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Nuclear Fuel Modelling During Power Ramp

Federico Cantini, Martina Adorni, Francesco D'Auria
GRNSPG, San Piero a Grado Nuclear Research Group, University of Pisa
Via Livornese 1291, San Piero a Grado (PI), Italy
f.cantini@ ing.unipi.it, m.adorni@ing.unipi.it, f.dauria@ing.unipi.it

ABSTRACT

Fuel rods operating for several years in a LWR can experience fuel-cladding gap closure as a result of the phenomena due to temperature and irradiation. Local power increase induces circumferential stresses in the cladding as a result of the different expansion in the cladding and the pellet. In presence of corrosive fission products (i.e. Iodine) and beyond specific stress threshold and level of burnup, cracks may grow-up from the internal to the external cladding surface, causing fuel rod failure. The phenomenon, known as pellet cladding interaction-stress corrosion cracking PCI/SCC, or PCI, has been identified as a problem since the 70's.

The PWR Super-Ramp experiment (part of OECD/NEA "International Fuel Performance Experiments (IFPE) database") twenty eight fuel rods behaviour has been simulated using TRANSURANUS code version "v1m1j11". Two sets ("Reference" and "Improved") of suitable input decks modelling the fuel rods, based on the available literature are used to run the simulations. Focus is given to the main phenomena which are involved or may influence the cladding failure. Systematic comparison of the code results with the experimental data are performed for the parameters relevant for the PCI phenomenon. Sensitivity calculations on fission gas release models implemented in TRANSURANUS code are also performed in order to address the impact on the results.

The results show the ability of TRANSURANUS version "v1m1j11" in conservatively predicting the rods failure due to PCI in PWR fuel and Zircaloy-4 cladding. Increased availability of experimental data would help to perform a deeper analysis.

1 INTRODUCTION

Stress corrosion cracking (SCC) can induce Pellet-cladding interaction (PCI) failures [1][2], which might occur in presence of local power increase. The relevance of PCI in nuclear technology is connected with the prevention of fuel failures due to SCC, involving the loss of integrity of the first and second barriers during normal, off normal and accident conditions.

TRANSURANUS [3][4][5] is a computer program for the thermal and mechanical analysis of fuel rods in nuclear reactors .The TRANSURANUS code consists of a clearly defined mechanical–mathematical framework into which physical models can easily be incorporated. The mechanical–mathematical concept consists of a superposition of a one-dimensional radial and axial description (the so called quasi two-dimensional or 1½-D model). The code was specifically designed for the analysis of a whole rod.

The aim of the study is to verify the ability of TRANSURANUS code version v1m1j11 in simulating the PCI phenomenon and predicting the cladding failure.

The experimental data used for the TRANSURANUS evaluation are part of the Studvisk PWR Super-ramp project. It address the behavior of 28 PWR rods subjected to power ramp at burnup range between 28 and 45 MWd/kgU.

2 PWR SUPER-RAMP

In the PWR Super-Ramp project, 28 fuel rods were ramp tested in order to contribute to the understanding of the PCI (Pellet Cladding Interaction) failure mechanism under power ramp condition[1]. Fuel rods were provided by two manufacturers: 19 by KraftWerk Union AG/Combustion Engineering (KWU/CE) and 9 by Westinghouse (W).

The KWU fuel rods (length about 300mm) were irradiated in the Obrigheim (Germany) power reactor, having a 2655mm core active length. The rods are divided in 4 groups. In Table 1 the main properties of the groups are summarized.

Table 1 KraftWerk Union AG/Combustion Engineering (KWU/CE) rods groups.

Group name	**Number of rods**	**Rod type**	**Nominal burnup (MWd/KgU)**
PK1	5	Standard rods	35
PK2	5	Standard rods	44
PK4	4	Standard rods plus Gd2O3 (4%)	33
PK6	5	Remedy rods. Large grain	36

The W rods (length about 980mm) in the Mol (Belgium) BR-3 reactor, having a 1000mm active core length. The rods are divided in 2 groups. In Table 2 the main properties of the groups are summarized.

Table 2 Westinghouse (W) rods groups.

Group name	**Number of rods**	**Rod type**	**Nominal burnup (MWd/KgU)**
PW3	5	Standard rods	30
PW5	4	Remedy rods, annular pellets	32

All the 28 rods were power ramped at the Studsvik R2 research reactor using suitable experimental rig to simulate PWR conditions. The power ramp test include the following steps:

- 24 hours long preconditioning phase at 25KW/m;
- Power ramp at a constant rate ranging between 150 and 600 KW/(mh) until the power reaches the preselected ramp terminal level (RTL);
- Holding phase at RTL held for about 12 hours or until the rod failure.

Ramping data are reported in Table 3.

Table 3 Rods ramping data.

Rod group	**Rod label**	**Conditioning terminal level (W/m)**	**Conditioning holding time (h)**	**Ramp terminal level (W/m)**	**Ramp rate (W/mh)**	**Hold time at RTL (minutes)**
PK1	PK1/1	25	24	41.5	540	720
	PK1/2	25	24	44	480	720
	PK1/3	25	24	47.5	510	720
	PK1/4	25	24	47.5	570	720
	PK1/S	25	24	42	360	720
PK2	PK2/1	25	24	41	510	720
	PK2/2	25	24	46	570	720
	PK3/3	25	24	49	510	720
	PK2/4	25	24	44	510	1
	PK2/S	25	24	44	510	720
PK4	PK4/1	25	24	39	480	720
	PK4/2	25	24	44.5	510	720

	PK4/3	25	24	50.5	660	720
	PK4/S	25	26	43	510	720
PK6	PK6/1	25	24	45	540	55
	PK6/2	25	24	40	540	720
	PK6/3	25	24	43	540	720
	PK6/4	25	24	44	600	60
	PK6/S	25	24	41	600	720
PW3	PW3/1	25	24	40	600	22
	PW3/4	25	24	37.7	540	12
	PW3/S	25	24	40.5	600	17
PW5	PW5/1	25	24	42.7	540	118
	PW5/2	25	23	40.3	540	26
	PW5/3	25	24	38.2	540	38
	PW5/4	25	24	38	510	72

The rods failed are 9 out of 28: PK6/1, PK6/4 and all the PW rods.

3 TRANSURANUS MODELS

3.1 Input decks

TRANSURANUS models developed in [8] have been used for the validation of version v1m1j11 of the code. The models were developed according to the original experimental data[6] for the 19 KWU rods and for the 9 W rods. Two sets of input decks have been developed in [7] and [8]: "Reference" and "improved".

The models selected in the "Reference" set are generally the ones standard for the transient to be simulated. Only the active part of the fuel is accounted for the simulation. For the reference calculations, the nominal geometrical values were used when available, the measured values are considered when nominal values are not specified (gas plenum length). The input deck of each rod within KWU group differs from the others in:

- <u>boundary conditions:</u> burnup, linear heat rate, ramp terminal level, cladding temperature histories;
- <u>geometry</u>: pellet diameter, cladding inner and outer diameter, gas plenum length;
- <u>physical proprieties</u>: enrichment, UO_2 grain size (PK6 rods are of large grain size), gadolinium content (PK4 rods contain Gadolinium), porosity.

The input deck of each rod within W group differs from the others in:

- <u>boundary conditions:</u> burnup, linear heat rate, ramp terminal level, ramp rate, clad temperature histories;
- <u>geometry:</u> pellet diameter, cladding inner and outer diameter, gas plenum length, kind of pellet standard (PW3 are standard while PW5 are annular);
- <u>physical proprieties:</u> enrichment, UO_2 grain size, porosity.

The KWU rods differ from W rods principally in:

- base irradiation performed into two different reactors.

fuel rods height: W rods are about three times longer than KWU rods.

The main properties of the rods are reported in Table 4.

Table 4 Main properties of fuel rods

Group label	Rod Label	Gas plenum length (mm)	Clad outer diam. (mm)	Clad Inner diam. (mm)	Pellet outer diam. (mm)	Pellet inner diam. (mm)	Clad grain size (µm)	UO2 grain size (µm)	Gd_2O_3 content (w%)	U235 enrich. (w%)
PK1	PK1/1	32.0	10.76	9.31	9.110	0	10.5	6	0	3.2
	PK1/2	32.2								
	PK1/3	32.4								
	PK1/4	32.3								
	PK1/S	32.1								
PK2	PK2/1	32.6	10.75	9.28	9.138	0	9.5	5.5	0	3.21
	PK2/2	33.0								
	PK2/3	32.5								
	PK2/4	32.6								
	PK2/S	32.8								
PK4	PK4/1	32.4	10.77	9.28	9.113	0	9.5	5.5	4.09 +/- 0.05	3.19
	PK4/2	32.8								
	PK4/3	32.4								
	PK4/S	32.8								
PK6	PK6/1	33.1	10.74	9.29	9.144	0	12.0	22	0	2.985
	PK6/2	32.5								
	PK6/3	33.0								
	PK6/4	33.0								
	PK6/S	32.9								
PW3	PW3/1	117.78	9.51	8.35	8.19	0		10.5	0	8.26
	PW3/2	119.32								
	PW3/3	118.66								
	PW3/4	104.90								
	PW3/S	116.48								
PW5	PW5/1	120.94	9.51	8.35	8.19	2.17		16.9	0	5.74
	PW5/2	121.36								
	PW5/3	121.48								
	PW5/4	121.88								

KWU rods have been modelled using 3 axial slices, while W rods have been modelled using 6 axial slices. The lengths of the slices are reported in Table 5 and Table 6.

Table 5 KWU rods axial slices

Group label	Slice 1 Height [mm]	Slice 2 Height [mm]	Slice 3 Height [mm]
PK1 PK2 PK4	104	80	128
PK6	90	80	145

Table 6 W rods axial slices

Rod label	Slice 1 Height [mm]	Slice 2 Height [mm]	Slice 3 Height [mm]	Slice 4 Height [mm]	Slice 5 Height [mm]	Slice 6 Height [mm]
PW3/1	169	169	13	209	209	209
PW3/4	171	171	13	211	211	211
PW3/S	170	170	14	209	209	209
PW5 group	169	169	14	208	208	208

The "Improved" set is based on the results obtained from the sensitivity analyses performed in [8]. An "improved" input-deck may be necessary to take into account the different designs and irradiation conditions among the six groups (PK1, PK2, PK4, PK6, PW3, PW5). Differences in design, material, fabrication procedure and base irradiation history may influence the rod behavior during ramping and the achieved rod data[1]. It must be mentioned that the gap geometry, the grain size, the cladding tensile and rupture stresses are characteristics of each group as well as the rod lengths and the active lengths of the NPP cores used for the base irradiation

3.2 Boundary conditions

The following boundary conditions have been implemented in order to allow the TRANSURANUS simulation both in base irradiation and ramp test:

- Linear heat rates (LHR) at 3 (KWU rods) or 6 (W rods) axial position;
- Cladding temperature histories at the same axial position;
- Neutron fast flux (one average value for each irradiation cycle);
- Coolant pressure (one value).

The rate of increase and decrease between different constant LHR spans has been selected as 6 KW/(mh). The power ramps have been derived from original experimental data [6] and from data available in [1]. **Error! Reference source not found.** shows, as an example, the LHR for the rod PK1/S during base irradiation (left) and power ramp (right).

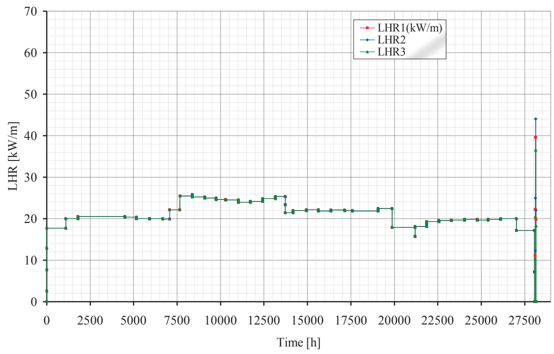

Figure 1: Rod PK1/S – LHR for base irradiation.

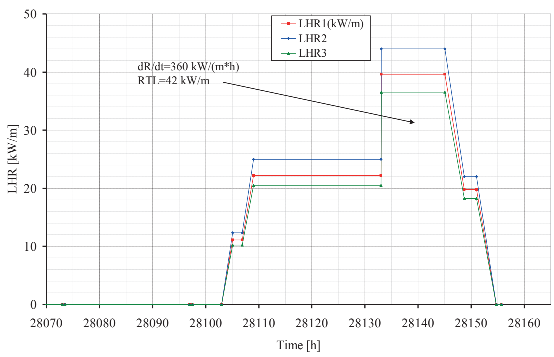

Figure 2: Rod PK1/S – LHR for power ramp (right).

3.3 Fission Gas Release sensitivity analysis

The objective of the sensitivity analysis is to verify the influence of the FGR models on the failure predictions. The “Reference” calculations are performed with Fgrmod 6 to treat intragranular bubbles gas behavior. It uses the URGAS algorithm with the diffusion coefficients of Hj. Matzke (thermal) and a constant athermal diffusion coefficient; the sensitivities concern:

- Fgrmod 4: URGAS algorithm with the diffusion coefficients of Hj. Matzke (thermal) and athermal diffusion coefficient according to data of R. White.
- Fgrmod 9: URGAS algorithm with diffusion coefficients of T. Turnbull.

The “Reference” model that considers intergranular gas bubbles behavior is igrbdm 3, the new model to treat the grain boundary FGR during power ramps. Three sensitivities are performed with the “Reference” model for intragranular behavior (Fgrmod 6) and setting the intergranular behavior to:

- Igrbdm 0: Fission gas behavior at grain boundaries not treated
- Igrbdm 1: Simple grain boundary fission gas behavior model (standard option)
- Igrbdm 2: Simple grain boundary fission gas behavior model

The solving algorithm used in the “Reference” simulation is Idifsolv 0: diffusion equation is solved by the URGAS-algorithm. The analyses are performed with the “Reference” models for intragranular and intergranular behaviors (Fgrmod 6, Igrbdm3 respectively) and changing the solving algorithm to:

- Idifsolv 6: Diffusion equation is solved by the FORMAS-algorithm with 6 exponential terms.

4 RESULTS

4.1 Cladding diameter change

Measurements related to the cladding creep down are provided by the PWR Super-Ramp Project database [6]. The maximum cladding diameter decrease (max creep down) is calculated as the greatest difference between several measures of the diameter prior to irradiation (PTI) and prior to ramp (PTR) at room temperature (20 C°) and atmospheric pressure. The same parameter is calculates using “reference” TANSURANUS simulation data and compared to the experimental in Figure 2. A systematic under-prediction of the experimental data can be observed.

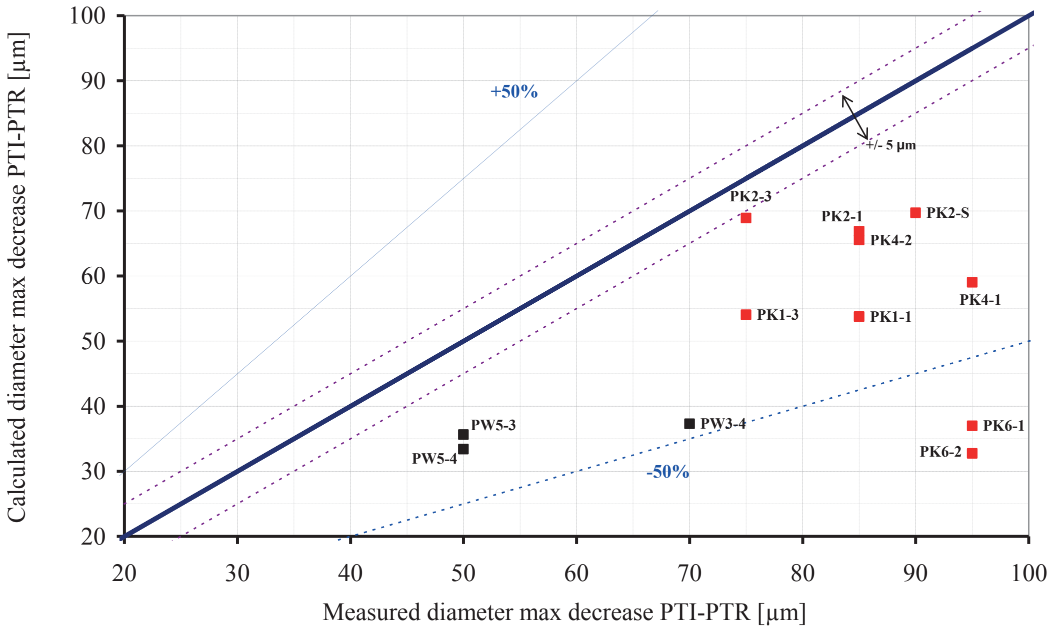

Figure 2: Cladding diameter change, base irradiation.
Comparison between experimental data and simulation.

Analogously, the maximum diameter increase is calculated from several diameter measurements performed prior to ramp (PTR) and after the ramp (AR) at room temperature and atmospheric pressure. Experimental measures are provided between ridges and at the ridges edges. Since TRANSURANUS is not able to simulate ridges, the experimental between ridges are used. Comparison between experimental and calculated data ("Reference") are shown in Figure 3

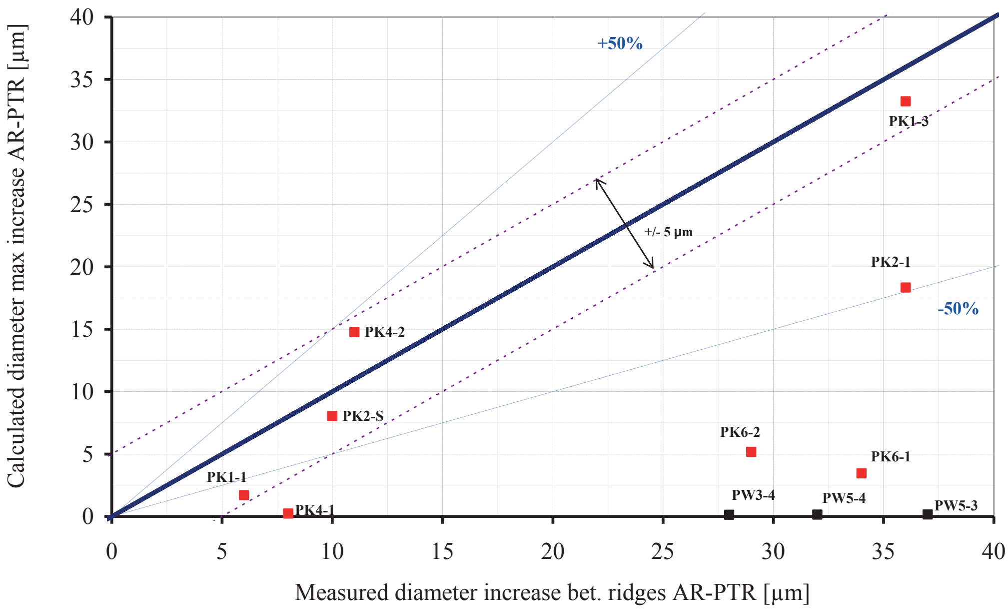

Figure 3: Cladding diameter change, ramp test.
Comparison between experimental data and simulation

4.2 Cladding outer corrosion

The comparison between the experimental data and the "Reference" TRANSURANUS simulation are reported in Figure 4

The KWU rods show highest values of cladding corrosion layer thickness. The simulation suggests that these rods reach the break-away point. The selected model is the EPRI standard model. The results under-estimation is about 50%, except PK2/4 and PK6/3, where greater errors are observed.

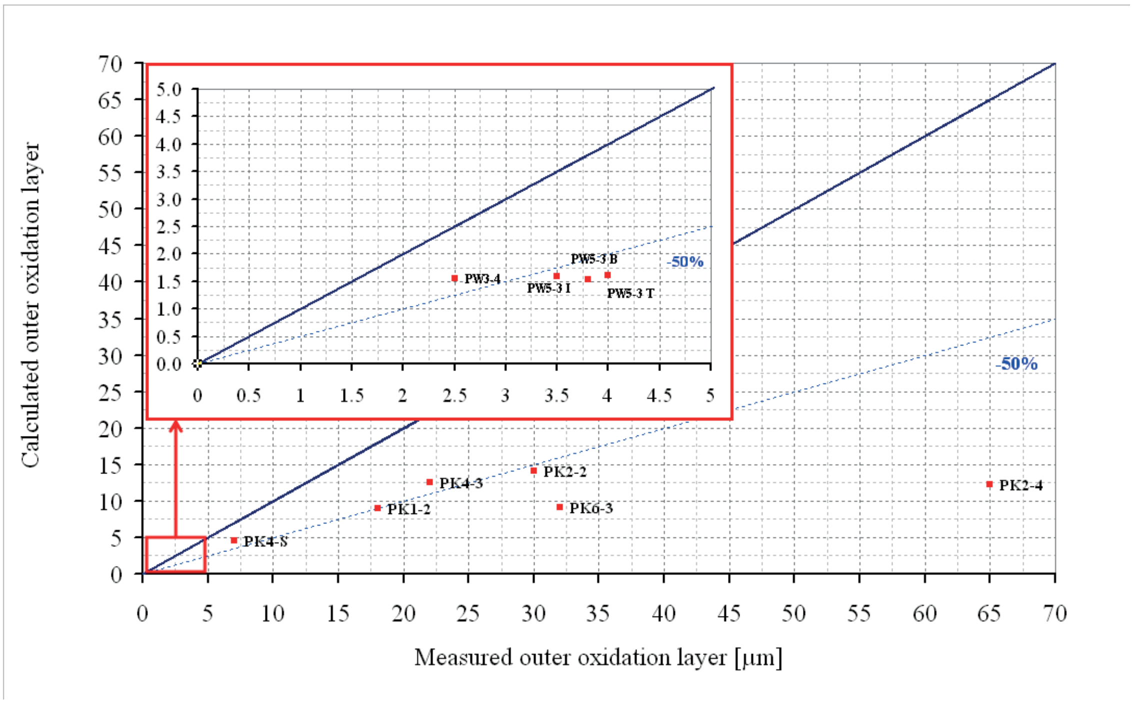

Figure 4: Cladding corrosion. Comparison between experimental data and simulation

4.3 Fission gas release

The comparison between the experimental data and the "Reference" TRANSURANUS simulation are reported in Figure 5.

Different predictions are observed:

- PK1 group results over predicted; the range of FGR is between 5 and 25%.
- PK2 group is generally slightly under predicted with the exception of PK2/S rod, this rod was ramped at a temperature 50° C lesser than the others. In this case, the range of FGR spreads between 10 up to 45 % (simulation and experiment).
- PK4 group results slightly under predicted with the exception of rod PK4/S that shows the worst under prediction (about -60%). The range of FGR is embedded in 10-30% (experiment and simulation)
- PK6 and PW3 groups highlight low values of FGR (less than 10%) both in the experiment and calculation. The trend is generally under predicted.

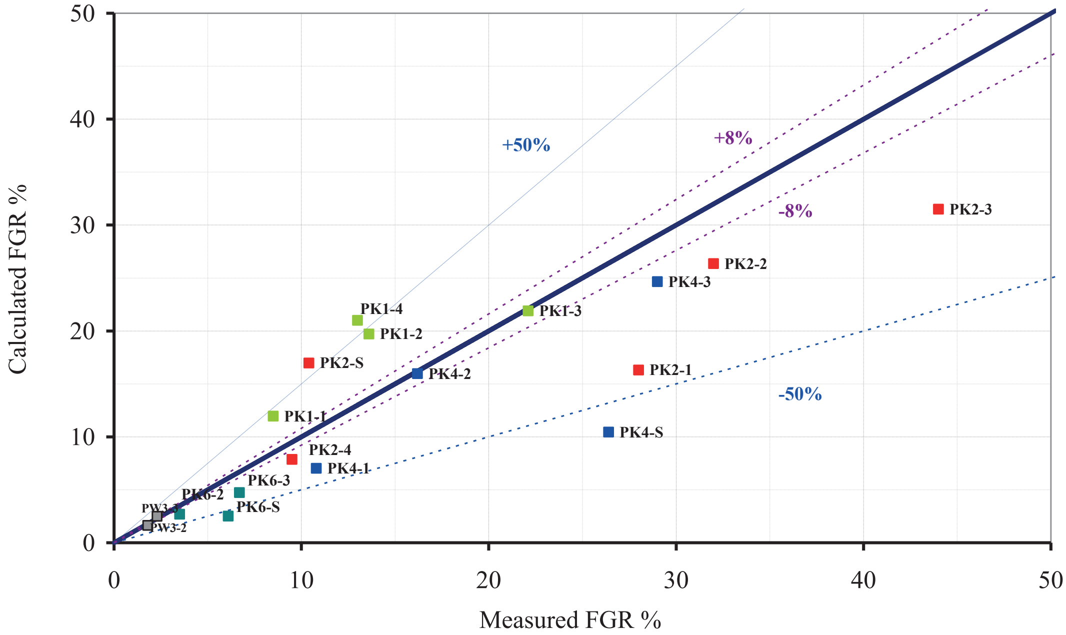

Figure 5: Fission gas release. Comparison between experimental data and simulation

4.4 Failure prediction

The TRANSURANUS failures predictions are summarized and compared with experimental data in Table 7. In the "Reference" calculation, the KWU rods are conservatively predicted even if the failures number is over-estimated, while the W rods are incorrectly predicted not-failed.

Table 7: Failure prediction. Comparison between experimental data and simulation

Rod group	Rod Label	Measured Burnup [MWd/kgU]	EXP F/NF	TU Reference F/NF	TU Improved F/NF
PK1	PK1-1	35.4	NF	NF	NF
	PK1-2	35.6	NF	F	NF
	PK1-3	35.2	NF	F	NF
	PK1-4	33.1	NF	NF	NF
	PK1-S	34.4	NF	NF	NF
PK2	PK2-1	45.2	NF	F	F
	PK2-2	45.1	NF	F	F
	PK2-3	44.6	NF	F	F
	PK2-4	41.4	NF	F	F
	PK2-S	43.4	NF	F	NF
PK4	PK4-1	33.7	NF	NF	NF
	PK4-2	33.8	NF	F	NF
	PK4-3	33.6	NF	F	NF
	PK4-S	32.5	NF	NF	NF
PK6	PK6-1	36.7	F	F	F
	PK6-2	36.8	NF	F	F
	PK6-3	36.5	NF	F	F
	PK6-4	33.6	F	F	F
	PK6-S	35.9	NF	F	F
PW3	PW3-1	38.1	F	NF	F
	PW3-4	36.6	F	NF	NF
	PW3-S	35.1	F	NF	F
PW5	PW5-1	40.5	F	NF	F
	PW5-2	39.9	F	NF	F
	PW5-3	41.4	F	NF	F
	PW5-4	39.2	F	NF	F

The “Improved” calculation corrects the results of 5 KWU rods without introducing false negatives (failed rods predicted not-failed). The W rods are now correctly predicted except PW3/4.

4.5 Fission gas release sensitivity analysis

The results of three different models for the intragranular bubbles gas behavior are plotted in Figure 6. The sensitivities highlight an increase of the calculated FGR comparatively with the reference case. The model Fgrmod4 better represents the behavior of PK2 and PK4 groups (PK1 result over predicted, PK6 and PW3 that have low FGR are predicted as in the reference). Model Fgrmod9 largely over predicts the FGR.

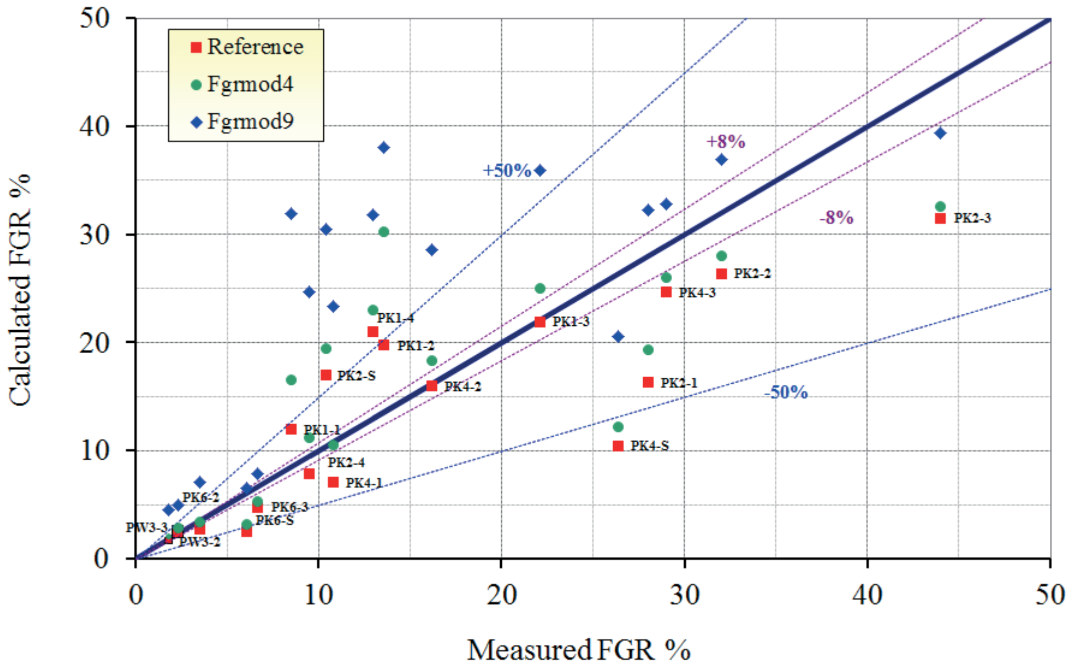

Figure 6: Fission gas release, intragranular behaviour models.
Comparison between experimental data and simulation

The results of four different models for the intergranular bubbles gas behavior are plotted in Figure 7. With the exception of the Igrbdm0 that assumes FGR without grain boundaries trapping in the other cases the predictions are lower compared to the reference one. Igrbdm 1 and Igrbdm 2 seem better represent the FGR of group PK1 while make worst the predictions in the other cases.

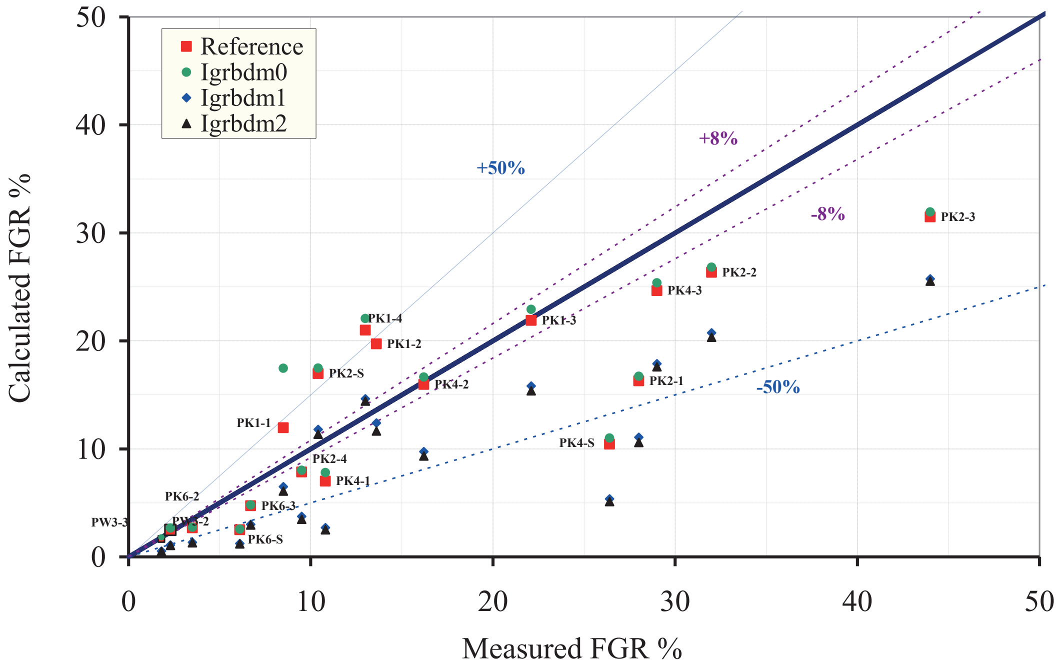

Figure 7: Fission gas release, intergranular behaviour models.
Comparison between experimental data and simulation

Best estimate predictions can be reached choosing different models, the result obtained are presented in Figure 8.

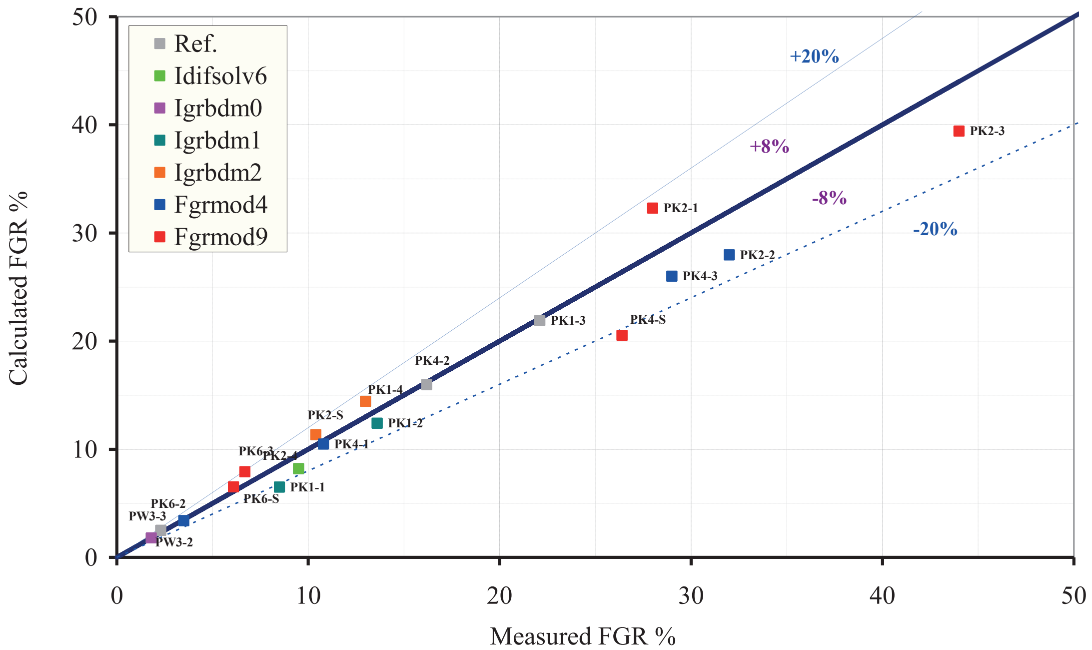

Figure 8: Fission gas release, best estimate results.
Comparison between experimental data and simulation

The different models seem to be not significant in term of failure prediction. Some model predicts the correct not failures of one rod (PK1/2). Results are summarized in Table 8.

Table 8 Failure prediction. FGR sensitivity analysis.

Rod group	Rod Label	Measured Burnup [MWd/kgU]	EXP F/NF	TU Ref. F/NF	TU Idifsolv6 F/NF	TU Igrdbm0 F/NF	TU Igrbdm1 F/NF	TU Igrbdm2 F/NF	TU Fgrmod4 F/NF	TU Fgrmod9 F/NF
PK1	PK1-1	35.4	NF	NF	NF	NF	NF	NF	NF	NF
	PK1-2	35.6	NF	F	NF	F	NF	NF	F	F
	PK1-3	35.2	NF	F	F	F	F	F	F	NF
	PK1-4	33.1	NF	NF	NF	NF	NF	NF	NF	NF
	PK1-S	34.4	NF	NF	NF	NF	NF	NF	NF	NF
PK2	PK2-1	45.2	NF	F	F	F	F	F	F	F
	PK2-2	45.1	NF	F	F	F	F	F	F	F
	PK2-3	44.6	NF	F	F	F	F	F	F	F
	PK2-4	41.4	NF	F	F	F	F	F	F	F
	PK2-S	43.4	NF	F	F	F	F	F	F	F
PK4	PK4-1	33.7	NF	NF	NF	NF	NF	NF	NF	NF
	PK4-2	33.8	NF	F	F	F	F	F	F	F
	PK4-3	33.6	NF	F	F	F	F	F	F	F
	PK4-S	32.5	NF	NF	NF	NF	NF	NF	NF	NF
PK6	PK6-1	36.7	F	F	F	F	F	F	F	F
	PK6-2	36.8	NF	F	F	F	F	F	F	F
	PK6-3	36.5	NF	F	F	F	F	F	F	F
	PK6-4	33.6	F	F	F	F	F	F	F	F
	PK6-S	35.9	NF	F	F	F	F	F	F	F
PW3	PW3-1	38.1	F	NF	NF	NF	NF	NF	NF	NF
	PW3-4	36.6	F	NF	NF	NF	NF	NF	NF	NF
	PW3-S	35.1	F	NF	NF	NF	NF	NF	NF	NF
PW5	PW5-1	40.5	F	NF	NF	NF	NF	NF	NF	NF
	PW5-2	39.9	F	NF	NF	NF	NF	NF	NF	NF
	PW5-3	41.4	F	NF	NF	NF	NF	NF	NF	NF
	PW5-4	39.2	F	NF	NF	NF	NF	NF	NF	NF

5 CONCLUSIONS

The ability of TRANSURANUS code version v1m1j11 in simulating the PCI phenomenon and predicting the cladding failure has been evaluated using the Studvisk PWR Super-Ramp data.

The simulation of 19 KWU rods shows a general under-prediction of the creepdown during the base irradiation and the diameter increase during power ramp. It is worth noting that the observed differences can be in part explained with the intrinsic limitation in TRANSURANUS geometric modelling: one-dimensional, plane and axisymmetric schematization characterized by plain strain condition, inability to model local geometry variation (i.e. ridges). A general under-prediction of the outer cladding oxidation is also observed. The prediction of the failure is correct for 7 out of 19 rods (Reference input decks). Improved input decks reach a correct prediction of 12 out of 19 rods All the errors are conservative.

The analysis of the W rods shows good results with the "improved" input decks, when an axial shape of the neutron fast flux is considered. & out of 7 rods are correctly predicted.

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