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SUPER-RAMP PK2 cases by START-3

Preliminary report

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Introduction

The Studsvik SUPER-RAMP Project, an internationally sponsored research project, investigated the failure propensity of typical LWR fuel in the form of test rods when subjected to power ramps, after base irradiation to high burn-up. The following information summarizing the project is abstracted from the Final Report of the SUPER-RAMP project (STSR-32).

The Project power ramped 28 individual PWR rods and 16 BWR rods. The PWR rods were all tested using high ramp rates. Due to different objectives for the BWR subprogramme, one set of the BWR rods was tested using a high ramp rate, and another set were tested with a very slow ramp rate.

All rods underwent a thorough examination programme, comprising characterisation prior to base irradiation, examination between base and ramp irradiation and examination after ramp irradiation.

This consisted of 6 groups of rods with variations in design and material parameters. The rods were base irradiated in a power reactor environment KK Obrigheim or BR-3 at time averaged heat ratings mainly in the range 14-26 kW/m to peak burn-ups in the range 33-45 MWd/kgU and were subsequently ramp tested in the research reactor R2 at Studsvik, Sweden. The result can be summarized as follows:

In this document some calculations are made on the PK2 group fuel rods. The rods were standard rods manufactured by Kraftwerk Union AG/Combustion Engineering (KWU/CE).
All these rods sustained ramping to power levels in the range 41 to 49 kW/m and power changes 16-24 kW/m without failure, in spite of large deformations, fuel restructuring and fission gas release particularly for the PK2 rods

SUPER-RAMP important details

Base irradiation

The KWU/CE test fuel rods were base irradiated in the commercial pressurized water reactor Obrigheim (KWO) in FR Germany. Reactor characteristics for KWO are given below:

Obrigheim Power Reactor Characteristics.
Values At 100 % reactor power

Average reactor power (thermal)	1045 MW
Average rod power	171 W/cm*
Coolant temperature at core inlet	283 C
Coolant temperature at core outlet	312"
Coolant velocity at 300 C	3.39 m/s
Mass flow rate	6833 kg/s
Average system pressure	14.5 MPa
Active length of core	2650 mm

* 193 W/cm since cycle 10 (August 78)

Ramp details

The power ramp tests were performed according to the following typical scheme, characterized by the following phases:

1. A conditioning phase, with a rather slow increase of the linear heat rating from an initial value to a selected value of 250 W/cm (the conditioning level) and holding at the value for 24 hours. **We have to add that there is no further clarification of the words “rather slow increase of LHR” in the documents, provided by IAEA. This leads to uncertainty in the irradiation history and, hence, is a possible cause of errors and unsatisfactory agreement of calculated and measured results.**

The objective of the conditioning was to adjust the rod conditions to the same conditioning level for all rods, thus equalizing the start-point of the ramp tests.

2. A ramping phase with a rapid increase of about 100 W/cm/min from the conditioning heat rating to a pre-selected ramp terminal level.

3. A holding period at the ramp terminal level of normally 12 hours.

Exceptions to this normal ramping scheme were the following tests:

- Rod PK2-S was deliberately tested with a coolant inlet temperature 50 C below normal.

- For rod PK2-4 the hold time at the ramp terminal level was intentionally interrupted after 1 min.

Table 1. OVERVIEW OF PWR TEST MATRIX

Rod ID	PK2-1	PK2-2	PK2-3	PK2-4	PK2-S
Burn-up MWd/kgU	45.2	45.1	44.6	41.4	43.4
Fluence *E25 /sq m	8.1	8.1	8.1	8.1	8.1
Pre-condition power kW/m	25	25	25	25	25
Hold time hrs	24	24	24	24	24
Ramp power kW/m	41.0	46.0	49.0	44.0	44.0
Ramp rate kW/m/min	8.5	9.5	8.5	8.5	8.5
Hold time mins.	720	720	720	1	720
Failed (Y/N)	NF	NF	NF	NF	NF

Figure 1 shows history of base irradiation and ramp.

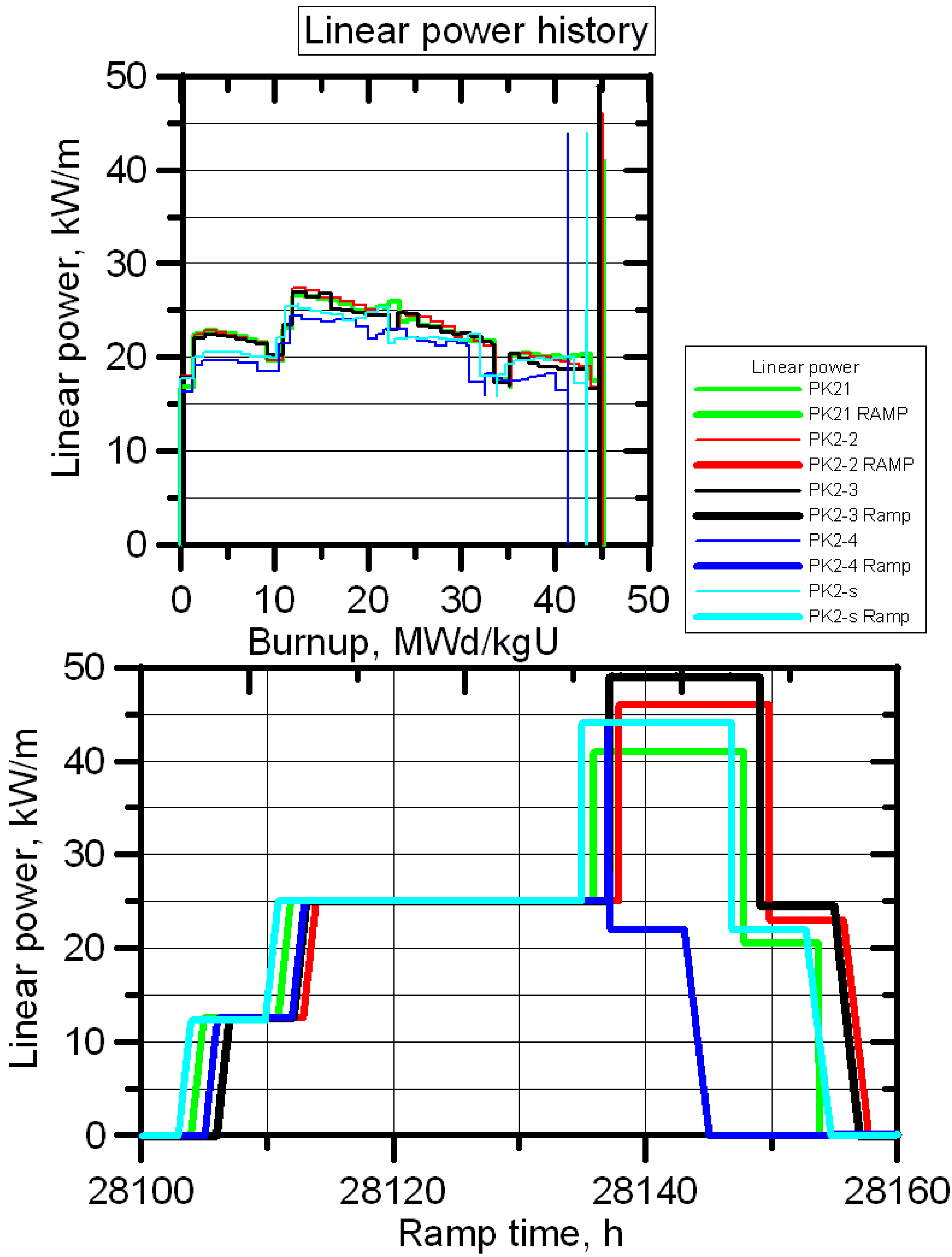

Fig.1 Linear power history.

We have to mention that STSR-32 gives a plot of PK2-1 rodlet elongation during the ramp and the details of the ramp irradiation. These differ from the history that was provide by IAEA in ASCII PK2-1.HIS file.

Pellet and cladding details

Table 2: Fuel rod characteristics

Variable					
Rod	**PK2-1**	**PK2-2**	**PK2-3**	**PK2-4**	**PK2-S**
Cladding material	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4
Overall rod length, mm	390.26	390.15	390.16	390.12	390.34
Fuel column length, mm	317.4	317.8	319.0	317.2	317.4
Plenum length, mm	32.6	33.0	32.5	32.6	32.8
Cladding OD, mm	10.753	10.752	10.752	10.751	10.754
Cladding ID, mm	9.283	9.283	9.283	9.283	9.283
He fill pressure, bar	22.5	22.5	22.5	22.5	22.5
Fuel weight, g	210	210	210	209	210
Pellet					
Enrichment, %	3.21				
Pellet density (g/cm3	10.36				
Outer pellet diameter, mm	9.14				
Inner pellet diameter, mm	-				
Average grain size, mkm	5.5				
O/U ratio	2.00 ± 0.01				
Portion of open porosity of pellet volume, %	3.16 ± 0.27				

We have to mention that inner diameter of cladding (9.38 mm) given in the file PRECHAR.PWR contradicts with the corresponding value of 9.28 given in the PDF file STSR-32, which seems to be correct.
The other important parameter that the SUPER-RAMP dataset lacks is the fuel pellet and cladding surface roughness.

Calculation results

Versus time and burnup

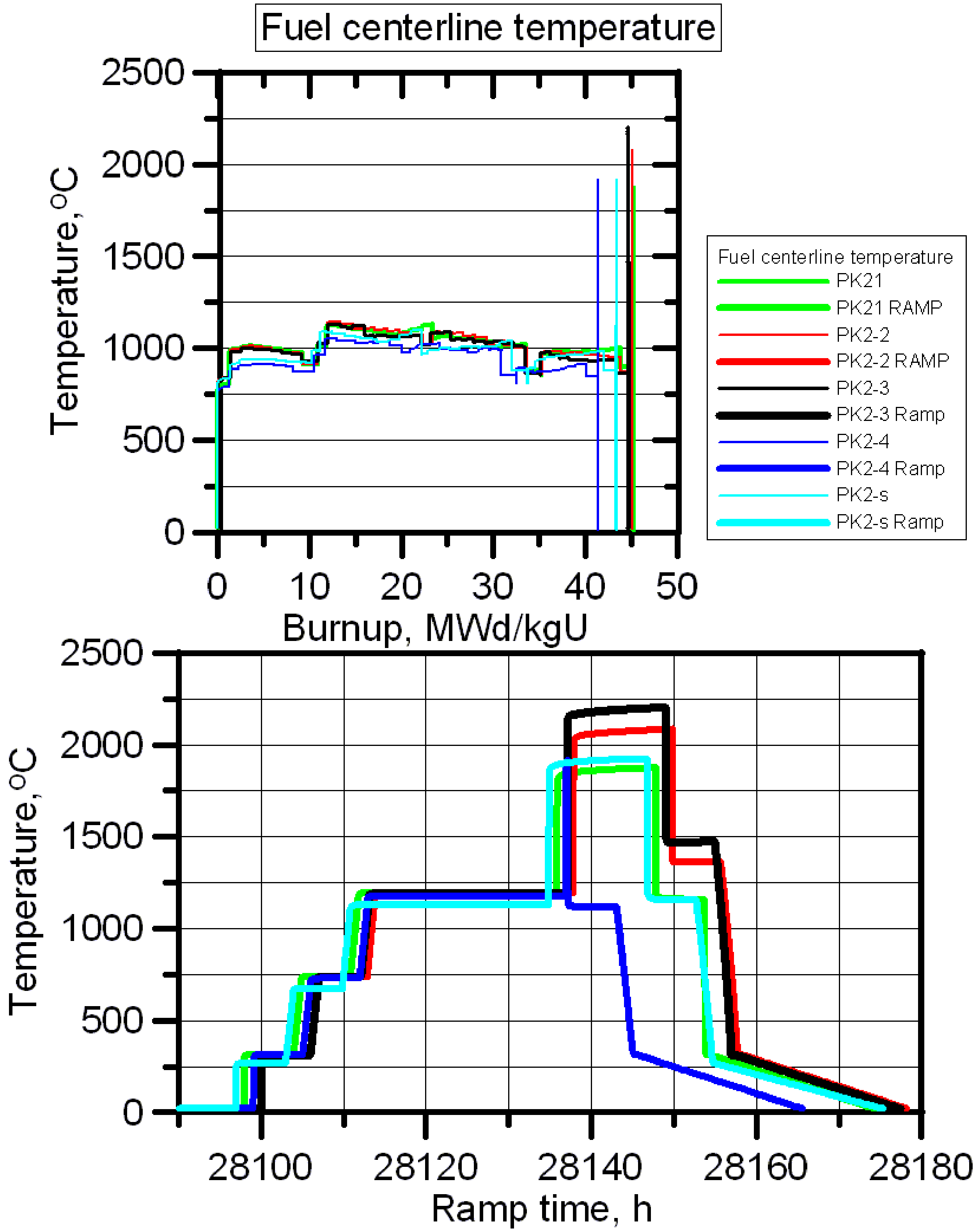

Fig.2 Calculated fuel centerline temperature versus burnup and time.

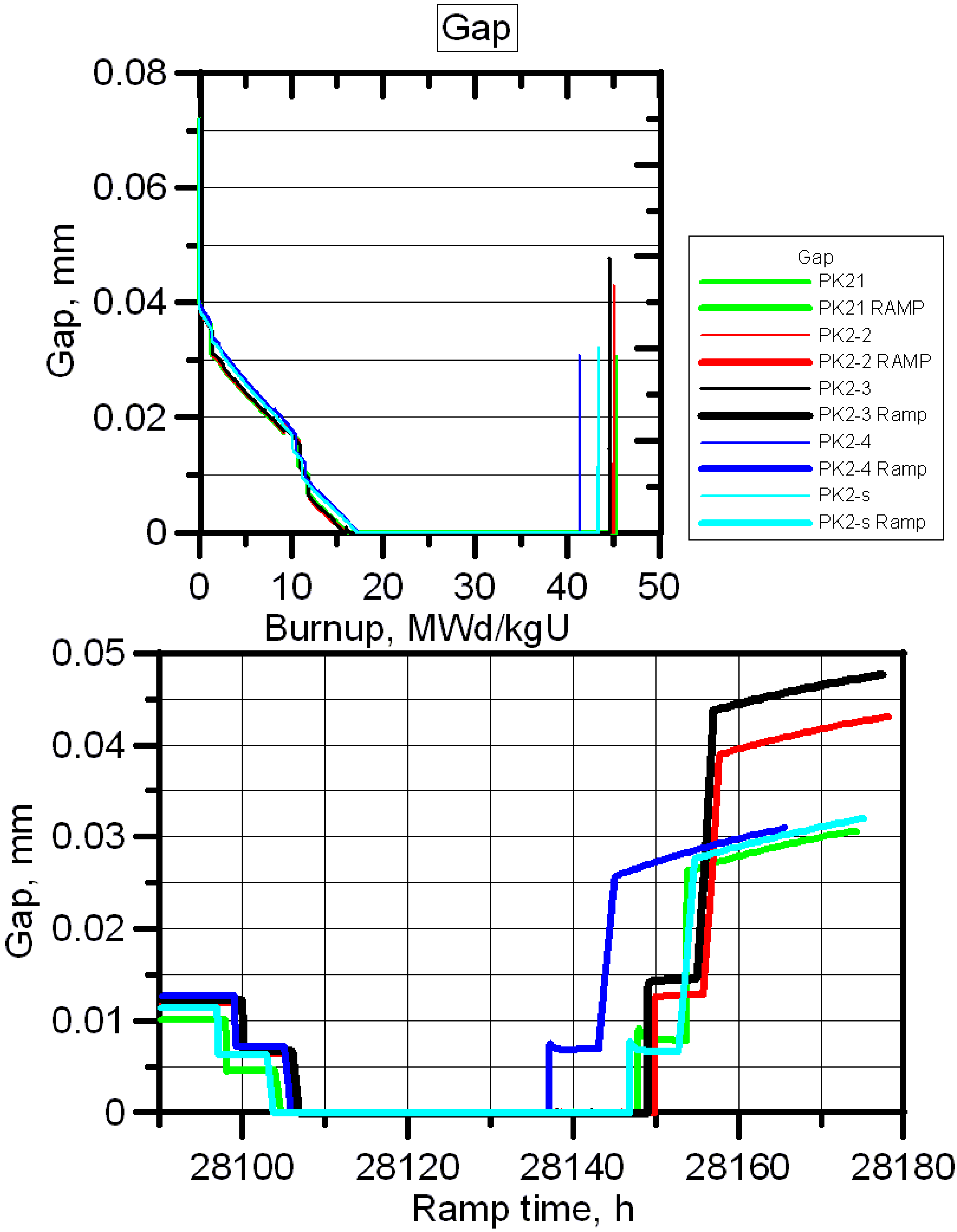

Fig.3 Calculated radial gap versus burnup and time.

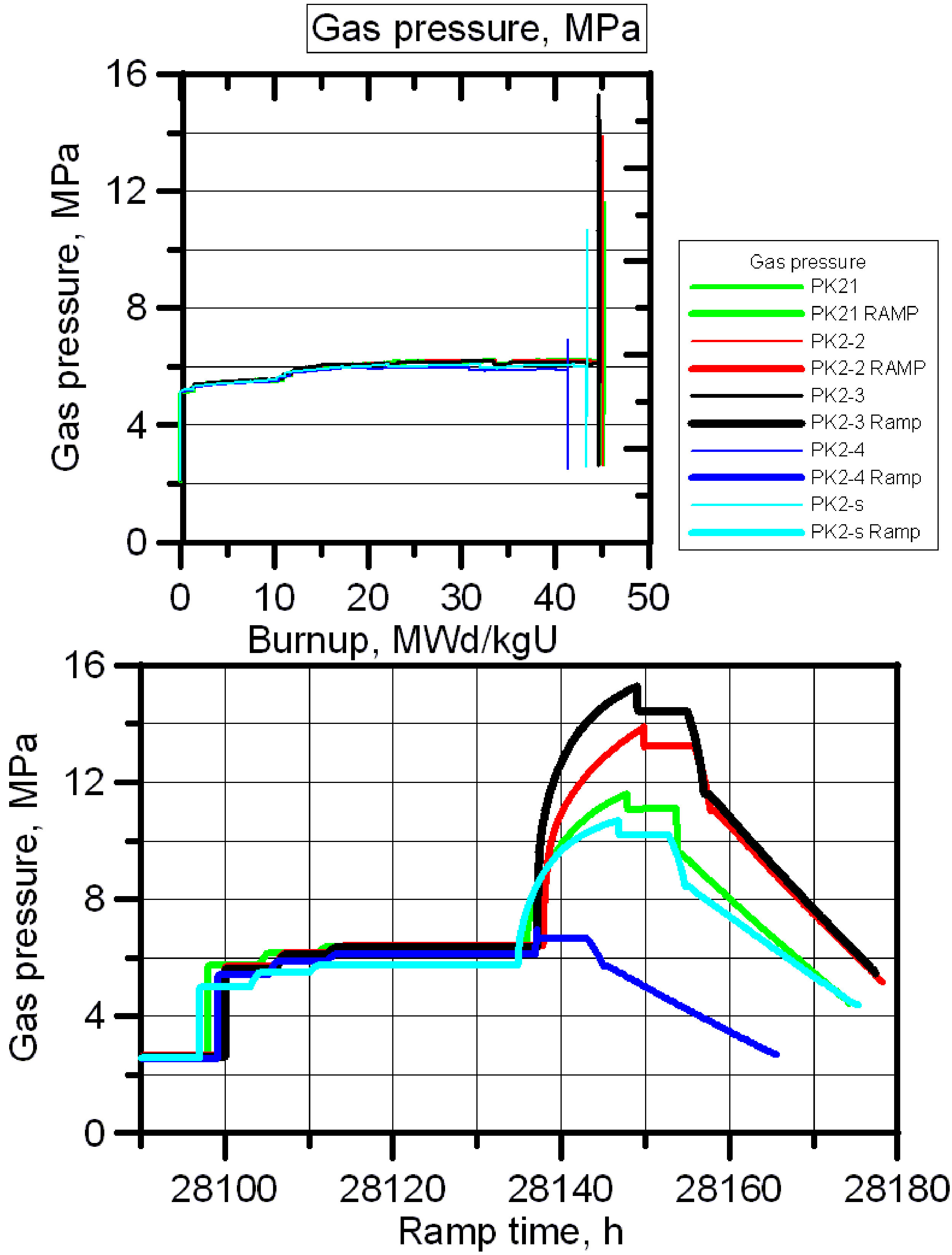

Fig.4 Calculated gas pressure versus burnup and time.

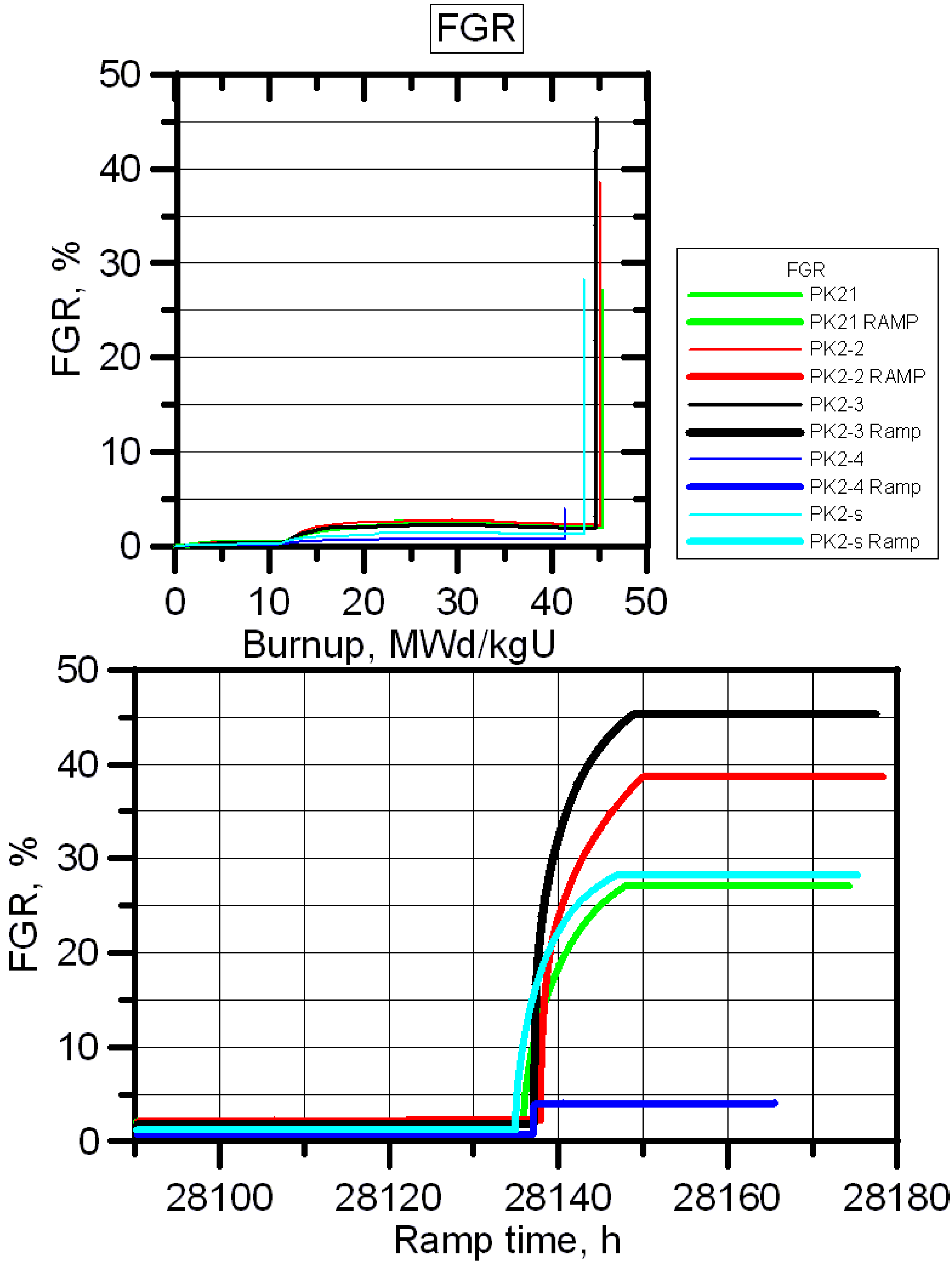

Fig.5 Calculated fission gas release versus burnup and time.

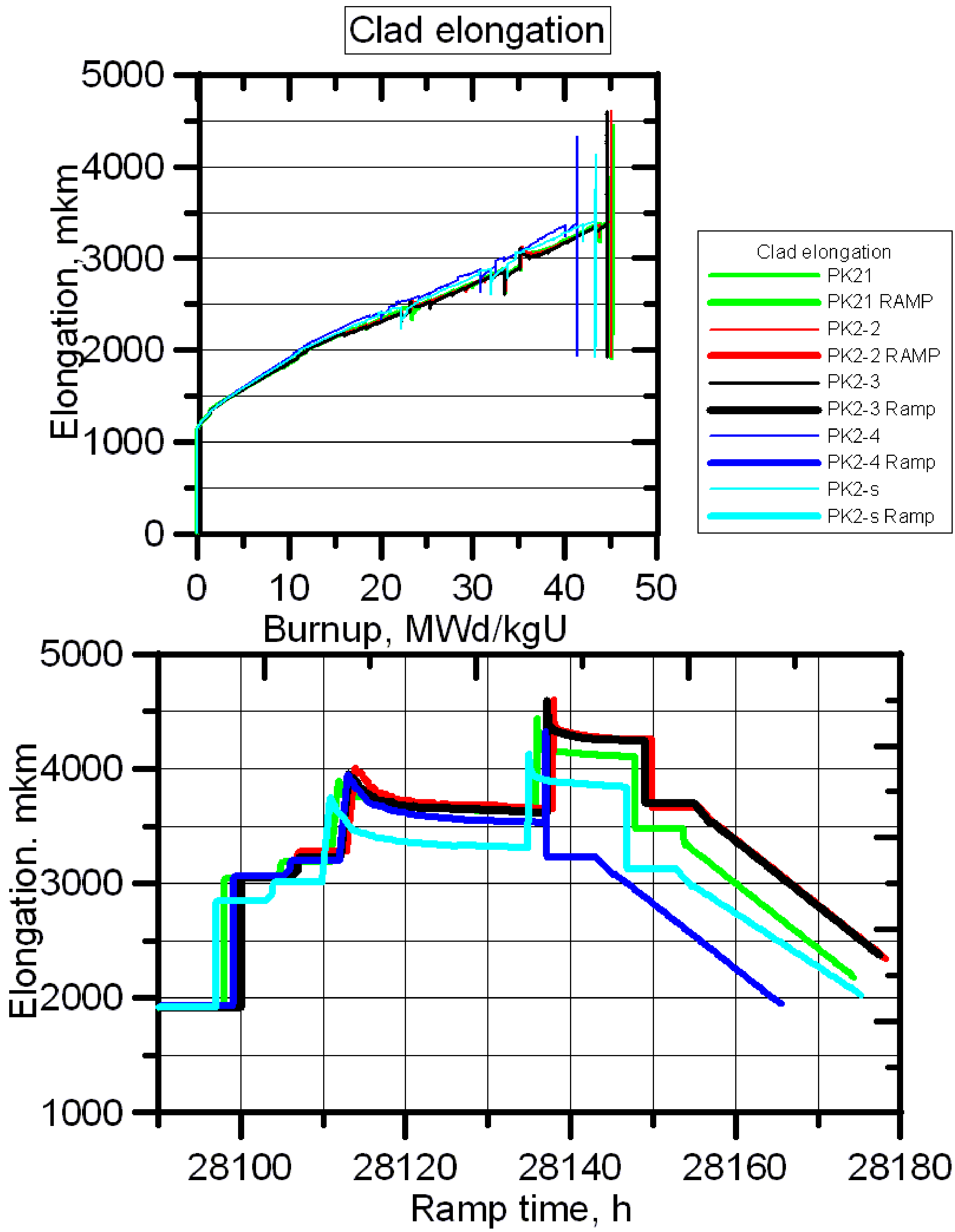

Fig.6 Calculated axial elongations versus burnup and time.

At the end-of-life

Diameters

Table 3.Initial clad outer diameters.

Rod	Initial diameter (mm)
PK2-1	10.754
PK2-2	10.752
PK2-3	10.752
PK2-4	10.750
PK2-s	10.754

Table 4. Calculated clad diameters after ramp (including oxide layer thickness), mm:

	Section 1	Section 2	Section 3
PK2-1	10.799	10.821	10.789
PK2-2	10.827	10.848	10.809
PK2-3	10.836	10.864	10.818
PK2-4	10.763	10.784	10.750
PK2-s	10.806	10.828	10.789

Table 5. Calculated diameters change (after base irradiation and after ramp, mkm)

Rodlet	Diameter change during base irradiation	diameter change during ramp
PK2-1	-11	78
PK2-2	-9	105
PK2-3	-12	124
PK2-4	-28	62
PK2-s	-13	87

Calculated values were taken in the 2nd axial zone.

Clad Oxidation

Table 6. Calculated clad oxidation thickness (mkm):

	Section 1	Section 2	Section 3
PK2-1	41	41.2	41.11
PK2-2	41	41.12	41.11
PK2-3	40	39.89	39.79
PK2-4	33	32.82	32.79
PK2-s	36.19	36.20	36.18

According to STSR-32 the average oxidation thickness for PK2/2 and PK2/4 is

Table 7. Oxidation thickness according to STSR-32, mkm

PK2/2	19-38
PK2/4	60-72

As we can see from the tables above, the coincidence is good for PK2/2 but it is also bad for PK2/4. The measured value of 60-72 mkm for PK2/4 seems very strange judging from the point of view of the adopted by the current version of START-3 oxide layer growth model [1] since there is no obvious specific reason for this rodlet to have such a thick (relative to other rodlets) oxide film.

Radial gap

Table 8.Calculated radial gap (mm) after ramp:

	Section 1	Section 2	Section 3
PK2-1	0.0259	0.0306	0.0223
PK2-2	0.0293	0.043	0.0287
PK2-3	0.0402	0.0476	0.0359
PK2-4	0.0243	0.0309	0.0214
PK2-s	0.0274	0.032	0.0262

Grain size

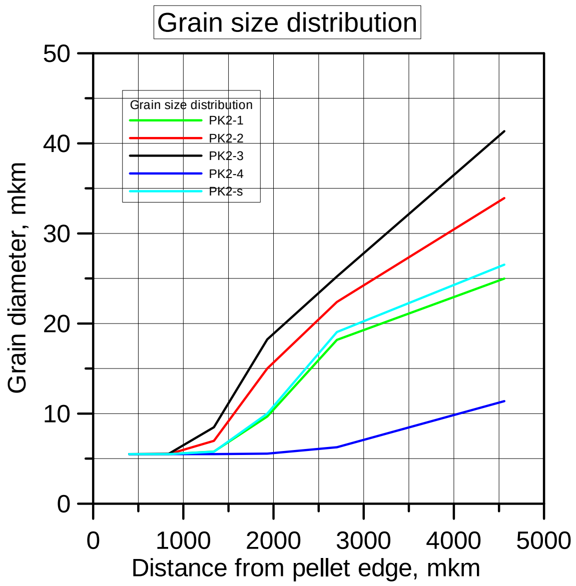

Fig.7 Grain diameter vs distance from fuel pellet edge.

FGR

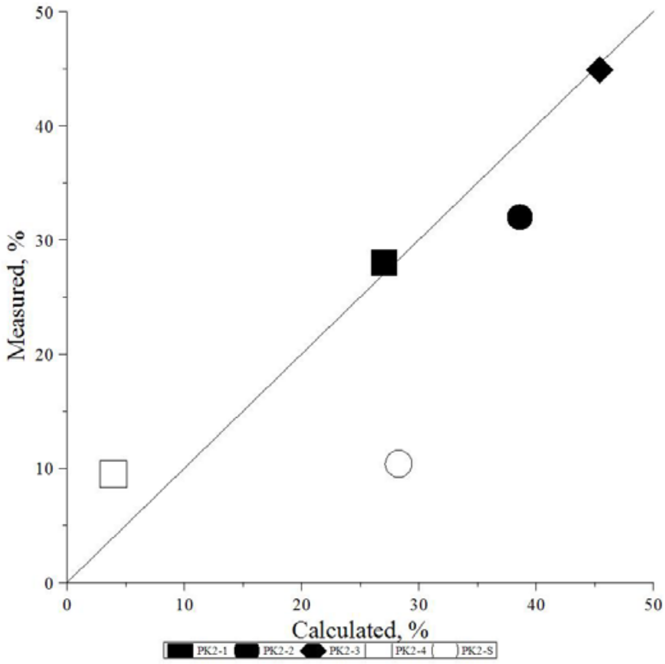

Figure 8. Fission gas release after ramp. Calculated value vs measured value.

Maximum fuel centerline temperature

Table 9. Maximum fuel centerline temperature.

Rod	Temperature, °C
PK2-1	1873
PK2-2	2083
PK2-3	2202
PK2-4	1918
PK2-s	1923

Elongations

According to STSR-32 report, the measured elongation for PK2-1 during the ramp equals approximately 400 mkm. As we can see from the Fig.6, this value is smaller than START-3 prediction of 821.8 mkm. The residual elongation for PK2-1 according to STSR-32 is 250 mkm and the START-3 prediction is 211.1 mkm. The calculations were performed on the fuel stack length only.

Summary and conclusion

Though the results of this paper seem to be based on the incomplete dataset (ambiguity in power raise rates, undefined fuel pellet and cladding surface roughness), we think that START-3 Zircalloy-4 model requires further improvements. In order to do them, we kindly ask IAEA to provide us with more detailed irradiation histories (more than 3 axial zones, power increase and decrease rates, fast neutron fluxes during the ramp) and other experimental data (measured diameter profiles before and after ramp and elongations during ramp), mentioned as available in the STSR-32 report, but not provided in the PK2 case.
Fission gas release model shows satisfactory results despite all the difficulties with input data and other parts of the code.

Literature

1. SKI Report 2005:41 'Models for Rod behaviour at High Burnup', Lars O. Kernkvist, Ali R. Massih