Paper for presentation at
“6 th International Conference on WWER Fuel Performance, Modelling
and Experimental Support,
19–23 September 2005, Albena Congress Center, Bulgaria,

"REVIEW OF WWER FUEL AND MATERIAL TESTS IN THE HALDEN REACTOR"

B. Volkov, e-mail: Boris.Volkov@hrp.no
E. Kolstad, e-mail: Erik.Kolstad@hrp.no
OECD Halden Reactor Project

Abstract

A review of the tests with WWER fuels and materials conducted in HBWR over the years of cooperation with Russia is presented. The first test with old generation WWER-440 fuel and PWR specification fuel was carried out from 1995 to 1998. Some differences between these fuels regarding irradiation induced densifications and pellet design as well as similar fuel thermal behaviour, swelling and FGR were revealed during the test. The data from this test are reviewed and compared with PIE recently performed to confirm the in-pile measurements.

The second test was started in March 1999 with the main objective to study different modified WWER fuels also in comparison with PWR fuel. The results indicated that all these modified WWER fuels exhibit improved densification properties relative to earlier tested fuel. In-pile data on fuel densification have been analysed with respect to as-fabricated fuel microstructure and can be used for verification of fuel behaviour models.

Corrosion and creep tests in the Halden reactor encompass WWER cladding alloys and some results are given. Prospective WWER fuel and material tests foreseen within the frame of the joint program of OECD HRP are also presented.

1. INTRODUCTION

The Institutt for Energiteknikk was established in 1958 and since then it has hosted the Halden Reactor Project (HRP) organised under the auspices of the OECD Nuclear Energy Agency which is promoting cooperation among the member countries in the area of nuclear fuels and materials development. The main tool of the HRP is the Halden Boiling Water Reactor (HBWR) of 20 MW power, which operates with heavy water as coolant and moderator. The reactor is employed to carry out the Joint Programme (sponsored by 18 member countries) and client experiments on bilateral basis. Active fuel testing began in the early 70-ties. Investigations related to materials behaviour have been carried out since the late 80-ties and this area is growing.

In 1995 Russia joined the HRP and this year marks the 10-th anniversary of the first WWER fuel test in the Halden reactor. The main objective of this first test was to investigate WWER fuel behaviour and reliability in comparison with PWR specification fuel. Based on the results obtained from this test, the WWER fuels have been modified and delivered by MSZ Electrostal for the next test which is carried on to date. Additionally, some structural materials developed in Russia and used in WWERs have also been investigated within the frame of the HRP joint program. A review of the results from the WWER fuel and material tests conducted over the last years as well as prospective WWER tests envisaged in the near future are presented in this paper.

2. EXPERIMENTAL CAPABILITIES AT THE HALDEN REACTOR

The first Instrumented Fuel Assembly (IFA) was loaded in HBWR in 1963. From that time a large number of in-pile tests have been performed in the Halden reactor constantly improving the in-pile instrumentation and methods of measurements. IFE develops and produces high precision measuring instruments and experimental equipments which allow in-pile fuel temperature, gas pressure, fuel and cladding elongation to be measured continually up to 10 years. A number of various experimental systems are also developed and applied to provide specific information on nuclear fuels and materials in-core behaviour. The capability of the reactor allows testing at the same time of up to 35 rigs including installations in 15 loops operated under required thermal-hydraulic conditions and specified water chemistry. More than 1000 raw signals simultaneously coming from the reactor are collected and treated by the data acquisition system using powerful computers installed at HRP. The data are recorded with one minutes intervals under normal operation conditions, however, for special purposes the acquisition system is capable of logging data with frequency from 0.5 sec to 25 ms, which is very important for transient tests.

The experimental systems and in-pile instrumentation allow the following tests to be conducted:

- instrumented fuel and material base irradiations under HBWR conditions (240 º C, 34 bar);
- instrumented fuel and material tests under typical BWR or PWR conditions with specified water chemistry in the test loops;
- ramp tests controlled by ^{3}He gas pressure system;
- tests with measurement of fuel rod diameter changes by means of a hydraulically driven diameter gauge;
- tests with hydraulically movement of the fuel rod inside the reactor core for ramp testing at high neutron flux during reactor operation;
- tests allowing gases from fuel rods to be swept for on-line analysis of the radioactive isotopes by gamma spectrometer;

- tests to study the stress corrosion cracking of reactor materials/internals under influence of different mechanical loads driven by a high gas pressure system;
- Dry out and LOCA simulation tests carried out in special designed water loops.

It should be noted that the in-pile instrumentation technology recently developed by HRP, enables fuel stack and cladding elongation, internal fuel rod pressure (fission gas release) and fuel centre temperatures to be simultaneously measured in the same rod.

3. WWER FUEL TESTS IN THE HALDEN REACTOR

3.1 First Comparative WWER/PWR Fuel Test in HBWR (IFA-503.1)

The main objective of the first test in IFA-503 was to compare the behaviour of WWER-440 fuel (produced by MSZ Electrostal until 1995) with PWR specification fuel. The test commenced in 1995 and concentrated on in-pile data on the fuel densification and swelling, thermal behaviour and fission gas release. The data revealed insignificant differences in the behaviour of these two types of fuel, with the exception of 2-2.5 times larger densification in the WWER fuel [1,2].

The test was finished in November 1998 at a burnup of about 26-28 MWd/kg UO_2. Later, one of the WWER rods was chosen for destructive PIE to verify the in-pile data. The rod was instrumented with a fuel elongation detector and pressure transducer allowing fuel dimensional and gas pressure changes as well as FGR to be evaluated. The as-fabricated data, evaluations derived from in-pile measurements at EOL and PIE data are compiled for comparison in Table 1.

For burnup determination, the so-called “Neodymium-148” method was applied. The burnup of 3.34 % of heavy atoms was determined corresponding to 30.6 - 32.2 MWd/kgU.

TABLE 1 Comparison between in-pile data and PIE

Parameter	As- fabricated	In-pile data at EOL	PIE
Burnup, MWd/kgU	-	30.0 - 31.6*	30.6 - 32.2
Internal pressure (STP), bar	5.60	5.60	5.57
Free volume, cm^3	5.3	5.27**	5.27
Total gas amount (STP), cm^3	29.2	29.4	28.8
Helium content (STP) cm^3	29.2	28.3-28.6	28.3
Helium deficiency,% from filled gas	-	2-4%***	3.1
Fission gas release,%	-	~1.0	< 1.0

* - burnup was estimated at the same axial level where the sample was cut for PIE;
**- free volume derived from EF measurements at EOL;
***- the data on He-absorption derived from free volume changes evaluated from rod pressure (PF) and fuel elongation (EF) measurements.

Comparison shows good agreement between the in-pile evaluations and PIE data which confirms not only the in-pile burnup determination, fuel dimensional changes and low FGR but also the He-gas deficiency revealed in the in-pile test [3]. Some polished and etched cross-sections of the irradiated WWER pellets are shown in Figure 1. Ceramography shows about 5 fully radial cracks from the centre hole to the outer edge of the pellet and some partial radial cracks as well. On both etched and polished cross-sections, an inhomogeneous microstructure was observed. This was indicated by the clusters of pores and grains across the pellet radius.

3.2 Modified WWER Fuels Testing in HBWR in Comparison with PWR Fuel (IFA-503.2)

In March 1999, the next test with modified WWER fuel pellets, produced by MSZ Electrostal using improved technology, was commenced and it is still going on. Three types of the modified WWER fuel pellets and PWR reference fuel are irradiated in IFA-503.2, with the main objective to compare fuel dimensional stability at BOL, thermal behaviour and FGR with burnup. The modified WWER fuels are also investigated to determine the fuel variant with the best in-pile performance. Power history and burnup development are shown in Figure 2.

Results of the in-pile measurements indicate that all modified WWER fuels exhibit improved densification properties in comparison with earlier tested WWER fuel (IFA-503.1). One of the WWER fuels (designated WWER-1 in the test) shows most stable dimensional behaviour which is close to PWR fuel. The fuel densification and subsequent swelling rates, derived from the fuel elongation measurements of the modified WWER fuels compared with PWR fuel and old generation WWER fuel (tested in IFA-503.1), are shown in Figure. 3.

The dimensional stability of all the fuels tested in IFA-503 was also considered with respect to their as-fabricated microstructure which was examined to investigate potential correlations to the observed densification of the different WWER and PWR fuels [2]. Porosity examination over a wide range of pore sizes was carried out for WWER-1 fuel. For examination of submicron porosity, SEM (Scanning Electron Microscope) was used whereas LOM (Light Optical Microscope) was applied for medium and macro porosity observation. Three scales of the magnification were chosen to cover the total range (0.1-100 μm) of the pore sizes in the fuel. All these examinations were linked together for overall pore size distribution to be evaluated for this fuel type. A quantitative 2D analysis of the ceramographical images was performed by special software for determination of frequency of the pores by logarithmic pore class intervals. Then these two-dimensional examinations were transformed to volume (3D) porosity distributions by sizes of pores using Saltykov's methods of quantitative stereology [4]. The data suggest that WWER-1 fuel has a two-waves porosity distribution where the peak at submicron pore sizes may be related to the basic porosity formation whereas the second one may by a result of the pore-former utilizations.

The porosity distributions were also examined for the other types of the WWER and PWR fuel but only in the medium pore-size range. Overall pore-size distributions of these fuels were evaluated by means of extrapolation of the data to the micro and macro pore-size ranges using data on immersion density of these fuels and assuming two-waves log-normal law of the distribution. The results of the porosity distributions for the four types of the fuel tested in IFA-503 are shown in Figure 4. The analysis confirmed an effect of sub-micron porosity on fuel dimensional stability at BOL.

The fuel stack elongation measurements of the flat pellets employed in the test revealed difference in the fuel shortening evaluated at HSB and at power. A so-called "in-pile dishing" is suspected to be developed in the initially flat-ended pellets [3]. This effect was more pronounced in the PWR type fuel than in the WWER fuels, probably due to higher fuel temperatures in the solid PWR pellets than in the hollow WWER pellets. Both fuel densification and local fuel creep may contribute to the fuel shortening at power. Estimate of the local fuel deformation due to creep in the flat-ended pellets was done in comparison with the in-pile fuel stack elongations measured at power in the PWR rod (with solid flat-ended pellets) and WWER rods (with hollow pellets) at BOL.

The estimation of local fuel pellet creep deformation was based on the correlation from the MATRPRO library [5]. It was assumed that initially flat ends of the pellets become convex at power due to axial thermal expansion correlated with radial fuel temperature distribution across the pellet. In this case, the pellet ends come into contact within a small area resulting in high compressive stresses generated by axial forces provided by built-in rod spring or early-in-life PCMI. At power these stresses are relaxed due to local fuel deformation of leading to contact area increase and fuel column shortening. The evolution of this relaxation process due to fuel creep in WWER hollow pellets and PWR solid pellets (without dishes) is shown in Figure 5. It should be noted that the contact area between pellets is uncertain but has a strong effect on axial creep deformation. Small contact area was simply chosen to demonstrate possible effect of creep on axial fuel stack shortening of different PWR and WWER fuel design. The lower values for the WWER pellets can be explained by lower temperature around the hole as well as the larger contact area between pellets because of contact ring for hollow pellets instead of circular contact between solid pellets.

The results of the calculations were considered together with the relative fuel stack shortening at power measured in the PWR and WWER fuel rods from IFA-503.2 for 1000 hours from BOL. Eliminating the fuel creep from the fuel stack shortening evaluated at power give similar value of residual fuel shrinkage at HSB as shown in Figure 6 for WWER and PWR fuel in IFA-503.2.

Gas pressures in all fuel rods also reflected some changes of the internal free volume due to fuel densification and subsequent swelling. However, the PF data analysis revealed somewhat larger densification than that derived from the fuel elongation measurements whereas the subsequent swelling rates were similar for all fuel types (~0.50 % / 10 MWd/kg UO_2). In this case, an effect of helium diffusion/absorption into fuel matrix has been considered as a potential cause of this phenomenon. The analysis showed that helium absorption, which may occur early in life, is about 2 - 4% of the fill gas. This effect was confirmed by PIE revealed as a He-gas deficiency at the end of irradiation of the WWER fuel in the first loading.

The fuel thermal performance study is based on fuel centreline temperature measurements which show reasonably consistent trends with the rod powers and other in-pile data. The fuel thermal performance has been consistent with irradiation-induced fuel densification and swelling evaluated from the fuel stack elongation measurements for the same type of fuel. The normalised fuel centreline temperatures have reflected a stable fuel thermal behaviour resulting in competition between gap conductance improvement (due to fuel relocation/swelling) and fuel conductivity degradation.

4. WWER CLADDING MATERIALS TESTING AT HALDEN

4.1 Comparative Corrosion Test

The purpose of the comparative test in the Halden reactor is to study the corrosion and hydriding behaviour of different zirconium-based cladding materials under PWR conditions at increased burnup levels (exceeding 40 - 50 MWd/kg U). Test assembly has been irradiated for 11 cycles. The test materials are in the form of fuelled tube claddings and coupons, and have been exposed to water chemistry, fluence and thermal hydraulic conditions similar to those found in commercial PWRs. Corrosion behaviour is assessed by means of interim inspections, comprising of weight gain and oxide thickness measurements and photography. Oxide thickness measurements on fuelled cladding segments are performed by means of an eddy current proximity probe (based on an individual calibration for each material type).The materials under investigation comprise 12 fuelled tube specimens and 51 unfuelled coupons in the form of 1/3 or 1/4 sectors of canning with length 30 mm are installed in two holders; one located above the fuel rods and the other running along the axial length of the rods.

The test rig is contained within a pressure flask and connected to a PWR loop. Water chemistry conditions similar to those in commercial PWRs are created by adding boron, lithium and hydrogen. Water chemistry conditions specified are 3 ppm Li, 1000 ppm B and 2 - 4 ppm H_2 maintained at constant levels throughout the irradiation cycles.

Temperature, pressure and flow rate in the loop system are also monitored continuously. The inlet temperature was in the range from 300 to 310°C, and a coolant heat-up of 8 - 9°C was measured at a coolant flow rate varying between 1.2 and 1.9 m/s at the coolant pressure of 155 bar.

Three vertically positioned vanadium detectors enable the axial flux and power distribution to be determined. In order to increase the fast flux level, booster fuel rods surround the test section, covering both the coupons and the fuel rods. Average fast neutron flux (> 1 MeV) levels within the fuel rod region have varied from approximately 2.0 to 3.5 10^{13} n cm^{-2} s^{-1}.

The alloys used as a structural materials in WWERs have also been included in the comparative test. Alloy E635 supplied by ABB Atom is being tested both as a coupon and as a fuelled cladding segment. Kurchatov Institute has provided a coupon of Zr-1%Nb (E110). These coupons are compared with standard Zr-4 coupons located at similar axial rig positions and exposed to similar coolant temperatures. A number of interim inspections provided information on weight gain of the samples during the test execution. These data was re-calculated to the oxide thicknesses shown in Fig. 7 for two pairs of coupons tested at low (inlet position) and high (outlet) temperatures.

The results obtained up to 1000 full power days corresponded to a test assembly burnup of about 44 MWd/kg U. The data indicated that the oxidation rates of the WWER E110 cladding are lower than Zr-4 which have been tested 900 fpd. The other Russian alloy E635 tested at higher temperature has exhibited an oxidation rate similar to those for Zr-4. However, it should be noted that the oxidation rates of both alloys began to increase after 700 fpd.

4.2 Comparative Creep Test

The Halden Project investigates the cladding in-pile creep of modern pre-irradiated alloys under varying stress levels in a high pressure loop. The rig, shown in Fig. 8, contains two test rods, one comprising of two and one of three clad sub segments. The lower rod consists of two segments of Zry-2 (BWR cladding) from fuel irradiated in a commercial reactor. The upper test rod consists of three segments, M5, ZIRLO and Zr-1%Nb (WWER cladding). The WWER cladding specimen was pre-irradiated in KFKI (Hungary) to the fast neutron fluence of about 0.7×10^{21} $n \cdot cm^{-2}$.

The rods are independently connected to an external gas pressurization system for controlling the stress conditions of the claddings. Each test rod is equipped with a 3-point contact diameter gauge scanning the outer surface of the rod including the mid- and end-plugs. The known magnitude of these mid- and end-plugs allow the cladding diameter measurements to be calibrated for each gauge run. The Zr-1%Nb sub-segment at the top of the rod is unfuelled and thus it sees constant temperature and has no gradients across the cladding wall.

Both internal and external rod pressures are monitored on-line and the data have been used to calculate the applied hoop stress in the clad segments. The stress history of the E-110 segment as well as mid-wall temperatures and fast fluxes are shown in Fig. 9.

Axial scans of the cladding segments have been taken with various intervals through the experiment, with the frequency increasing just after a stress change.

Steady state secondary creep data have been clearly seen in E-110 segment after approximately 3200 fph at tensile stresses as shown in Fig. 10. A quite low outward creep rate of the unfuelled WWER segment is observed, probably, due to benign conditions in terms of stress, temperature and fast flux.

5. PROSPECTIVE WWER FUEL AND MATERIAL TESTS IN THE HALDEN REACTOR

The joint research program for the coming three-year period is based on the priorities of HRP member organizations with respect to improving the nuclear fuel utilization. Over the past ten years the WWER tests at HRP mostly investigated safety and reliability aspects of the Russian-line production nuclear fuels and materials, whereas the WWER tests planned for forthcoming period are more related to the researches and developments aiming at utilization of the modern nuclear fuel.

5.1 New Generation WWER-1000 Fuel

In the next three-year program period 2006 – 2008, it is planned to test the following WWER-types of fuel in the Halden reactor:

- Gd-doped fuel, with 5% absorbing isotopes, utilizing in WWER-1000;
- New generation WWER-1000 fuel with alumosilicate and niobium oxide additives enhancing grain size of the fuel matrix;
- Reference WWER specification fuel pellets for comparison with new fuel types.

The tentative test matrix requires high power rating (35-40 kW/m) at BOL and target burnup of about 60 MWd/kgU. The in-pile test will provide information on behaviour of Gd-doped and large grain pellets in comparison with typical UO_2 WWER fuel. The objective of the Gd-doped fuel testing is to study the effect of a burnable absorber on fuel behaviour at BOL and to determine thermal properties of this fuel type with burnup. The test with new generation WWER-1000 fuel is focusing on FGR and PCMI behaviour at high burnup. It is also supposed to study in-pile fuel thermal performance, dimensional behaviour at BOL. The irradiation test in the HBWR will be complemented with PIE to examine formation of the rim structure in this fuel type.

It should be noted that advanced in-pile instrumentation of the test fuel rods consist of 5 detectors (fuel stack elongation (EF), cladding elongation (EC), internal fuel rod pressure (PF) and fuel centre temperatures (2x TF – thermocouples)), which enable the entire fuel performance to be examined during irradiation.

5.2 LOVIISA WWER Fuel Re-irradiation Tests in HBWR

A three parties project with the leadership of Fortum Nuclear Services Ltd (FNS, Finland) supporting by JSC TVEL (Russia) and Paks NPP (Hungary) was initiated in 2002 with the aim to investigate reliability of WWER-440 fuel at enhanced level of burnup.

According to the project, advanced WWER-440 fuel assembly irradiated in the Loviisa NPP to a burnup of 43.7 MWd/kgU was subjected to poolside inspection. Afterwards several rods with average burnup of around 49 MWd/kgU were detached from this assembly and sent to Studsvik for destructive Post Irradiation Examinations. After completion of PIE, several rod segments irradiated to a burnup of 50-55 MWd/kgU were delivered to OECD Halden Reactor Project for re-instrumentation and in-pile testing in HBWR.

For the period 2006-2008, the following in-pile tests are planned to be carried out using these re-instrumented fuel rod segments:

- Clad creep out or so-called “lift-off” test conducted under controlled internal gas pressure in the test rod. The pressurization of the rod is up to 400 bars. With a loop pressure of about 150-160 bar and coolant temperature of 300 - 320 ^{o}C, the overpressure can reach about 200-250 bar. One test will require about 2000 hours of irradiation;
- Ramp test to study FGR and PCMI behaviour. The rig will be operated under HBWR conditions (coolant 34 bar, 240 °C) and the test rods surrounded by a He-3 coil. The coil enables power reduction down to 40 - 50 %. The schedule of the test and rod instrumentation will be dependent on the requirements which are under discussion.
- LOCA simulation tests similar to the series which have been performed at HRP recently to investigate fuel fragmentation and relocation at the balloon region. The single test rod will be instrumented with cladding surface thermocouples, cladding extensometer, pressure sensor, fuel thermocouple and fast response neutron detectors. Target peak clad temperatures may be between 800 and 1200 °C for the tests planned.

Post Irradiation Examination will be an important part of these tests.

6 SUMMARY

The current WWER fuel test IFA-503 in the Halden reactor will be completed at the end of 2005.

Over the ten years of WWER fuel testing in the Halden reactor the following results were achieved:

- The data base on in-pile WWER fuel behavior was extended;
- The data provided information on the dimensional stability of WWER fuel;
- The effect of fuel microstructure on in-pile densification was evaluated;
- One of the modified WWER fuels exhibit in-pile behaviour similar to PWR specification fuel;
- corrosion rates of E110 and E635 alloys are evaluated in comparison with standard Zr-4 and other modern cladding materials;
- in-pile creep of E110 alloy was evaluated in comparison with other cladding materials;

The following experiments with WWER-1000 fuels are planned to be carried out in the Halden reactor from 2006:

- new generation WWER-1000 fuel pellets with large grains;
- commercial Gd-bearing fuel with 5% absorbing isotopes;

A series of tests with WWER fuel irradiated in Loviisa NPP will also be performed in the coming three-year period (2006-2008).

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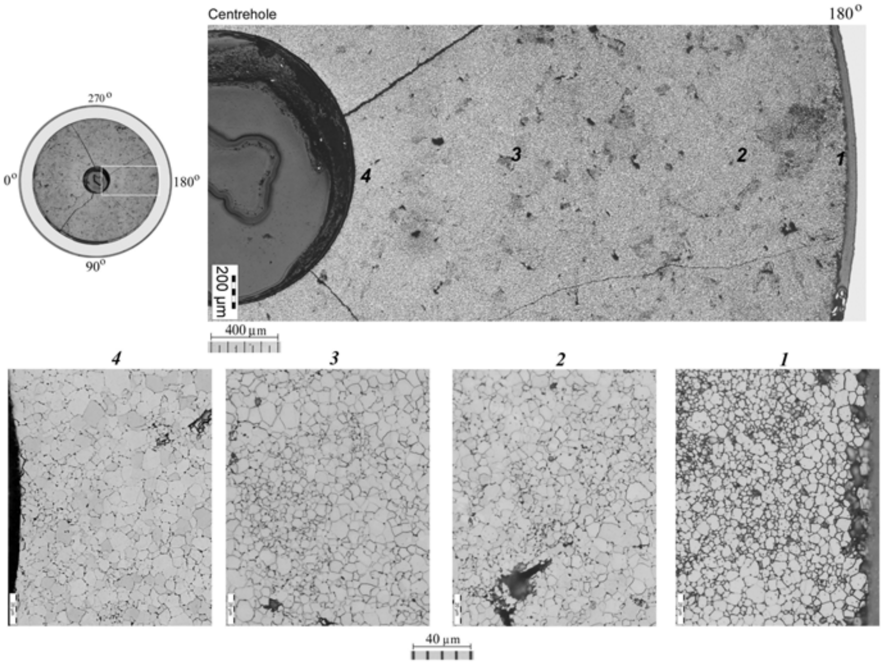

Figure 1 Cross section of the WWER-1 pellets after irradiation in IFA-503.1

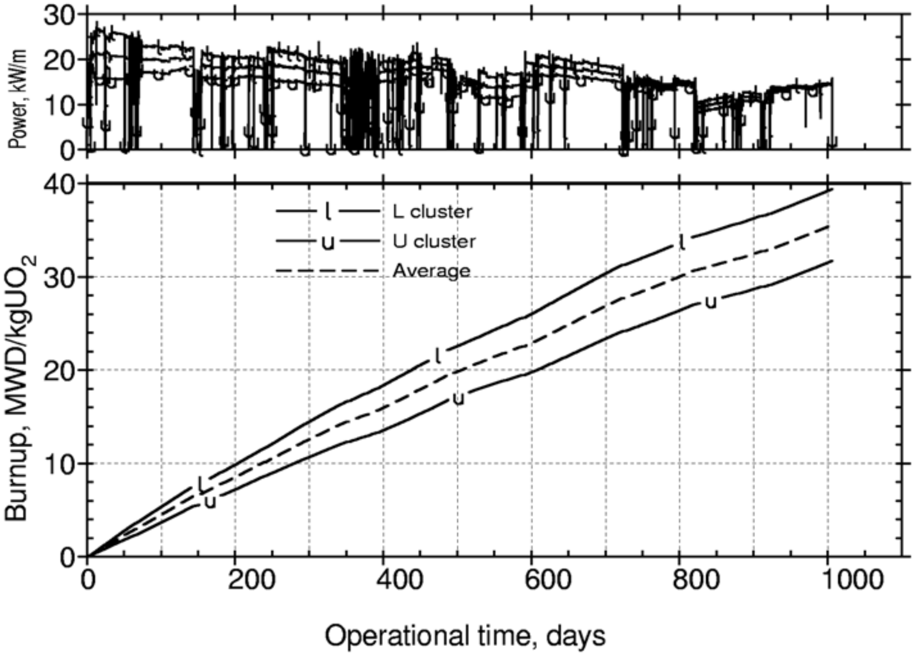

Figure 2 Power history and burnup inventory in IFA-503.2

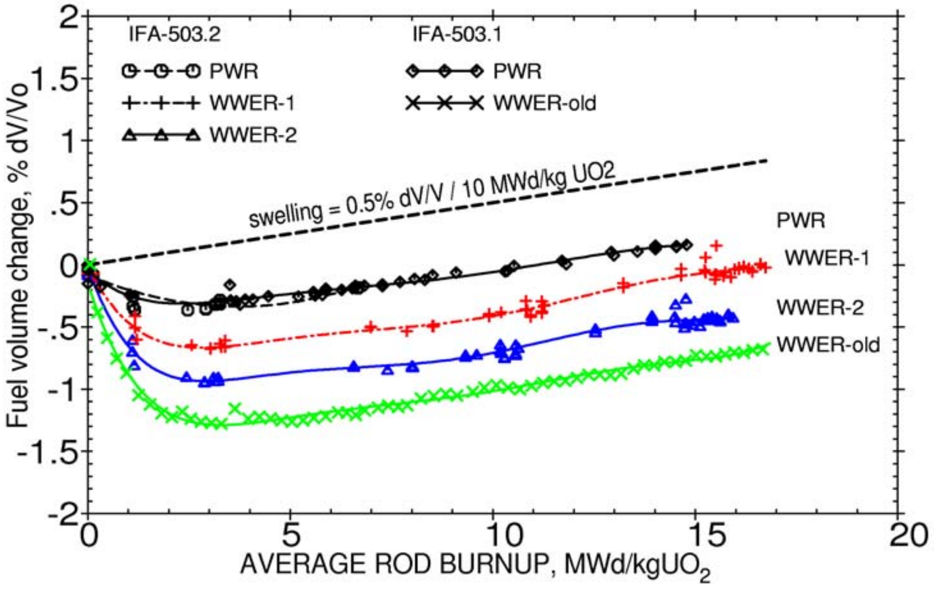

Figure 3 Fuel densification and swelling of different fuels irradiated in IFA-503

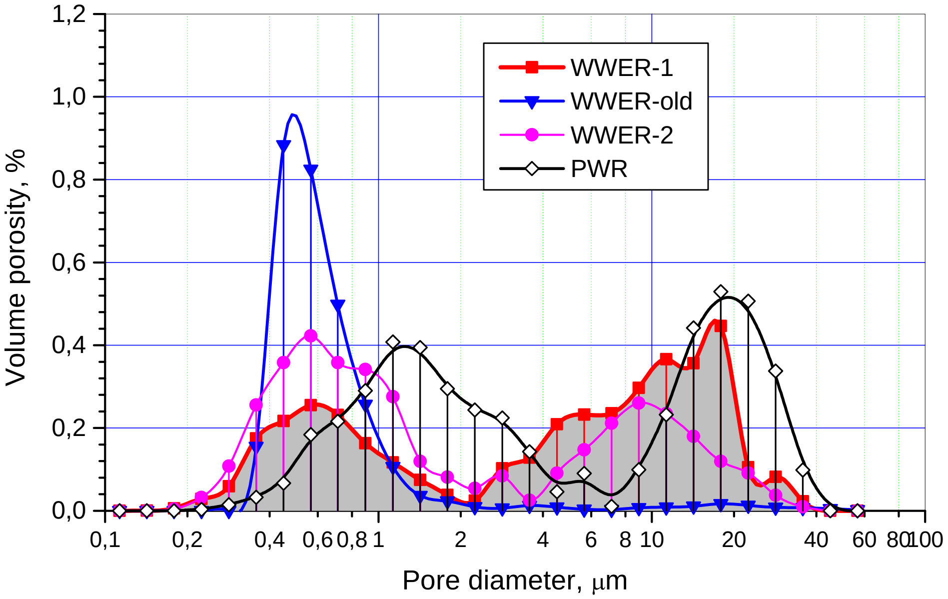

Figure 4. Volume porosity distribution evaluated from microstructural examination of reference fresh pellets of the fuels irradiated in IFA-503

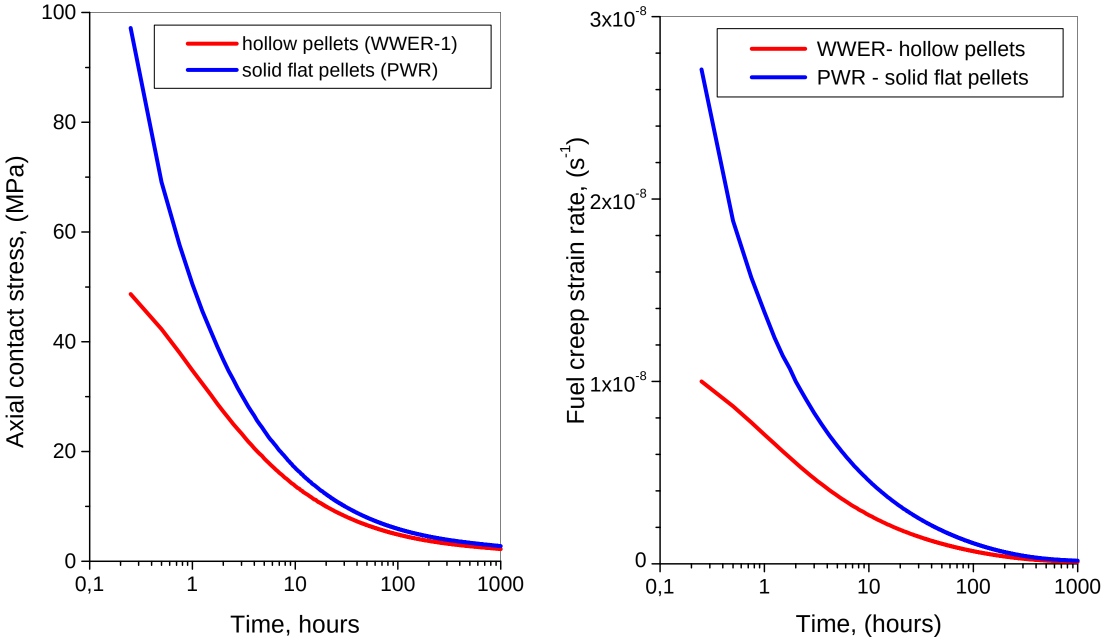

Figure 5. Estimated axial contact stress and subsequent strain of the WWER hollow pellets in comparison with those of the PWR solid flat pellets employed in IFA-503.2 test

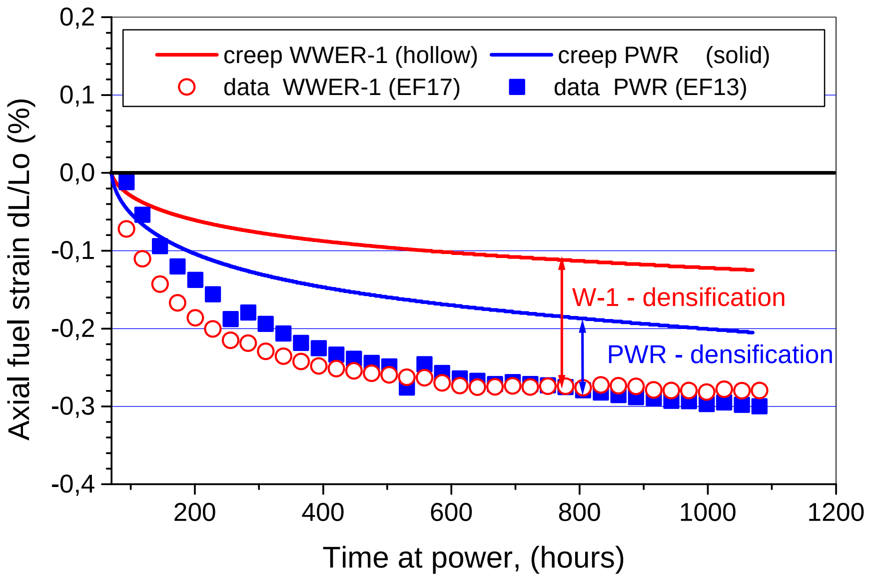

Figure 6. Axial fuel stack shrinkage due to fuel creep and densification estimated for WWER and PWR fuel rods from the test IFA-503.2

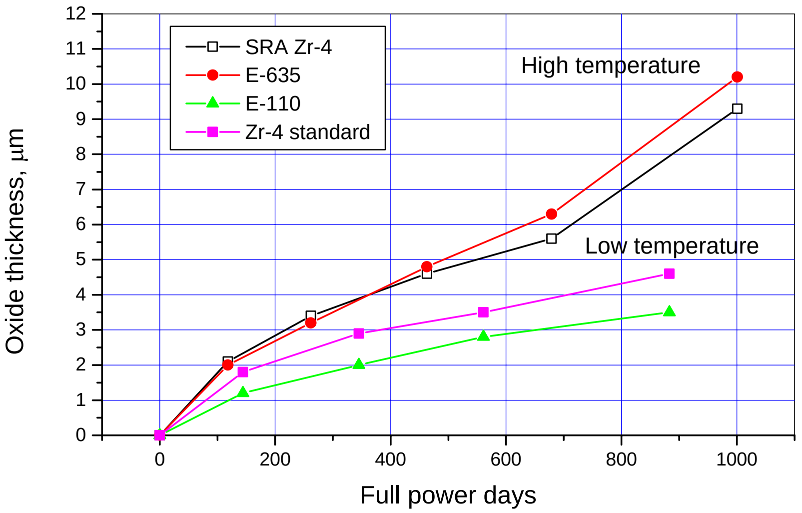

Figure 7. Corrosion rates measured for the WWER materials E110 and E-635 in comparison with Zr-4

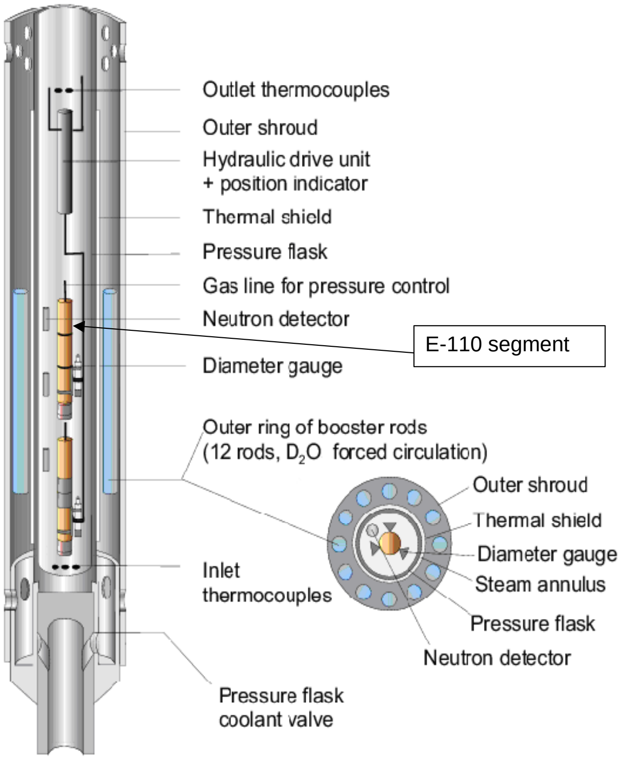

Figure 8 Principal rig design for in-pile measurements of the fuel rod diameter during irradiation (creep test with E110 cladding alloy)

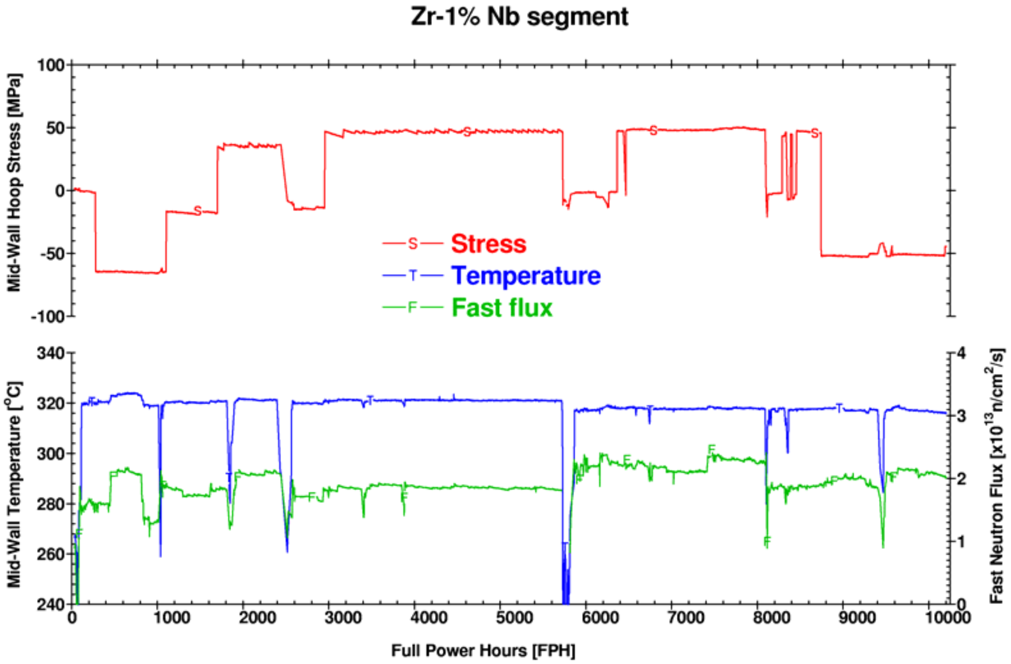

Figure 9. Mid-wall stresses, temperatures and fast neutron flux history determined for cladding creep test with E110 alloy.

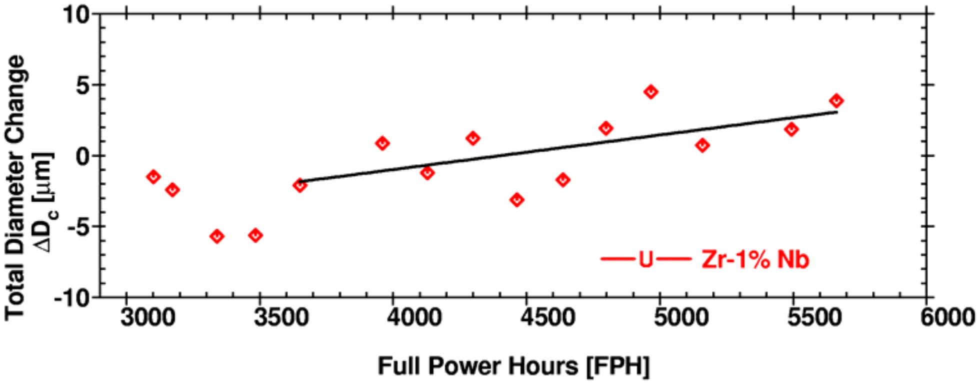

Figure. 10. Data on diameter changes of the E110 cladding sample during creep test. (Creep rate corresponds to about 20 micron / year)