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Results of Calculation of VVER-440 Fuel Rods (Kolskaya-3 NPP) at High Burnup

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The results of calculations of thermal physical characteristics of fuel rods of two fuel assemblies (#1 and #2), which were operated within 5 - 8 and 5 - 9 core fuel loadings of Unit 3 of the Kolskaya NPP, respectively, are presented in this paper. These fuel assemblies were unloaded off the reactor, being actually tight (as the cladding tightness control of the NPP showed), and sent for the post-irradiation testing.

1 Calculation Code

PIN-04 (PIN-micro) code [1] was previously used in the USSR to validate the working ability of VEER-440 fuel rods during normal operation regimes. Now for the same purposes we also use the code for fuel rod thermal physical calculations PIN-mod1 (PIN-mod2) [2, 3], which is designed for modeling of VVER reactor type fuel rods behavior in a quasisteady-state operation. In comparison with PIN-micro code, the following models have been changed in PIN-mod2 code: the model of fission gas release out of fuel, calculation of relocation, fuel-cladding gap thermal conductivity calculation, burnup influence on the fuel thermal conductivity has been taken into account, as well as the effect of fuel creep upon the increase of fuel diameter, the approach for setting the input data while calculating burnup has been changed, etc. The average burnup along the fuel cross-section is used as a parameter to consider the influence of burnup on the increase of the fission gas release out of fuel. We would like to note that in [3] it is shown that for VVER fuel rods when, for example, average burnup along the cross-section is 48 *MWd/kg*, burnup in the thin layer of the outer fuel surface can reach 96 *MWd/kg*. During calculation of the temperature field the values of the relative power density reduction along the fuel radius, depending upon the achieved burnup, were used [3]. It was considered that fast neutron flux along the fuel rod height is proportional to thermal load; proportion coefficient was calculated depending upon the linear thermal load and the achieved burnup [3].

In PIN-type codes while modeling behavior of fuel and cladding, their presentation in the form of integral coaxial cylinders (in respect to the height) values of geometric and structural parameters are used.

2 Input Data for Calculations

The results of neutron physical calculations for Kolskaya NPP and the reactor system operational history (power, coolant inlet temperature, coolant heating in the reviewed fuel assemblies) were used for developing power history of fuel rods.

Figure 1 presents the simplified power history of the 3rd Unit of the Kolskaya NPP reactor system within the time period since 24.09.1986 (beginning

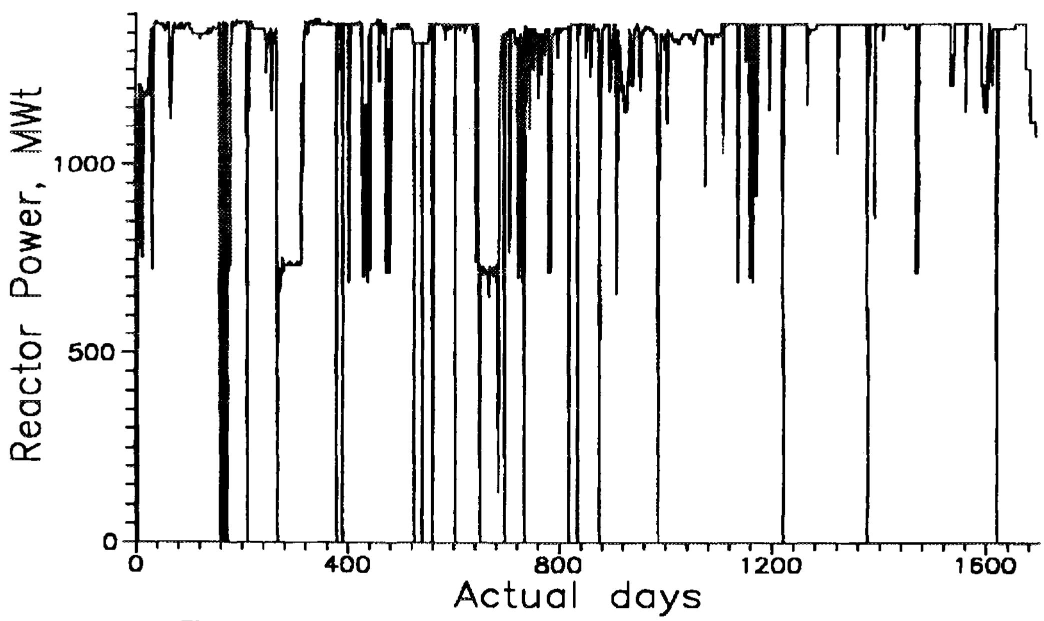

Figure 1 Kolskaya-3 NPP reactor system power history

of the 5th fuel cycle) till 12.10.1991 (end of the fuel cycle). Periods between the planned annual fuel reloading were excluded in Fig. 1. Power history of fuel rods was presented in actual (and not in effective) days of operation. In this case planned annual reloadings were excluded. For the simplification of presentations of figures, it was considered that the coolant temperature is 200°C, when the reactor is at zero power. Short-term distortions of the power density along the height of the fuel assembly, which van take place if the power of the unit is changed, are not presented in the power history of fuel rods.

Several power histories of fuel rods were developed for each fuel assembly. In all cases the relative profile of power density along the height of the fuel rod was obtained out of the results of neutron-physical calculations for fuel assembly.

Case 1.

While developing power history of the "average within the fuel assembly" fuel rod, the average value of the linear thermal load for any of the presented moments of time was obtained as the quotient when fuel assembly power at a given moment of time was divided by 124 (number of fuel rods within the fuel assembly) and by 243 *cm* (the length of fuel stack).

Cases 2 - 4 relate to one specific fuel rod, which has maximum deep (for its fuel assembly) burnup. The fuel rod power history was developed in two ways: with and without consideration of the excess coefficients.

Case 2.

When the fuel rod power history was developed without consideration of excess coefficients, the average value of the linear thermal load $q_{lav}^{(2)}$ was obtained as a quotient when fuel rod power at this particular time moment was divided by 243 *cm.*

Case 3.

While developing fuel rod power history with the consideration of excess coefficient in respect to burnup, average value of linear thermal load:

$$q_{lav}^{(3)} = 1.04\, q_{lav}^{(2)}$$

Case 4.

While developing fuel rod power history with the consideration of excess coefficient in respect to linear load, average value of linear thermal load:

$$q_{lav}^{(4)} = k\, q_{lav}^{(2)}$$

where k is a coefficient considering technological deviations during fuel fabrication and errors of neutron-physical calculations; it was considered that $k=1.06$. In this case it was considered that k influence caused only the increase of linear load 1.06 times; k influence was not considered while calculating burnup and fission products storage in the fuel.

After thus developing fuel rod power history (in effective operation days), consideration of the real power history of the reactor system was introduced.

In the calculations it was considered that:

- the initial fuel stack length is 242 *cm*;
- fuel pellets are flat ended;
- fuel enrichment for U-235 is 4.4%;
- fuel grain average size is 7 μm;
- fuel mass for case 1 is 1095 *g* for the fuel rod out of fuel assembly 1 and for the fuel rod out of assembly 2 it is 110 *g*; for cases 2 - 4 fuel mass is 1091 *g* for the fuel rod out of fuel assembly 1 and for the fuel rod out of fuel assembly 2 it is 1098 *g*;
- outer diameter of the cladding is 9.15 *mm*;
- filling gas - helium (98%) when the filling pressure P_{fil} .

As the parameters of fuel pellets and cladding have the spread in values, than three calculations were carried out for the variants:

MAX : maximum effective gap:
inner cladding diameter D_{ci}=7.8 *mm*;
pellet outer diameter D_{po}=7.54 *mm*;
pellet linear diameter d=1.2 *mm*;
fuel initial density p=10.43 g/cm^3 ;
filling gas pressure P_{fill}=0.5 *MPa*;
volume of the gas collector v_g=3.7 cm^3.

AVER: average effective gap:
D_{ci}=7.76 *mm*; D_{po}=7.565 *mm*;
d=1.6 *mm*; p=10.58 g/cm^3;
P_{fill}=0.6 *MPa*; v_g=4.2 cm^3.

MIN : minimum effective gap:
D_{ci}=7.72 *mm*; D_{po}=7.59 *mm*;
d=2.0 *mm*; p=10.73 g/cm^3;
P_{fill}=0.7 *MPa*; v_g=4.7 cm^3.

It is important to note that MAX or MIN case can not be realized in reality within the height of the fuel rod (in this case the requirements of the regulatory documents relating to the mass and height of the fuel stack will not be satisfied), but it is possible that one pellet with the parameter values described as MAX variant gets into the very part of the cladding, the value of the inner diameter of which corresponds to the MAX value. Due to small heat leaks between the pellets in the axial direction [2] maximum temperature in this pellet will be close to the MAX variant. That is why as a real case we are to review the results of calculations for the AVER case, basing on the results of calculations of the fuel rod local characteristics (maximum temperature, maximum fuel/cladding interaction) for MIN and MAX cases as the limiting possible ones.

It was considered that the limiting value of the fuel resintering is restricted by the density 10.68 g/cm^3, at the same time change of fuel diameter can not exceed 0.4%.

Figures 2 and 3 present dependencies of the average linear thermal load of the fuel rod out of fuel assemblies 1 and 2, respectively, vs. time for Case 2.

Figures 4 and 5 present calculated maximum fuel temperature of the fuel rod out of fuel assemblies 1 and 2, respectively, vs. time for Cases 2 AVER.

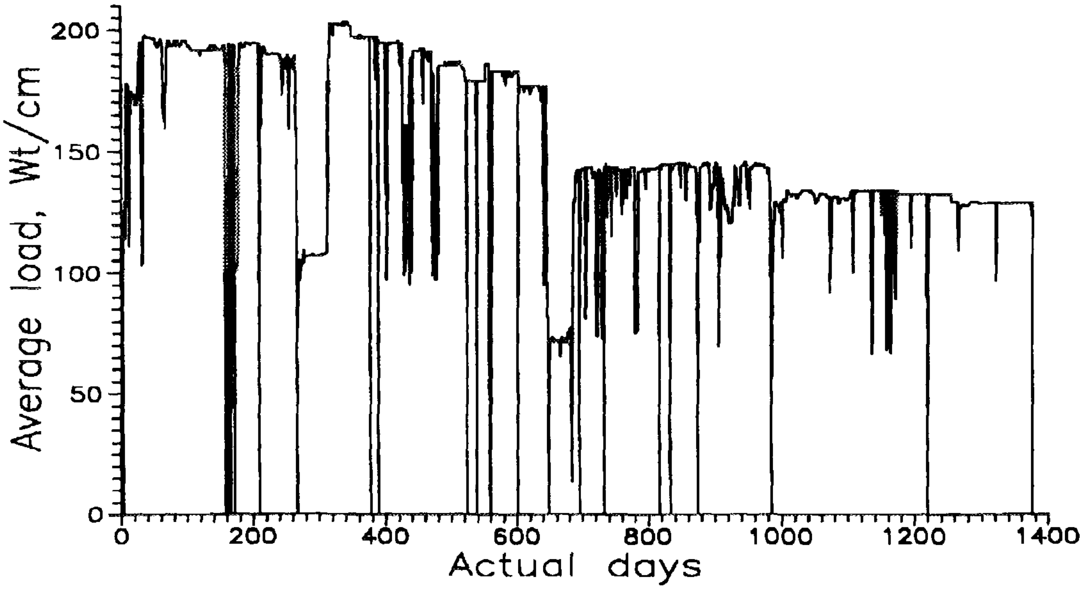

Figure 2 Dependence of the fuel rod average linear thermal load (assembly # 1) vs. time for case 2.

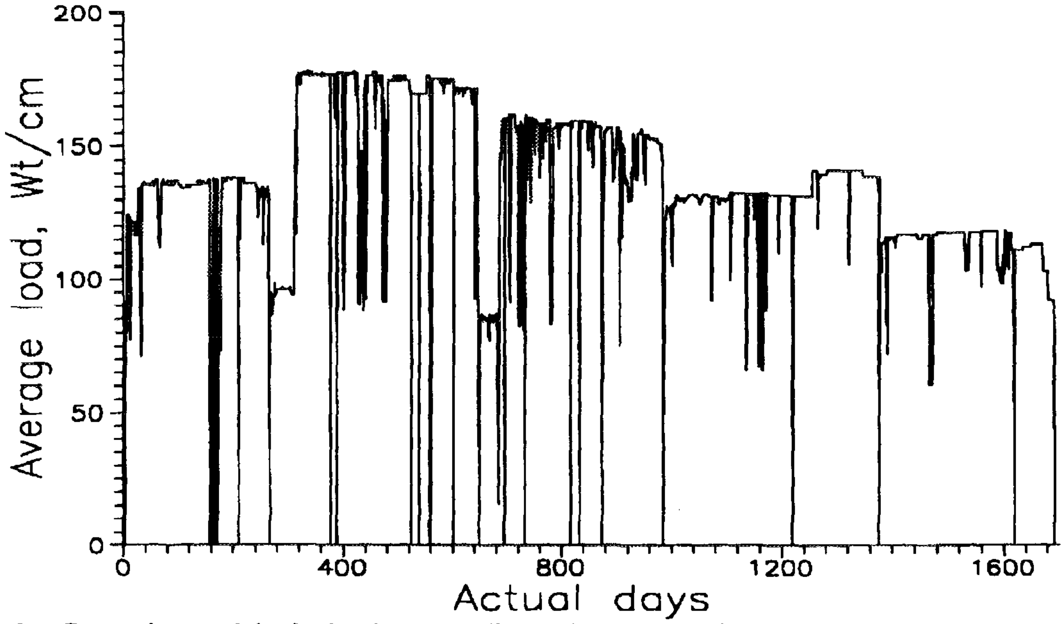

Figure 3 Dependence of the fuel rod average linear thermal load (assembly # 2) vs. time for case 2.

Figures 6 and 7 present calculated fuel temperature of the fuel rod out of fuel assemblies 1 and 2, respectively, vs. time for Cases 4 AVER.

Figures 8 and 9 present calculated values of the gas media pressure inside the cladding of the fuel rod out of assemblies 1 and 2, respectively, vs. time for Cases 2 AVER.

The calculation results are listed in Tables 1 - 2.

3 Conclusions

The results of thermal-physical calculations of fuel rods of fuel assemblies, which have achieved deep burnup during 4-year (> 46 *MWd/kg*) and 5-year (> 48 *MWd/kg*) fuel cycles of the 3rd Unit of Kolskaya NPP are presented in the paper.

For the calculations the average fuel rod in the fuel assembly and the fuel rod with the maximum burnup were selected.

The preliminary comparison of the calculation results with the results of post-irradiation examinations [5] (fission gas release from 0.7 to 1.3% for the fuel rods of fuel assembly 1; form 1.5 to 3.7% for the fuel rods from assembly 2; pressure inside the cladding at the end of campaign at normal conditions is from 0.87 to 1.13 *MPa* for the fuel rods of fuel assembly 1 and from 0.95 to 1.4 *MPa* for fuel rods of fuel assembly 2; decrease of the cladding radius is up to 35 μm for the fuel rods of fuel assembly 1 and up to 30 μm for the fuel rods of fuel assembly 2, etc.) showed very satisfactory agreement. In the future the improvement of the model for calculation fission gas release and creep of the cladding is planned on the basis of the results of post-irradiation examination.

The results show that the fuel rod completely preserves its working ability; fuel temperature does not exceed 1300°C; fission gas release does not

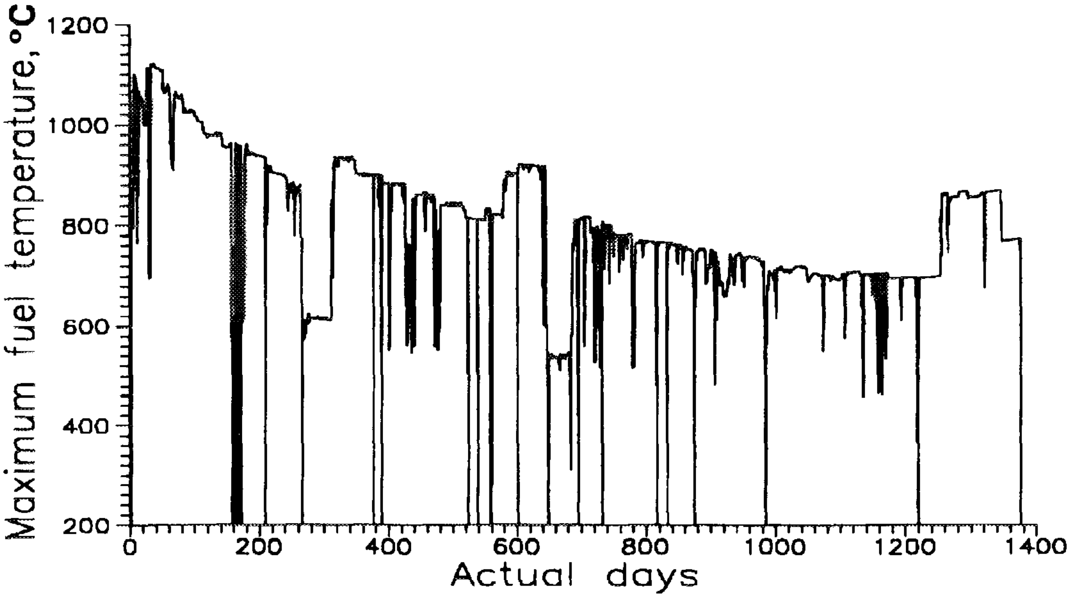

Figure 4 Calculated maximum fuel temperature (assembly # 1) vs. time for case 2 - aver.

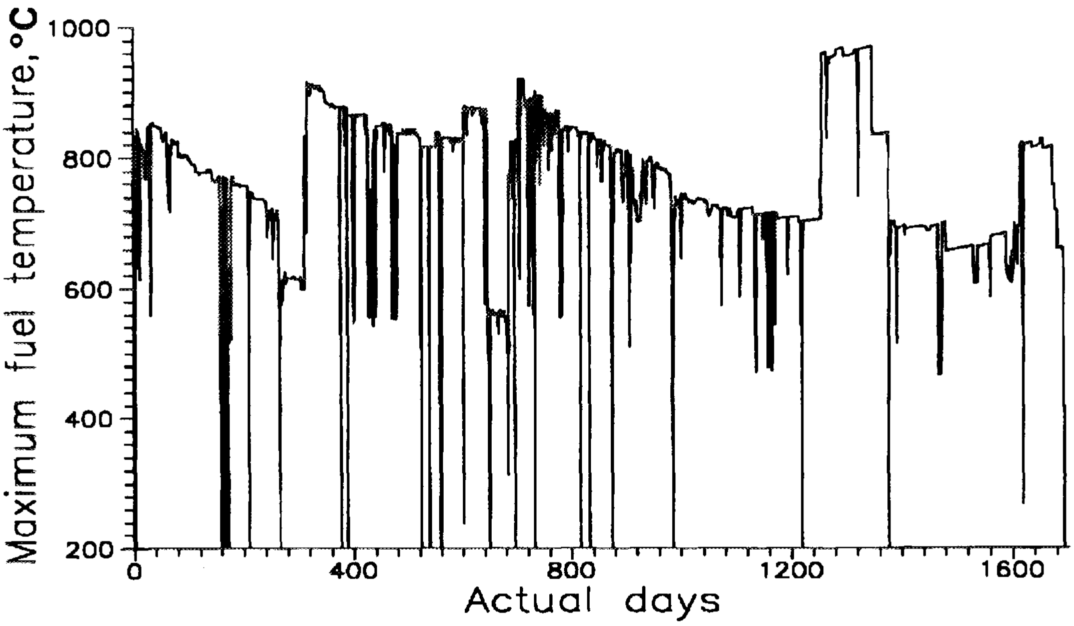

Figure 5 Calculated maximum fuel temperature (assembly # 2) vs. time for case 2 - aver.

Table 1 Calculated values for assembly # 1

	Case 1	Case 2			Case 3	Case 4
variant	aver	max	aver	min	aver	aver
Burn	46.7	52.2	52.2	52.5	54.3	52.2
B_{max}	51.8	57.9	57.8	58.1	60.1	57.8
T_{max}, °C	954	1265	1121	932	1154	1171
T_t, °C	545	649	571	509	583	586
F, %	0.56	3.13	1.17	0.73	1.96	1.53
P_{max}, *MPa*	2.02	3.02	2.63	2.70	3.19	2.87
P_c, *MPa*	0.83	1.11	1.00	1.06	1.17	1.07
dr_f, μm	35		48		53	
dr_c, μm	-27		-32		-34	

Table 2 Calculated values for assembly #2.

	Case 1	Case 2		Case 3	Case 4
variant	aver	max	aver	aver	aver
Burnup	49.6	57.8	57.7	60.0	57.7
B_{max}	55.8	64.9	64.8	67.4	64.8
T_{max}, °C	917	1209	972	1028	1036
T_t, °C	511	631	549	563	565
F, %	0.63	7.88	2.99	5.56	4.00
P_{max}, *MPa*	2.07	5.55	3.60	5.12	4.17
P_c, *MPa*	0.87	2.03	1.41	1.97	1.60
dr_f, μm	44		64	70	
dr_c, μm	-24		-30	-31	

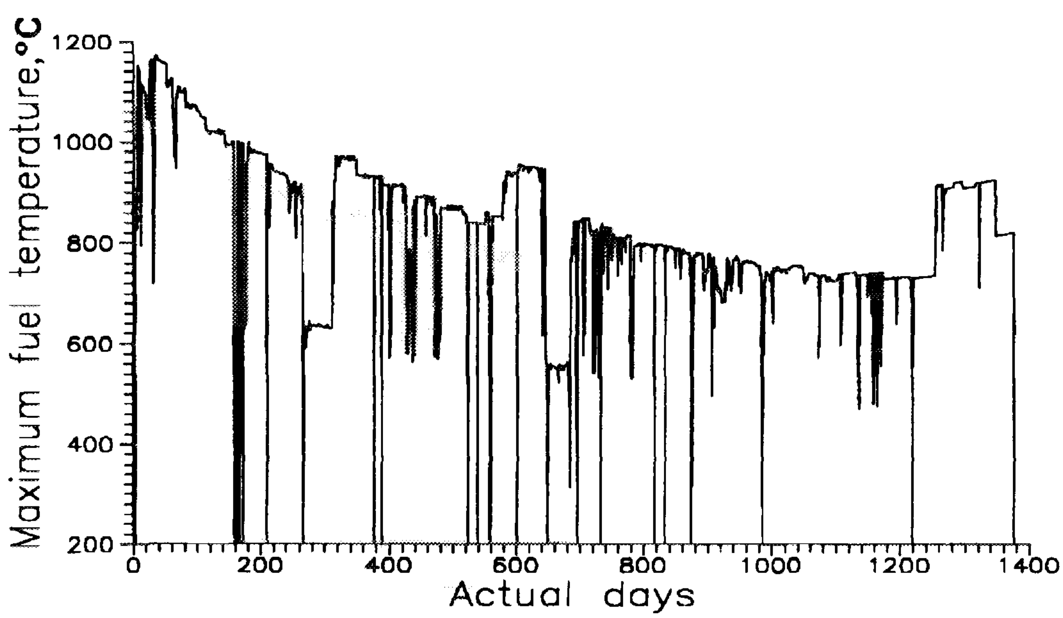

Figure 6 **Calculated maximum fuel temperature (assembly # 1) vs. time for case 4 - aver.**

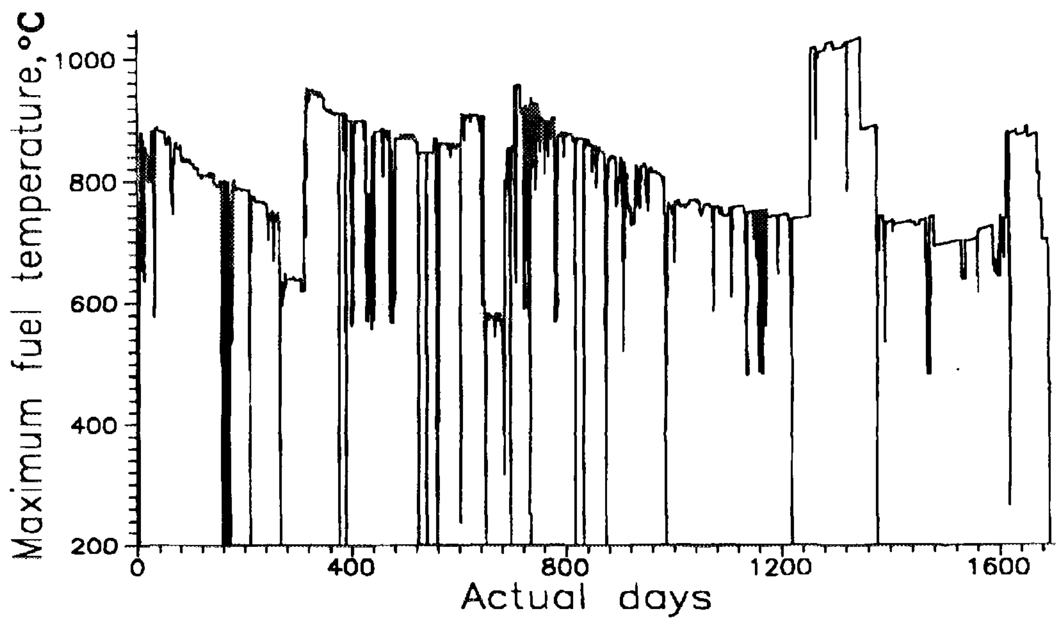

Figure 7 **Calculated maximum fuel temperature (assembly # 2) vs. time for case 4 - aver.**

exceed 4%; maximum gas pressure inside the cladding does not exceed 4 *MPa*; gas pressure inside the cladding at the end of campaign does not exceed 2 *MPa*.

References

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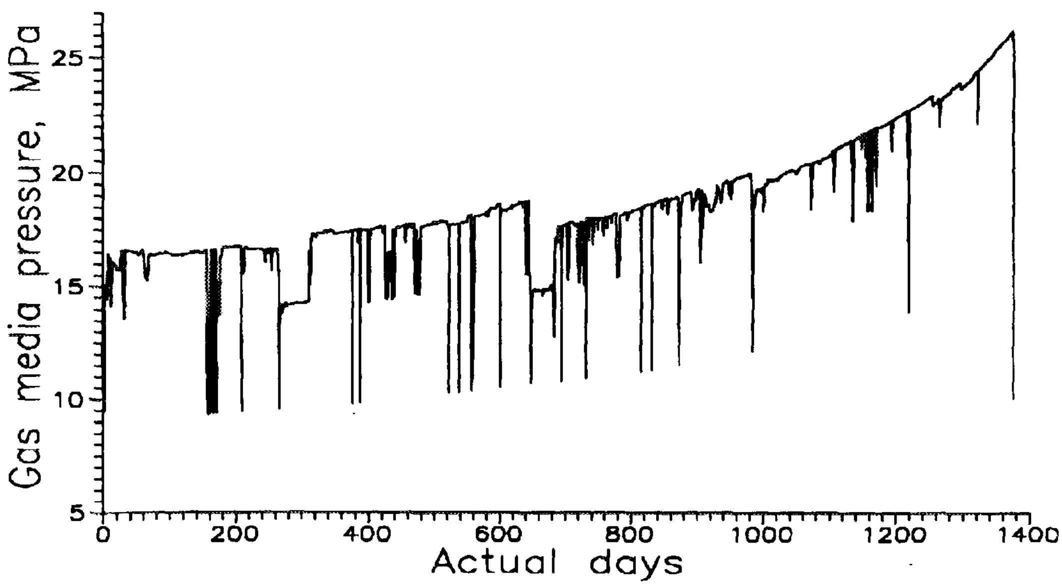

Figure 8 Calculated fuel rod gas pressure (assembly # 1) vs. time for case 2 - aver.

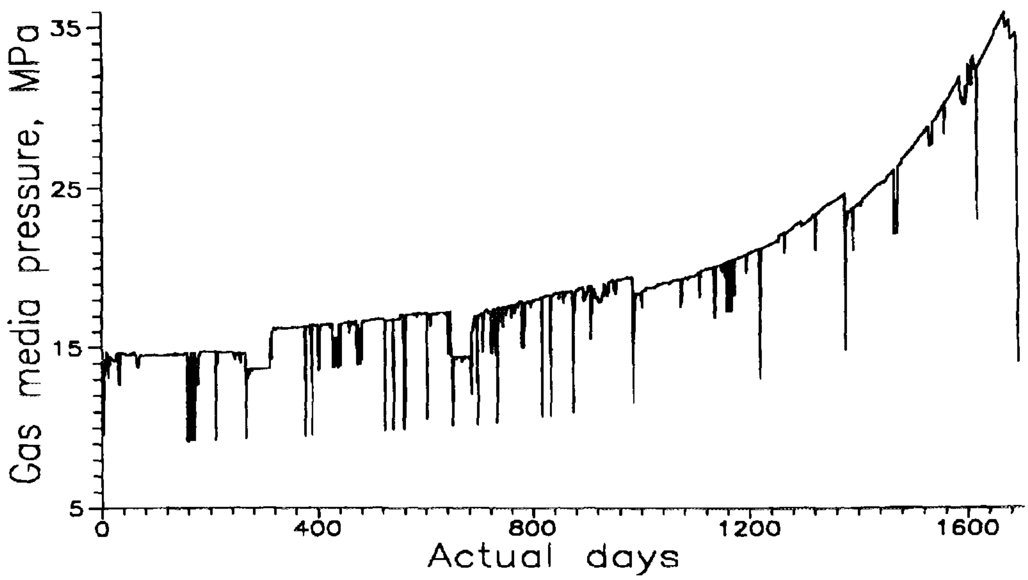

Figure 9 Calculated fuel rod gas pressure (assembly # 2) vs. time for case 2 - aver.

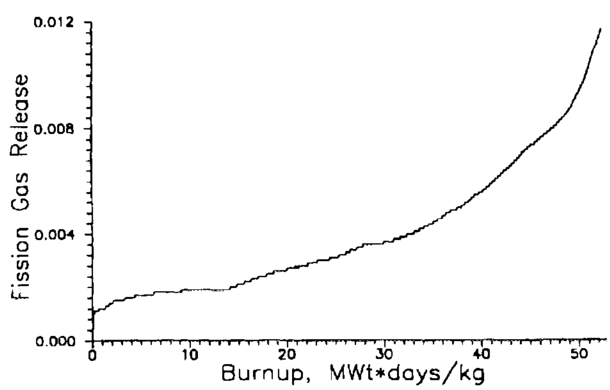

Figure 10 Fission gas release (assembly # 1) vs. burnup for case 2 - aver.

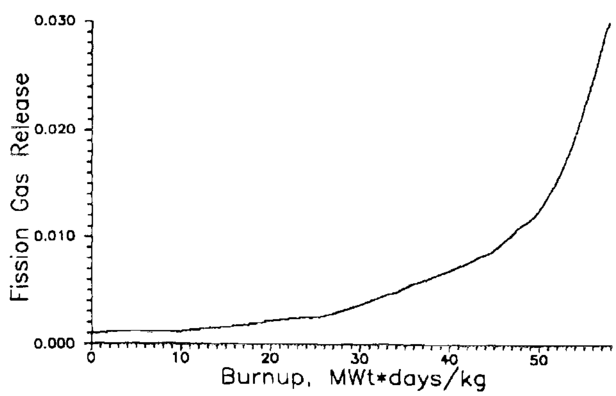

Figure 11 Fission gas release (assembly # 2) vs. burnup for case 2 - aver.