SENSITIVITY OF FUEL ROD BEHAVIOUR ON AS FABRICATED CHARACTERISTICS AND ON OPERATING CONDITIONS

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ABSTRACT

BELGONUCLEAIRE's gradually increasing in-reactor experience has enabled the continuous development and assessment over the years of a coherent set of specifications and drawings for UO_2-PuO_2 and UO_2 fuel for LWR's.

On the basis of this experience, design codes have been developed, benchmarked and are thereafter applied to cover completely the whole range of fuel specifications and irradiation histories. The sensitivity of the fuel rod behaviour on as fabricated characteristics and on operating conditions (steady and transient) is outlined through calculation results of the COMETHE III-J computer code.

INTRODUCTION

BELGONUCLEAIRE's gradually increasing in-reactor experience has enabled the continuous development and assessment over the years of a coherent set of specifications and drawings for UO_2-PuO_2 and UO_2 fuel for LWR's.

The adequacy of the products manufactured according to the resulting set of specifications has been evidenced through the supply of demonstration assemblies and core reloads for power reactors (BWR's and LWR's), their surveillance during irradiation and their performance evaluation by on-site investigations and hot cell post-irradiation examinations. Moreover fresh fuel samples taken from production batches or fuel rods pre-irradiated in the BR 3 plant are irradiated in material test reactors. On the basis of the experience gained, design codes have been benchmarked and are thereafter applied to cover completely the range of parameters and irradiation histories to be encountered or evaluated.

The paper outlines the approach followed by BELGONUCLEAIRE in fuel performance modeling and gives some examples of the sensitivity of fuel rod performance on as fabricated characteristics under steady state and transient (ramping) conditions.

DATA ACQUISITION

Power Reactor Experience

The 35,750 fuel rods, representing the cumulated BN experience and embracing all the designs utilized in LWR's have been introduced in power plants (demo assemblies and reload cores), covering a wide range of power ratings and burnups. The diversity of designs and specifications has enabled to obtain a complete view of the problems involved and to reach adequate solutions. A proportion of the fuel is fully characterized for the purpose of implementing the data base (e.g. BR 3 fuel, cf. Table I).

Test Reactor Experience

Irradiations in material testing reactors are continuously performed to assess particular details of the specifications, to prove the validity of the choice of the characteristics under extrapolated conditions (situations likely to be met but usually not encountered in a power plant or potential future operation modes) to define margins, to investigate accidental conditions (to be considered in the safety evaluation for the licensing procedure) or to complete collection of design data over the full range required for fuel reloads. Table II presents the irradiation classified according to their main objective. Since many irradiations fulfil several objectives simultaneously, the total number of data points is over 400. Table I compares the main characteristics of the Zircaloy clad fuel rods irradiated in BR 3 together with those of the fuel rods of the same specifications irradiated in BR 2.

NEEDS FOR ADEQUATE DESIGN TOOLS

Because of our trial and error approach, the experimental data bases can be used for the design of the fuel rods only through accurate calculation models qualified on their experimental results.

The COMETHE code evaluates the mechanical and thermal behaviour of fuel rods under irradiation. It has been benchmarked on results of Uranium and Plutonium fuels irradiated in thermal conditions as well as in fast reactor conditions. As a result, the code has been qualified and is now utilized by 40 organizations over the world. It includes as input options every single characteristics of the fuel pellet and fuel rod, retrievable from the Quality Control results or available from previous characterizations performed at a process qualification stage. It can therefore assess the effect of any departure from product or process specification.

Other codes and calculation techniques relating to fuel characteristics and behaviour are also applied to perform the required licensing analyses : e.g. the clad collapsing due to initial ovality and creep-down are evaluated by CREBUCK and CUIC ; the effect of local agglomerates of fissile material under transient conditions is assessed by THEATRE 3 and SPARTAN.

SENSITIVITY STUDIES UNDER STEADY STATE CONDITIONS

As a justification of the specification of 10.72 mm outer diameter Zircaloy 4 clad fuel rods typical of the 14 x 14 and 15 x 15 PWR in operation

TABLE I

CHARACTERISTICS OF BN FUEL RODS WITH ZR4 CLADDING MANUFACTURED FOR BR 3

Irradiation in	Total	BR3		BR2 P3	BR2 CEB9	BR3		BR3			BR 2 P5	BR 2 CEB12	BR3	BR3		
Type of rods	U + Pu + Gd	ZO/U	ZO/Pu P + V	ZO/Pu P + V		Z/U	Z/Pu	G/U	G/Pu	G/Gd	G/Pu		g/U	go/U	go/Pu	go/Gd
Number of rods	1.826	216+6	36+1	2	2	112	212	200	243	8	2	2	28	378	304	74
Well charact'd	358	42	37	2	2	28	58	22	41	2	2	2	28	50	44	18
Pie in progress	50	24	19	2	2	-	-	1	2	-	-	-	-	-	-	-
Pie planed	272	42	37	2	2	-	-	2	41	2	2	2	28	50	44	18
Cladding																
OD mm		8.7				8.7		10.75					9.4	9.5		
W/OD %		6.2				6.2		6.3					6.7 - 7.1	6.6		
F_n								0.1					<0.3	de 0,10 à 0.32		
UTS (20°C) kg/mm²								80					annealed	str.rel. >60 & ann. >50		
Ductility (20°C) %								17						>17 >22		
OD surface treatment		autoclaved						autoclaved						pickeled B ground		
Springs		Inconel X				Inconel X		carbon steel						carbon steel		
Pellets																
O/M		2,00 ± 0.02		P1,991;V1,999				2.00 + 0.015 - 0.005	1.99 + 0.02 - 0.01	2.00 + 0.015 - 0.005	1.99 + 0.02 - 0.01		2.0 + 0.015 - 0.005	2	2.0 + 0.01 - 0.02	2
Immersion density TD %		90		91		90	90	93 ± 1.5	92 - 93	93 ± 1(bulk)	89 - 91		9? ± 1	94.5 ± 1.5 (6>96(8rods)	94.0 ± 1.5	94.5 ± 1.5
Dish volume (v/o per dish)		1. - 1.5		1.0		1.0	1.5	1.0		0.5	1		0.5	0.75		
L/D		1.37		1.375		1.34		1.4	1.46		1.43			1.5		
Gas mm³/g		<50		P 20 ; V 90			10 - 100	<50	200 & 700	<60	<100		<60	<40		<40
C ppm		P < 60 ; V 40					20 - 60	<100	20 - 160	<100	<100		<100	<150	<150	<150
N "		<100		P 40 ; V 10			10 - 100	<30	50 - 90	<60 N^{tot}	<30		<60 N^{tot}	<30	<75	<30
Cl(+ F) "		<10		P 8 ; V 20			2 - 8	<20	10 & 50	<40	<20		+ F,<40	<40	<40	<40
H "		<60					0.7 - 5	<10 (excl. H_2O)	10 & 40 (incl. H_2O)	<10 (excl. H20)	<27 (incl. H20)		<10 (incl. H20)	<5 (incl. H_2O)	<5 (incl. H_2O)	<5 (incl. H_2O)
H_2O "		<20		P 30 ; V 50		<15	≤9 - 15	<20	2 - 15	<15	5 & 24		<15	<4	<10 (*)	<4
Fuel rod																
Initial cold diameter gap um		120 & 180		100 - 210		150		230	170 & 235	230	230			200	260	200
Filling gas kg/cm²		1				20	1	20	1 & 20	20	1		20	20	20 & 1	20
He %		He/P & He + Ar/V							≥90		≥90		≥90	≥90	grade A	≥90
Ar "									≤5		≤5		≤5	≤5	-	≤5
H "							1 - 13		3							
Irradiation																
Peak irradiation condition (W/cm)		320	330	820	460	350	330	580	600		700	600	470	470	490	530
(W/cm²)		115	119			130	120	170	180		207	178	160	157	164	178
Power tilt		1.1		1.1	1.0	1.11		1.21		1.20	1.1	1.0	1.16	1.14	1.15	1.14
Peak pellet burnup (GWd/tM) target		60	50	60	95	56	64.5	50	60	40	60	60	37	39.4	43	40
actual		60	50	65	95	[illegible]	61.6	56.1	56.1	27	56	56	44.9	47.3	51.6	48.0
Number of power cycles																
- target																
- actual																
Days at power (EFPD)		824	640	808	790	1018	1018	1018	1018	[illegible]	595	611	684	500	500	500

(*) including proportional rest gas and moisture of pin internals.

TABLE II

FUEL RODS IRRADIATED IN TEST REACTORS

Clad	Fuel	Main purpose						Total
		Heat transfer	Densification	Specification limits	Burnup	Power changes	Fission gas release	
SS	Pu	39	–	–	13	2	14	68
Zr 4	U	6	–	–	–	1	–	7
Zr 4	Pu	–	24	2	4	–	–	30
Zr 2	U-Gd	4	–	–	1	–	–	5
Zr 2	Pu	12	16	6	–	3	–	37
Total		61	40	8	18	6	14	147

in Belgium, various parameters were investigated : rod length, plenum volume, cladding thickness and anisotropy, pellet clad diametral gap, pellet density, densification behaviour, grain size, pre-pressurization, power rating histories.

On the basis of core configurations and most likely assembly reshuffling patterns, various possible rod histories (Figure 1) have been considered in order to select the worst conditions with regard to the following design criteria : pellet clad mechanical interaction, maximum fuel temperature (LOCA) and maximum inner gas pressure. Calculations have been performed for the three power rating histories and mean core power of 226 W/cm, first with nominal rod characteristics and then with the worst combination of tolerances [1].

POWER RAMPING PARAMETRIC STUDY

The behaviour of a 15 x 15 TIHANGE type fuel rod has been evaluated during transient reactor operation like power ramp at reactor start-up. The characteristics of the studied fuel rod are listed in Table III. The fuel rod has been assumed to be irradiated in low power rated core zone during the two first cycles and shuffled in a higher power rated core zone for the third cycle irradiation. The reference power history (q') is plotted in Figure 2A. The same figure shows the evolution of the fuel central temperature (Tc), the fractional fission gas release (f) and the inner gas pressure (Pg). The clad mechanical response is exhibited in Figure 2B, i.e. the equivalent stress (σ_{eq}), the contact pressure (Pc) and the hoop and axial creep strains ($\varepsilon_{\theta c}$, ε_{zc}).

The strong interaction between the expanding fuel and the Zircaloy clad at BOC 3 induces tensile stresses so that the equivalent stress exceeds the threshold stress limit for SCC adopted as design criterion [2]. Such a situation may not be tolerated as the integrity of the fuel rod is endangered. Therefore, power increase rate limitations have to be imposed to enable the clad to accommodate the fuel thermal expansion by progressive creep. The impact of different power ramps (Figure 3) on the stress-strain cladding response has been investigated at starting of cycle 3. The ramps ③ and ④ may be accepted as the threshold stress for SCC is not exceeded whislt the ramps ① and ② induce too high stresses and so have to be rejected. It is generally believed that steady state power periods are essential to enable stress relaxation by clad creep. The comparison of calculation results obtained for ramps ③ and ④ demonstrates that steady state power periods are not required and can even be detrimental. During the steady period, burn-up is accummulated and the fuel swells. Although being very low, this swelling is significant enough to have an effect on the pellet-clad interaction. In addition, the gain in energy by considering the ramp ③ instead of ramp ④ is more than 50 %, which is a decisive consideration.

CONCLUSIONS

Experimental data accummulated over the last 18 years cover fuel in a large range of as fabricated characteristics, power ratings and burn-up for both UO_2 and UO_2-PuO_2 fuels.

The sensitivity of design tools to fuel characteristics included in the specification allows to justify the selected nominal characteristics and to assess permissible tolerances. Their sensitivity to the operating conditions allows for a better understanding of the operational restrictions, in particular during power ramp when returning to full power after a refuelling shut-

TABLE III

Fuel Rod Characteristics

Array = 15x15

Clad O.D. = 10.72 mm

Clad thickness = 0.62 mm

Diametral gap = 190 µm

Fuel bulk density = 93.5 % TD

Active length = 3642 mm

down or after a period of operation at a lower power level.

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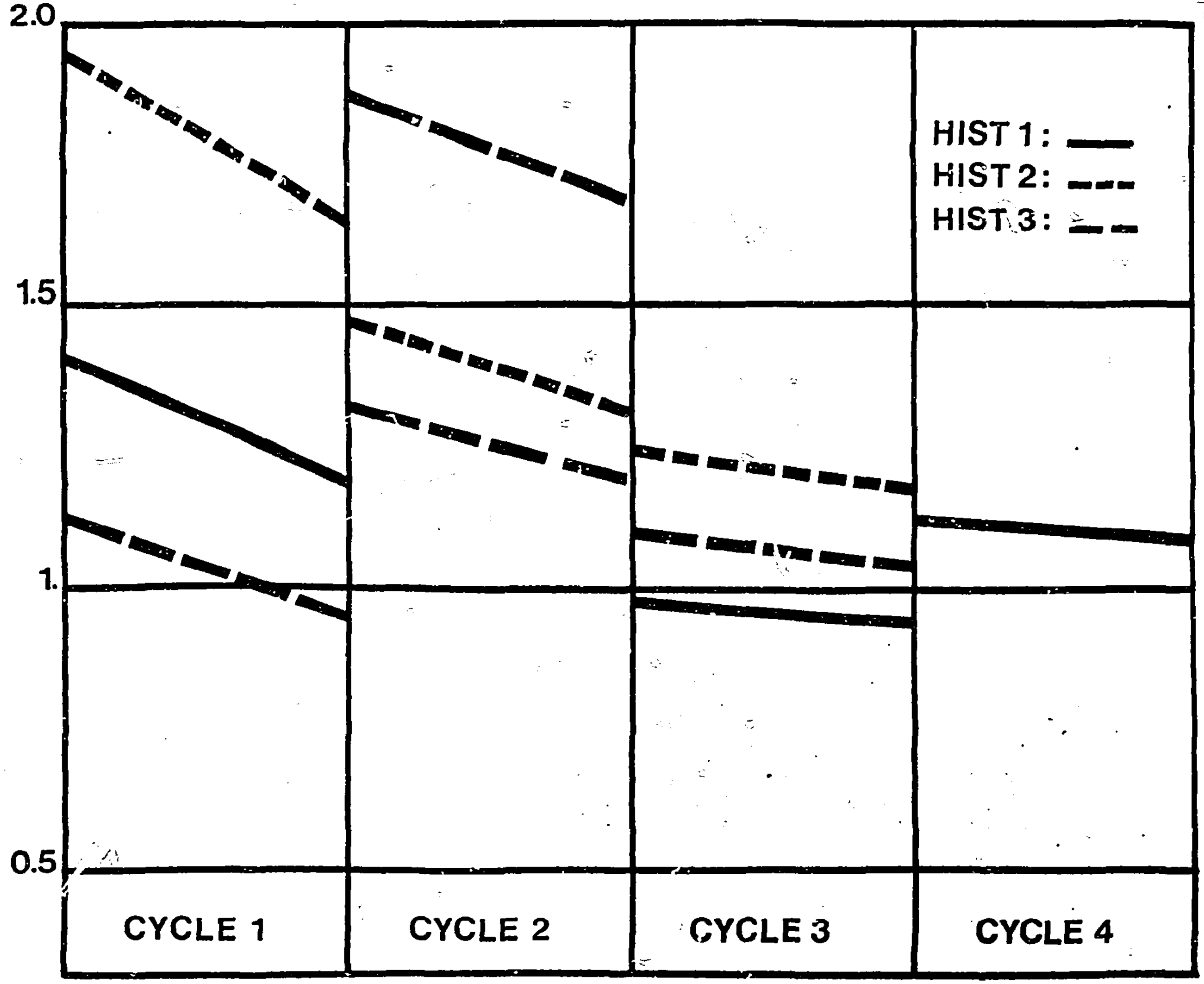
2. P. BOUFFIOUX, J. VAN VLIET, P. DERAMAIX, M. LIPPENS, "Potential Cause of Failures Associated with Power Changes in LWR's", Paper presented at the KTG/ENS/JRC Meeting held at Petten (Holland), November 30 - December 1,1978.



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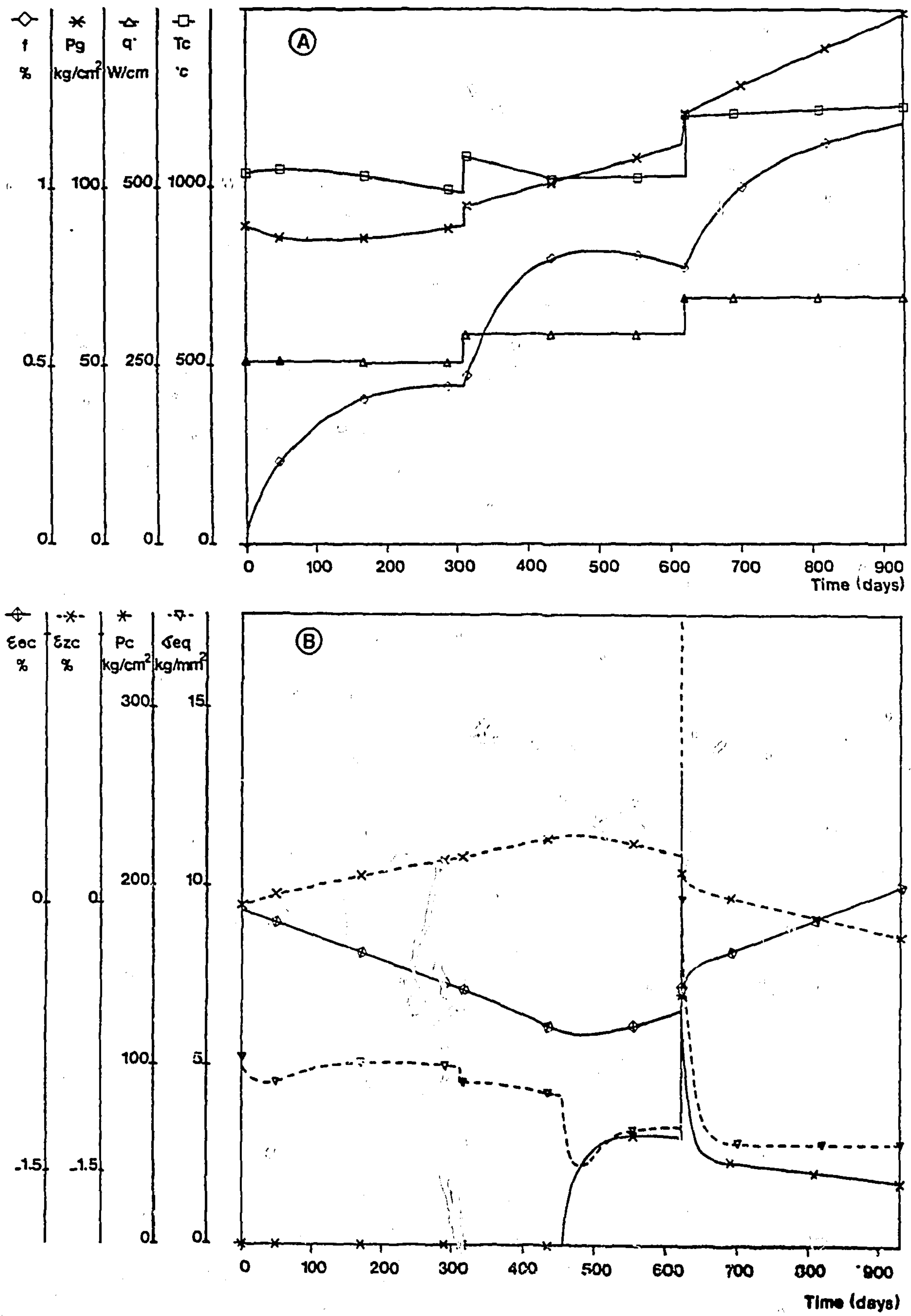
POWER HISTORIES **FIG.1**



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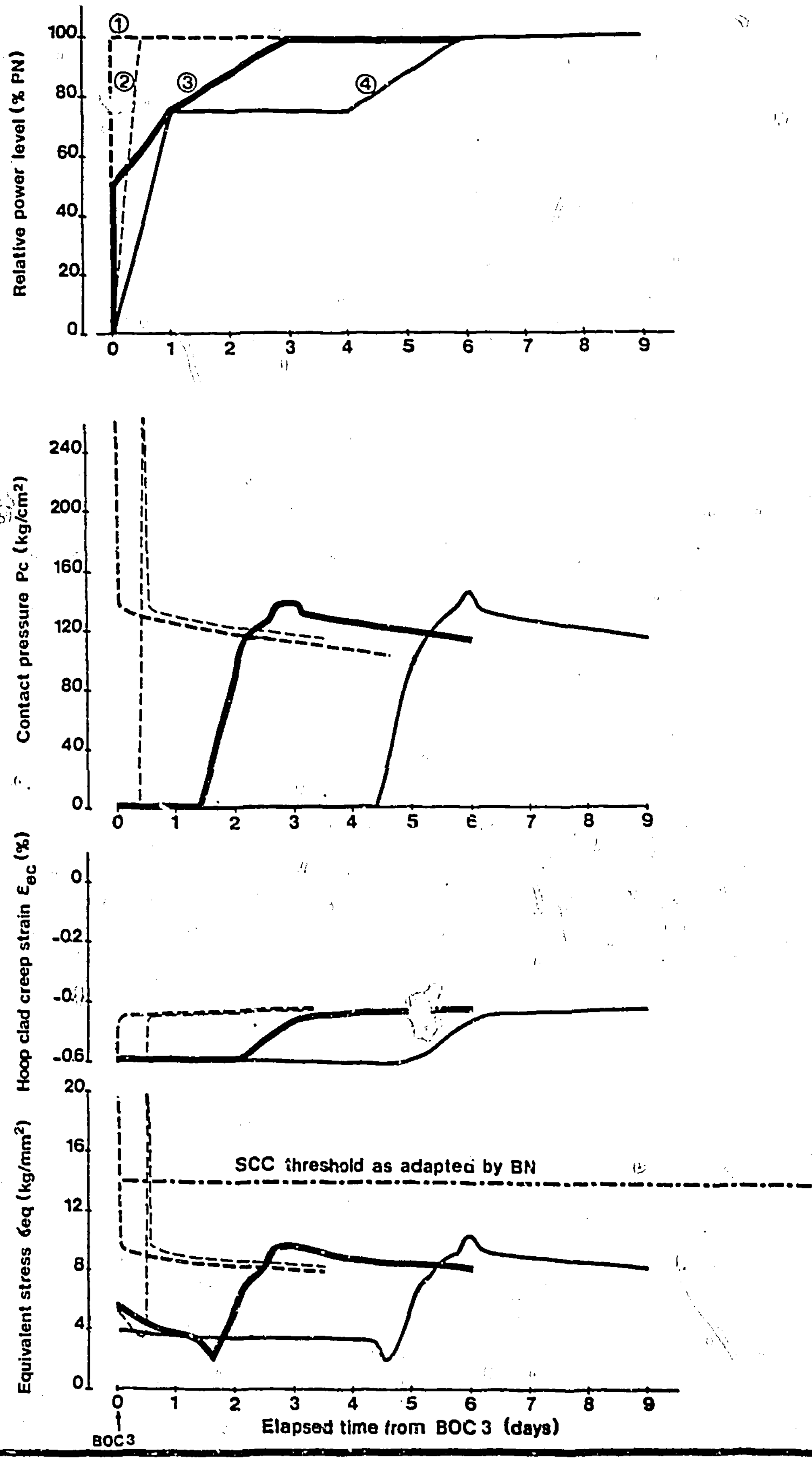
15x15 FUEL ROD Power ramp analysis
Reference case

FIG.2



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Starting-up at beginning of cycle 3

Pellet-clad interaction at ridge location (Maximum)

FIG.3

Comparison of clad mechanical responses to different power ramps