Assessment of the prediction capability of the TRANSURANUS fuel performance code on the basis of power ramp tested LWR fuel rods

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Abstract

The present work is aimed at assessing the prediction capability of the TRANSURANUS code for the performance analysis of LWR fuel rods under power ramp conditions. The analysis refers to all the power ramp tested fuel rods belonging to the Studsvik PWR Super-Ramp and BWR Inter-Ramp Irradiation Projects, and is focused on some integral quantities (i.e., burn-up, fission gas release, cladding creep-down and failure due to pellet cladding interaction) through a systematic comparison between the code predictions and the experimental data. To this end, a suitable setup of the code is established on the basis of previous works. Besides, with reference to literature indications, a sensitivity study is carried out, which considers the "ITU model" for fission gas burst release and modifications in the treatment of the fuel solid swelling and the cladding stress corrosion cracking. The performed analyses allow to individuate some issues, which could be useful for the future development of the code.

Keywords: Light Water Reactors, Fuel Rod Performance, Power Ramps, Fission Gas Burst Release, Fuel Swelling, Pellet Cladding Interaction, Stress Corrosion Cracking.
1. INTRODUCTION

High efficiency and reliability of the nuclear fuel is a basic requirement for the safe and economic operation of commercial power reactors. Due primarily to the safety aspect of the fuel rod, which is the first and the second barrier (i.e., the fuel matrix and the surrounding cladding tube, respectively) to the release of radioactive fission products, the design has to ensure the integrity of the cladding under both normal and abnormal operation conditions [1, 2]. For this design purpose, qualified simulation tools are widely used [3], since they provide a quantitative insight into the relevant details of the complex in-reactor behaviour of the nuclear fuel [4]. A great effort is being spent in order to extend the prediction reliability of these codes to the fuel behaviour under onerous operating conditions, and in particular for transient and accident conditions [5].

Indicators of fuel performance are those parameters, related to the correct operation (thermal and dimensional stability) and to the integrity of the fuel rod, which must be maintained under control in order to assure the respect of the design limits. Among them:

- the amount of fission products released out of the fuel pellets and filling the rod free volume (pellet-cladding gap + upper plenum), which acts on the thermal conductance of the gap as well as on the rod internal pressure. This phenomenon (Fission Gas Release – FGR) is enhanced during rapid power increases due to the rise of the fuel temperature and to the pellets micro-cracking that means new paths for the release of fission gas (burst release). A literature survey of the burst release effect is given in Ref. [6], where the "ITU model" – developed at the Institute for Transuranium Elements (ITU) and adopted in this work – is also described;
- the dimensional changes occurring in both the fuel and the cladding, leading to a progressive closure of the gap until an intimate contact is reached after a certain burn-up [4];
- the cladding rupture due to chemically assisted crack propagation (Stress Corrosion Cracking – SCC) under Pellet Cladding Interaction (PCI) [1, 4, 7, 8].

In the framework of the IAEA Coordinated Research Project FUMEX-III [5], the present work is aimed at assessing the prediction capability of the TRANSURANUS fuel performance code [9] in simulating burn-up, FGR, cladding creep-down and PCI failures of power ramped LWR fuel rods, on the basis of a systematic comparison with experimental data from all the irradiation tests of the Studsvik BWR Inter-Ramp [10] and PWR Super-Ramp [11] Projects. Moreover, modifications in the treatment of the fuel solid swelling and in the threshold stress intensity factor for SCC (K_{ISCC}), adopted in the PCI cladding failure criterion, have been attempted in order to identify some issues for the future improvement of the code.

2. IRRADIATION EXPERIMENTS

The Inter-Ramp and Super-Ramp Projects were carried out in order to study the power ramp performance of LWR fuel at moderate burn-up. During these Experiments, fuel rods featured by different design parameters (see Tables 1 and 2 for details) were subjected to power ramps in the Studsvik reactor R2 after base irradiation. An example of base irradiation and power ramp is shown in Figure 1, which refers to the Super-Ramp PK1-2 rod.

Thanks to an extensive experimental analysis program and to detailed records of fuel rods power histories [10, 11], a wide dataset is available from these two Projects, which has been included in the OECD/NEA-IAEA International Fuel Performance Experiments (IFPE) database [12] and constitutes a useful validation framework. It comprehends burn-up, FGR and dimensional changes measurements as well as cladding failure data.

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1 Inter-Ramp rods were base-irradiated to linear powers in the range 20-40 kW/m and raised from 25-40 kW/m (Conditioning Power Level - CPL) to 40-65 kW/m (Ramp Terminal Level - RTL) during the ramp, with power rates of about 65-90 W/(m·s) [10]; Super-Ramp rods experienced 9-27 kW/m during the base irradiation and were ramped from 25 kW/m (CPL) to 38-50 kW/m (RTL) with rates of 100-180 W/(m·s), indicatively [11].
Through the Inter-Ramp Project 20 BWR fuel rods were tested at burn-up of 8-20 MWd/kgU, while during the PWR Super-Ramp Project 28 rods were irradiated in a burn-up range comprised between 28 and 45 MWd/kgU. Only 26 out of 28 Super-Ramp rods have been considered in the analyses of the present work, excluding the PW3-2 and PW3-3 rods due to lack in consistency found in their power histories [13].

3. MODELLING

During the life-time in reactor, the fuel rod is subjected to intricate physical, mechanical and chemical phenomena, involving both the fuel (relocation, densification, solid and gaseous swelling, creep, thermal expansion) and the cladding (creep, irradiation growth, thermal expansion, corrosion), often mutually interdependent [1, 2, 4]. Consequently, in fuel rod performance analyses several mutually interacting models are involved, and integral codes are necessary for a global description of such a complex and coupled system [3]. In the present work, simulations have been performed by means of the fuel performance code TRANSURANUS (version v1m1j06, currently available at Politecnico di Milano), which is a computer program for the thermal and mechanical analysis of various fuel rod types [9]. Based on the idealization of the rod geometry (assumed to be axial-symmetric) and the superposition of a one-dimensional radial and axial description (the so-called 1½D approach), its mechanical-mathematical framework allows to analyze, at reasonable computer cost, the whole fuel rod during a complicated, long power history (unlike 2D or 3D codes used for the analysis of local effects). Moreover, TRANSURANUS is featured by a flexible structure into which various physical models can easily be incorporated. A review of the present status of the code as well as of the activities for its development and verification is given in [14].

The calculations described in this paper have been carried out coherently with the power histories and coolant conditions from beginning-of-life to the end of ramp test, using the manufacturing specifications and pre-irradiation characterization data of the analysed rods [10, 11]. Various models are available in the code for the treatment of the relevant phenomena occurring in the fuel rod: a suitable setup ("reference setting" – see Table 3) for the test conditions of interest has been established, on the basis of previous code assessments [13, 15-17], taking into account the different specificity of BWR and PWR type rods.

A sensitivity study has been performed starting from the "reference setting" and testing some modifications introduced on purpose in the version v1m1j06 of the code [18]: in particular, the "ITU model" for fission gas burst release (derived from Koo et al. [19]) has been implemented, together with modifications of the fuel solid swelling model and of the $K_{SCC}$ parameter adopted in the cladding failure criterion. All the details of the "ITU model" can be found in [6], while the other two modifications attempted in the present work are briefly described here below.

3.1 Fuel swelling

The standard TRANSURANUS option (used in the "reference setting") for UO$_2$ swelling treatment is the MATPRO LWR model [20], which considers both the contributions due to gaseous fission products (gaseous swelling) and to solid fission products (solid swelling).

According to this approach, the solid swelling is simply proportional to the burn-up [18]:

$$\Delta V_{\text{solid}} = b \cdot \Delta bu$$  \hspace{1cm} (1)$$

where $\Delta V_{\text{solid}}$ represents the increment of fractional increase in volume due to solid fission products in a time step with a burn-up increment $\Delta bu$, and $b$ is an empirical coefficient depending on the density of heavy metals ($^{235}\text{U} + ^{238}\text{U}$) in the fuel. By considering typical UO$_2$ density and heavy metal fraction, a solid swelling rate of about 0.06%/ (MWd/kgU) is predicted by this model. According to recent indications [3], a modification has been attempted in the present work by
multiplying the coefficient \( b \) by a factor of 1.5. In this way (“modified solid swelling” setting), the solid swelling rate is increased to about 0.09%/\((\text{MWd/kgU})\), that compares well with the theoretically predicted solid swelling rate of 0.08 to 0.1%/\((\text{MWd/kgU})\) suggested in [21]. The impact of such modification will be discussed in Section 4.

As concerns the gaseous swelling, TRANSURANUS calculates the fuel volume increment due to fission gases through a temperature and burn-up dependent simple formula (details can be found in [18]), but neglects this contribution when contact pressure between pellets and cladding arises. Indeed, quantifying and taking into account the inhibition of the phenomenon due to the hydrostatic stress field in the fuel under PCI conditions is still an open issue during power ramps [8, 17, 22, 23]. It is worth noting that gaseous swelling turns into fuel porosity increment and thermal conductivity degradation (hence affecting temperature-dependent phenomena such as the FGR) and directly influences the cladding failure propensity (together with the solid swelling), once contact with the fuel pellets occurs.

### 3.2 PCI failure criterion

The PCI failure criterion is established in TRANSURANUS by the stress corrosion cracking SPAKOR model [18, 24], which assumes that inter-granular phase of crack growth is due to the chemical environment and is independent of the mechanical stress state, while trans-granular fracture phase is described by means of Linear Elastic Fracture Mechanics. The initial propagation of the crack starts only if critical conditions on local burn-up, cladding temperature, strain rate and hoop stress are satisfied. During subcritical propagation (inter-granular phase), the crack growth rate is assumed to decrease exponentially with the crack length, until a critical length is reached that depends on the cladding hoop stress and on the threshold stress intensity factor for SCC \( (K_{\text{ISCC}}) \). At this point, the trans-granular fracture phase starts and proceeds up to failure, that TRANSURANUS considers to occur if the crack length exceeds the cladding wall thickness or the length at which the net section stress exceeds the ultimate strength \( (\sigma_u) \). Major details of the PCI failure criterion of the code can be found in [18, 24].

It is worth noting that the SPAKOR model applies to both Zircaloy-2 (Zy-2) and Zircaloy-4 (Zy-4) cladding materials, indifferently, and adopts the following simple correlation to evaluate \( K_{\text{ISCC}} \):

\[
K_{\text{ISCC}}[\text{MPa} \cdot \text{m}^{1/2}] = 24.5 - 0.0325 \cdot \sigma_{ys}
\]  

where \( \sigma_{ys} \) is the cladding yield strength (expressed in MPa) at room temperature. As concerns the simulations of the present work, in the "reference setting" the \( \sigma_{ys} \) and \( \sigma_u \) data of the specific rod types have been employed where available (i.e., for Zy-4 claddings) [11], obtaining \( K_{\text{ISCC}} \) values within the range 4.3÷7.1 MPa·m\(^{1/2}\). Nevertheless, lower values can be found in literature for Zy-2 and Zy-4 [4, 8, 25-31]; in the light of these indications, and to take into account the experimentally ascertained temperature dependence of this parameter, an alternative correlation [29] has been implemented (“modified \( K_{\text{ISCC}} \)” setting):

\[
K_{\text{ISCC}}[\text{MPa} \cdot \text{m}^{1/2}] = 2.2742 \cdot 10^2 - 6.6881 \cdot 10^{-1} \cdot T_{\text{tip}} + 4.9617 \cdot 10^{-4} \cdot T_{\text{tip}}^2
\]  

where \( T_{\text{tip}} \) is the local temperature at the crack tip expressed in Kelvin. It is well known that \( K_{\text{ISCC}} \) in Zircaloys depends on several other factors than temperature, such as the material composition (Zy-2/Zy-4), texture, heat treatment, fast neutron dose and iodine concentration; however, these influences are neglected in the above approach. Thanks to the employment of correlation (3), \( K_{\text{ISCC}} \) values in the range 2.4÷5.6 MPa·m\(^{1/2}\), obtained for the Super-Ramp and Inter-Ramp test conditions, compare well with literature data [4, 8, 25-31]. This modification is expected to alter the crack propagation time evolution and the cladding failure predictions, as discussed in the next Section.
4. RESULTS

The results obtained by means of the "reference setting" in terms of fuel burn-up, FGR, cladding creep-down and PCI failure are hereinafter presented and compared with the experimental data. The assessment of the code against these integral quantities is also performed through a sensitivity study, by evaluating the impact of the above discussed alternative options ("ITU model" for burst release, "modified solid swelling", "modified KISC")", where significant. Some other modelling issues, which may require further investigation in the future, are individuated as well.

4.1 Fuel burn-up

Within the Inter-Ramp and Super-Ramp Projects, burn-up measurements have been performed on 10 rods (6 and 4, respectively). Pellets have been analysed by means of the Nd-148 method [10, 11]. In Table 4, the comparison between the computed and the measured burn-up at the end-of-life is reported. All predictions (obtained by means of the "reference setting") lie in the acceptability band of ±10%. Slight discrepancies can be due to the uncertainty associated to the chemical analyses based on Nd isotopes and the uncertainty associated to the recorded power histories.

4.2 Fission gas release

The ratio of the computed to the measured FGR values at the end-of-life is reported in Figure 2 for 26 Super-Ramp and Inter-Ramp rods, namely all those ones provided by FGR experimental data. Both the "reference setting" and the "ITU model" calculations are presented: the results are in agreement with those obtained in [6, 32]. Computations indicate that a systematic under-estimation occurs when the "reference setting" is used (such a conclusion was also drawn by analysing other power ramp tests [6, 14]); however, the average ratio of the computed to the measured FGR values (equal to about 0.6) lies within the acceptability band (0.5±2, namely a factor of 2) commonly considered as satisfactory for integral irradiation experiments, and 65% of rods fall inside this band. Besides, it can be noticed that the results are less dispersed and more accurate for high FGR (measured values above 10%): it is the typical situation of the Super-Ramp rods, where the fission gases are mainly released during the power ramp [17].

Taking into account the burst release effect, by means of the "ITU model", allows to overcome the systematic under-prediction of the code, better evaluating the total (i.e., at the end-of-life) FGR: the average computed-to-measured ratio is now about 1.3 (thus closer to the centre of the acceptability band), and 85% of rods lie within the band. Again, FGR is more properly predicted in the zone where measured values are above 10% (the average computed-to-measured ratio is about 1.2 in this zone, while 100% of rods fall within the band): it is an indication of the efficiency of the "ITU model", since burst release primarily occurs during the ramp, that gives the dominant contribution to the total FGR in this zone. However, further investigation could be useful to better interpret the discrepancies for low (<10%) measured FGR: an over-estimation (average factor 1.5) is noticed when the "ITU model" is employed in this zone (typically for Inter-Ramp cases, where in general the base irradiation contributes for the most part to the total FGR [17]). For a better assessment, analyses based on a larger number of cases are advisable, and extending the burn-up range is required to verify the burst release "ITU model" at high burn-up as well [6]. Finally, for power ramp conditions, a more adequate treatment of gaseous swelling, which could affect the FGR through degradation of fuel thermal conductivity and tends to be inhibited by the high hydrostatic stress resulting from PCI, seems needed [17]. On this subject, a collaboration between Politecnico di Milano and ITU has been established to develop a stress-dependent swelling approach.
4.3 Cladding creep-down

Estimation of the dimensional changes of the cladding is currently recognized as a key issue for the modelling of the fuel rod behaviour [5]. In this work, reference has been made to the creep contribution to the diameter change (creep-down), experimentally available for 12 Super-Ramp and all the 20 Inter-Ramp rods from measurements performed at the end of base irradiation [10, 11]. The TRANSURANUS code adopts a simple model for the treatment of LWR cladding creep [18, 33], which does not consider the primary creep stage. However, it is worth noting that creep-down is an integral quantity, whose correct evaluation not only relies on the constitutive correlation, but also depends on the overall working conditions (stress state, eventual imposed strain, temperature, neutron fluence).

As a result of the simulations performed in this work, a systematic under-estimation of the cladding creep-down has been found (Figure 3), namely about 40% on average for both the Super-Ramp and Inter-Ramp Experiments ("reference setting"). It has been ascertained that several rods experience gap closure already during the base irradiation, and that the under-estimation is independent of this eventuality; hence, the noticed discrepancy is not strictly related to the contact pressure, which is mainly driven by the fuel dimensional changes. An improvement of the creep modelling, based on literature indications [20, 34, 35] and able to take into account the primary creep contribution (that could overcome the under-estimation), could be useful in this sense. However, it is premature to draw definitive conclusions, in consideration of the integral character of creep-down, of the limited number of analysed experimental data, and of the measurement inaccuracy by which some of these data are affected [11].

The effect of the "modified solid swelling" model (see Par. 3.1) has been also evaluated: in this case, the values of creep-down calculated by TRANSURANUS are lower with respect to the "reference setting" calculations (Figure 3). As expected, such behaviour is found for those rods experiencing gap closure during the base irradiation: in fact, a higher fuel diameter due to a larger amount of incompressible solid swelling (that corresponds to the "modified solid swelling" setting) leads to a higher cladding strain imposed by the fuel and then to a different creep deformation.

4.4 PCI cladding failure

In the context of fuel utilization at extended burn-up – which is the tendency of current R&D programs [5, 8] – there are many incentives to progress on PCI modelling and to adopt fuel performance codes in order to allow reliable predictions of rod damaging. Another important suggestion for the simulation of PCI arises from the evaluation of the failure threshold curves, which are established by means of power ramp tests and represent the power limits against rod damaging [4, 7, 36]). These limits are of great importance for fuel vendors and nuclear power plant operators for the establishment of the operating constraints [8]. A preliminary evaluation of the failure threshold curves (in terms of the maximum power and the maximum power change with respect to burn-up) by means of TRANSURANUS is described in Ref. [15], which is based on a limited number of rods from the Super-Ramp Project.

In the present work, the TRANSURANUS failure predictions for 26 Super-Ramp rods and 20 Inter-Ramp rods have been carried out and compared with the experimental data. This analysis has pointed out that the code gives correct failure predictions in the 50% of cases for both the Experiments ("reference setting"), thus suggesting that some improvements in the PCI modelling are needed to reliably evaluate the failure threshold curves.

It has been verified – through the "modified $K_{ISC}$" calculations – that the threshold stress intensity factor $K_{ISC}$ plays a relevant role in the failure predictions. To give some details, 7 more failures are predicted in the "modified $K_{ISC}$" case with respect to the "reference setting" simulations, and the percentage of correct predictions is about 48%. In addition to this, it is well-known that the cladding failure propensity is an integral parameter, mainly determined by the amount of tensile stress due to the occurrence of contact pressure and therefore influenced by the dimensional changes of both the
fuel and the cladding. For example, it has been ascertained (by the "modified solid swelling" simulations) that the effect of solid swelling is significant, due to the increased contact pressure because of the higher fuel deformation. In particular, 6 more failures are predicted in the "modified solid swelling" case with respect to the "reference setting" simulations, and the percentage of correct predictions is 41%.

The gaseous swelling (neglected by TRANSURANUS in contact conditions – see Par. 3.1) is actually expected to give a non-negligible contribution to the fuel deformation and contact pressure also under PCI conditions. Moreover, the cladding creep should influence the failure propensity of fuel rods through two opposite effects [4]: on the one hand, creep-down anticipates gap closure and PCI occurrence; on the other, creep relaxation of stress in the cladding is in favour of rod integrity.

It is worthwhile to mention that a simplified "no-slip condition" [18] for the axial Pellet Cladding Mechanical Interaction (PCMI) has been adopted in this work (Table 3), while a more physical analysis would require the axial friction forces to be accounted for.

Finally, it has been noticed that the conservative character of the failure predictions seems to depend on the cladding composition (Zy-2/Zy-4). As a matter of fact, the code misses to predict cladding rupture in only 2 out of 26 Super-Ramp (Zy-4) cases, while 9 non-conservative predictions have been found among the 20 Inter-Ramp (Zy-2) cases ("reference setting" simulations).

All things considered, the results obtained in the present work suggest that different values of $K_{ISCC}$ should be adopted for Zy-2/Zy-4; anyway, in order to improve the prediction capability of TRANSURANUS in view of the evaluation of the failure threshold curves, attention should be paid to the failure criterion and to the axial PCMI as well as to the cladding creep and the fuel (gaseous + solid) swelling behaviour.

5. CONCLUSIONS

The prediction capability of the TRANSURANUS code for evaluating the fuel rod (LWR type) behaviour under power ramp conditions has been assessed on the basis of experimental data available from the BWR Inter-Ramp and PWR Super-Ramp databases (46 irradiation histories, in the burn-up range of 8-45 MWd/kgU). The analyses, which have been focused on some integral quantities, have firstly pointed out that burn-up predictions are accurate. As regards end-of-life FGR computations, an under-estimation with respect to the experimental data has been noticed, in particular for FGR lower than 10%. On the whole, the average ratio of computed to measured FGR lies within the commonly recognized acceptability band. The sensitivity study has pointed out that the "ITU model" for burst release leads to a significant improvement, both in terms of average deviation and number of rods falling within the acceptability band, although further investigation would be useful in the lower FGR range as well as at high burn-up. As concerns the cladding creep-down at the end of base irradiation, a systematic under-prediction has been observed, also for those rods whose gap remains open: this could be due to the fact that the primary creep stage is neglected in the simulations, although further studies are required before drawing such a conclusion. Percentages of correct failure predictions are around 50% at the current state of things. For the purpose of evaluating the failure threshold curves, some refinements of the cladding failure criterion are advisable, together with further investigation on the fuel swelling under power ramp conditions.

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REFERENCES


Table 1 - Main features of the BWR Inter-Ramp rods

<table>
<thead>
<tr>
<th>Group</th>
<th>Pellet type</th>
<th>Cladding heat treatment*</th>
<th>Diametral gap size [µm]</th>
<th>UO₂ density [%TD]**</th>
<th>Average burn-up [MWd/kgU]</th>
<th>Enrichment [wt% ²³⁵U]</th>
<th>Av. grain size [µm]</th>
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<tr>
<td>LR</td>
<td>Standard RX</td>
<td>150</td>
<td>95</td>
<td>8.5-10.3</td>
<td>2.82</td>
<td>8.3</td>
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<tr>
<td>LS</td>
<td>Standard SR</td>
<td>150</td>
<td>95</td>
<td>8.2-10.4</td>
<td>2.82</td>
<td>8.3</td>
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<td>TR</td>
<td>Standard RX</td>
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<td>10</td>
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<td>Standard RX</td>
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<td>7.9</td>
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<td>Standard RX</td>
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<td>95</td>
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<td>3.50</td>
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<tr>
<td>HS</td>
<td>Standard SR</td>
<td>150</td>
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<td>16.6-19.3</td>
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<td>3.50</td>
<td>8.4</td>
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*RX = recrystallised Zircaloy-2, SR = cold-worked + stress relieved Zircaloy-2; **design specifications.

Table 2 - Main features of the PWR Super-Ramp rods

<table>
<thead>
<tr>
<th>Group</th>
<th>Pellet type</th>
<th>Cladding heat treatment°</th>
<th>Diametral gap size [µm]</th>
<th>UO₂ density [%TD]°°</th>
<th>Average burn-up [MWd/kgU]</th>
<th>Enrichment [wt% ²³⁵U]</th>
<th>Av. grain size [µm]</th>
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<tr>
<td>PK1</td>
<td>Standard SR</td>
<td>191-200</td>
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<td>PK2</td>
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<td>94°°</td>
<td>41–45</td>
<td>3.21</td>
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<td>PK4</td>
<td>Standard + Gd₂O₃ Large grain SR</td>
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<td>33–34</td>
<td>3.19</td>
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<td>PK6</td>
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<td>34–37</td>
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°SR = cold-worked + stress relieved Zircaloy-4; °°calculated from pellet density measurements; °°°calculated from measurements of pellet weight and dimension.

Table 3 - Main choices of the "reference setting"

<table>
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<tr>
<th>TU input variable</th>
<th>Value</th>
<th>Meaning*</th>
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<td>MODCLAD (1÷20)</td>
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<td>Standard LWR settings for cladding material</td>
</tr>
<tr>
<td>MODFUEL (1÷5; 7÷20)</td>
<td>20</td>
<td>Standard LWR settings for UO₂ fuel properties</td>
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<tr>
<td>MODFUEL (6)</td>
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<td>Fuel thermal conductivity according to Harding and Martin correlation for UO₂</td>
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<tr>
<td>IDENSI</td>
<td>2</td>
<td>Fuel densification is calculated by a simplified empirical model##</td>
</tr>
<tr>
<td>IRELOC</td>
<td>8</td>
<td>Fuel relocation is calculated by the modified FRAPCON-3 model</td>
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<tr>
<td>IXMODE</td>
<td>0</td>
<td>A no-slip condition is assumed for axial PCMI</td>
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<tr>
<td>IZENKA</td>
<td>1</td>
<td>Central void formation is taken into account</td>
</tr>
<tr>
<td>IGRBDM</td>
<td>1</td>
<td>Simple grain boundary fission gas behaviour model</td>
</tr>
</tbody>
</table>

*For all the details, see the TRANSURANUS Handbook [18]; ##the requested input parameters (denpor and denbup) are the same established in [17].
Table 4 - Comparison between measured and calculated burn-up

<table>
<thead>
<tr>
<th>Project</th>
<th>Rod</th>
<th>Measured [MWd/kgU]</th>
<th>Calculated [MWd/kgU]</th>
<th>Error [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inter-Ramp</td>
<td>LR3</td>
<td>9.6</td>
<td>10.3</td>
<td>7.3</td>
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<td></td>
<td>TR1</td>
<td>11.2</td>
<td>11.2</td>
<td>0.0</td>
</tr>
<tr>
<td></td>
<td>LS3</td>
<td>8.5</td>
<td>9.2</td>
<td>8.2</td>
</tr>
<tr>
<td></td>
<td>HR2</td>
<td>18</td>
<td>18.3</td>
<td>1.7</td>
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<tr>
<td></td>
<td>HR5</td>
<td>21</td>
<td>21.4</td>
<td>1.9</td>
</tr>
<tr>
<td></td>
<td>HS2</td>
<td>17.7</td>
<td>18.9</td>
<td>6.8</td>
</tr>
<tr>
<td>Super-Ramp</td>
<td>PK2-2</td>
<td>46.4</td>
<td>43.1</td>
<td>-7.1</td>
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<td></td>
<td>PK4-3</td>
<td>36.3</td>
<td>33.3</td>
<td>-8.3</td>
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<td>PW5-3</td>
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<td>40.8</td>
<td>-3.3</td>
</tr>
</tbody>
</table>

Figure 1. Input power history in axial peak position during a) base irradiation and b) power ramp for the Super-Ramp PK1-2 rod.

Figure 2. Comparison between FGR data, measured at the end-of-life, and TRANSURANUS (v1m1j06 version) calculated values.
Figure 3. Comparison between cladding diametral creep-down data, measured at the end of base irradiation, and TRANSURANUS (v1m1j06 version) calculated values.