Performance Evaluation of the Loviisa Advanced Type Fuel Rods

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1. Introduction

1.1. The New Type Fuel Assemblies

For the Loviisa WWER-440 power plant, operated by Fortum Power and Heat, the Russian fuel vendor has supplied six lead assemblies of an advanced type which feature profiling of the fuel enrichment, de-mountability of the assembly and a reduced shroud wall thickness. The manufacturer has used a new pellet fabrication technology, which is expected to provide more stable fuel performance under operation. The pool side examination programme of the fuel rods has been initiated [1] including visual inspections, as well as diameter and length measurements between cycles. At end-of-life, the fission gas release data, estimated from ^{85}Kr activity in the plenum, will be exploited for comparing the new and the old fuel.

1.2. The ENIGMA Fuel Performance Code

Complementary fuel rod evaluations and testing of the fuel performance models are done by VTT Energy. The British ENIGMA code predicts the behaviour of a fuel rod in normal operation and during operational transients. Earlier, extensive validation of the code has been performed against Finnish power reactor data, and some essential parts have been updated resulting in a set of models that represent the recent experience and experimental findings. Specific WWER material properties have been incorporated in the code; among these are the cladding creep and irradiation growth. The sources of experimental WWER data behind the validation of ENIGMA are listed in Table 1.

2. Irradiation and Pool Side Measurements

The six lead assemblies were loaded into the Loviisa 1 reactor in 1998. The highest rod average burnups are now above 40 MWd/kgU. Two of the assemblies have been inspected with the pool side equipment ATULA; one (for brevity denoted by FA33) after the first irradiation cycle and another (FA32) after the second cycle. The average burnups were then ca.11 and 26 MWd/kgU,

Table 1. WWER-data being used by VTT for ENIGMA validation

Source	In-pile data					PIE data				Remarks
	Reactor	Temp	dL	pre	pool	FGR	Dim	PCI	misc	
Loviisa pre-char. rods	LO2				X	^{85}Kr	prof dL	prof gap	gam	40-50 MWd/kgU
SOFIT-1.1	MR	TC	X			punc	prof	cera	gam	
Halden IFA-503.1	HBW R	ETM	X	X		punc	prof	cera		PIE to be done irra d. underway
Halden IFA-503.2	HBW R	ETM	X	X						
Kola high burnup rods	Kola					punc ^{85}Kr	prof dL	cera gap	gam	>50 MWd/kgU
Re-instrumented rods	MIR	RTC		X		punc	prof	cera	gam	>50 MWd/kgU
Advanced fuel ass.	LO1				X	^{85}Kr	prof dL	prof gap	gam	irrad. underway

Explanation:
TC – Thermocouple
RTC – Re-instrumented Thermocouple
ETM – Expansion Thermometer
pre – pressure measurement
pool – pool-side measurements
dL – length measurement
cera – ceramography
gam – gamma scanning
prof – profilometry
gap – clad squeezing

respectively. This far the results of the visual inspections, assembly dimensional changes and the profilometry data from the corner rods are available [1]. Further inspections will be carried out for the three cycle rods (FA34) in 2002, whereas one assembly will be irradiated for four cycles and inspected thereafter.

An example of the power histories encountered by the rods can be seen in Figure 1. The highest linear heat rates stay much lower than those allowed in the Loviisa reactor, and low fuel temperatures are therefore to be expected. The fast neutron flux is of the order of $7.5\text{-}9.5{\cdot}10^{17}$ n/m^2s (>1 MeV). Typical coolant temperature is 265°C and 300°C in the assembly inlet and outlet, respectively. The clading temperature of the rods rises to about 300-330°C. These rather stable reactor conditions lead to cladding creep-down, which in the middle region of the fuel rods ranges to 15-35 μm during each cycle. In terms of creep vs. burnup this equals to about 20 μm per 10 MWd/kgU.

3. Appropriate Performance Models

The corner rods subjected to examinations have an enrichment of 3.3%, slightly lower compared to the 3.6% used in the old design. The initial rod dimensions, gap size and fill gas pressure are unchanged. Compared to the old design, possible performance differences might be related to the fuel densification, swelling and fission gas release. The fuel rod processes are strongly interrelated, however, and therefore even small fuel microstructure changes may lead to different thermal-mechanical performance. Because of the intermediate rod diameter measurements (profilometry), special interest is in the calculated creep behaviour against the measured values.

3.1. The ENIGMA Creep Calculation Formalism

The mechanical treatment is isotropic and symmetric as regards to hoop stress. In the von Mises material model the multidimensional stress field is reduced to an effective (or equivalent) stress σ_f, which is then fed to a uniaxial creep correlation for the calculation of effective strain $\varepsilon_f = f\sigma_f$). The expression for the effective stress in the isotropic case is:

$$\sigma_f = \sqrt{0.5(\sigma_r - \sigma_\theta)^2 + 0.5(\sigma_\theta - \sigma_z)^2 + 0.5(\sigma_z - \sigma_r)^2} \qquad (1)$$

There are two phases of creep to account for in normal operation. Primary and secondary parts can be separated in the calculation:

$$\varepsilon^c = \varepsilon^c_p + \varepsilon^c_s \qquad (2)$$

The ENIGMA creep formalism encloses simultaneously both hardening and softening of the material. Under stress, hardening in one loading

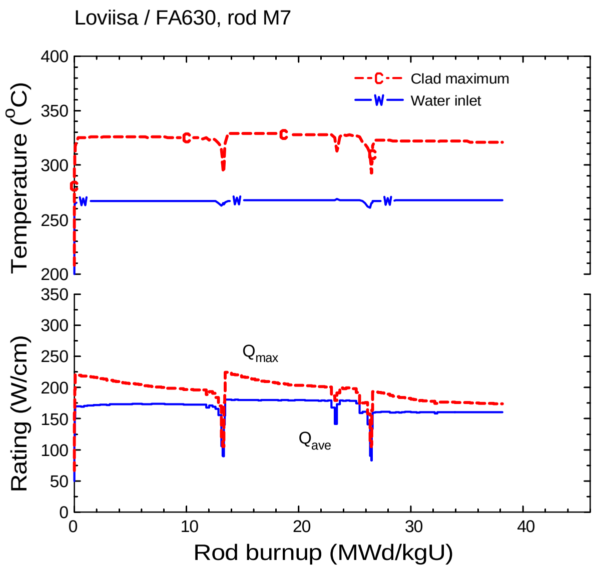

Figure 1. Example of the irradiation conditions for the advanced fuel rods

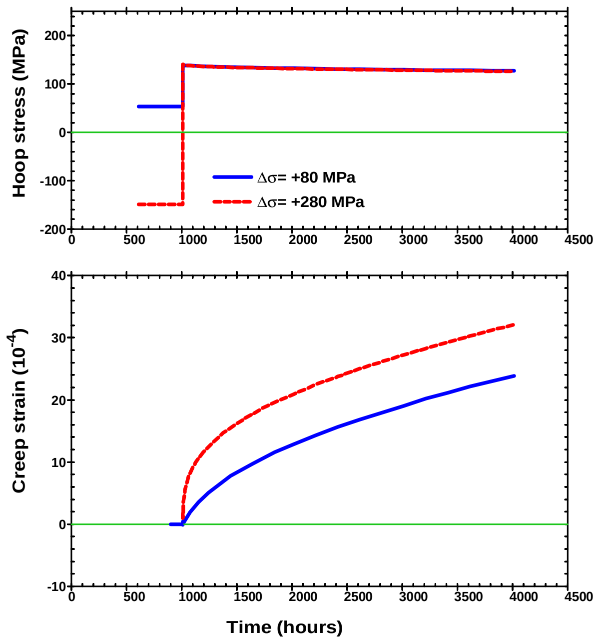

Figure 2. Effect of stress change on clad primary creep in the ENIGMA model

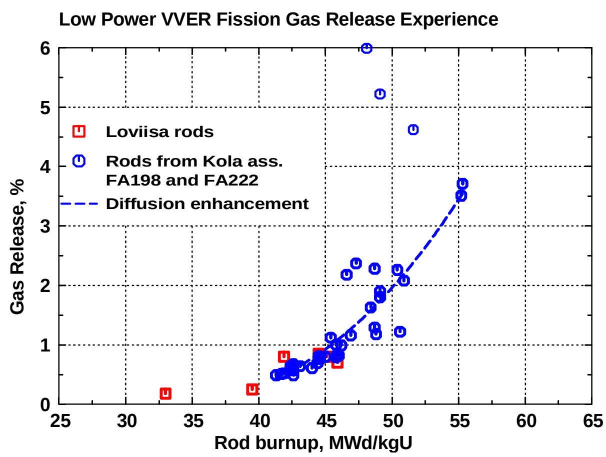

Figure 3. Puncture data show enhanced FGR above 40 MWd/kgU

direction can be accompanied by an equal amount of softening in the reversed direction. This means that increments of creep in compression decrement the hardness in tension and vice versa. The feature has implications in cases where, after a considerable creep strain, the hoop stress changes its sign, primary creep, more prominent than what the new stress level would suggest, occurs. Principal situation is the onset of hard pellet-clad contact, when pellet swelling and thermal expansion start to push the cladding outwards. Figure 2 illustrates the stress reversal effect on the primary creep.

3.1.1. The creep correlation

There is a general target of finding a feasible creep model for all cladding types (including BWR) used in Finnish reactors. Two kinds of creep models have been previously used in the ENIGMA for WWER cladding strain estimates. Both of these have been successful in predicting the rod upper part creep where the temperature is the highest. However, there has been a tendency to underestimate the creep-down in the low stress and low temperature regions like bottom parts of the rods. Recently an LWR cladding creep correlation was tuned on the basis of in-pile measurements of the Halden Reactor Project [2], and one may try to use the same correlation with slightly modified parameters for the WWER purpose, too. It describes the strain as a function of stress, temperature and fast flux in a simple way, and includes both primary and secondary components:

$$\varepsilon_p^c = A_0 \sigma_f^n \exp\left(\frac{-Q_0}{T}\right) \cdot t^m \qquad (3a)$$

$$\varepsilon_s^c = A_1 \phi \sinh(a\sigma_f) \exp\left(\frac{-Q_0}{T}\right) \cdot t \qquad (3b)$$

The variables have their usual meanings:
ε – the effective strain;
σ_f – the effective stress, [MPa];
ϕ – the fast neutron flux, [n/m^2/s];
t – time, [s];
T – the wall average temperature, [K], and
$a, A_o, A_1,$
$Q_o = 7070K, m = 0.33, n = 1.5$
are material specific or other constants.

The thermal creep part is enclosed in (3) with the assumption that neutron flux is close to the normal values for LWRs. It should be noted that equal temperature relationships are used for the primary (3a) and secondary (3b) creep rates. In the latter, the hyperbolic sine function combines the different low stress and high stress dependencies. In the low stress region, a linear relationship seems appropriate, whilst in the high stress region, an exponential dependency is more realistic.

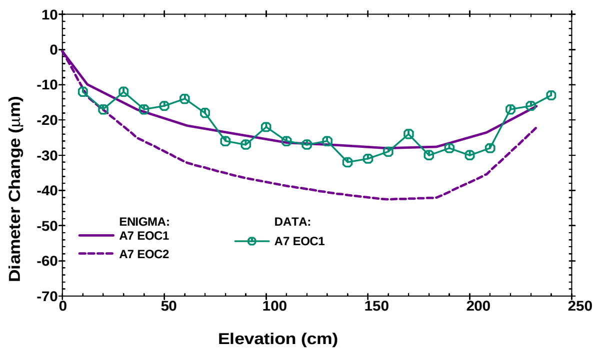

Figure 4. WWER cladding creep strain: Loviisa-1, FA 633, rod A7

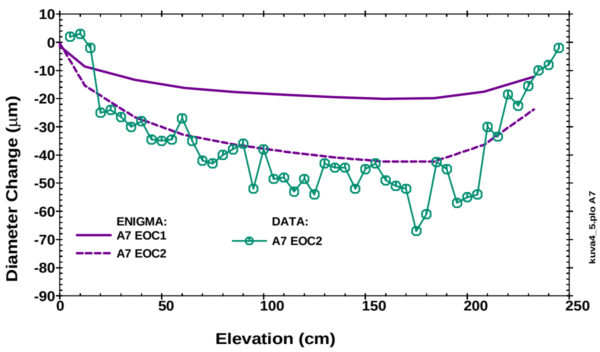

Figure 5. WWER cladding creep strain: Loviisa-1, FA 632, rod A7

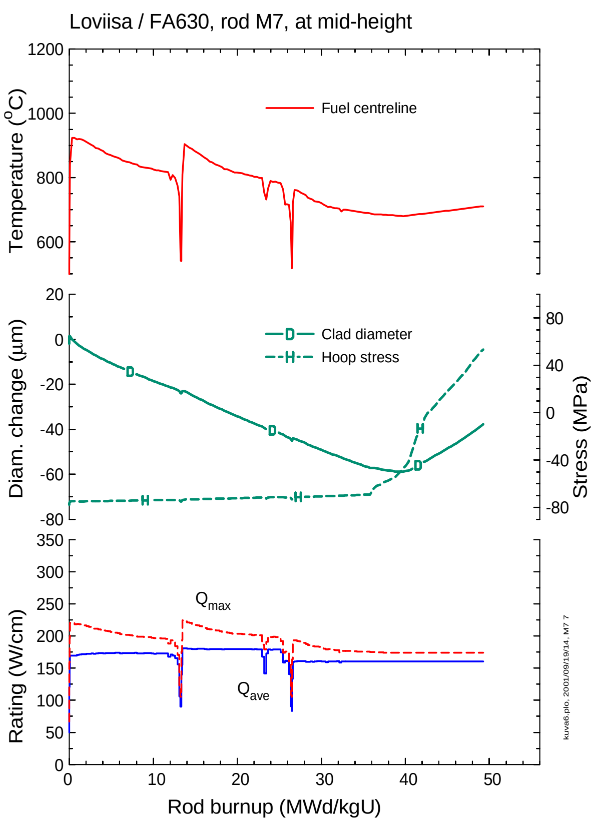

Figure 6. Prediction for selected fuel rod parameters up to 50 MWd/kgU

3.2. Fuel Densification

The old type of fuel had a porosity of 3-3.5%, and the average grain size was in the range 5-8 μm. Despite the low porosity, the fuel still densified rather much in the operation, from 1% to 1.5% of volume. Since then the pellet manufacturing process has been modified, and the new rods contain fuel, expected to be less prompt to in-pile sintering. Signs of the improved stability have been seen in the second PWR/WWER irradiation test in Halden [3].

In the calculations we use 96% TD for the initial density and 10 μm for the average grain size of the advanced fuel. With these values the model gives a densification of about 0.5% ΔV within the first weeks of operation.

3.3. The Fission Gas Release Model

The current fission gas release model is due to White [4]. For special applications the calculation of radial power and burnup distribution of the fuel has been renewed at VTT by introducing RAD99 model where heavier plutonium isotopes than ^{239}Pu are taken into account. This produces a steeper burnup gradient towards the pellet edge and thereby mimics the development of RIM by means of decreased thermal conductivity and increased fission gas generation. Enhanced fission gas release is assumed to occur after certain burnup by including a special multiplier into the Xe and Kr diffusion coefficients. This reproduces the observed FGR upswing that has been observed above 40 MWd/kgU (rod average burnup), but contains no mechanistic modelling of the phenomena. On the other hand, the temperature dependence of the gas release is preserved down to low temperatures, which fits with the current finding that not only the outer fuel part contributes to the release. Some of the fission gas release data originated from WWER fuel rods are shown in Figure 3 together with the effect of the diffusion amendment.

4. Examples of Calculations

4.1. Cladding Creep-Down, Diameter Changes

Partially recrystallised cladding material is expected to creep less than typical cold worked Zircaloy.

The formula (3) was applied in the calculations by only adjusting the factor A_1 for the Zr1%Nb material. Most of the time there is a compressive load with hoop stress of about 74 MPa (equivalent to $\sigma_f \approx 60$ MPa).

The calculated creep-down is typically 10-25 μm after the first cycle and 20-45 μm after the second cycle with correlation parameters (Equation 3) similar to those for the partially recrystallised PWR segments in [2]. This correlation improves the prediction especially at the lower section of the rod where clad temperature remains below 310°C. The comparison between the measured and calculated diameter profiles in Figures 4 to 5 shows that there is good a general correspondence. The exception is rod A1 from FA32 where the maximum diameter decrease at the end of the second cycle is over 70 μm, whilst the predicted maximum decrease is 50 μm. Local burnup there is close to 30 MWd/kgU. Small (stochastic) variation in the mechanical properties may cause the difference seen between this rod and the others.

With prolonged irradiation, the time of gap closure and stress reversal can be estimated. Figure 6 (middle graph) shows that a strong contact develops only after rod average burnup of about 35-40 MWd/kgU. Before this, the gap may be physically closed because of the relocated fuel fragments, and even bonding of the fuel and clad can exist, but there is still room for expansion within the fuel. When all free space has been consumed, cladding hoop stresses turn positive and slight ridge formation can be awaited. The third cycle data of gap closure and reversed hoop strain will be an interesting validation exercise for the model and ENIGMA.

4.2. Fission Gas Release

Relatively low temperatures are predicted for all of the rods, the peak temperature not exceeding 1000°C. In those conditions the fission gas release is of athermal origin, and less than 1%. Comparison with the threshold curve derived in the Halden Project gives a margin of several hundred degrees to fission gas release onset, Figure 6 upper graph. Towards rod burnups of 50 MWd/kgU (one assembly will be irradiated for four cycles), increased gas release is expected even though the linear powers (temperatures) would stay low. A gas release fraction between 1% and 3% is probable due to the saturation of the xenon and Krypton concentrations in certain fuel regions.

5. Conclusions

In the Loviisa-1 WWER-440 power plant, six lead assemblies of an advanced type are being irradiated. The basic modifications comprise profiling of the fuel enrichment, de-mountability of the assembly and a reduced shroud wall thickness. The currently ongoing pool side examination programme includes visual inspections, diameter and length measurements between each operation cycle, and finally the fission gas release determination from the ^{85}Kr activity of the plenum. Evaluation of the fuel rod performance and testing of WWER specific models is possible with the ENIGMA code.

The diameters of the corner rods have decreased 15-30 μm during the first cycle and up to 70 μm after the second cycle (with rod burnups of 24-30 MWd/kgU), while the calculated creep-down is typically 10-25 μm and 20-45 μm after the first and second cycle respectively. The comparison between the measured and calculated diameter profiles shows a good general equivalence. A new correlation adjusted for Zr1%Nb cladding improves the prediction especially at lower section of the rod where clad temperature stays below 310°C. The gap closure and reversed hoop strain is to be awaited during the third cycle. The calculated temperatures stay low, and therefore low fission gas release fractions are anticipated as well.

References

[1] M. Pihlatie. Operational Experience and Main Results of the Poolside Inspections up to Two Irradiation Cycles on the Advanced Fuel Assemblies in Loviisa. Proc. 4-th Int. Seminar WWER Fuel Performance, Modelling and Experimental Support, Albena, Bulgaria, 1-5 Oct., 2001, INRNE-BAS, 2002.

[2] K. Ranta-Puska. Analysing Clad Creep Behaviour in IFA-585 Experiments with the ENIGMA Code and Utilising the Results for Lift-off Estimations. OECD Halden Project Work Report HWR-677, January, 2001.

[3] B. Volkov, et al. The Thermal and Mechanical Behaviour of Modified WWER Fuel Compared with Specification Fuel in IFA-503.2. OECD Halden Proj. Work Rep. HWR-637, Dec., 2000.

[4] R. White, M. Tucker. A new Fission-Gas Release Model. J. Nucl. Mat: 118, 1-38, 1983.