Journal of Nuclear Materials 385 (2009) 387–391

Contents lists available at ScienceDirect

Journal of Nuclear Materials

journal homepage: www.elsevier.com/locate/jnucmat

Pressurized heavy water reactor fuel behaviour in power ramp conditions

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ARTICLE INFO

PACS:
06.30.Bp
06.60.Ei
28.41.Bm
28.41.Qb
46.50._a
61.82.Bg
62.20.mm
61.66._f
68.47.De
81.70.Tx

ABSTRACT

In order to check and improve the quality of the Romanian CANDU fuel, an assembly of six CANDU fuel rods has been subjected to a power ramping test in the 14 MW TRIGA reactor at INR. After testing, the fuel rods have been examined in the hot cells using post-irradiation examination (PIE) techniques such as: visual inspection and photography, eddy current testing, profilometry, gamma scanning, fission gas release and analysis, metallography, ceramography, burn-up determination by mass spectrometry, mechanical testing. This paper describes the PIE results from one out of the six fuel rods. The PIE results concerning the integrity, dimensional changes, oxidation, hydriding and mechanical properties of the sheath, the fission-products activity distribution in the fuel column, the pressure, volume and composition of the fission gas, the burn-up, the isotopic composition and structural changes of the fuel enabled the characterization of the behaviour of the Romanian CANDU fuel in power ramping conditions performed in the TRIGA materials testing reactor.

1. Introduction

The Institute for Nuclear Research (INR) of Piteşti has a set of nuclear facilities consisting of a 14 MW TRIGA reactor and LEPI (Romanian acronym for Post-Irradiation Examination Laboratory) which allow the investigation of the behaviour of the nuclear fuel and materials under various irradiation conditions. LEPI is an alpha–gamma hot-cell facility that became operational operation in December, 1983. It includes two heavy concrete hot-cells, three steel hot-cells and one lead hot-cell.

The nuclear fuel used in the CANDU-6 reactor at Cernavoda NPP consists of 37 rods assembled in the shape of bundle by means of end plates. This fuel bundle has 495 mm length, 103 mm in diameter and 24 kg weight. A CANDU fuel rod contains sintered cylindrical pellets of natural uranium dioxide (UO_2) stacked within a Zircaloy-4 cladding tube sealed at both ends by end plugs. A pallet is 492 mm long and 13.08 mm in diameter.

In order to check and improve the quality of the Romanian CANDU fuel, a significant number of experimental fuel rods have been tested to different power histories in the TRIGA reactor. Some of the most important tests have been performed in power ramping conditions.

In CANDU the "ramp" event is particularly important because this type of reactor uses the "on-power" re-loading during which the CANDU fuel experiences relatively severe ramps all the time.

This paper describes the post-irradiation examination (PIE) tests performed to assess the quality of the fuel rods.

2. Description of fuel

The fuel rods used for power ramping tests have been manufactured by the Institute for Nuclear Research. The fuel rods have dimensions that are representative of a typical commercial CANDU-6 fuel rod. An exception to this is the rod length, which is limited to about 300 mm in fuel stack length in order to keep the fuel within a reasonably flat flux in the test site. The fuel rod enrichment has been set at 5.75 wt% ^{235}U in order to achieve the desired powers for the power ramping test.

The power ramping was performed on the assembly of six CANDU fuel rods under the following conditions:

- Rod linear power in pre-ramp: 43 kW m^{-1};
- Rod linear power in ramp: 63 kW m^{-1};
- Power increase rate in ramp: 0.025 kW m^{-1} s^{-1};
- Maximum temperature on the fuel rod sheath in pre-ramp: 553 K;
- Maximum temperature on the fuel rod sheath in post-ramp: 603 K;
- Cooling water composition: pH 9.5–10.5 and content of oxygen = 10–50 ppb;
- Total time of irradiation: 284 days.

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doi:10.1016/j.jnucmat.2008.12.026

3. Post-irradiation examination

After irradiation, the fuel rod was kept in the reactor pool for 3 months to allow for radioactive decay. Afterwards, the fuel rod was transferred to the INR hot-cells where it was subjected to detailed examinations.

3.1. Profilometry

Profilometry is a step-by-step measurement of diameter along the fuel rod at regular intervals using a remotely operated profilometer controlled by computer. The profilometer consists of a vertical fuel rod-positioning machine equipped with SLO-SYN step-motors and diameter-measuring equipment using two opposed SCHLUMBERGER inductive transducers also controlled by computer. The measurement accuracy is ±5 μm and has been obtained by linearization of transducers.

The fuel rod was characterized dimensionally both pre-irradiation and post-irradiation in the following manner:

- Measurement of outer diameter axial distribution;
- Measurement of length between end-caps;
- Measurement of bow.

The diameter measurement was performed by a step-by-step measurement along the fuel rod at 1 mm intervals in three planes (0° plane includes the bearing pad, 120°, 240°). The diametrical profile as the average of the three scans as well as the bending profile have been plotted and are shown in Fig. 1(a) and (b). For the irradiated fuel rod, this figure shows the ridging of the sheath at pellet interfaces due to pellet end swelling, which is common for CANDU power reactor fuel.

The parameters which highlight the dimensional changes of this fuel rod are the following:

- Average diameter before irradiation: 13.068 mm;
- Average diameter after irradiation: 13.117 mm;
- Average diameter at top of ridges: 13.162 mm;
- Average diameter at bottom of ridges: 13.086 mm;
- Average circumferential ridge height: 38 μm;
- Maximum diametrical expansion: 145 μm (1.1%);
- Length before irradiation: 307.04 mm;
- Length after irradiation: 307.42 mm;
- Axial elongation: 380 μm (0.1%);
- Bow: 208 μm.

The diameter expansion of 1.1% remains acceptable for such a fuel rod.

3.2. Gamma scanning and tomography

The gamma scanning equipment consists of a vertical fuel rod-positioning machine equipped with SLO-SYN step-by-step-motors, a collimator set in the hot-cell shielding wall, a PGT intrinsic Ge detector and a multi channel analyzer. The equipment is controlled by computer.

The axial gamma scanning was performed using a horizontal collimator slit having an aperture size of 0.5 mm. The gamma acquisition along the fuel rod was performed at regular intervals of 0.5 mm; the acquisition time per step was 200 s. Fig. 2(a) shows the fuel rod axial gross gamma activity profile. A prominent depression of count rate at fuel pellet interfaces is observed, which means that there is no pellet interaction. This gamma activity profile highlights a practically symmetric loading of the fuel rod.

A method of tomographic reconstruction based on a maximum entropy algorithm has been developed according with Refs. [1,2]. The data acquisition was done while the fuel rod was moved transversally step-by-step at regular interval of 0.25 mm after every rotation of 72° in front of a vertical collimator slit of 50 mm in height and 0.25 mm in aperture. Fig. 2(b) shows, qualitatively, the tomographic image of the radial ^{137}Cs gamma activity distribution in the flux peaking area cross section of the fuel rod. This tomography indicates that the ^{137}Cs isotope migrated from middle to periphery of the fuel rod and redistributed according to the temperature profile. In Fig. 2(b), it is observed that the peak of Cs activity is pellet periphery oriented.

The ^{137}Cs isotope was used as a burn-up monitor. For an accurate determination of the burn-up, the gamma self-absorption coefficient was calculated using the distribution of ^{137}Cs activity in the cross section of the fuel rod. The burn-up of the fuel rod is 8.77 MW d kg^{-1} (for 192 MeV fission of U). The fuel rod burn-up determined by mass spectrometry is 9.0 MW d kg^{-1}. These results are in agreement.

3.3. Fission gas measurement

The pressure and volume of fission gas inside the fuel rod and the fuel rod internal void volume have been measured. The used technique is to pressurize a standard volume with helium at a measured pressure and to expand it into the unknown volume (V_u) of the various components in the system. So the V_u volume is calculated using the isothermal gas transformation law. The fuel

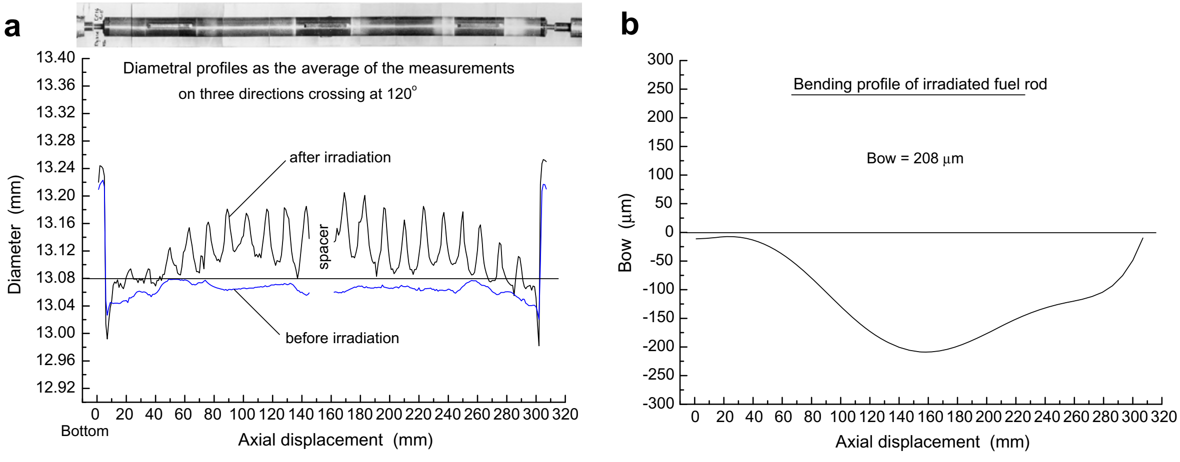

Fig. 1. Profilometry on the CANDU fuel rod irradiated in the INR–TRIGA reactor in a power ramping test.

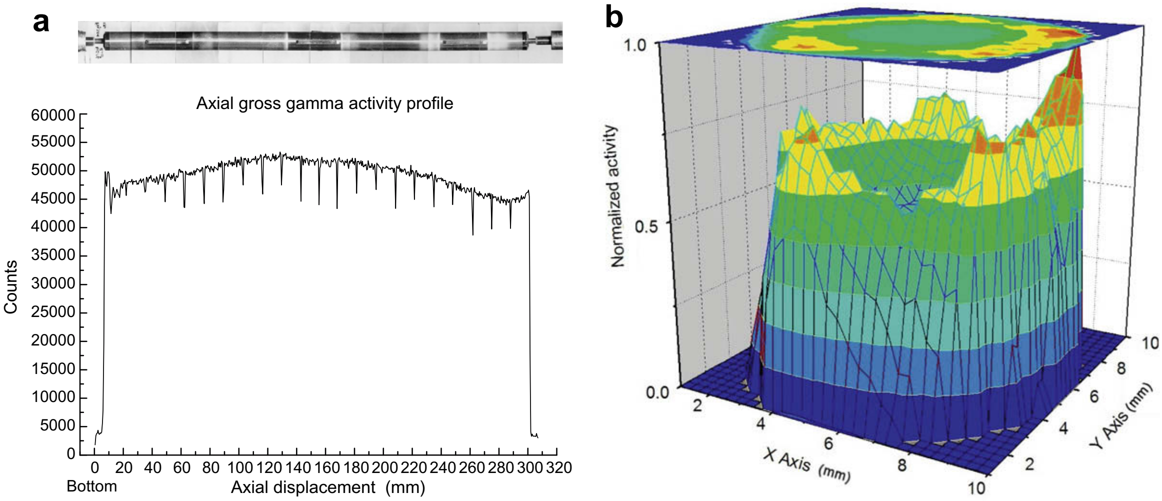

Fig. 2. Axial gamma scanning (a) and tomography (b) on a CANDU fuel rod irradiated in the INR–TRIGA reactor in a power ramping test.

rod was punched mechanically using a hardened steel tip. The fission gas was released into the combined internal fuel rod and V_u volume. The combined volume was determined in the same manner as V_u volume. The internal void volume of the fuel rod was determined by subtracting the V_u volume from the combined internal fuel rod and V_u volume and it was 1.73 cm^3. The total released fission gas volume at STP conditions was 7.29 cm^3 and the fission gas pressure 482 kPa. The fission gas composition was analyzed using a QMS Gas Analyzer installed at the outside of the hot cell. The xenon, krypton and helium concentrations were 59%, 14% and 27%, respectively.

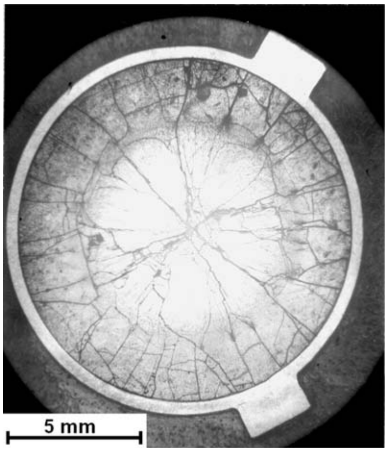

Fig. 3. Cross section macrography.

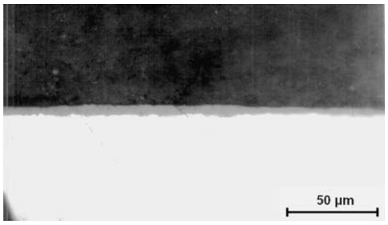

Fig. 4. Sheath oxidation layer.

3.4. Metallographic/ceramographic examination

A LEITZ MM5RT optical microscope having a magnification up to x500 was used for macro graphic and micro-structural analysis of irradiated fuel rod samples and for micro hardness measurements. It is equipped with an image analyzer and a Vickers micro

Fig. 5. Sheath hydriding features.

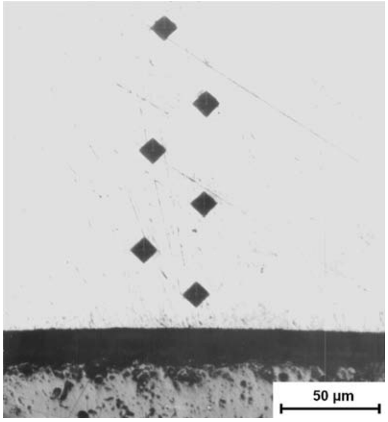

Fig. 6. Vickers micro hardness.

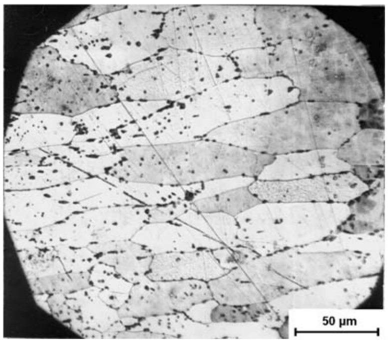

Fig. 7. Columnar grains at pellet center.

hardness tester (maximum load 2 N). A computerized analysis system is used for the quantitative determination of structural features, such as grain and pore size distribution.

The preparation of the samples included precise cutting, vacuum resin impregnation, sample mounting with epoxy resin in an acrylic resin cup, mechanical grinding and polishing, chemical etching according to Ref. [3].

Fig. 3 illustrates the macrography in the cross section of the irradiated fuel rod. Characterization of sheath oxidation is shown in Fig. 4. Oxide layer thickness is 5 μm. Fig. 5 shows the hydride precipitations orientated parallel to the sheath surfaces. A content of hydrogen of about 120 ppm was estimated by means of hydruration charts, see Ref. [4]. The Vickers hardness measured on the surface of the sheath cross section is 2589 MPa (see Fig. 6).

Micrographies have been taken from the center, mid-radius and periphery of the sample and they show the fuel restructuring (columnar and equiaxial grains) due to power ramp conditions. Fig. 7 shows columnar grains at center of the fuel, Fig. 8 shows grown equiaxial grains at mid-radius and Fig. 9 shows as sintered grains at periphery. The size of the columnar grains is 57 μm and the size of the equiaxial grains is 24 μm being measured by the linear interception method.

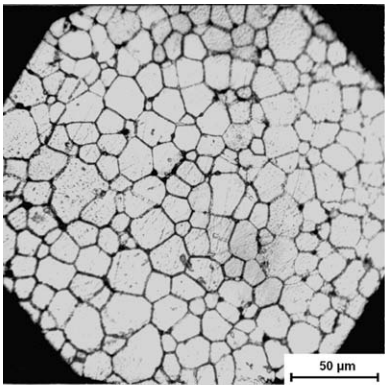

Fig. 8. Grown equiaxial grains at pellet mid-radius periphery.

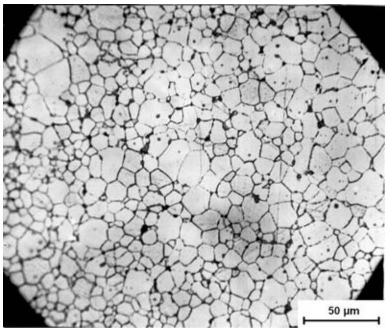

Fig. 9. As sintered grains at pellet periphery.

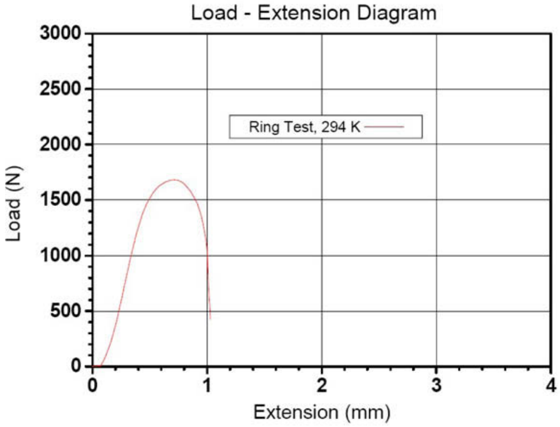

Fig. 10. Load–extension diagram.

3.5. Mechanical properties determination

After the preliminary tests three 5 mm long pieces were cut from fuel rod to obtain ring samples for tensile testing. The samples were prepared according to the shapes and dimensions given in Refs. [5,6]. The samples have to be tested to evaluate the changes in the mechanical properties. In the Post Irradiation Examination Laboratory (PIEL) the tensile testing machine is an INSTRON 5569 model. The machine uses the Merlin software to process the data obtained from the mechanical tensile tests.

The tests have been performed to record or evaluate the following mechanical characteristics:

- the strain–stress diagrams and load–extension (Figs. 10 and 11);
- the yield strengths (offset method at 0.2%);
- the elastic limit;
- the ultimate tensile strength of the samples.

Strain–stress and load–extension diagrams were obtained by means of Merlin Software, using a preload of 25 N.

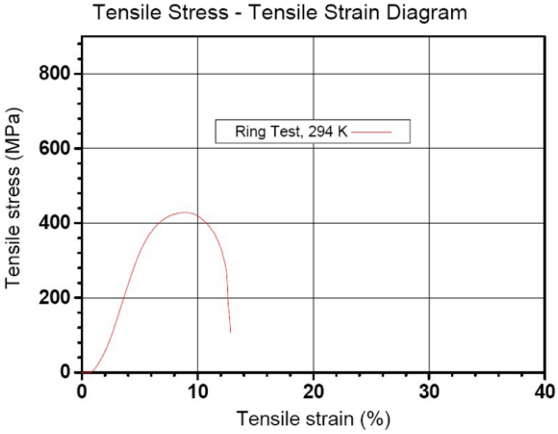

Fig. 11. Strain–stress diagram.

The tests were performed on fuel element sheath samples at 294 K, in a controlled tensile strain mode, with a rate of 0.05 min^{-1}.

The tests were performed according to the procedures and standards given in Ref. [7].

The primary data of the tensile tests were used to establish the relationship between the parameters of irradiated fuel sheath material. The Merlin software was used to process the tensile experimental diagrams.

4. Conclusion

The power ramping experiment performed on CANDU-6 fuel rods was successful, as they did not fail during the severe ramps performed in the TRIGA reactor.

The tomography indicates that the ^{137}Cs isotope migrated from middle to periphery of the fuel rod and redistributed according to the temperature profile. The peak of ^{137}Cs activity is located near the pellet periphery.

The pressure of fission gas inside the fuel rod was 482 kPa which is normal for CANDU fuel.

The results obtained by non-destructive and destructive examinations concerning the integrity, dimensional changes, oxidation, hydriding and mechanical properties of the sheath, the fission-products activity distribution in the fuel column, the pressure, volume and composition of the fission gas, the burn-up and structural changes of the fuel are typical of CANDU-6 fuel.

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