RING TESTS ON CANDU FUEL ELEMENTS SHEATH SAMPLES

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This work is a study of the behaviour of CANDU fuel elements out of service from power nuclear plant. The tests are made on ring samples took from fuel cladding in Institute for Nuclear Research (INR) Pitesti. Zircaloy-4 is the material used for CANDU fuel sheath. The importance of studying its behaviour results from the fact that the mechanical properties of the CANDU fuel sheath suffer modifications during normal and abnormal operation. In the nuclear reactor, the fuel elements endure dimensional and structural changes as well as cladding oxidation, hydriding and corrosion. These changes can lead to defects and even to the loss of integrity of the cladding. This paper presents the results of examinations performed in the Post Irradiation Examination Laboratory (PIEL) from INR Pitesti:

- Dimensional and macrostructural characterization;
- Microstructural characterization by metallographic analyses;
- Determination of mechanical properties;
- Fracture surface analysis by scanning electron microscopy (SEM).

During 20 years of post-irradiation examination, the Post-Irradiation Examination Laboratory from INR Pitesti has obtained the operational and professional ability to evaluate the performance of the CANDU nuclear fuel from Cernavoda NPP. The obtained data could be used to evaluate the security, reliability and nuclear fuel performance, and for CANDU fuel improvement.

Keywords: CANDU, fuel, cladding, examination, post-irradiation

1. Introduction

The development of the nuclear energy in our country by commissioning of new units at Cernavoda Nuclear Power Plant (NPP) requires the nuclear fuel safety strengthening and implicitly the development and improvement of the techniques to investigate the processes which take place in the fuel, in order to obtain relevant information concerning:

- Maneuver regimes or power ramping;

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- Nuclear fuel behaviour in accident conditions generated by reactivity and loss of coolant;
- Fuel element failure kinetics and fission product release.

These items of information are needed to evaluate the performance of nuclear fuel and materials in NPP in order to check the concordance with safety criteria.

Safety criterion used in power ramping conditions is the “Pellet-Clad Mechanical Interaction” (PCMI) which is related to the stress on the cladding produced by the UO2 pellet expansion in a short period of time. This situation can result in an interaction and if the stress is high enough and the cladding ductility low enough the „ramp” situation can lead to a clad failure. In the CANDU reactor this type of event is of particular importance because this type of reactor uses the „on-power” re-loading and therefore the CANDU fuel endures fairly severe ramps all the time.

In order to check and improve the quality of the Romanian CANDU fuel, the power ramping tests have been performed on experimental fuel elements in the TRIGA SSR (Steady State Reactor) reactor of INR Piteşti and their behaviour has been analysed by post-irradiation examination (PIE) in the hot cells.

The accident conditions which can lead to fuel melting are the loss of coolant or an excessive power in the fuel. If these accidents happen, the cool ability of reactor core and its geometry must be preserved, thus avoiding the fuel dispersion and major consequences on the environment. The accidents studied to prevent these possibilities are:

- Loss of Coolant Accident (LOCA)
- Reactivity Insertion Accident (RIA)

In order to investigate the Romanian CANDU fuel behaviour in the accident conditions, simulated LOCA and RIA tests have been performed on experimental fuel elements in the TRIGA ACPR (Annular Core Pulsed Reactor) reactor of INR Piteşti.

The purpose of this work is to determine by post-irradiation examination, the behaviour of CANDU fuel, irradiated in the 14 MW TRIGA reactor. The results of the post-irradiation examination are:

- Visual inspection of the cladding;
- Profilometry (diameter, bending, ovalization) and length measuring;
- Determination of axial and radial distribution of the fission products activity by gamma scanning and tomography;
- Microstructural characterization by metallographic and ceramographic analyzes;
- Mechanical properties determination;
- Fracture surface analysis by scanning electron microscopy.

A transportation cask and the necessary devices for bundle handling were designed and manufactured at INR Piteşti. The cask will be used for transporting the fuel bundles from Cernavoda NPP to Post-Irradiation Examination Laboratory.

The obtained data could be used to evaluate the security, reliability and nuclear fuel performance, and for CANDU fuel improvement.

The irradiation of a fuel element can lead to defects in the cladding. This is due mainly to a combination of a strain quite high and a low ductility of the cladding material. In CANDU reactors, the fuel elements are subjected to power ramps severe enough when reloaded during the functioning of the reactor.

The CANDU reactors from Cernavodă Nuclear Power Plant (NPP) are using as nuclear fuel bundles of 37 elements each, assembled by some edge grids. This bundle has a length of 495 mm, a diameter of 103 mm and weight of 24 kg. The CANDU fuel element contains cylindrical pellets of UO_2 syntherized, placed into a Zircaloy-4 tube (also known as sheath or cladding), closed at both edges with end caps. It has a length of 492 mm and a diameter of 13.08 mm. [1]

In order to check and improve the quality of the Romanian CANDU fuel, power ramp tests on experimental fuel elements were performed in our TRIGA SSR reactor. The irradiated fuel elements were further subjected to examination in the PIEL laboratory.

During the irradiation, the fuel elements suffer dimensional and structural changes, and also modifications of the cladding surface aspect, as result of corrosion and mechanical processes. This can lead to defects and even the integrity of the fuel element can be affected.

Fuel sheath examination by tensile testing was a study object for cladding producer and customer as well. The most important studies are: “A Study of Ring Test for Determination of Transverse Ductility of Fuel Element Canning” [2] and “A Critical Analysis of the Ring Expansion Test on Zircaloy Cladding Tubes” [3] done at Studsvik Nykőping Sweden which presents the requirements for this type of tests, general consideration of design and the main conclusions are also presented. Chalk River Laboratory, Ontario Canada also developed a test named “Development of a Closed End Burst Test Procedure for Zircaloy Tubing” regarding ring tests and JAERI Laboratory from Japan another study called “Optimization of Sample Geometry in Modified Ring Tensile Test” [4].

2. The aspect of the cladding surface

After irradiation, the fuel rod was kept in the reactor pool for three months, for cooling. The fuel rod was then transferred to the INR hot cells where it was subjected to detailed examinations.

An image of the fuel element is given in Fig. 1. It was obtained using a periscope, coupled with an OLYMPUS digital camera. The aspect of the cladding surface indicates a normal behaviour of the fuel element.

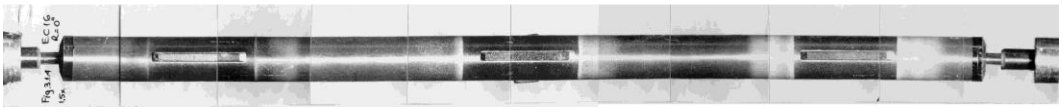


Fig. 1: Fuel element CANDU tested in the power ramp

3. Profilometry

The diametrical profile, diametrical increasing, ovality and the arrow of fuel element were determined. In Fig. 2 is presented the average diameter profile of the fuel element. The average diameter is 13.149 mm. The average diametrical increasing is 0.087 mm (0.67 %), with respect to the diameter before irradiation.

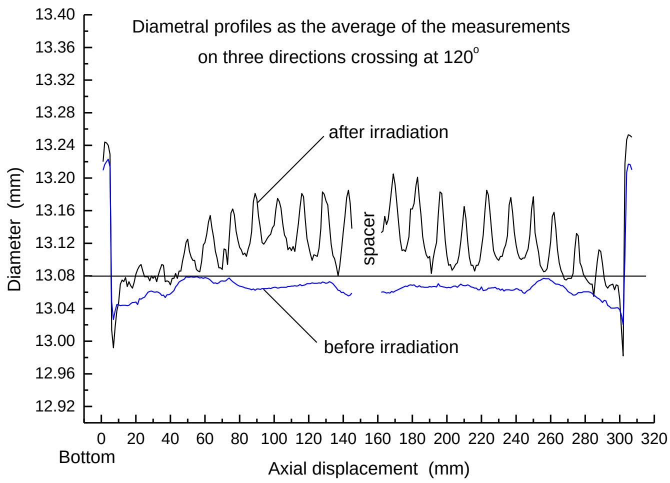


Fig. 2: Average diameter profile after irradiation

After visual examination of the fuel rod we choose the zone with the greatest ovality for detailed analyse of diameter. Ovality profiles of the fuel element for two different positions on the vertical axis, Z = 32 mm and Z = 88 mm, are presented in Fig. 3.

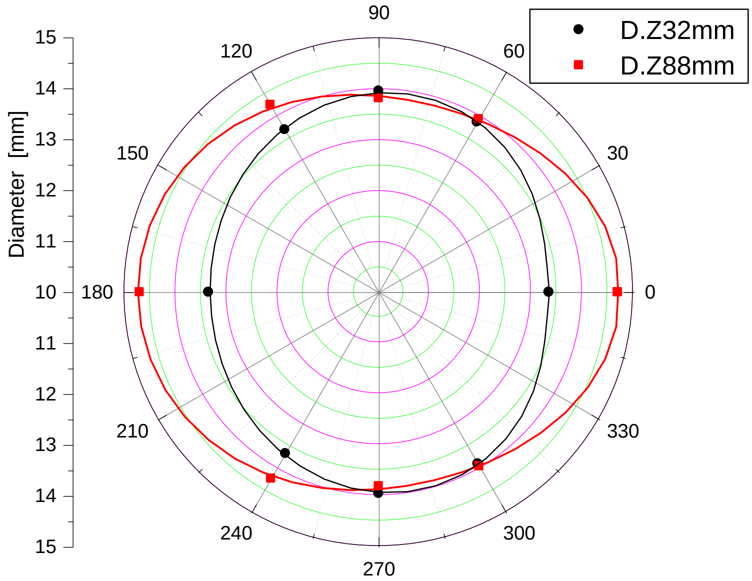


Fig. 3: Ovality profiles for Z = 32 mm and Z = 88 mm

The graphic representation was made based on the measurements performed at these positions in three directions (0°, 120° and 240°). The profiles of bending are presented in Fig. 4 which shows the profilometry results obtained from the experimental CANDU fuel element shown in Fig. 3 which has been irradiated in the TRIGA pulse reactor in a simulated RIA test. It is observed the prominent sheath swelling due to fuel swelling and a violent fission gas release which have determined sheath cracking. The sheath ovalization profile has been obtained by circumferential diameter measurements.

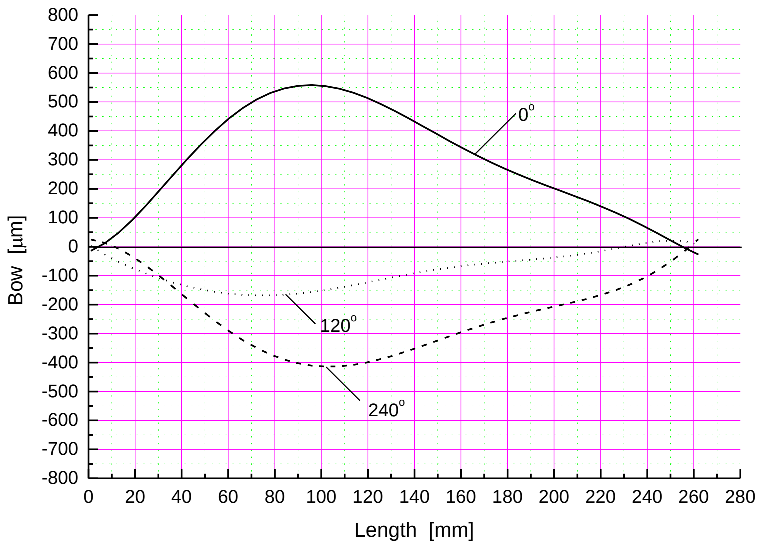


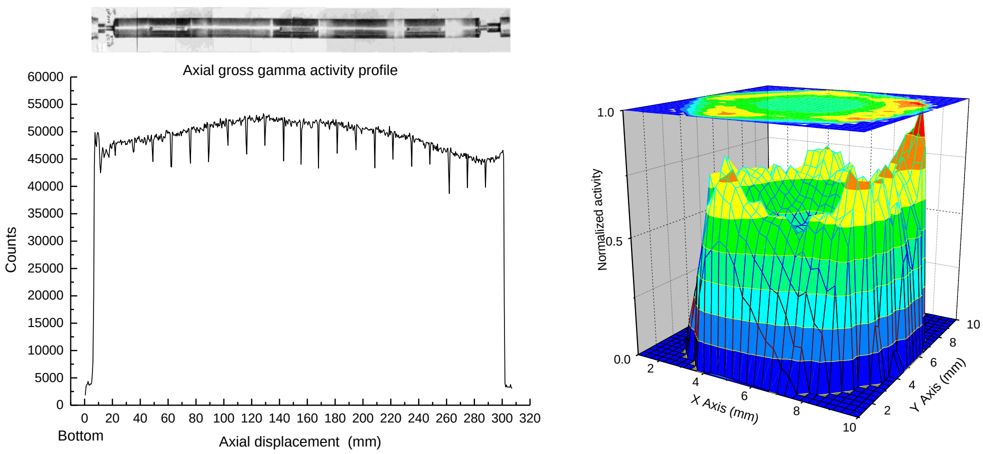
Fig. 4: Profiles of bending after irradiation

4. 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, in the hot cell shielding wall, a PGT intrinsic Ge detector and a multi-channel analyzer.

For axial gamma scanning, the slit of the collimator was horizontal, having an aperture 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. 5a shows the fuel rod axial gross gamma activity profile. A prominent depression of count rate at fuel pellet interfaces is observed, which means there is no interaction between the pellets. This gamma activity profile highlights practically a symmetric loading of the fuel rod.

A method of tomographic reconstruction based on a maximum entropy algorithm has been developed as described in ref. [5], [6]. The data acquisition was done while the fuel rod was moved transversally step-by-step at regular intervals of 0.25 mm after every 72º rotation in front of a vertical collimator slit (which is 50 mm high and has a 0.25 mm aperture). Fig. 5b shows, qualitatively, the tomographic image of the radial distribution of ^{137}Cs gamma activity in the cross section of the fuel rod, in the flux peaking area. This tomography indicates that the ^{137}Cs isotope migrated from the middle to the periphery of the fuel rod and was redistributed according to the temperature. The temperature is higher in the center and falls outward up to 300 °C, where the fuel element is in contact with the coolant.



a. b.

Fig. 5: Axial gamma scanning (a) and tomography (b) on a CANDU fuel rod irradiated in the INR TRIGA reactor in a power ramping test

The ^{137}Cs isotope was used as burn up monitor. For an accurate determination of the burnup, 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 MWd(kgU)$^{-1}$ (for 192 MeV fission of U). The fuel rod burns up determined by mass spectrometry is 9 MWd(kgU)$^{-1}$ (for 192 MeV fission of U). These results are in good agreement.

5. Metallographic and Ceramographic Examination

A LEICA TELATOM 4 optical microscope having a magnification up to x1000 was used for macrographic and microstructural analysis of the irradiated fuel rod. A computer-assisted analysis system is used for the quantitative determination of structural features, such as grain and pore size distribution.

The preparation of the samples includes precise cutting, vacuum resin impregnation, sample mounting with epoxy resin in an acrylic resin cup, mechanical grinding and polishing, chemical etching, ref. [7].

The analyses by optical microscopy provide information concerning:

- the aspect of pellet fissure (Fig. 6);
- the structural modifications of fuel and the sizes of the grains (Fig. 7);
- the thickness of the oxide layer and the cladding hydriding.

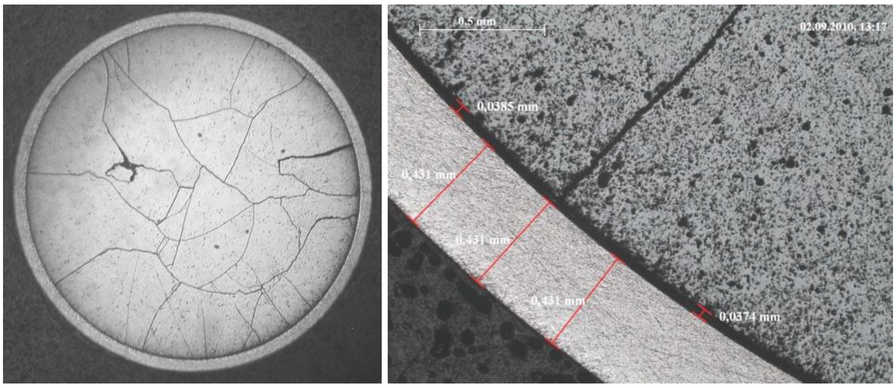
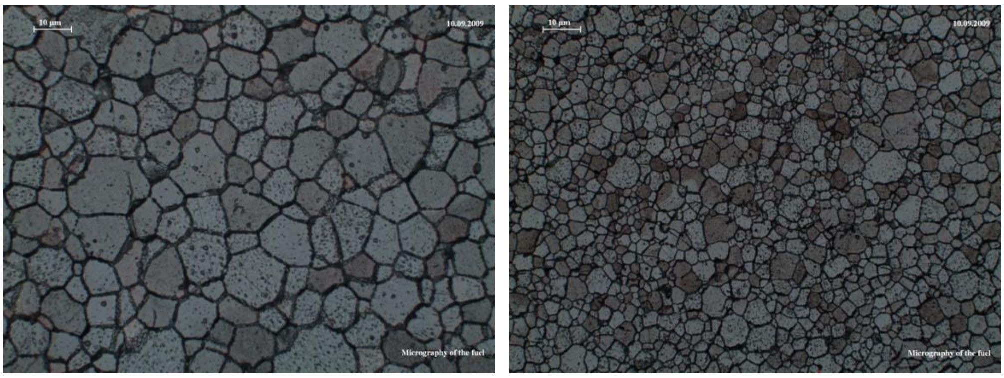


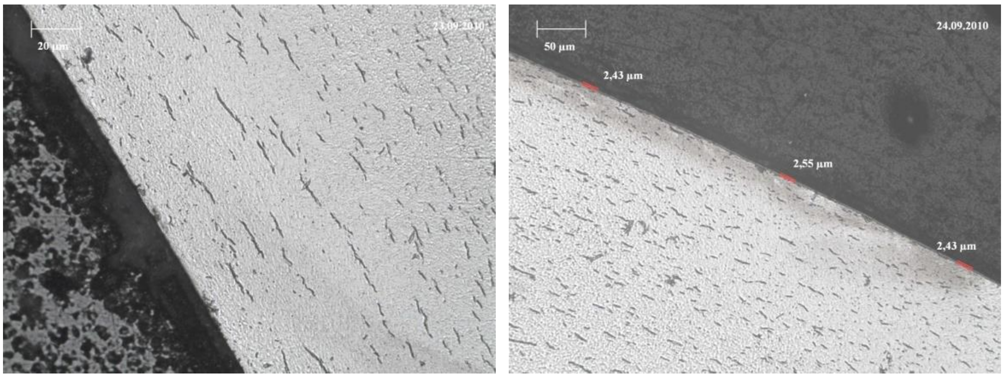
Fig. 6: The cross section of the fuel pellet

The cross section of the fuel pellet (x8) presents radial and circular fissures on the whole section. The cladding doesn't present nonconformities, the thickness of this being 0.431 mm. There are no visible effects on fuel sheath, due to mechanical or chemical interactions.



a) Equiaxial grains b) Unaffected grains

Fig. 7: The structural modifications in the fuel pellet



a) Cladding hydrading b) Outer oxide layer on cladding

Fig. 8: Cladding aspect

The hydride precipitates are orientated parallel to the cladding surface. A content of hydrogen of about 120 ppm was estimated by means of hydriding charts ref. [8]. The fuel element presents on the outer side of the cladding a continuous and uniform zirconium oxide layer (Fig. 8). The thickness of the cladding oxide layer is 2.5 μm.

6. Determination of mechanical properties

After the preliminary tests, three ring samples (5 mm long each) were cut from the fuel rod, for further tensile tests (Fig. 9). The samples were prepared according to the shapes and dimensions given in ref. [9], [10].

The samples are tested in order to evaluate the changes of their mechanical properties as a consequence of irradiation. The tensile testing machine used is an INSTRON 5569 model. The machine uses the Merlin software for data acquisition and analysis.

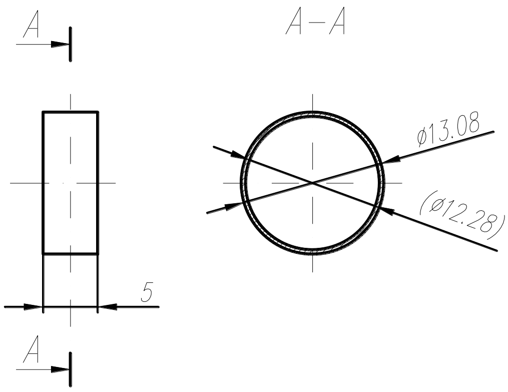


Fig. 9: Ring test sample

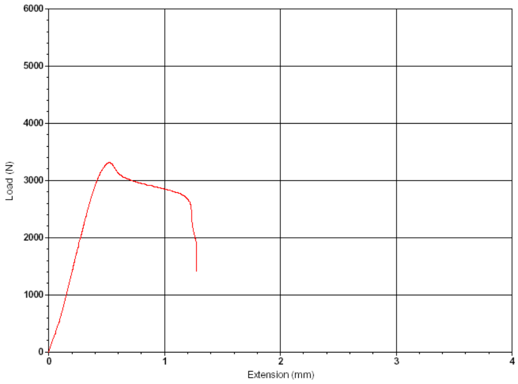


Fig. 10: Load-extension diagram

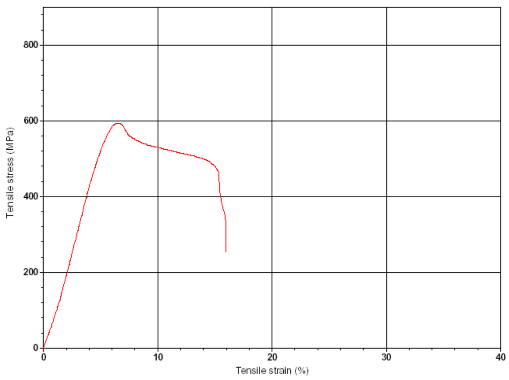


Fig. 11: Strain-stress diagram

The tests were done under the following conditions: constant testing temperature (300°C), 25N preload and constant tensile strain (v=0.05 min^{-1}). The tests results are presented in Table 1:

Table 1

Obtained results on irradiated cladding samples

Test	Testing speed (min^{-1})	Yield strength $\sigma_{0,2}$ (MPa)	Ultimate tensile strength σ_m (MPa)	Plastic elongation (%)	Total elongation (%)
Ring 1	0,05	498,7	597,5	6,3	29,4
Ring 2	0,05	348,9	428,2	2,6	10,1
Ring 3	0,05	679,4	700,0	1,2	4,2
Medium values		509,0	575,2	3,4	14,6

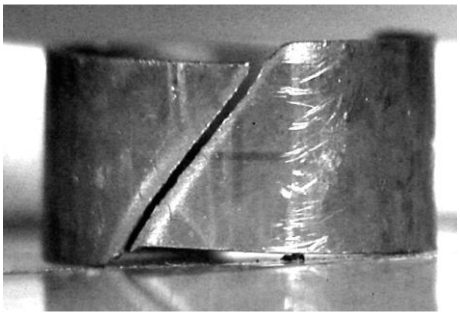


Fig. 12: Ring sample after test

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

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

The tests were done according to the procedures and standards are given in ref. [9], [10] and [11]. The aspect of the rupture and the relevant mechanical properties reveal the effect of irradiation on cladding material. It could be observed the increase of yield strengths and ultimate tensile strength with around 50% and the decrease of elasticity properties with the same value as effect of irradiation.

The aspect of the ring sample after the test is presented in Fig. 12.

The obtained results were compared with values presented in international research papers presented by specialists from other labs [12], [13] in similar conditions which validate the utility and reliability of the data from INR Pitesti.

7. Fracture surface analysis by scanning electron microscopy (SEM)

For sample analysis, an electron microscope model TESCAN MIRA II LMU CS with Schottky Field Emission and a variable pressure was used. The magnification range is 4x ÷ 1,000,000x. An outstanding depth field, much higher than in the case of optical microscopy characterizes the scanning electron microscopy (SEM). This makes SEM very appropriate for analyzing fracture surfaces of zircaloy 4 cladding resulted from the tensile test.

Because of the ring shape of the sample, for rupture surface visualization, the sample was split into two parts, which were mounted in microscope chamber as in Fig. 13.

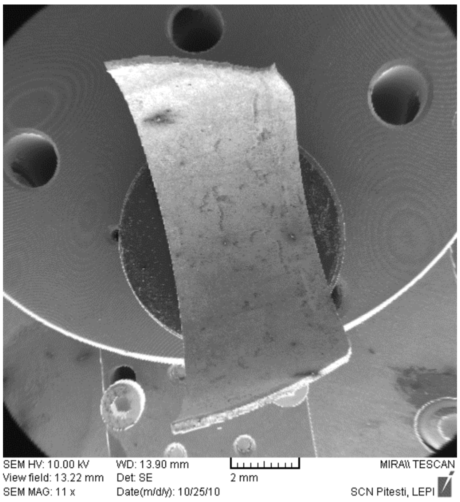
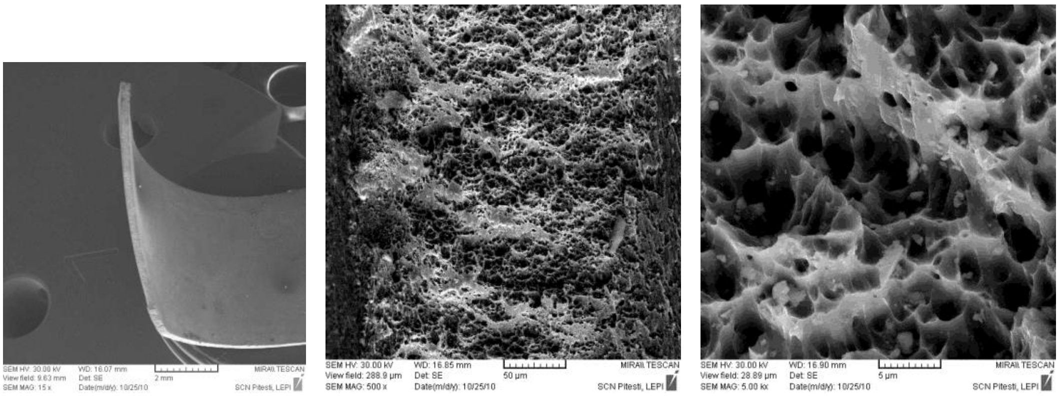


Fig. 13: Sample fixture on the electronic microscope table

Both sides of the tensile fracture were analyzed on each half of the ring. The dimples from the central zone are rather deep, whereas the ones on the outer side are tilted and smaller.

The central zone of the fracture presents equiaxial dimples (Fig. 14).



a) general view x15 b) x500 c) x2000

Fig. 14 The aspect of the central zone of the fracture

8. Conclusions

After irradiation, the fuel rod was kept in the reactor pool, for cooling and then it was transferred to the INR-PIEL hot cells where it was subjected to detailed examinations.

First of all, visual inspection of the cladding was done. The aspect of the cladding surface indicates a normal behaviour of the fuel element.

The diametrical profile, diametrical increasing, ovality and the arrow of fuel element were determined.

The tomography indicates that the ^{137}Cs isotope migrated from middle to periphery of the fuel rod and was redistributed according to the temperature profile.

By metallographic and ceramographic examination we determined that the hydride precipitates are orientated parallel to the cladding surface. A content of hydrogen of about 120 ppm was estimated. The cladding doesn't present nonconformities. The fuel element presents on the outer side of the cladding a continuous and uniform zirconium oxide layer 2.5 μm thick.

After the preliminary tests, three ring samples were cut from the fuel rod and were subject to tensile test on an INSTRON 5569 model machine in order to evaluate the changes of their mechanical properties as a consequence of irradiation.

Scanning electron microscopy was performed on a microscope model TESCAN MIRA II LMU CS with Schottky FE emitter and variable pressure. The analysis shows that the central zone has deeper dimples, whereas, on the outer zone, the dimples are tilted and smaller.

The post-irradiation examination of CANDU fuel elements manufactured at Pitesti and tested in the TRIGA material testing reactor has been performed in the hot cells from INR Pitesti since 1984 as part of the Romanian research program for the manufacturing, development and safety of the 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 and structural changes of the fuel have enabled the CANDU fuel behaviour characterization after its testing in the TRIGA reactor both in normal operation and in accident conditions.

During 20 years of post-irradiation examination, the Post-Irradiation Examination Laboratory from INR Pitesti has obtained the operational and professional ability to evaluate the performance of the CANDU nuclear fuel from Cernavoda NPP.

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