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Mixed-oxide (MOX) fuel performance benchmark

Summary of the results for the PRIMO MOX rod BD8

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Work carried out within the

Expert Group on Reactor-based Plutonium Disposition (TFRPD)

Under the auspices of the OECD/NEA

Working Party on Scientific Issues in Reactor Systems (WPRS)

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Foreword

The OECD/NEA Nuclear Science Committee has established an expert group that deals with the status and trends regarding reactor physics, nuclear fuel performance and fuel cycle issues related to the disposition of weapons-grade plutonium as mixed-oxide (MOX) fuel. The objectives of the group are to provide NEA member countries with up-to-date information on, and to develop consensus regarding, core and fuel cycle issues associated with weapons-grade plutonium disposition in thermal water reactors (PWRs, BWRs, VVER-1000s and CANDUs) and fast reactors (BN-600s). These issues concern core physics, fuel performance and reliability, and the capability and flexibility of thermal water reactors and fast reactors to dispose of weapons-grade plutonium in standard fuel cycles.

The activities of the NEA Expert Group on Reactor-based Plutonium Disposition are carried out under the auspices of the NEA Working Party on Scientific Issues of Reactor Systems (WPRS). A major component of these activities is benchmark studies.

MOX fuel behaviour benchmarks finalised or in progress are as follows:

- Halden Reactor Project (HRP) MOX fuel irradiation experiment benchmark (completed);
- Belgonucléaire and SCK•CEN PRIMO ramped MOX fuel rod performance benchmark (finalised with this publication);
- United States Department of Energy weapons-grade MOX fuel irradiation experiment irradiated at the advanced test reactor (ATR) benchmark (started);
- Kurchatov Institute MOX fuel rod behaviour in fast power pulse conditions (started).

The following benchmarks relative to the reactor physics activities of the expert group are completed or in progress:

- VENUS-2 MOX core benchmarks, carried out jointly with the WPRS (completed);
- VVER-1000 LEU and MOX computation benchmark (completed);
- KRITZ-2 benchmarks, carried out jointly with the WPRS (completed);
- benchmark using dosimetry data from the VENUS-2, MOX core experiments (completed);
- VVER-1000 in-core self-powered neutron detector calculation benchmark (started);
- VENUS-7 weapons-grade MOX core benchmark (started).
- VENUS-9 weapons-grade MOX core benchmark (started).

It should be noted that the PRIMO data was compiled, reviewed and integrated into the International Fuel Performance Experiments (IFPE) database, including the detailed comparative tables from the present study. Inquiries about the availability of the full data should be addressed to programs@nea.fr.

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Abstract

Within the framework of the Expert Group on Reactor-based Plutonium disposition (TFRPD) within the Working Party on Scientific Issues in Reactor Systems (WPRS), a fuel modelling code benchmark test for MOX fuel was initiated, with irradiation data on the MOX rod BD8 (programme PRIMO) provided by the Studiecentrum voor Kernenergie • Centre d'Études de l'Énergie Nucléaire (SCK•CEN) and Belgonucléaire (BN). This report summarises the provided data and fuel characteristics for the irradiation, and presents the calculation results provided by the contributors.

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Chapter 1: Introduction

1.1 PRIMO MOX fuel benchmark

The PWR Reference Irradiation of MOX Fuels (PRIMO) programme was started in October 1986. It was jointly organised by the Studiecentrum voor Kernenergie • Centre d'Études de l'Énergie Nucléaire (SCK•CEN) and Belgonucléaire (BN) and was co-sponsored by ten participants including fuel vendors, utilities, nuclear centres and national authorities. The PRIMO programme sought to investigate MOX fuel, with the following major objectives:

- irradiation of MOX fuel rods to different burn-up stages, following power histories representative of those of PWR power plants, to determine their behaviour as far as their mechanical, thermal and neutronic properties are concerned;
- execution of a ramp test programme, to determine the failure threshold of MOX fuel rods and to obtain mechanical and thermal data under ramp conditions;
- fast power transients on a MOX rod to simulate a class II incident in a PWR.

Non-destructive examinations were performed on all the rods of the programme after base irradiation and, when appropriate, after a subsequent power excursion test. All the non-failed rods and rodlets were then punctured.

Destructive tests consisting of ceramographies, density measurements, EPMA, SIMS and thermal diffusivity measurements were performed on selected rods of the programme in five laboratories:

- SCK•CEN-Mol for puncture, ceramography and density measurement;
- CEA-Saclay for puncture, ceramography and EPMA;
- AEA Technology-Windscale for ceramography, density and thermal diffusivity measurements;
- PSI-Würenlingen for EPMA and SIMS measurements;
- STUDEVIK for the examinations of the rodlets (from rod SL).

BN and SCK•CEN provided the PRIMO data (fabrication and irradiation) for rod BD8 for use by the Expert Group as a MOX fuel performance benchmark. The full benchmark specification is provided in Appendix A.

1.2 Rod BD8 characteristics

Fuel rod BD8 was manufactured at BN in 1983, using the Micronized Master Blend (MIMAS) process presently used for industrial production of LWR MOX fuel. The use of the Master Blend principle leads to fuel presenting some Pu-rich particles, which can locally reach the content of ~30% PuO₂ of the primary blend, depending on the size of the agglomerates and on the U-Pu interdiffusion during sintering.

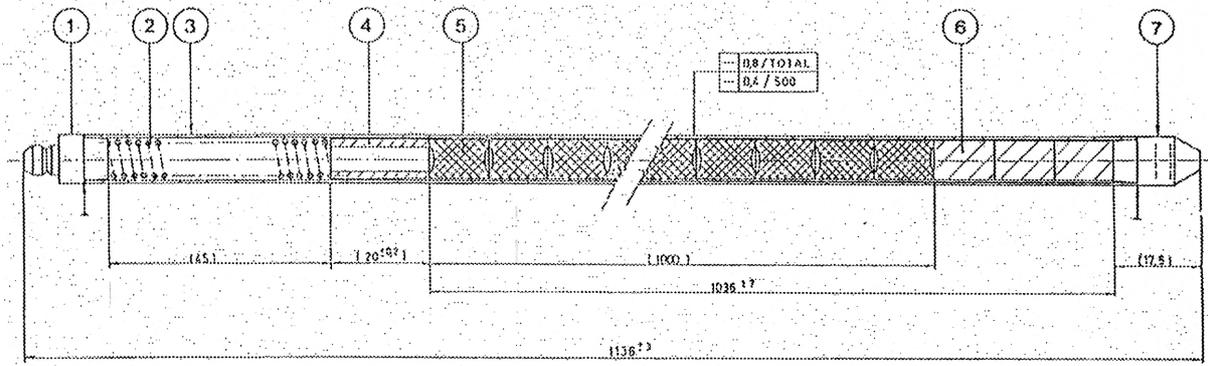
The general schematic of rod BD8 is illustrated in Figure 1.

Rod BD8 had an overall length of 1 136 mm with a MOX fuel stack height of ~1 005 mm. The comprehensive fuel, cladding and fuel rod characteristics are presented in Appendix A.

Figure 1: General layout of rod BD8

(1) upper end plug, (2) hold down spring, (3) cladding tube, (4) spacer tube, (5) MOX fuel pellets, (6) UO₂ blanket pellets, (7) lower end plug

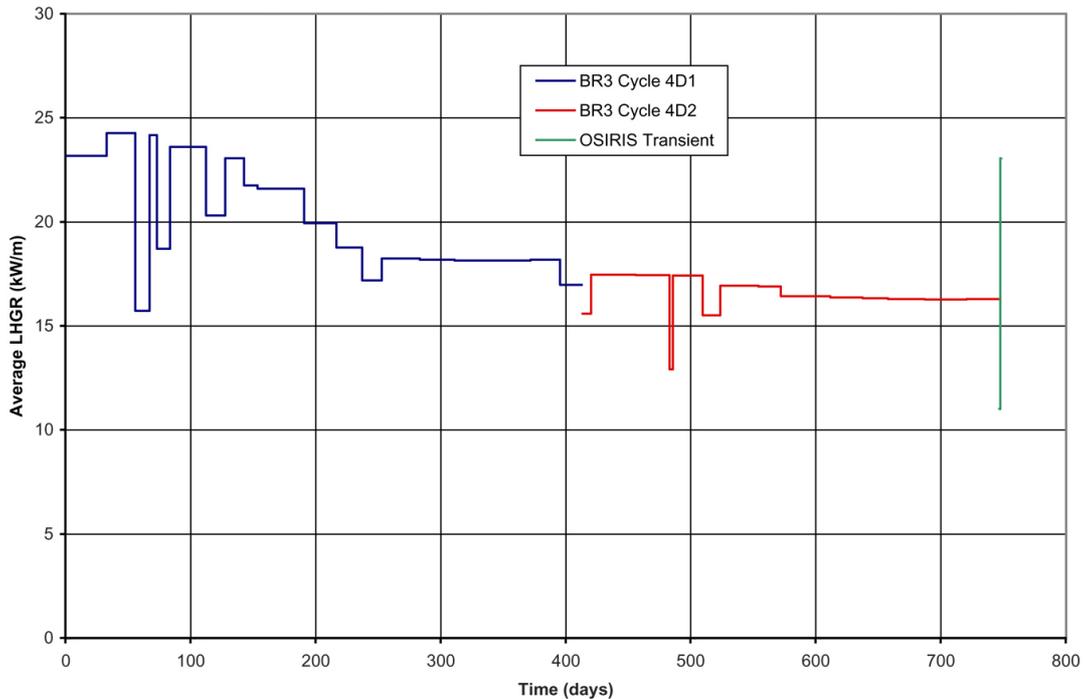
Dimensions indicated in mm



1.3 Rod BD8 irradiation data

Fuel rod BD8 was base irradiated in the BR3 reactor of SCK•CEN for cycles 4D1 and 4D2. During cycle 4D1 (from 13/07/1984 until 11/11/1985, 413 days at power) the rod accumulated an average burn-up of 18.0 GWd/tM, corresponding to a peak pellet burn-up of 22.9 GWd/tM (calculated values). During cycle 4D2 (from 03/07/1986 until 30/06/1987, 333 days at power) the average rod burn-up reached 30.1 GWd/tM, corresponding to a peak pellet burn-up of 38.0 GWd/tM (calculated values). Average rod power during the base irradiation for rod BD8 is illustrated in Figure 2.

Figure 2: Rod BD8 average linear power during base irradiation in the BR3 reactor in cycles 4D1 and 4D2



After base irradiation in the BR3 reactor, the BD8 rod was transported to CEA-Saclay for irradiation in the OSIRIS reactor (ramp power excursion). The irradiation device used for the PRIMO ramps was the ISABELLE 1 loop, installed on a movable structure of the core periphery. The power variations were obtained by inwards/backwards movements of the loop in the core water.

The preconditioning phase for rod BD8 occurred at a peak power level of 189 W/cm with a hold time of 27 hours. The subsequent power excursion rate amounted to 77 W/(cm min), reaching a terminal peak power level of 395 W/cm that lasted for 20 hours. The ramp power excursion for rod BD8 in the OSIRIS reactor is illustrated in Figure 3. The accuracy of the determination of the peak pellet linear power is 6% for 1 σ confidence level.

Figures 2 and 3 show the average rod power for BD8, and the reactor(s) core power shapes (that is, the axial power peaking factors) for the base and ramp irradiations are given in Figure 4.

The detailed rod power histories (cycles 4D1 and 4D2) and the BR3 and OSIRIS reactor operating conditions during the irradiations are tabulated in Appendix A.

1.4 Contributing organisations and codes for the rod BD8 benchmark exercise

The following countries and organisations, and the respective fuel performance codes that they employed, have provided contributions to the MOX fuel rod BD8 benchmark exercise.

Organisation	Fuel modelling code
KAERI (Korea)	COSMOS (Lee, 2007)
Kurchatov Institute (RF)	FRED (Mikityuk, 1999)
NNL (UK)	ENIGMA (White, 1991)
ORNL (USA)	FRAPCON-3 (Lanning, 1997) TRANSURANUS (Lassmann, 1992)
PSI (Switzerland)	FALCON (Rashid, 2005)
SCKCEN (Belgium)	FEMAXI-V (Suzuki, 2000)
VNIINM-Bochvar (RF)	START-3 (Medvedev, 2001)

1.5 Rod BD8 PIE

Rod BD8 was not instrumented. The available results for the rod BD8 post-irradiation examinations (PIE) are presented in Appendix A.

Figure 3: Ramp power excursion for rod BD8 in the OSIRIS reactor

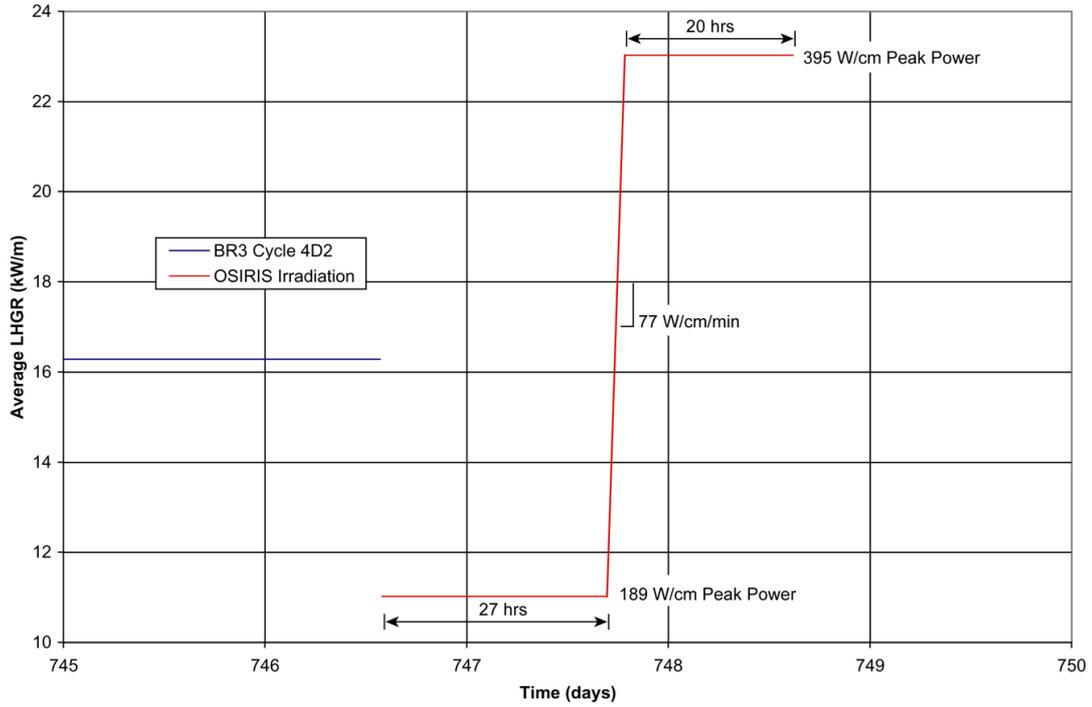
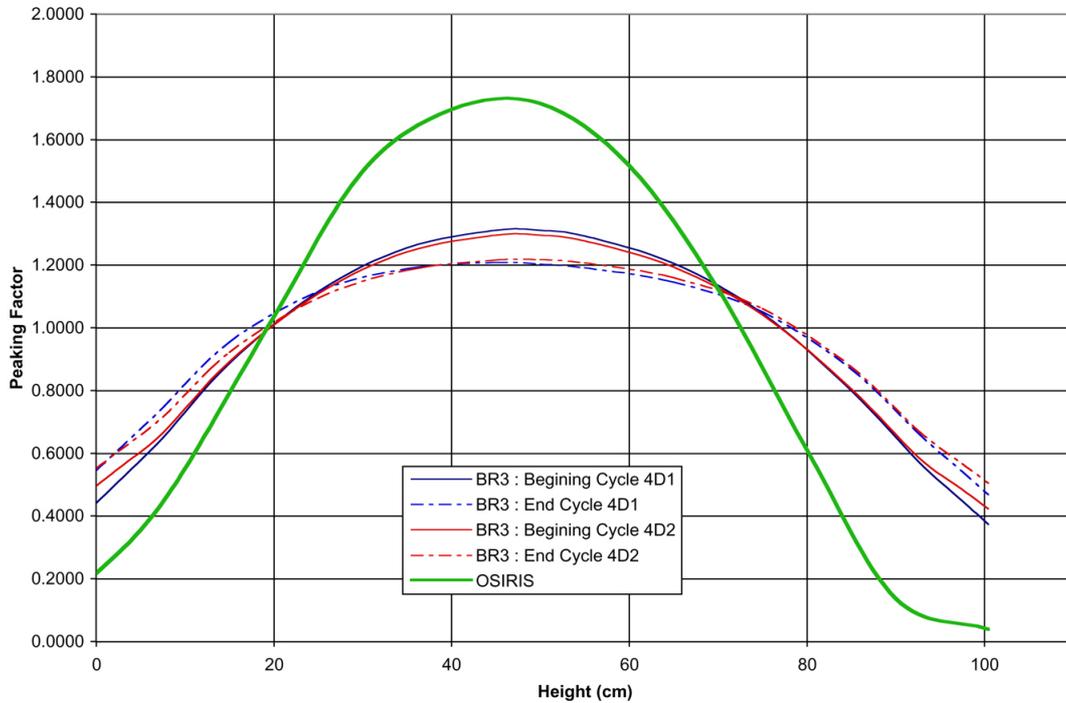


Figure 4: Core power shapes for the BD8 irradiation in the BR3 and OSIRIS reactors



Chapter 2: Calculation results

Rod BD8 was not instrumented; therefore, the calculated results (fuel rod centreline temperature and rod pressure) of the benchmark participants are presented as code-to-code comparisons. Puncture results are available from the PIE of BD8 (and a sister rod which did not undergo the power ramp); thus the predicted fission gas release (pre- and post-ramp) will be compared with the puncture data.

2.1 Fuel burn-up

A comparison of the calculated average burn-up for rod BD8 is presented in Figure 5. Also shown in this figure is the fuel burn-up computed by SCK•CEN using CONDOR and the PIE determination of burn-up. In Figure 5, the CONDOR calculation also includes the maximum rod burn-up (upper error bar at EOL); and the FRAPCON-3 simulation shows (at EOL) the minimum-mean-maximum fuel burn-up.

There is excellent agreement with all the participant simulations, the SCK•CEN experimental analyses (with CONDOR), and the PIE results. Basically, the calculated rod average burn-ups indicate that all benchmark participants are modelling the rod geometry and heating histories (including the axial power distribution) correctly.

2.2 Rod midplane centreline temperature

The calculated fuel midplane centreline temperatures (prior to the ramp) are illustrated in Figure 6. Through ~253 days of irradiation, all code predictions are within a ~125°C band; after 253 days, the Russian Federation (RF) codes (FRED and START-3) yield results that are higher and diverge from the remaining codes (whose results are within a ~100°C band). The ENIGMA prediction is consistently the lowest of the predictions. Prior to the ramp at the end of cycle 4D2, the RF codes predict fuel temperatures of 1 130°C (FRED) and 1 175°C (START-3), while all other code predictions are within 960-1 062°C. The peak fuel temperatures during the ramp phase of the test are given in Figure 7; except for the START-3 prediction (highest) and the ENIGMA prediction (lowest), all code predictions are within a ~250°C band.

2.3 Rod internal pressure

The calculated fuel pin internal pressures (pre-ramp) are given in Figure 8. There are two distinct groupings with ~1 MPa difference in the predictions.

2.4 Rod fission gas release

Figure 9 shows the calculated and experimental FGR for Rod BD8. For the base irradiation in BR3, the experimental FGR value is 0.5% (from a sister rod of BD8 which did not undergo the ramp phase). The RF code predictions for the FGR during the base irradiation in BR3 are all greater than 2%, and the FALCON results are ~1.5%; all other code simulations are close to the experimental value. The PIE of BD8 (after the ramp) yielded a FGR of 11.2%. The START-3 and FALCON (second simulation) predict the highest values of 15.4 to 16.2%; the lowest prediction is that of TRANSURANUS at 4.7% with ENIGMA at 7.2%. All other code predictions are within the range of 9.1 to 13.7%.

Figure 5: PRIMO Rod BD8 calculated average fuel burn-up

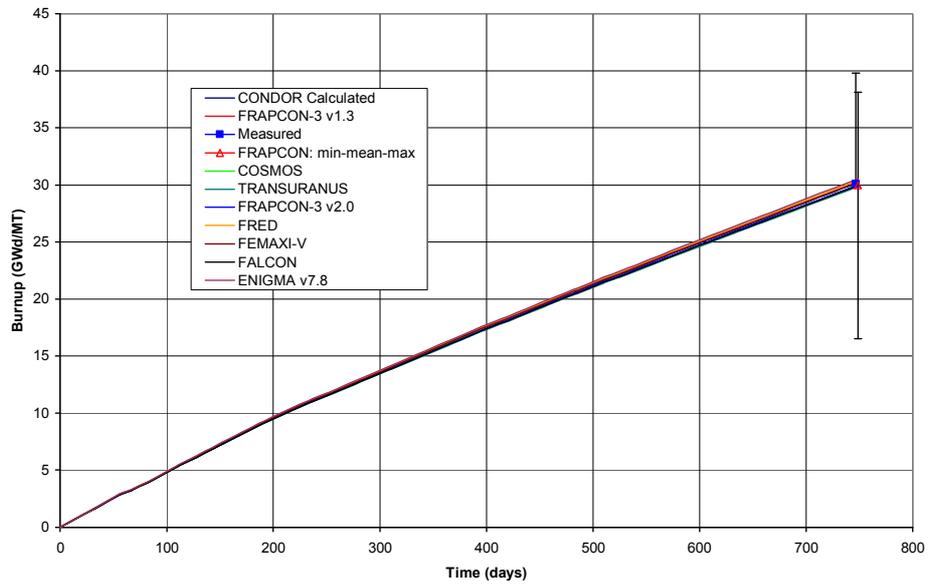


Figure 6: PRIMO rod BD8 calculated fuel midplane centreline temperature, prior to ramp

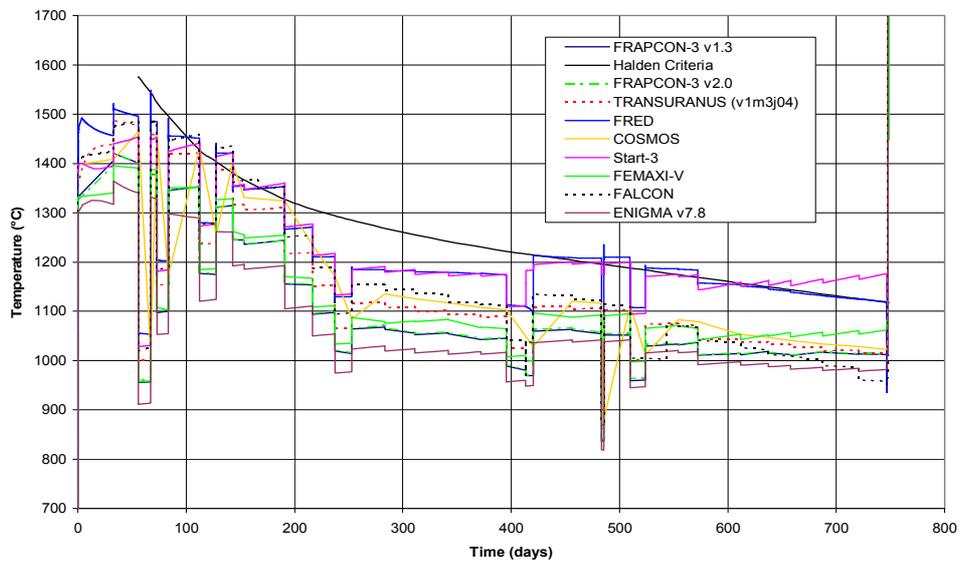


Figure 7: PRIMO rod BD8 calculated fuel midplane centreline temperature during ramp

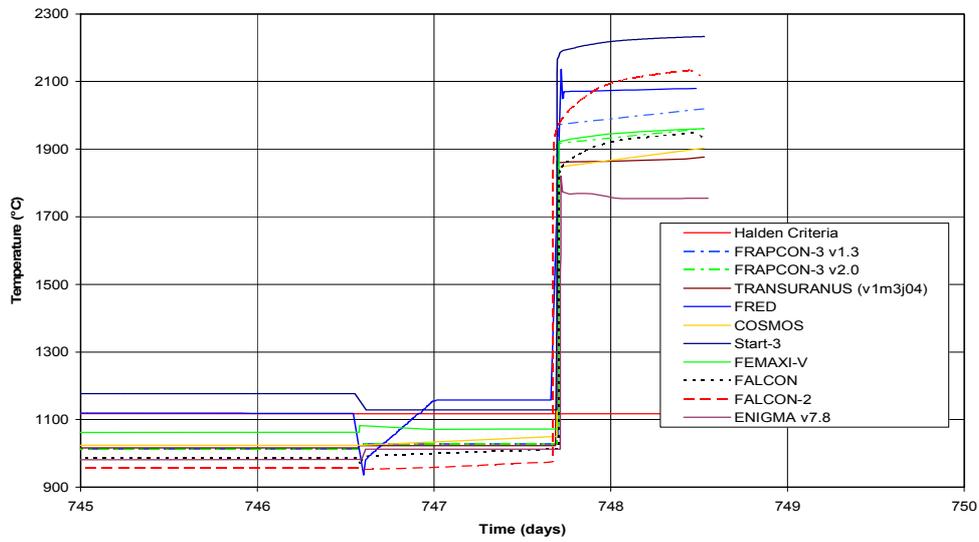


Figure 8: PRIMO rod BD8 calculated fuel pin internal pressure

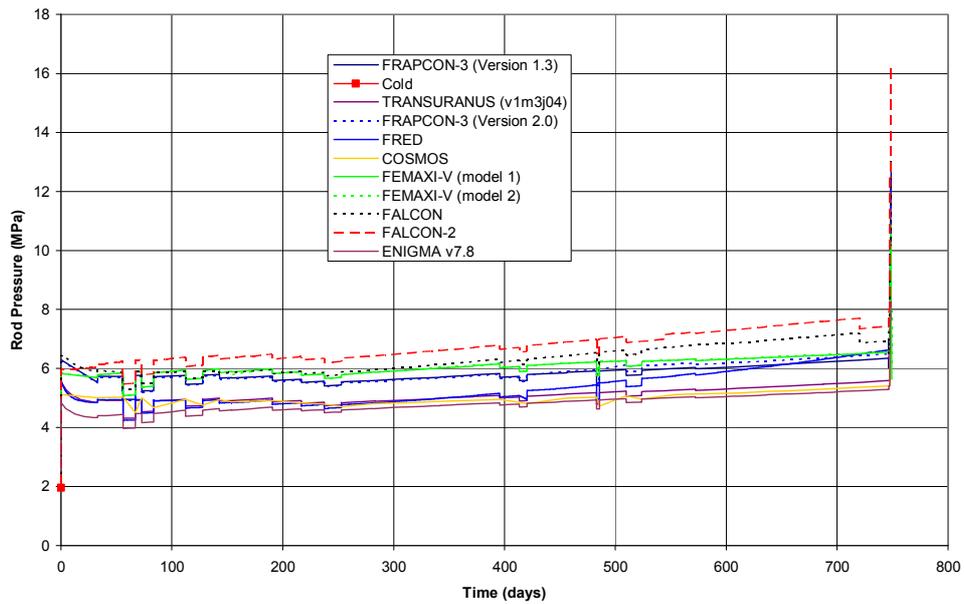
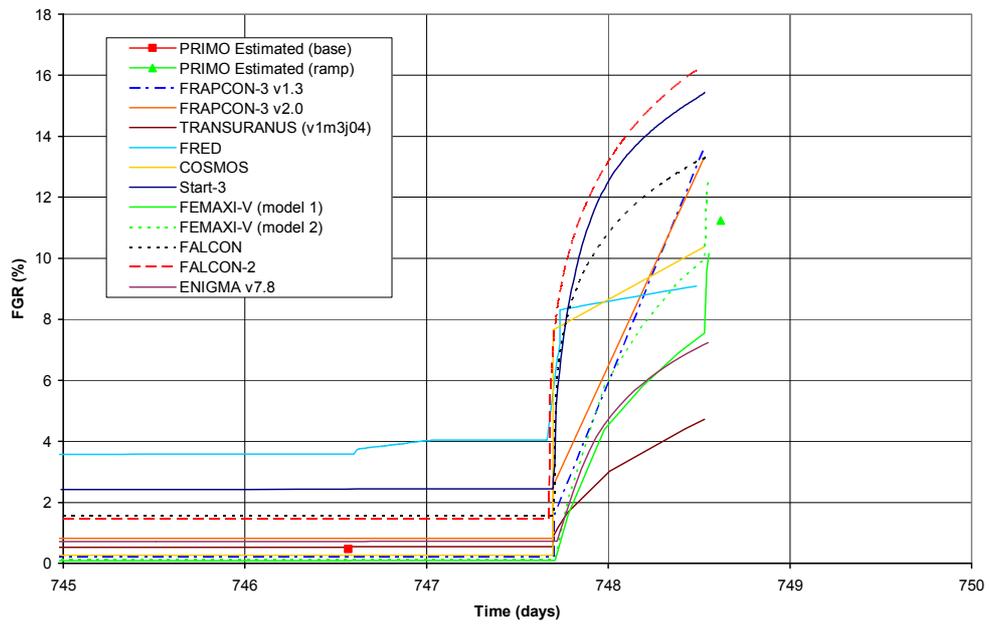


Figure 9: PRIMO rod BD8 experimental and calculated fuel fission gas release



Chapter 3: Observed code prediction differences

Generally in a fuel performance code, the models that can most significantly impact the fuel pin simulation are the fuel thermal conductivity and the fission gas release (FGR) models. FGR feedback via lowered gap conductance affects the fuel pin thermal response and, in turn, the fuel thermal condition is a significant parameter in the FGR model.

The codes exercised by the participants in these benchmark problems employ a diverse set of thermal conductivity and FGR models, as shown in the following table.

Code	Thermal conductivity model	FGR model
COSMOS	Heterogeneous	UO ₂
ENIGMA	0.92 * k _{UO2}	UO ₂
FEMAXI	Baron	UO ₂
FRAPCON-3	Duriez/NFI	Massih/Forsberg
FRED	MATPRO	FASTGRASS
START-3	Duriez	Two-stage diffusion
TRANSURANUS	Duriez/ITU	URGAS

In addition, fuel models such as: 1) relocation; 2) densification; 3) swelling, are also varied amongst these codes. Even before significant FGR impacts the predicted thermal response, these codes will predict a range of thermal responses just because of the differences in the thermal conductivity models and the relocation-densification-swelling models. After appreciable FGR (which has wide variations in predictions as shown in this benchmark), the thermal predictions are further perturbed.

The observed ranges in the predicted thermal and FGR responses are reasonable given the variety and combination of thermal conductivity and FGR models employed in these codes.

Chapter 4: Summary

The OECD/NEA Nuclear Science Committee has established an Expert Group that deals with the status and trends of reactor physics, nuclear fuel performance, and fuel cycle issues related to the disposition of weapons-grade plutonium as MOX fuel. The activities of the NEA Expert Group on Reactor-based Plutonium Disposition are carried out in close co-operation with the NEA Working Party on Scientific Issues in Reactor Systems (WPRS). A major part of these activities includes benchmark studies.

This report describes the results of the PRIMO rod BD8 benchmark exercise, the second benchmark by the TFRPD relative to MOX fuel behaviour. The corresponding PRIMO experimental data have been released, compiled and reviewed for the International Fuel Performance Experiments (IFPE) database.

The observed ranges (as noted in the text) in the predicted thermal and FGR responses are reasonable given the variety and combination of thermal conductivity and FGR models employed by the benchmark participants with their respective fuel performance codes.

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Appendix A: Benchmark specification

PWR reference irradiation of MOX fuels (PRIMO)

Data on a ramped MOX fuel rod

Released to TFRPD by Belgonucléaire and SCK•CEN

Prepared by

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19 July 2002

Modified by

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Based on data from SCK•CEN, and
Dr. Paul van Uffelen, ITU

26 January 2004

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- Destructive examination

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- BD8_4D1_fflux_IFPE.dat
- BD8_4D1_LHR_IFPE.dat
- BD8_4D2_fflux_IFPE.dat
- BD8_4D2_LHR_IFPE.dat

PWR reference irradiation of MOX fuels (PRIMO)

Summary of the PRIMO project as abstracted from the Final Report (PR 92/42 A, Dec. 2001).

The programme PRIMO was started in October 1986. It was jointly organised by (Studiecentrum voor Kernenergie • Centre d'Études de l'Énergie Nucléaire (SCK•CEN) and Belgonucléaire (BN) and was co-sponsored by ten participants including fuel vendors, utilities, nuclear centres and national authorities.

The PRIMO programme was an investigation on MOX fuel, with the following major objectives:

1. Irradiation of MOX fuel rods at different burn-up stages and following power histories representative of those of PWR power plants to determine their behaviour as far as their mechanical, thermal and neutronic properties are concerned. The data obtained have been used for the following purposes:
 - for licensing, to demonstrate the ability of the MOX fuel rods to sustain irradiation conditions comparable to uranium dioxide fuel;
 - to better understand the influence of the fabrication process on the fuel behaviour;
 - to benchmark the thermomechanical codes for MOX fuel calculation.

The major phenomena to be examined were:

- mechanical behaviour under steady-state operation: pellet-cladding interaction, ridging effects, fuel swelling and creep;
 - thermal behaviour: fission gas release, grain growth, fuel diffusion effects;
 - nuclear behaviour: burn-up radial profiles and U/Pu isotopic effects, versus burn-up.
2. Execution of a ramp test programme, to determine the failure threshold of MOX fuel rods and to obtain mechanical and thermal data under ramp conditions. The ramp tests were carried out on selected fuel types, at various burn-up levels. The ramp conditions (ramp rate, terminal power and hold time) were chosen such that the results can be compared to existing results on UO₂ fuel.
 3. Fast power transients on a MOX rod to simulate a Class II incident in a PWR.

These objectives were met through extensive post-irradiation examinations (PIE) in hot cells.

The base programme includes four different “types” of fuels:

- Fuel rods manufactured at BN using UO₂-PuO₂ powder mechanical blending process, referred to as the “reference” process. This fuel consists in a UO₂ matrix in which UO₂-PuO₂ rich zones, containing up to 100% PuO₂ are distributed.

Four rods of this type of fuel, designated “R” (reference), are included in the PRIMO programme: three from the old BR3/A4 fabrication campaign (1975), designated “RA” rods; and one from the more recent BR3/4D campaign (1983), designated as the “RD” rod.

- Fuel rods manufactured at BN, using the Micronised Master Blend (MIMAS) process presently used for industrial production of LWR MOX fuel. The use of the Master Blend principle leads to fuel presenting some Pu-rich particles, which can locally reach a content of ~30% PuO₂ of the primary blend, depending on the size of the agglomerates and on the U-Pu interdiffusion during sintering.

Eight rods of this type of fuel, named “B” (Belgonucléaire) are included in the PRIMO programme: six from the first BR3/4D MIMAS fabrication campaign (1983), named “BD” rods; two from a more recent BR3/4E fabrication campaign (1985), named “BE” rods.

- Fuel rods manufactured at BN (Dessel plant) for Mitsubishi Heavy Industries (MHI), using the MIMAS process as well, but following specifications of Mitsubishi (mainly different pellet size, cladding properties, and gap size).

Three rods of this type of fuel, named “M” (Mitsubishi) are included in the PRIMO programme. They were fabricated together with the BR3/4E campaign and are referred to as “ME” rods.

- One fuel rod manufactured at BN (Dessel plant) for FRAGEMMA, using the MIMAS process as well, but following FRAGEMMA specifications. This type of fuel rod has been named the “SL” type.

The first 15 MOX fuel rods were base irradiated in the BR3 reactor of SCK•CEN up to the following peak burn-ups:

- RA type: between 45 and 60 GWd/tM;
- RD type: around 40 GWd/tM;
- BD type: between 20 and 45 GWd/tM;
- BE type: between 20 and 25 GWd/tM;
- ME type: between 20 and 25 GWd/tM.

The irradiation of the first fuel rods started in July 1976 with BR3/core 4A and ended in June 1987 with the final shutdown of the reactor at the end of core 4D2.

The SL-type fuel rod was irradiated during two cycles in the Saint-Laurent B1 PWR reactor.

After base irradiation, selected fuel rods were submitted to a power excursion: one transient test in the BR2 reactor of SCK•CEN, five ramps in the OSIRIS reactor at Saclay, two ramps in the R2 reactor at Studsvik under the following conditions:

- BR2 transient on 1 RA rod: 410 W/cm peak power;
- OSIRIS ramps on 3 BD rods: 350, 390 and 395 W/cm peak power;
- OSIRIS ramps on 2 ME rods: 360 and 425 W/cm peak power;
- R2 ramps on 2 SL rodlets: 415 and 476 W/cm peak power.

Non-destructive examinations were performed on all the rods of the programme after base irradiation and, when appropriate, after a subsequent power excursion test. All the non-failed rods and rodlets were then punctured.

Destructive tests consisting of ceramographies, density measurements, EPMA, SIMS and thermal diffusivity measurements were performed on selected rods of the programme in five laboratories:

- SCK•CEN-Mol for puncture, ceramography and density measurement;
- CEA-Saclay for puncture, ceramography and EPMA;
- AEA Technology-Windscale for ceramography, density and thermal diffusivity measurements;
- PSI-Würenlingen for EPMA and SIMS measurements;
- Studsvik for the examination of the rodlets (from rod SL).

The present reports provide the results obtained on rod BD8.

Pre-characterisation data were extracted from the Final Report (PR 92/42 A, Dec. 2001), and the Topical Report PR 89/15 from the PRIMO programme

Pre-characterisation for BD8

Rod ID: F5790

Generalities

A general fuel rod drawing is shown in Figure 1. The fuel rods coming from the 4D fabrication campaign are composed of the following parts:

- cladding tube;
- lower end plug;
- upper end plug;
- fuel pellet hold-down spring (cf. Figure 2);
- spacer tube (cf. Figure 3);
- uranium dioxide blanket pellets;
- mixed-oxide fuel pellets.

Fuel pellets

Three UO₂ blanket pellets were loaded at the bottom end of the mixed-oxide fuel stacks. The UO₂-depleted fuel pellets were manufactured by Franco Belge de Fabrication de Combustible (FBFC) in its Dessel plant.

All the MOX fuel pellets were manufactured by BN in its Dessel plant (sintering conditions: 1 700°C, 1 h under Ar+5%H₂).

Pellet characteristic	Unit	UO ₂ (blanket)	MOX
Fabrication lot no.	–	BN lot 6303	BN lot IL123
Diameter*	mm	8.044	8.034, 8.041, 8.046
Length*	mm	12.115	8.591, 9.222, 9.667
Shoulder* (1)	mm	–	0.811, 0.864, 0.903
Shoulder* (2)	mm	–	0.830, 0.883, 0.934
Dish volume* (1)	%	–	0.947, 1.011, 1.095
Dish volume* (2)	%	–	0.903, 0.961, 1.051
Chamfer width* (1)	mm	0.5	0.274, 0.293, 0.324
Chamfer width* (2)	mm	0.5	0.297, 0.323, 0.344
Chamfer height* (1)	mm	0.5	0.102, 0.135, 0.166
Chamfer height* (2)	mm	0.5	0.132, 0.149, 0.170
O/M	–	1.994	1.991
²³⁵ U/U	w/o	0.3502	0.576
Fissile Pu/(U + Pu)	w/o	–	7.894
Fraction of ²³⁸ Pu	w/o**	–	0.159
Fraction of ²³⁹ Pu	w/o**	–	75.259
Fraction of ²⁴⁰ Pu	w/o**	–	20.994
Fraction of ²⁴¹ Pu	w/o**	–	2.820
Fraction of ²⁴² Pu	w/o**	–	0.768
Mean UO ₂ grain size***	Micron	14	5
Mean PuO ₂ -rich particle size	Micron	–	21.3
Max. PuO ₂ -rich particle size	Micron	–	<214
Pellet surface roughness	Micron	–	<2
Geometrical density	%TD	95.4	94.64
Density increase during resintering test**** (12h, 1 710°C)	%	–	1.1-1.9

(1) One side of pellet, (2) Side opposed to (1), * Minimum, mean, maximum value, ** Isotopic composition as of October 1983, *** Mean linear intercept, **** Obtained from pellet diameter and height change.

Cladding

The cladding tube used for the BD8 rod fabrication is seamless Zircaloy 4 manufactured by Mannesman Röhrenwerke (Germany) from an ingot provided by TWCA, Oregon (USA). The Zircaloy 4 cladding tubes were delivered in one lot identified as BNT262. The tube was stress-relieved at 460°C during 2.5 hours.

Cladding characteristic	Unit	Value
Outer diameter	mm	9.492-9.510
Inner diameter	mm	8.232-8.250
Minimum wall thickness	mm	0.58
Inner surface transversal roughness	Micron	3.2
Outer surface transversal roughness	Micron	2.8
Tensile test temperature	°C	400
Ultimate strength	Kg/mm ²	39.3
Yield strength	Kg/mm ²	31.4
Elongation	%	17.8
Hydride orientation (Fn)	–	0.04-0.15
Grain size	Micron	10
Hydrogen content	ppm	11
Oxygen content	ppm	1 170
Vickers Hardness (HV200)	–	205-214

Fuel rod parameters

The main rod parameters are summarised in the following table.

Rod characteristic	Unit	Value
Diametral gap	Micron	200
Pre-pressurisation (He)	Kg/cm ²	20
Uranium weight*	g	412.6
Plutonium weight	g	46.4
(U,Pu)O ₂ weight	g	521.1
UO ₂ blanket weight	g	19.4
Fuel length	mm	1 004.7
Blanket length	mm	36.5
Plenum length**	mm	40.0
Total rod length	mm	1 136.16
Rod free volume***	cm ³	5.7

* In (U,Pu)O₂ column only.

** Spacer tube excluded.

*** Free volume in upper plenum + fuel cladding gap + dishes + chamfers + free space in upper plug. Open porosity is not included.

A spring made of stainless steel was placed in the upper plenum free space, in order to avoid the formation of inter-pellet gaps during the rod handling.

A spacer ring made of Zy4 and 20 mm long was inserted in the fuel rod between the spring and the top fuel pellet.

Irradiation

Base irradiation in BR3

Rod BD8 (F5790) was base irradiated in the BR3 reactor of SCK•CEN. During cycle 4D1 (from 13/07/1984 until 11/11/1985, 413 days at power) the rod accumulated an average burn-up of 18.0 GWd/tM, corresponding to a peak pellet burn-up of 22.9 GWd/tM (calculated values). During cycle 4D2 (from 03/07/1986 until 30/06/1987, 333 days at power) the average rod burn-up reached 30.1 GWd/tM, corresponding to a peak pellet burn-up of 38.0 GWd/tM (calculated values).

The fast fluence, i.e. the fluence of neutrons with energies above 1 MeV, reached in rod BD8 after cycle 4D1 was on average 1.70×10^{21} neutrons/cm², whereas the maximum fluence amounted to 2.14×10^{21} neutrons/cm². After cycle 4D2, the fast fluence was on average 2.93×10^{21} neutrons/cm², whereas the maximum fluence amounted to 3.69×10^{21} neutrons/cm².

The BR3 reactor nominal power was 40.9 MWth (11.4 MWe). The BR3 primary circuit operates at a constant pressure of 140 atm.

The average temperature of the coolant is maintained at 262.5°C (power changes are compensated by adapting the mass flow rate).

Coolant inlet temperature is 255°C, and the mean outlet temperature at nominal power reaches 270°C.

The PRIMO rods were loaded in dismantable assemblies with a 17 × 17 configuration called “go” type, especially designed for the BR3-core 4.

The square lattice of the “go” assemblies is characterised by:

- rod diameter: 9.5 mm;
- rod pitch: 13.5 mm;
- hydraulic diameter: 14.93 mm.

in cold conditions.

The mean flow area for the 17 × 17 assembly is 42.65 cm²; using a mean mass flow rate of 9.4 kg/s for the “go”-type assembly, one can deduce a mass flow rate at core inlet of 220 g/cm² s.

Power transient in OSIRIS

After base irradiation in the BR3 reactor, the BD8 rod was transported to CEA-Saclay for irradiation in the OSIRIS reactor. This reactor is a 70 MWth pool reactor of the open core type. It is fuelled with 7 to 8 w/o ²³⁵U enriched UO₂ and cooled with light water. The irradiation device used for the PRIMO ramps was the ISABELLE 1 loop, installed on a movable structure of the core periphery. The power variations are obtained by inwards/backwards movements of the loop in the core water.

The fuel rod placed in a sample holder is contained in a tube pressurised at 150 bar nominal value and cooled with light water at a 200 g/s flow rate.

The fuel rod cladding outer surface temperatures were $342 \pm 5^\circ\text{C}$ (which is the water saturation temperature at 150 bar pressure).

The preconditioning of rod BD8 occurred at a peak power level of 189 W/cm with a hold time of 27 hours. The subsequent power excursion rate amounted to 77 W/(cm min), reaching a terminal peak power level of 395 W/cm that lasted for 20 hours. The accuracy on the determination of the peak pellet linear power is 6% for 1 σ confidence level.

Figure 1: General layout of the rod BD8 from the PRIMO programme

(1) upper end plug, (2) hold down spring (cf. Figure 2), (3) cladding tube, (4) spacer tube (cf. Figure 3), (5) UO_2 - PuO_2 fuel pellet, (6) UO_2 blanket pellet, (7) lower end plug

Dimensions are indicated in mm

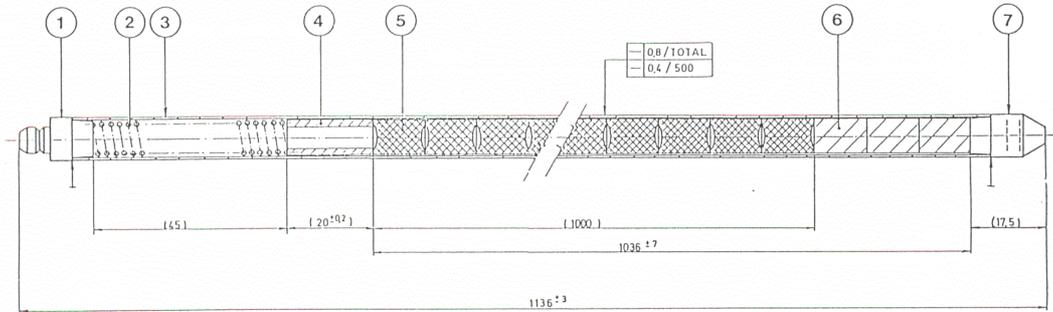


Figure 2: Hold down spring for rod BD8 from the PRIMO programme. The total number of coils amounts to 23, and the spring is made of stainless steel.

Dimensions are indicated in mm

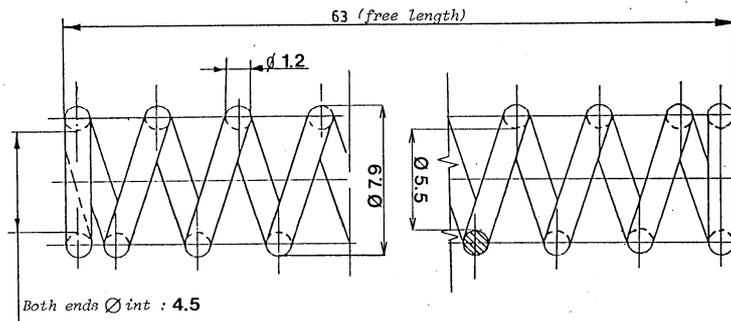
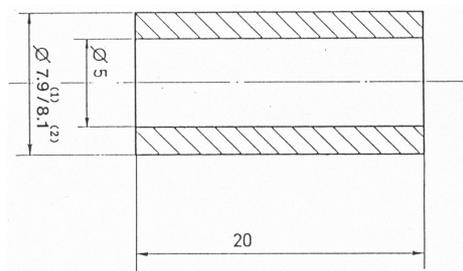


Figure 3: Spacer tube, made of Zr4, for the rods of the PRIMO programme

(1) RD and BD design, (2) BE design

Dimensions are indicated in mm



The irradiation history data were extracted from the Final Report (PR 92/42 A, Dec. 2001), and the Topical Report PR 89/14 from the PRIMO programme

Introduction

Rod BD8 (F5790) was base irradiated in the BR3 reactor of SCK•CEN. During cycle 4D1 (from 14/07/1984 until 11/11/1985, 413 days at power) the rod accumulated an average burn-up of 18.0 GWd/tM, corresponding to a peak pellet burn-up of 22.9 GWd/tM (calculated values). During cycle 4D2 (from 03/07/1986 until 30/06/1987, 333 days at power) the average rod burn-up reached 30.1 GWd/tM, corresponding to a peak pellet burn-up of 38.0 GWd/tM (calculated values). The maximum linear heat generation rate during the base irradiation was 316 W/cm.

The fast fluence, i.e. the fluence of neutrons with energies above 1 MeV, reached in rod BD8 after cycle 4D1 was on average 1.70×10^{21} neutrons/cm², whereas the maximum fluence amounted to 2.14×10^{21} neutrons/cm². After cycle 4D2, the fast fluence was on average 2.93×10^{21} neutrons/cm², whereas the maximum fluence amounted to 3.69×10^{21} neutrons/cm².

The BR3 reactor nominal power was 40.9 MWth (11.4 MWe). The BR3 primary circuit operated at a constant pressure of 140 atm.

The average temperature of the coolant was maintained at 262.5°C (power changes were compensated by adapting the mass flow rate).

Coolant inlet temperature was 255°C, and the mean outlet temperature at nominal power reached 270°C.

The PRIMO rods were loaded in dismountable assemblies with a 17 × 17 configuration called “go” type, specially designed for the BR3-core 4.

The square lattice of the “go” assemblies is characterised by:

- rod diameter: 9.5 mm;
- rod pitch: 13.5 mm;
- hydraulic diameter: 14.93 mm;

in cold conditions.

The mean flow area for the 17 × 17 assembly was 42.65 cm²; using a mean mass flow rate of 9.4 kg/s for the “go”-type assembly, one can deduce a mass flow rate at core inlet of 220 g/cm² s.

Rod power history

Base irradiation

The linear power has been calculated by means of the 2-D diffusion code CONDOR. The geometrical X-Y model covered the entire core for cycles 4D1 and 4D2. An R-Z model was used in order to obtain information on the axial behaviour. Nuclear constants were obtained from the cell code PANTHER. The axial effects such as variation of water density, variation of xenon density are taken into account in the results.

For the rod power calculations, the actual reactor power history of each cycle has been subdivided into constant core power time steps. The step lengths have been made small enough to ensure that the power does not vary more than a few per cent around the given mean values. The power history of the BD8 rod is summarised in the tables below. The complete data, including the fast flux data, are provided separately in the following files:

- BD8_4D1_LHR_IFPE.dat: Evolution of the linear heat generating rate in twelve axial zones along the fuel rod during cycle 4D1.
- BD8_4D2_LHR_IFPE.dat: Evolution of the linear heat generating rate in twelve axial zones along the fuel rod during cycle 4D2.

- BD8_4D1_FFLUX_IFPE.dat: Evolution of the fast flux in twelve axial zones along the fuel rod during cycle 4D1.
- BD8_4D1_FFLUX_IFPE.dat: Evolution of the fast flux in twelve axial zones along the fuel rod during cycle 4D2.

The first two files provide a histogram of time and mean power for a fuel column length divided in 12 axial zones nodes. The geometrical axial layout is given in Figure 4. Each time step has entries on 3 lines:

1. Time step nr., time (days), DT (days), Bu (GWd/tM).
2. Power (kW/m) nodes 1-6.
3. Power (kW/m) nodes 7-12.

The time corresponds to the elapsed time from the beginning of the cycle until the end of the current time step, expressed in a number of days. The parameter DT indicates the duration of the current time step, expressed in days. The parameter Bu corresponds to the burn-up at the end of the current time step and is expressed in GWd/tM. The linear heat generating rates are expressed in kW/m.

The two last files provide a histogram for the fast neutron (>1 MeV) fluence for a fuel column length divided in 12 axial zones. Each time step has entries on 3 lines as well:

1. Time step nr., time (days), DT (days), AVGFLU (*E12 n/cm²/s).
2. Fast flux [10¹²n/(cm² s)] nodes 1-6.
3. Fast flux [10¹²n/(cm² s)] nodes 7-12.

The time and DT denote the same as in the previous two files. The parameter AVGFLU corresponds to the rod averaged fast flux level and is expressed in 10¹²n/(cm² s). The fast flux level in each axial zone is provided in lines 2 and 3, and is expressed in 10¹²n/(cm² s) as well. At the very end of the file, the average fluence of the rod is expressed in 10¹⁹n/cm².

Table 1: Irradiation history of rod BD8 during cycle 4D1 in the BR3 reactor

Time (days)	Burn-up (GWd/tM)	LHR (kW/m)
33.02	1.7	23.18
55.86	2.9	24.26
67.32	3.3	15.72
73.13	3.6	24.16
83.69	4.0	18.71
112.39	5.5	23.61
127.72	6.2	20.30
142.96	6.9	23.06
153.80	7.4	21.74
190.83	9.2	21.60
216.66	10.3	19.93
237.24	11.1	18.76
252.94	11.7	17.18
283.38	12.9	18.24
311.20	14.0	18.18
342.08	15.3	18.15
371.99	16.4	18.14
395.52	17.4	18.17
413.26	18.0	16.97

Table 2: Irradiation history of rod BD8 during cycle 4D2 in the BR3 reactor

Time (days)	Burn-up (GWd/tM)	LHR (kW/m)
7.12	18.3	15.59
42.88	19.6	17.45
69.92	20.6	17.44
72.44	20.7	12.91
96.56	21.6	17.41
110.56	22.1	15.50
140.81	23.2	16.93
158.88	23.9	16.90
198.34	25.3	16.42
224.13	26.2	16.37
244.45	26.9	16.32
274.07	28.0	16.28
307.24	29.2	16.27
333.31	30.1	16.28

Power transient in OSIRIS

After base irradiation in the BR3 reactor, the BD8 rod was transported to CEA-Saclay for irradiation in the OSIRIS reactor. This reactor is a 70 MWth pool reactor of the open core type. It is fuelled with 7 to 8 w/o ²³⁵U enriched UO₂ and cooled with light water. The irradiation device used for the PRIMO ramps was the ISABELLE 1 loop (simulating PWR coolant temperature and pressure conditions), installed on a movable structure of the core periphery. The power variations are obtained by inwards/backwards movements of the loop in the core water.

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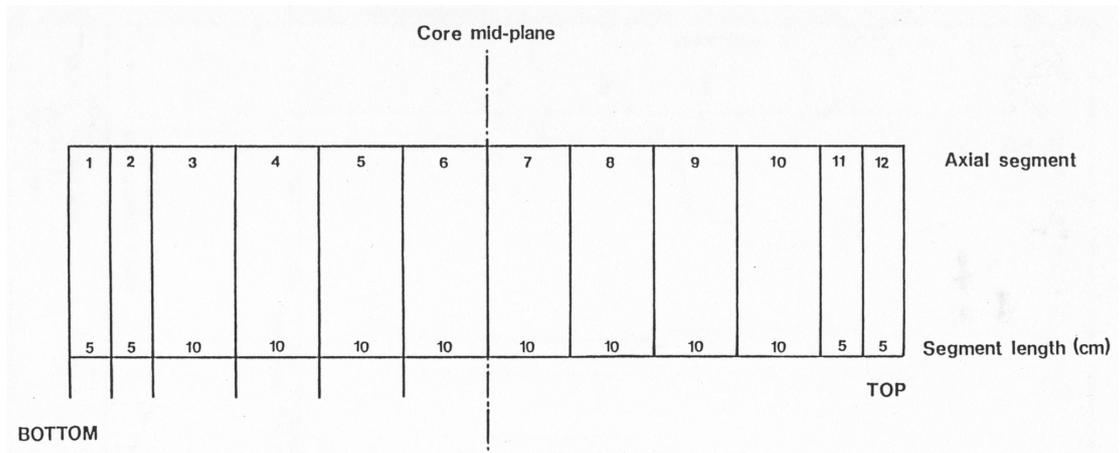
The relative OSIRIS axial power profile for the rod BD8 irradiation is given in Table 3.

Table 3: OSIRIS relative axial power profile for the rod BD8 irradiation

Axial position (mm)	OSIRIS profile	OSIRIS (W/cm)	Axial peaking factors
0	0.00022	48.67816	0.21686
50	0.00036	79.65517	0.35486
100	0.00056	123.90805	0.55200
200	0.00105	232.32759	1.03499
300	0.00152	336.32184	1.49828
400	0.00172	380.57471	1.69542
500	0.00174	385.00000	1.71513
600	0.00154	340.74713	1.51799
700	0.00114	252.24138	1.12371
800	0.00062	137.18391	0.61114
900	0.00014	30.97701	0.13800
1 000	0.00004	8.85057	0.03943

Figure 4: Subdivision of rod BD8 from the PRIMO programme in axial segments

Dimensions are indicated in cm



Non-destructive data were extracted from the Final Report (PR 92/42 A, Dec. 2001) from the PRIMO programme

Non-destructive examination for BD8

Rod ID: F5790

Examination programme

The non-destructive examinations were used to characterise the fuel rod evolution on three points:

- the cladding creep, irradiation growth and integrity;
- the fuel integrity;
- the pellet-cladding mechanical interaction.

Five non-destructive tests were included in the non-destructive examination programme:

- visual examination of the cladding;
- dimensional controls (diameter and rod length measurements);
- neutron radiography;
- gamma scanning (spectrometry, burnothèque);
- eddy current test.

Examinations after base irradiations

Cladding integrity

The cladding soundness was inspected through visual examination and eddy current test. Visual examinations showed a uniform dark-grey aspect, with scarce scratches resulting from rod manipulations. No anomaly was observed.

Eddy current tests were performed in order to detect cladding abrupt defects such as incipient cracks as well as longer and more gradual defects. The laboratory device used differentially coupled encircling coils to provide eddy current scanings. The frequencies selected for possible detection of defects were between 100 and 500 kHz. Neither primary defects nor ridging could be observed on rod BD8.

Cladding creep and fuel-cladding mechanical interaction

The cladding outer diameters were measured at CEA-Saclay by recording two diameters at 90°, giving the mean diameter and the azimuthal local ovalisation, all with an accuracy of ±5 micron. The rod BD8 showed cladding creep down along the fuel column length, without any sign of pellet-cladding mechanical interaction.

From the diameter records, one has calculated the cladding diameter change or hoop strain ($\epsilon\theta$), and the ovalisation (Δ) at twenty axial positions along the fuel column by means of the following expressions:

$$\epsilon\theta(\%) = (D_a - D_0)/D_0 \cdot 100$$

$$\Delta = \text{maximum recorded diameter} - \text{minimum recorded diameter}$$

where $D_a = (\text{the maximum diameter} + \text{the minimum diameter})/2$ and D_0 is the reference diameter (mean diameter at or close to the plenum level). At a maximum fast fluence ($E > 1 \text{ MeV}$) of $3.69E21 \text{ n/cm}^2$, the rod mean hoop strain amounts to -0.24%, whereas the local hoop strain at the maximum flux axial level was -0.32%. The rod mean ovalisation was about 13 micron, whereas the local ovalisation at the maximum fast flux level was 18 micron.

Cladding irradiation growth

In order to determine the Zircaloy irradiation growth for rod BD8, rod length measurements were performed at CEA-Saclay using neutron radiography examination. The accuracy on the results is estimated at 1 mm.

The rod elongation for rod BD8, defined as:

$$\varepsilon z(\%) = (L1 - L0)/LF*100$$

where L0 is the length before irradiation, L1 is the length after irradiation and LF is the fuel column length, amounts to 0.21% after a mean fast neutron fluence ($E > 1$ MeV) of $2.93E21$ n/cm².

Fuel column length change

The fuel column length was measured with the same technique (and accuracy) and its change was -0.13% for rod BD8.

Fuel burn-up

“Burnothèque” examination was performed in order to determine the maximum and mean burn-up by means of non-destructive gamma spectrometry on ¹³⁷Cs. The measured rod burn-up of rod BD8 was 30.1 GWd/tM whereas the peak pellet burn-up was 39.7 GWd/tM. The uncertainty on the experimental burn-up values is estimated to be less than 6%. These values are in good agreement with the calculated burn-up, according to which the average rod burn-up reaches 32.2 GWd/tM and the peak pellet burn-up 38.0 GWd/tM.

Examinations after transient or ramp

The preconditioning of rod BD8 occurred at a peak power level of 189 W/cm with a hold time of 27 hours. The subsequent power excursion rate amounted to 77 W/(cm min), reaching a peak terminal power level of 395 W/cm that lasted for 20 hours. This transient allowed the following observations:

- There was no significant cladding mean diameter change.
- There was the appearance of a ridging effect in the high power zone, ranging between 10 and 17 microns (diametral).
- There was a beginning of dish filling observable on the neutron radiography as a result of gaseous swelling and fuel plasticity during ramp.
- There was caesium migration towards the coolest zones of the pellets. The caesium axial profile is represented in Figure 5.
- There was a sound aspect of the cladding and the fuel column, with no anomaly.

Destructive examination for BD8

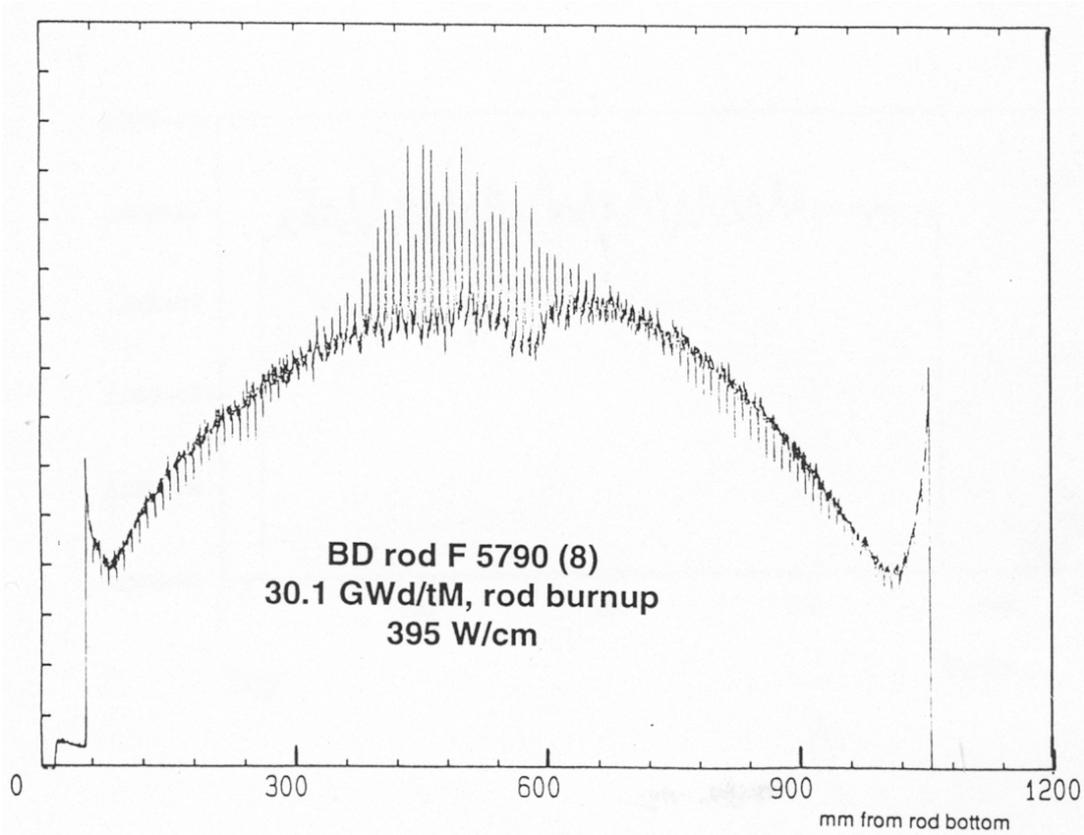
For a BR3 maximum linear heat generating rate of 316 W/cm during the base irradiation, followed by a ramp in OSIRIS reaching a peak power of 395 W/cm, kept during 20 hours, the following parameters were measured:

- Total collected gas: 171.3 cm³ at STP.
- Volume of Xe + Kr: 46.9 cm³ at STP.
- Gas composition (vol.%): 72.62 He, 25.71% Xe, and 1.67% Kr.
- Xe/Kr volume ratio: 15.4.
- Total FGR: 11.24%.
- Estimated Xe + Kr release after base irradiation: ≤1.98 cm³ at STP, or 0.47%.
- Estimated Xe + Kr release after the power ramp: 44.92 cm³ at STP, or 10.77%.

The fission gas release (FGR) fraction is calculated assuming a gas (Xe + Kr) generation rate of 30 cc STP/MWd. The FGR after base irradiation of the ramped rod has been estimated on the basis of the FGR of a sibling non-ramped rod, i.e. rod BD10 (F5784).

Ceramographic examination after the ramp revealed that grain growth occurred in the central part of the pellets (until $r/R_{\text{pellet}} = 0.4$). The mean grain size reached about 10 μm in that region.

Figure 5: Aspects of caesium migration in the ramped rod BD8 of the PRIMO programme



BD8_4D1_fflux_IFPE.dat

1	0.04	0.04	60.90			
	27.5	36.5	51.2	65.8	74.5	79.0
	78.7	74.8	66.4	53.3	37.9	29.0
2	166.88	166.84	58.70			
	28.2	37.1	51.1	63.9	70.4	73.6
	73.6	70.8	64.2	53.0	38.4	29.6
3	333.68	166.80	58.30			
	29.8	38.8	52.5	64.0	68.6	70.7
	70.9	69.1	64.1	54.3	40.0	31.0
170.2						

BD8_4D1_LHR_IFPE.dat

1	33.02	33.02	1.70			
	11.82	15.04	20.56	25.85	29.08	30.39
	29.95	27.94	24.23	18.56	13.34	10.39
2	55.86	22.84	2.90			
	12.57	15.95	21.66	27.05	30.29	31.57
	31.11	29.10	25.35	19.57	14.15	11.05
3	67.32	11.45	3.30			
	8.21	10.39	14.07	17.52	19.58	20.39
	20.09	18.81	16.42	12.72	9.22	7.21
4	73.13	5.82	3.60			
	12.65	16.00	21.66	26.94	30.08	31.31
	30.85	28.90	25.25	19.57	14.20	11.12
5	83.69	10.56	4.00			
	9.83	12.42	16.79	20.86	23.28	24.23
	23.87	22.37	19.55	15.17	11.02	8.64
6	112.39	28.69	5.50			
	12.50	15.76	21.25	26.32	29.32	30.49
	30.04	28.18	24.67	19.20	13.97	10.99
7	127.72	15.33	6.20			
	10.83	13.63	18.32	22.63	25.17	26.14
	25.76	24.19	21.21	16.55	12.08	9.52
8	142.96	15.25	6.90			
	12.36	15.53	20.84	25.71	28.57	29.66
	29.22	27.45	24.10	18.82	13.77	10.86
9	153.80	10.84	7.40			
	11.71	14.69	19.68	24.24	26.90	27.92
	27.50	25.84	22.72	17.77	13.02	10.28
10	190.83	37.03	9.20			
	11.72	14.67	19.60	24.08	26.68	27.66
	27.26	25.63	22.57	17.69	13.00	10.29
11	216.66	25.83	10.30			
	10.92	13.61	18.14	22.20	24.55	25.41
	25.05	23.57	20.82	16.37	12.08	9.58
12	237.24	20.58	11.10			
	10.39	12.92	17.16	20.92	23.05	23.81
	23.47	22.13	19.62	15.48	11.46	9.12
13	252.94	15.71	11.70			
	9.60	11.91	15.77	19.16	21.05	21.71
	21.39	20.22	17.97	14.23	10.57	8.42
14	283.38	30.44	12.90			
	10.32	12.77	16.84	20.37	22.29	22.93
	22.59	21.41	19.10	15.20	11.33	9.05
15	311.20	27.82	14.00			
	10.46	12.90	16.91	20.31	22.08	22.66
	22.32	21.21	19.04	15.27	11.45	9.18
16	342.08	30.88	15.30			
	10.63	13.08	17.03	20.29	21.91	22.42
	22.09	21.05	19.02	15.38	11.60	9.34
17	371.99	29.91	16.40			
	10.83	13.27	17.16	20.28	21.77	22.21
	21.88	20.93	19.01	15.50	11.76	9.51
18	395.52	23.53	17.40			
	11.02	13.46	17.31	20.31	21.70	22.07
	21.74	20.86	19.04	15.63	11.93	9.67
19	413.26	17.74	18.00			
	10.38	12.64	16.20	18.93	20.17	20.50
	20.21	19.45	17.83	14.74	11.30	9.20

BD8_4D2_fflux_IFPE.dat

1	0.04	0.04	54.80			
	24.7	32.8	46.1	59.2	67.1	71.1
	70.8	67.4	59.7	48.0	34.1	26.1
2	129.88	129.83	54.70			
	26.0	34.2	47.3	59.4	65.9	69.2
	69.0	66.3	59.8	49.1	35.4	27.2
3	259.67	129.79	54.70			
	27.2	35.6	48.5	59.8	64.9	67.4
	67.5	65.4	60.0	50.3	36.7	28.4
	292.9					

BD8_4D2_LHR_IFPE.dat

```
1      7.12      7.12      18.30
   8.60 10.35 13.91 17.30 19.36 20.20
 19.90 18.62 16.23 12.56  9.18  7.56
2     42.88     35.75     19.60
   9.78 11.72 15.67 19.37 21.57 22.45
 22.12 20.74 18.16 14.15 10.39  8.59
3     69.92     27.04     20.60
   9.90 11.82 15.74 19.36 21.50 22.33
 22.00 20.66 18.16 14.22 10.49  8.68
4     72.44      2.52     20.70
   7.35  8.77 11.67 14.34 15.89 16.50
 16.26 15.28 13.44 10.55  7.78  6.44
5     96.56     24.13     21.60
   9.94 11.86 15.76 19.34 21.42 22.22
 21.90 20.58 18.14 14.25 10.52  8.72
6    110.56     14.00     22.10
   8.88 10.60 14.06 17.23 19.04 19.74
 19.46 18.30 16.15 12.71  9.39  7.80
7    140.81     30.25     23.20
   9.75 11.64 15.41 18.82 20.76 21.49
 21.19 19.94 17.64 13.92 10.32  8.57
8    158.88     18.07     23.90
   9.79 11.68 15.44 18.79 20.66 21.37
 21.06 19.85 17.61 13.94 10.37  8.61
9    198.34     39.46     25.30
   9.60 11.44 15.07 18.26 20.03 20.68
 20.38 19.24 17.11 13.61 10.15  8.44
10   224.13     25.79     26.20
   9.66 11.49 15.07 18.19 19.89 20.50
 20.20 19.13 17.07 13.65 10.22  8.51
11   244.45     20.31     26.90
   9.60 11.42 14.96 18.03 19.70 20.32
 20.09 19.10 17.15 13.82 10.40  8.68
12   274.07     29.63     28.00
   9.55 11.35 14.83 17.85 19.49 20.12
 19.96 19.06 17.24 14.02 10.62  8.89
13   307.24     33.17     29.20
   9.66 11.47 14.90 17.82 19.36 19.95
 19.80 18.98 17.26 14.16 10.79  9.05
14   333.31     26.08     30.10
   9.86 11.64 15.02 17.82 19.27 19.80
 19.65 18.89 17.27 14.27 10.96  9.24
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