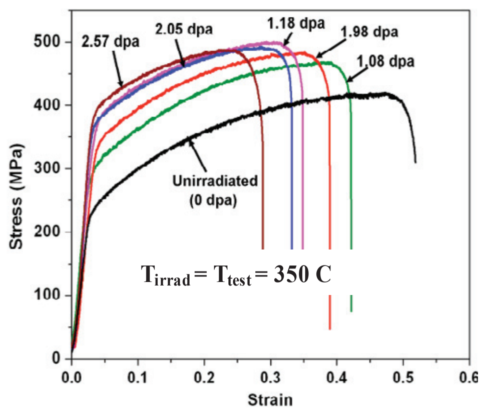


Fig.15. Stress-strain plots of the irradiated grid plate specimens of various dpa tested at 350C superimposed on that of virgin material.

7. PFBR MOX fuel test irradiation

One experimental fuel pin with MOX fuel pellets of PFBR composition was irradiated in FBTR for 13 days at a linear power of 400 W/cm for understanding the beginning of life gap closure behaviour. This experiment was aimed to optimize the duration of pre-conditioning of the MOX fuel in PFBR at lower linear power (400 W/cm) before raising the linear power to the design value of 450 W/cm. X-ray radiography and metallography of fuel pin cross-sections were the main examinations carried out.

The examination of experimental MOX fuel pin of 6.6 mm diameter necessitated either modifications in the existing setup or design and fabrication of new equipment. For example, the X-radiography pin holder was modified for holding a larger pin diameter, while the vacuum impregnation system (for immobilizing the fuel column during slicing of metallographic sample) was redesigned and fabricated for handling larger diameter pins. The fuel-clad surfaces prepared by metallographic procedures were replicated to measure the fuel-clad gap. Fig. 16 shows the typical photomosaic of the fuel pin cross-section at the centre of the fuel column. The measurements at different locations indicated that the apparent gap (without taking into account crack area) had reduced from average pre-irradiation value of 75-110 microns to uniformly around 12 - 13 microns in all the locations. The gap reduction during beginning of life indicates the feasibility of increasing the linear power of PFBR fuel to the design value after the initial pre-conditioning of approximately 20 EFPD.

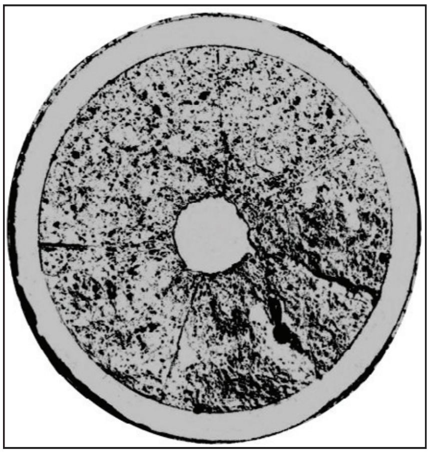


Fig.16: Typical photomosaic of fuel pin cross-section at the centre of the fuel column (120 mm from the top of the fuel column

8. Conclusions

The stage wise performance evaluation of the plutonium rich mixed carbide fuel of the Fast Breeder Test Reactor (FBTR), starting from beginning-of-life performance studies and subsequently at various stages of burn-up has led to a comprehensive understanding of the fuel and structural material behaviour. Compared to the initial design limit of 50 GWd/t, the burn-up has now reached 165 GWd/t without any fuel pin failure. PIE has thus played a crucial role in validating the design and choice of the unique fuel material of FBTR, thereby increasing its in-reactor life.

Besides the driver fuel assembly, the irradiation behaviour of various other core material like the control rod (B_4C material with stainless steel clad) has provided valuable information regarding the dimensional changes in the cladding and outer sheath, pellet integrity, B_4C swelling and depletion behavior.

With FBTR being used as a test bed for irradiation experiments, PIE serves as a vital link between connecting the designers and reactor operating personnel. PIE of MOX fuel of PFBR composition irradiated in FBTR for 13 days was useful in studying the evolution of the fuel-clad gap at the beginning of life. This information was very crucial for increasing the linear power of PFBR fuel to the design value after the initial pre-conditioning. As a part of life extension of FBTR, the irradiation experiment carried out to study the changes in mechanical properties of grid plate material after low dose irradiation indicated sufficient retention of residual ductility of the SS316 grid plate.

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