A PCI FAILURE IN AN EXPERIMENTAL MOX FUEL ROD AND ITS SENSITIVITY ANALYSIS

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Abstract

Within our interest in studying MOX fuel performance, the irradiation of the first Argentine prototypes of PHWR MOX fuels began in 1986 with six rods fabricated at the α Facility (CNEA, Argentina). These experiences were made in the HFR-Petten reactor, Holland. The goal of this experience was to study the fuel behaviour with respect to PMCI-SCC. An experiment for extended burnup was performed with the last two MOX rods. During the experiment the final test ramp was interrupted due to a failure in the rod. The post-irradiation examinations indicated that PCI-SCC was a mechanism likely to produce the failure. At the Argentine Atomic Energy Commission (CNEA) the BACO code was developed for the simulation of a fuel rod thermo-mechanical behaviour under stationary and transient power situations. BACO include a probability analysis within their structure. In BACO the criterion for safe operation of the fuel is based on the maximum hoop stress being below a critical value at the cladding inner surface; this is related to susceptibility to stress corrosion cracking (SCC). The parameters of the MOX irradiation, the preparation of the experiments and post-irradiation analysis were sustained by the BACO code predictions. We present in this paper an overview of the different experiences performed with the MOX fuel rods and the main findings of the post-irradiation examinations. A BACO code description, a wide set of examples which sustain the BACO code validation, and a special calculation for BU15 experiment attained using the BACO code including a probabilistic analysis of the influence of rod parameters on performance are included.

1. INTRODUCTION

Within our interest in studying MOX fuel performance, the irradiation of the first Argentine prototypes of PHWR MOX fuels began in 1986 with six rods fabricated at the α Facility (CNEA, Argentina) [1]. These experiences were made in the HFR-Petten reactor, Holland. The goal of this experience was to study the fuel behaviour with respect to PMCI-SCC.

An experiment for extended burnup was performed with the last two MOX rods. During the experiment the final test ramp was interrupted due to a failure in the rod. The post-irradiation examinations indicated that PCI-SCC was a mechanism likely to produce the failure [2]. That analysis was predicted with the calculated stresses [3].

In a nuclear reactor, the materials at the fuel are at relatively large temperatures and suffer the effect of an aggressive chemical and radiation environment. Therefore, mechanical solicitation might sometimes be near the limits of materials endurance. Design, within engineering concepts, has to consider parts performance as well as their in-service mechanical coupling. This coupling requires computer codes for valid results to be obtained.

At the Argentine Atomic Energy Commission (CNEA) the BACO code was developed for the simulation of a fuel rod thermo-mechanical behaviour under stationary and transient power situations [4]. Our modelling approach is based on using simple models, which are however sustained on physically sound ideas. BACO include a probability analysis within their structure covering uncertainties in input, fabrication, parameters and models and, therefore, output ones may include statistical dispersion [5]. In BACO the criterion for safe operation of the fuel is based on the maximum hoop stress being below a critical value at the cladding inner surface. This is related to susceptibility to stress corrosion cracking (SCC) [6].

The BACO code validation is sustained for a wide set of irradiation. We select some examples of BACO validity tests. The complete benchmarking is done with each new improvement included in the code. The selection made in this paper corresponds to exigent and very well known international blind tests and open literature.

The parameters of the MOX irradiation, the preparation of the experiments and post-irradiation analysis were sustained by the BACO code predictions [7].

This paper presents an overview of the different experiences performed with the MOX fuel rods and the main findings of the post-irradiation examinations. A BACO code description, and a special calculation for BU15 experiment attained using the BACO code including a probabilistic analysis of the influence of rod parameters on performance are included.

2. BACO CODE

2.1. BACO code description

The BACO code structure and models in its present versions are described by Marino et al., including steady state and transient thermal analysis. The number of instructions is at present (version 2.40) around one thousand FORTRAN 77 sentences. Data post-processing improves the code's performance and analysis of results.

On modelling the UO_2 pellet, elastic deformation, thermal expansion, creep, swelling, densification, restructuring, cracks and fission gas release are included. While for the Zry cladding, the code models elastic deformation, thermal expansion, anisotropic plastic deformation, and creep and growth under irradiation. The modular structure of the code easily allows the incorporation of different material properties. It can be used for any geometrical dimensions of cylindrical fuel rods with UO_2 pellets (either compact or hollow and with or without dishing) and Zry cladding.

Fuel rod power history and either cladding or coolant external temperatures are inputs to the program. Rod performance is numerically simulated using finite time steps. The code automatically selects time steps according to physical criteria. Temperature distribution in the pellet and cladding, main stresses at pellet and cladding, radial and axial crack pattern in the pellet, main strains and hot geometry of pellet and cladding, change in porosity, grain size and restructuring of the pellet, fission gas release to the free volume in the rod, trapped gas distribution in the fuel and in the UO_2 grain boundary, internal gas pressure and current composition of the internal gas, are calculated. The output contains the distribution along the rod axis of these variables.

2.2. Numerical treatment

We assume cylindrical symmetry for the fuel rod; our model is bidimensional and angular coordinates are not considered. However, angular dependent phenomena, as radial cracking, are simulated via some angular averaging method. For the numerical modelling the hypotheses of axial symmetry and modified plane strains (constant axial strain) are adopted. With respect to the axial dependence of the fuel rod behaviour, arising mainly from the axial dependence of power generation in the reactor, the fuel rod is divided in axial sections.

The mechanical and thermal treatment and the pellet, cladding and constitutive equations are very well described in Reference [4].

2.3. BACO code validity tests

2.3.1. *Experimental irradiation at the NRX reactor*

The work reported by Notley [9] was used for testing the BACO code and results were reported in References [4, 8]. In Notley's work six Zircaloy-sheathed UO_2 fuel elements were irradiated at power outputs between 760 and 600 W/cm to a burnup of about 5500 MWd/tonU. Then two of them and another two new rods were irradiated at lower powers for a further 1250-1700 MWd/tonU. The experiment was irradiated in the X-2 loop of NRX reactor. All elements were destructively examined and some of them were measured during the irradiation. The predicted and measured rod radius change $\Delta R/R$, fission gases released, columnar and equiaxed grains and central hole are provided by Notley [9] and calculated with BACO [4, 8].

We find that, although the predictions of $\Delta R/R$ and the gas released are approximately correct, the measured rod dimension changes are somewhat larger than predicted at low power and low burnup and are overpredicted by the code at high power and burnup. We discussed in reference [8] that this might be related to burnup and power ranges to which the code parameters have been fitted.

2.3.2. *CANDU fuelograms*

In CANDU reactors, fuel reshuffling is done under reactor operation. During reshuffling, the fuel undergoes a power ramp due to the power distribution along the fuel channel. For this reason, it is interesting to study the behaviour of a CANDU fuel under fast (10-20 min long) power ramps. The Linear Heat Generation Rate (LHGR) before the ramp, the burnup at which the ramps occurs, and the ramp height, cover a wide range. AECL has published bounds for safe operation, based on actual experience of power ramping due to fuel reshuffling in nuclear power stations. Usually [10, 11], the maximum power increase and maximum power such that fuel operation below those values present no failures, are given as a function of burnup.

The experimental bounds for power increase and maximum power corresponding to the Pickering Stations are plotted by Penn et al. [12]. Power histories simulating reshuffling were simulated with the BACO code. In the Code, the criterion for safe operation was based on the maximum hoop stress at the cladding inner surface. This is related to susceptibility to stress corrosion cracking. BACO results are in good agreement with AECL data; even the mispredictions can be explained on a physical basis [7]. A CANDU fuel rod simulation including statistical analysis was included in the Reference [13].

2.3.3. *A classical blind test (EPRI NP-369)*

The cases reported at the EPRI NP-369 [14] were a set of classical blind tests for fuel codes. The new versions of codes have evolved since then. The idea of this point is to show how the BACO code solves now the problem of relocation and compares with experiments.

Relocation of pellet fragments into the gap has an important effect on gap conductance. We use Broughton and MacDonald correlation [15] and a phenomenological model that includes circumferential cracks models relocation. Case C of EPRI [14] benchmarking is simulated and the experimental result satisfactorily reproduced.

In case C of the EPRI report on fuel code benchmarking, all codes predicted a central temperature larger than the experimental one. This fact agrees with the Broughton and Mac Donald (BMD) [16] observation that the fuel temperature calculated with large hot gaps gives poor agreement with experiments when the Ross and Stoute [17] or similar models are used for gap conductance. Case C of EPRI is then used for testing whether including a model of partial gap closing by pellet relocation proposed by BMD provides a better agreement between predictions and experiments.

In Figure 1 the EPRI benchmarking temperature for case C is plotted against burnup. The good agreement between experiments and BACO code predictions can be seen by using either the BMD model or its modifications when two cracks are included. We also include the performance of the COMETHE III J code. It can be seen that it overpredicts the central temperature. The same happens if in the BACO code the modified BMD model is not included.

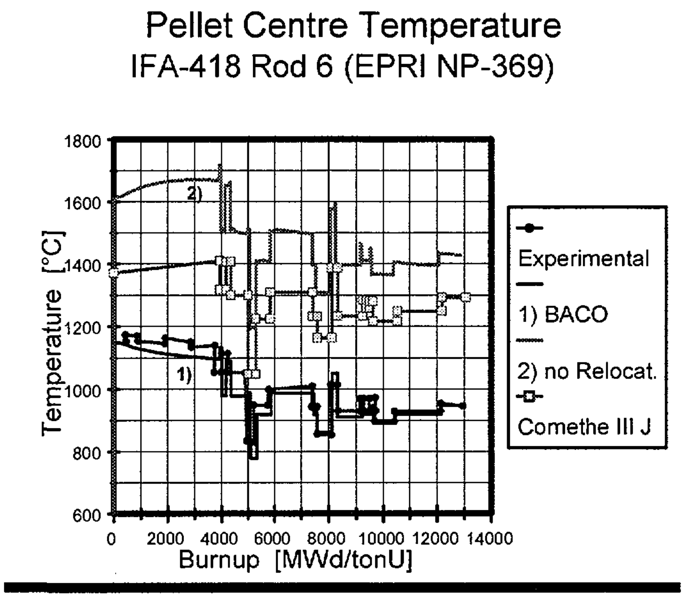

FIG. 1. Pellet centre temperature measurement and BACO calculations for the EPRI Case C.

It is shown that allowing for pellet relocation in the pellet-cladding gap provides gap heat conductance expressions that fit better to experiments than usual laws based on Ross-Stoute [17]. The perfect agreement obtained in the cases studied must however be taken with care. The gap conductance parameters are taken from BMD [16] model, which does not include a full pellet relocation model as proposed by us above. Then, although we found those values were sensible, a proper fit to experiment should be done for obtaining the best available model of gap conductance within our modified version of BMD model.

2.3.4. Co-ordinated research project on fuel modeling at extended burnup

The IAEA's CRP FUMEX (Co-Ordinated Research Project on Fuel Modelling at Extended Burnup) is a blind-test consisting of a set of experiments in order to compare fuel performance with code predictions. The OECD-HALDEN reactor (Norway) provided data. A set of instrumented fuels allowed to follow the evolution of some parameters (pellet centre temperature, inner pressure of the rod, cladding elongation, fission gas release, cladding diameter). The experiments include PIE analysis. The final burnup reached for the fuels were intermediate (25 MWd/kgU) and high (50 MWd/kgU) [18, 19].

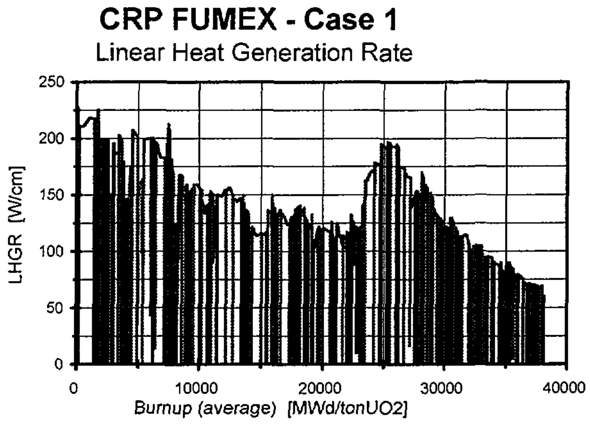

FIG. 2. Linear Heat Generation Rate (power history) as a function of the averaged burnup of the fuel rod.

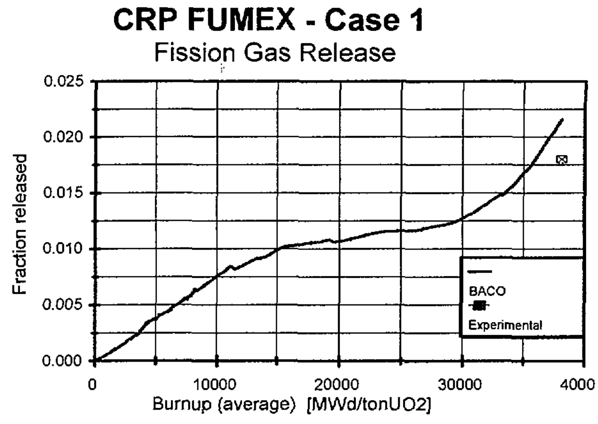

FIG. 3. Fraction of fission gas released as a function of averaged burnup of the fuel rod.

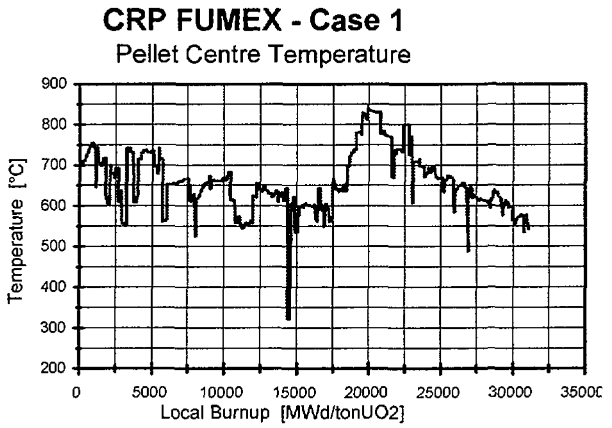

FIG. 4. Pellet centre temperature. CRP FUMEX Case 1. "On line" temperature measured at the OECD Halden Reactor.

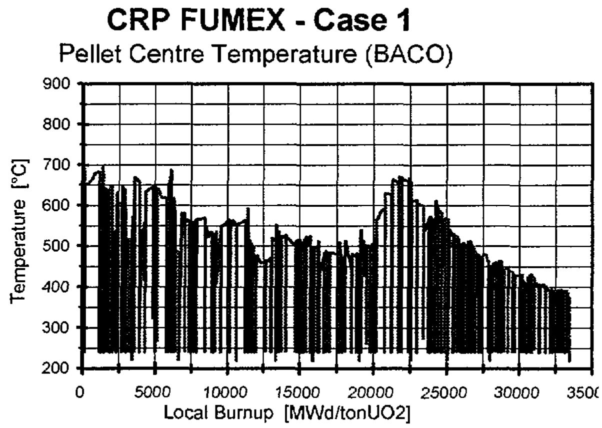

FIG. 5. BACO output for the pellet centre temperature. CRP FUMEX Case 1.

As an example of the BACO Code performance during the CRP FUMEX we include some of our results for the first exercises (FUMEX Case 1). Figure 2 shows the power history (input data) for that case. The calculated fraction of fission gases released at EOL (End Of Life) is 2.2 % (and the predicted value during the "blind test" stage was 2.5 %). The experimental value was 1.8 % (see Figure 3).

Figure 4 shows the measurement during irradiation of the temperature at the pellet centre (top of the nuclear fuel rod). The temperature monitoring was made every 15 minutes during all the irradiation. The measurement error was ± 50 °C. Figure 5 includes the BACO code calculation [20].

2.4. Sensitivity analysis with BACO

The uncertainties in the results of a validated fuel computer code with an experimented user are due to different sources:

(1) Input data of the codes:
   (a) Neutronic and reactor data,
   (b) Power history of the irradiation,
   (c) Fuel data:
      (i) Dimensional data
      (ii) Material properties

(2) Internal data of the code (and code structure):
   (a) Code parameters,
   (b) Modelling,
      (i) Physical constants,
      (ii) Parameters of the model,
      (iii) Field of application of the model.

Also, we can join these data around the point of view of its influence on the uncertainties:

- Modelling and its empirical or theoretical parameters,
- Data provided by direct measures due to in reactor irradiation and fuel test, and
- Fuel manufacturing data and fuel design data.

We can request the best models for our codes and then we solve the first point. We can request the best measurements during irradiation, material testing and reactor parameters. This means that an improvement of code results due to modelling and reactor and material data is possible.

The same does not happen with the parameters of the fuel due to the manufacturing process. The tolerances of fuel dimensions are a consequence of that process and they are sustained by the basic design. Then we must include the treatment of this source of uncertainties.

The first and easy way to analyse these topics is the definition of a set of worst cases. Those cases are the coupling of the variations of fuel parameters taking account of the tolerances in order to produce the worst situations (as maximum stress, maximum strains, extreme temperatures, etc.).

Some fuel performance codes include a probability analysis within their structure covering uncertainties in input, fabrication, parameters and models [21, 22, 23]. The BACO code (version 2.4) includes statistical analysis including statistical dispersion of the fuel rod parameters.

A BACO´s statistical analysis of power fuel reshuffling in the Atucha I NPP (an Argentinean PHWR) is included in the Reference [24], where we analyse the susceptibility of hoop stress during fuel reshuffling at different powers and burnups. Here we reproduce the recommendation of the designer for fuel reshuffling with simplified rules. The fuel for the Atucha I NPP are analysed in the Reference [5] and, the fuel for the EMBALSE NPP (CANDU type, Argentina), are analysed in Reference [13]. These exercises show, on the one hand, the sensitivity of the predictions concerning such parameters and, on the other, the potentiality of the BACO code for a probability study.

3. MOX FUEL IRRADIATIONS

3.1. MOX fuel description

The original design of the rods was made for utilization of the MZFR reactor (Karlsruhe, Germany). Due to the decommissioning of that reactor another place to make the irradiation tests had to be found. We carried out the experiments in the HFR-Petten reactor. Figure 6 shows the basic rods (rod A.1.2 and rod A.1.3). We intended to emulate extended burnup with Iodine doped pellets. Figure 7 shows the rods A.3 and A.4 and indicates the position of the doped pellets. Table I presents the characteristics of the rods taking into account the parameters specified by usual design. Those data are a coupling of MZFR, Atucha I and Petten fuel parameters. The Pu of these MOX fuels and the extension in burnup are compatible with our PHWR NPPs and they imply an increment of three times the actual burnup.

The fuel rod tolerances included in the Table 1 are provided by basic design. The real process at the α Facility Laboratory (CNEA, Argentine) allows a most detailed performance for those fuel rod parameters. Finally, the labs lead to a narrow band within tolerance and we can analyse the

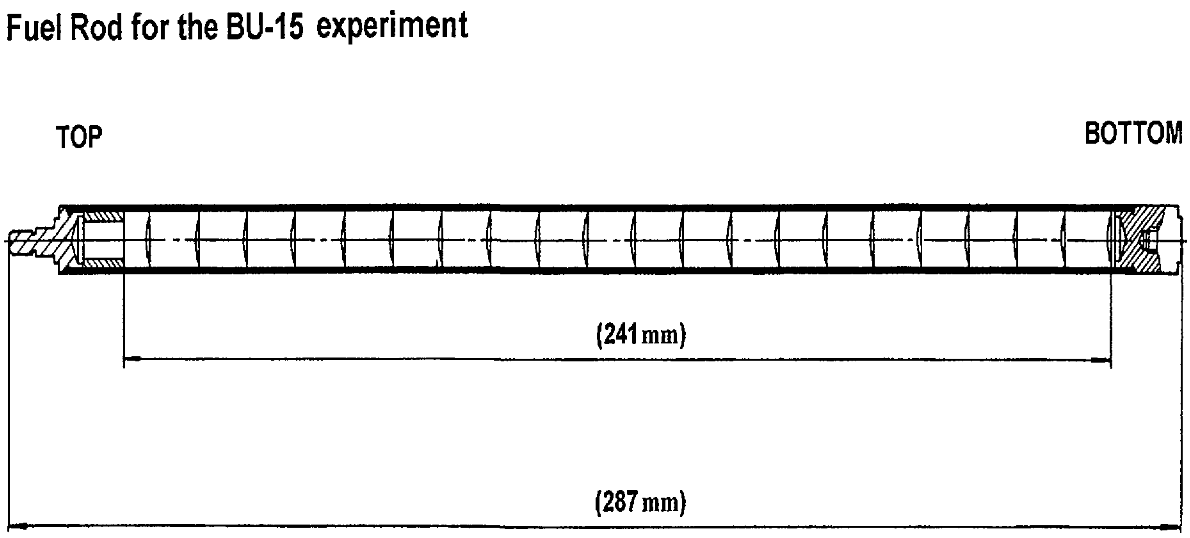

FIG. 6. MOX fuel rod type A.1 for the BU15 experiment.

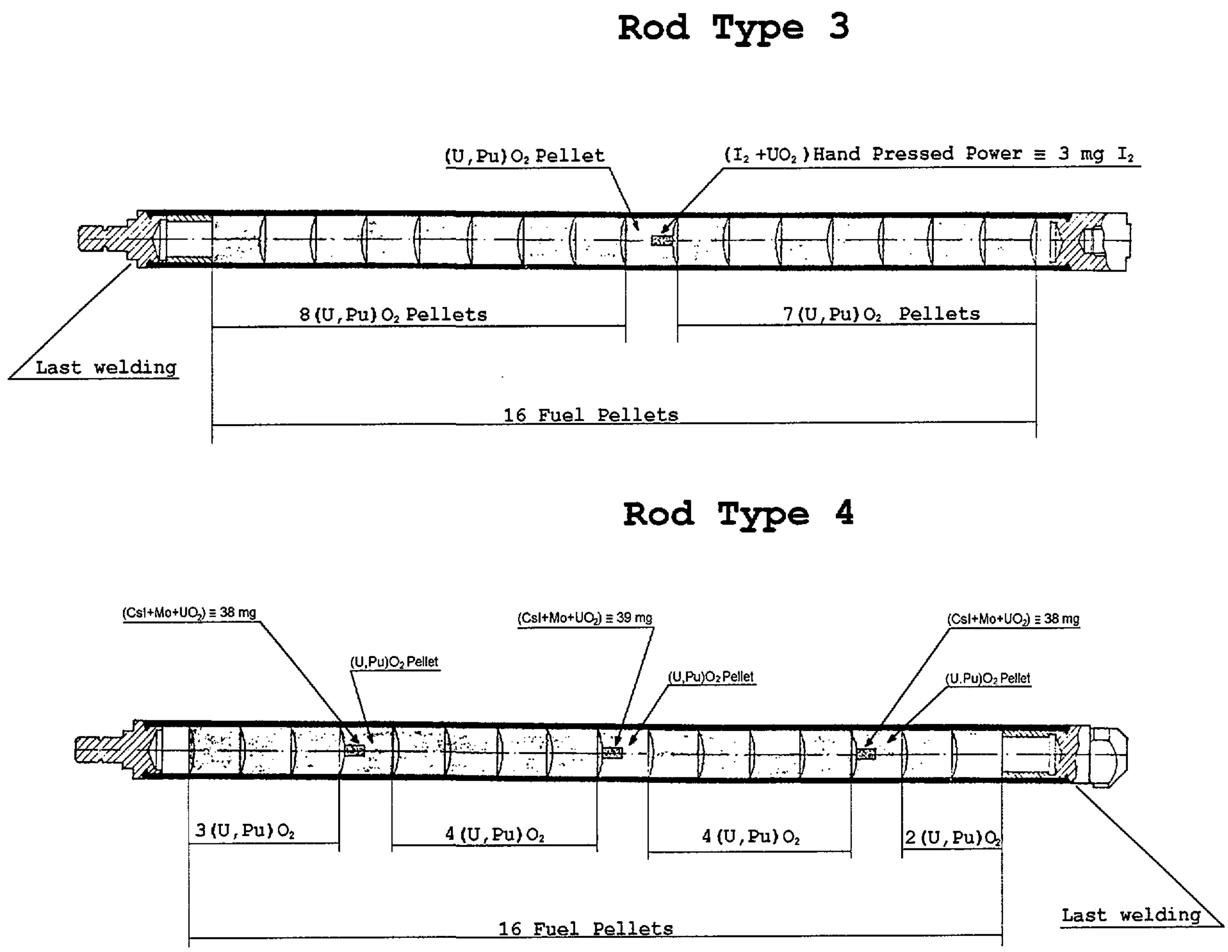

FIG. 7. MOX fuel pins type A.3 and A.4.

TABLE I. MOX FUELS IRRADIATED AT PETTEN REACTOR AS-FABRICATED FUEL RODS PARAMETERS FUEL RODS TYPE A.1, A.3 Y A.4

Rod	A.1	A.3	A.4
Length [cm]	24.1 cm	17.9	17.95
Number of rods	4	1	1
Pellets number	21	16	16
Compensating pellets number	1	-	-
Doped pellets number	-	1	3
Dope material	-	I	CsI Mo
Dope material (mg)	-	3	6.604 ± 0.170 1.373 ± 0.036
Simulated burnup (MWd/ton(M))	-	13761 ± 4933	14800 ± 390
Filling gasses	He	He	He
Filling pressure (atmospheres)	1.15	1.15	1.15

PELLETS

Pellet diameter	1.040 ± 0.001 cm
Pellet height	1.12 ± 0.01 cm
Density	10.52 ± 0.04 gr/cm^3
$Pu_{fiss}/U+Pu_{met}$	0.53 %
Enrichment (U^{235}+Pu)	1.25 %
O/M relation	2.00
Dishing volume	25. ± 5. mm^3

CLADDING

Cladding material	Zry-4
Cladding inner diameter	1.17 cm
Cladding thickens	0.06 cm

TABLE II. MOX FUELS IRRADIATED AT PETTEN REACTOR.
SOME STATISTICAL PARAMETERS OF THE FUEL RODS

PELLETS	Main Value (or specification)	Standard Deviation	Minimum Value	Maximum Value
Pellet diameter (cm)	1.0402	0.0005	1.0390	1.0414
Pellet height (cm)	1.1217	0.0204	1.1000	1.3000
Density (gr/cm^3)	10.522	0.048	10.350	10.650
...	...	...	...	...
CLADDING				
Cladding inner diameter	1.170 cm	...	1.166	1.174
...	...	...	...	...

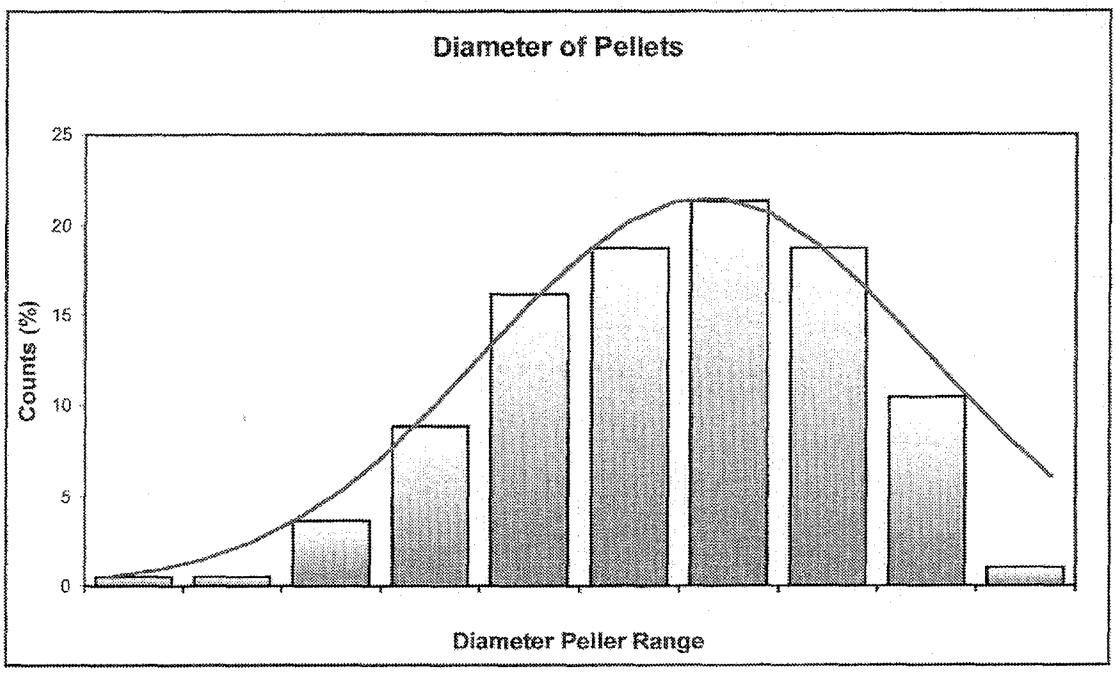

FIG. 8. Histogram of the pellet diameters. The columns are between the maximum and minimum specified values [$Ø_p$ = (1.040 ± 0.001) mm]. The curve is the associated Gauss distribution.

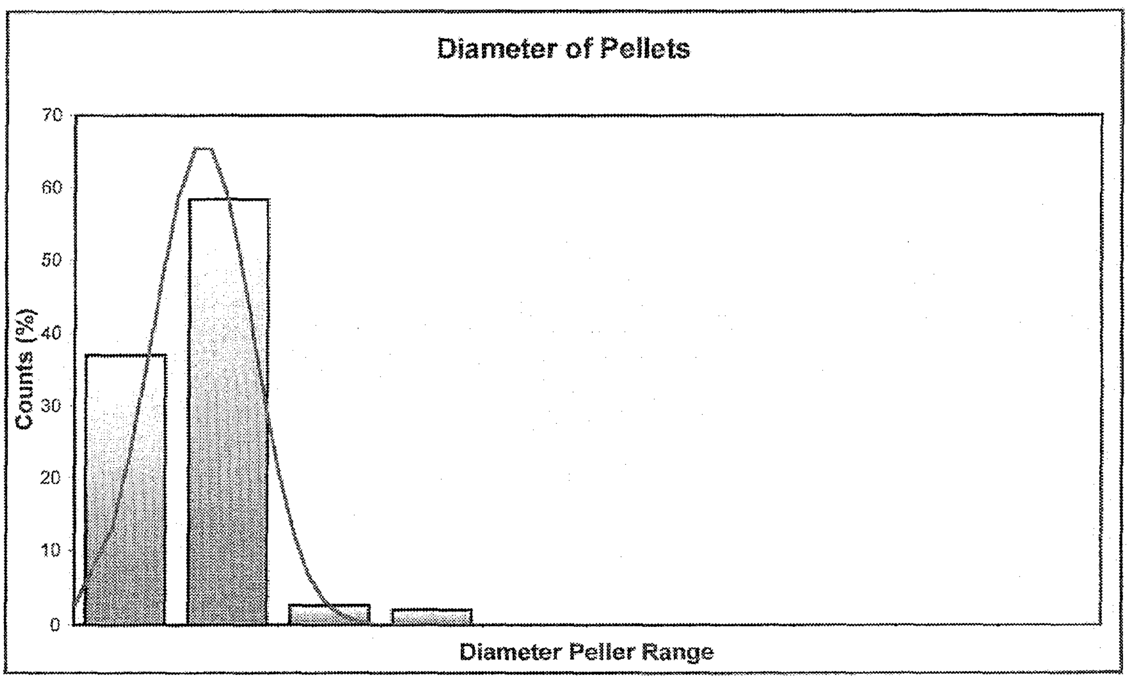

FIG. 9. Histogram of the pellet heights. The ten columns are between the maximum and minimum specified values [h_p = (11. ± 1.) mm]. The curve is an approximated Gauss distribution.

probability distribution of the fuel parameters with a big sample (pellet density, dishing volume, etc) or taking the measurements of all the pellets (pellet diameter, pellet height, etc.). Table II includes some statistical parameters of the fuel and maximum and minimum values found. Figure 8 shows the histogram for the pellet diameter distribution. Figure 9 includes the histogram for the pellet height. Both histograms are between the specified maximum values. The figures include the Gauss distribution curves.

The first irradiation was with the A.1.4 rod. We intended to calibrate the HFR reactor instruments. That experience verified the response of the HFR reactor detection systems with our MOX rod.

3.2. Doped MOX fuel rods A.3 (with Iodine) and A.4 (with CsI)

The experiment started with the irradiation of two fuel rods similar to pathfinder, both containing Iodine compounds as dope to simulate the effect of extending burnup [26].

Fuel rods A.3 and A.4 have been employed for this second experiment. This irradiation, carried out during 15 days consisted in a power cycling between 120 and 290 W/cm and a final ramp to reach 400 W/cm [27]. This final power level lead to a "hoop stress" of about 170 MPa which is likely to induce microcracks in the inner surface of the cladding, representing an incipient defect produced by stress corrosion cracking, without reaching the failure threshold of the tubing.

Rod A.4: CsI and Mo were mixed with UO_2, the mixture was introduced in central holes drilled in three pellets and hand pressed.

Rod A.3: Pure iodine was introduced in one pellet with the aim of comparing its effect on the cladding with that of combined iodine (CsI).

BACO code was employed to design the power history. The experimental power history was in good accordance with the proposed power history. Rod A.3 and A.4 behaved as expected:

- There were no failures in the cladding,
- No fabrication defects have been detected, and
- Microcracks in the inner surface of the cladding were detected during PIE.

There were good agreements between post-irradiation and BACO code output especially about stress analysis and gap dimension [3].

3.3. BU15 experiment

The third irradiation experiment was carried out with fuel rods A.1.2 and A.1.3, both similar to the pathfinder. A burnup of approximately 15 000 MWd/ton(M) was reached in both rod lets. This required 531.5 irradiation days. Rod A.1.3 was submitted to EOL power ramp.

Rods A.1.2 and A.1.3 behaved during the stationary phase as it was expected:
- No rod failures were detected,
- No fabrication defects were evident.

During the EOL power ramp, rod A.1.3 behaved as follows:
- A maximum power of 390 W/cm was reached,
- The power ramp had to be finished earlier than planned due to an increase in activity in the coolant circuit,

- Visual inspection of the fuel rod revealed the existence of a small circular hole in the cladding,

- This kind of failure is likely to be produced by SCC.

The aim of this step of the experiment was to study the stresses in the rod submitted to the power transient in order to determine the causes of the failure.

3.3.1. Irradiation power history

The BU15 irradiation experiment was carried out with fuel rods A.1.2 and A.1.3, both similar to the pathfinder. The power history for this irradiation test was proposed upon calculations made with the BACO code. The target duration of the irradiation was an average burnup of 15 000 MWd/ton(M), more or less twice the final burnup for Atucha I fuel. After the steady state irradiation at an average power level of 230 W/cm one of the rods was submitted to a power ramp [28].

During the steady state irradiation both rods were assembled together through a threaded coupling. This system permits an easy disassemble of the two rods in any irradiation stage.

The average burnup of 15 000 MWd/t(M) was reached in two main steps:

a) Up to a burnup of 8100 MWd/t(M) the rods were irradiated in different locations of the HFR core. The irradiation device consisted of an aluminium fuel rod capsule located in either of two different carriers. The first of these consisted of an aluminium structure in the core section while the other had a hollow stainless steel structure having the possibility of containing pressurized BF_3 gas for power control purposes. The primary pressure in the fuel rod capsule was kept at 145 bar, and

b) Up to a burnup of 15 000 MWd/t(M) both rods were irradiated in the Pulse Side Facility (PSF) of the reactor. The irradiation device consisted of an aluminium fuel rod capsule located in a special PSF capsule carrier which can be translated parallel to the HFR core box wall. Also the primary pressure in the fuel rod capsule was kept at 145 bar.

After termination of the bulk irradiation phase both fuel rods were disassembled in the Petten hot cells and the A.1.3 rod was prepared for insertion into a standard LWR fuel testing capsule consisting of an aluminium capsule located in a similar carrier as described in *b*), which allowed to adjust it to a specified power level. In this case the primary system pressure in the fuel rod capsule was kept during the ramp test phase at 115 bar (e.g. PHWR system pressure).

The ramp proposed consisted of two parts:

1) A short period of pre-irradiation power level in the PSF position which will be used for the ramp test in order to determine the experiment power versus PSF position characteristics, and

2) The ramp test starting with a ramp rate of 50 $Wcm^{-1}min^{-1}$ from nearly zero power to maximum 420 W/cm linear fuel rod power and followed by a 6 hours steady state holding at this maximum power.

The ramp test was performed following the proposed schedule. However, instead of the anticipated high power level of 420 W/cm the maximum fuel power reached only a maximum ramp power level of approximately 380 W/cm, due to the low quantity of remaining fissile material in the fuel rod.

Rods A.1.2 and A.1.3 behaved during the stationary phase as it was expected:

- No rod failures were detected,
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During the EOL power ramp, rod A.1.3 behaved as follows:

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- The power ramp had to be finished earlier than planned due to an increase in activity in the coolant circuit,
- Visual inspection of the fuel rod revealed the existence of a small circular hole in the cladding.

During operation for the entire experiment the total power production was determined via a heat balance approach by measuring the increase in temperature of the capsule coolant water and then the real power fuel rod production by difference with the measured nuclear heating of an empty capsule structure.

The power history includes the normal cycle operation, which means reactor shutdowns, variations from the planned average power in the fuel rods of approximately 100 W/cm and the final ramp test for the A.1.3 fuel.

The total time for this test was 1020.2 calendar days, corresponding to 26 irradiation periods (531.5 days) and 26 periods on which the fuels rods were not irradiated due to changes of irradiation capsules, reactor shutdowns or non availability of a suitable core position.

After a cooling time of 49 days and 9 hours rod A.1.3 was submitted to a power ramp.

Before the ramp test a preconditioning irradiation was performed, increasing power rate at 7.2 W/cm.min during 42 minutes until an average fuel rod power of 250/260 W/cm was reached. This last power level was held 39 minutes. The fuel rod power was then reduced during 5 minutes at a power rate of 48 W/cm.min until an average fuel rod power of 10 W/cm was attained, after 10 minutes, the power level was increased at a rate of 43 W/cm.min during 8 minutes to a final average rod power of 350/360 W/cm. This power was held during 52 minutes.

At approximately 28 minutes operation at the ramp power level an increase of the activity of the primary water was detected. This activity peak decreased after 6 minutes to its previous level. As this event gave reason for assumption of a fuel rod failure the experiment was 24 minutes after the start of the activity release shutdown at a power rate of 175 W/cm.min during 2 minutes.

It is noteworthy that at the beginning of life, the A.1.2 and the A.1.3 fuel rods reached a maximum peak fuel rod power of 430 W/cm and 380 W/cm, respectively. The average for the assembly was 330 W/cm. These maximum peak power levels, during that pre-irradiation period, where higher or equal than the maximum power reached during the final ramp test for the A.1.3 fuel rod.

3.3.2. Post-irradiation analysis

After finishing the low level irradiation in the HFR, the irradiation device was disassembled and some non-destructive examinations were done on both pins. These examinations were repeated on fuel rod A.1.3. after the transient. Destructive examinations were performed on both rods (for A.1.2 after the low level irradiation and for A.1.3. after the transient). The main findings of the post-irradiation examinations are described in the following paragraphs.

The irradiation-induced effects were studied in detailed post-irradiation-examination, which was performed at the HFR site in Petten, and later at the hot cells of the Karlsruhe Nuclear Research Centre.

We do not give too much detail about this aspect of the experiment because the aim of the report is about BACO code comparison. We indicate only the main post-irradiation analysis made with the pins, as follows (references [2, 25, 29, 30 and 31]):

- Visual inspection
- Eddy current check
- Neutron radiography
- Gamma scanning
- Betatron-radiography
- Dimensional control [32]
- Sectioning diagram
- Ceramography
- Scanning Electron Microscopy (Fig. 17 and 18 show the microcracks of the rod A.1.3)

The analysis of the experiment made with the BACO code and a complete description of the calculation was presented at the IAEA's TCM on "Water Reactor Fuel Element Modelling at High Burnup and Experimental Support" [3]. At that meeting it was suggested that additional post-irradiation examinations and more details of the failure condition would be needed to confirm the failure as being due to PCI. That report was continued in the IAEA's TCM on "Recycling of Plutonium and Uranium in Water Reactor Fuel" [2]. At that meeting we presented a new set of post-irradiation examinations (not available at the previous meeting). The results attained show a PCI-SCC failure in the A.1.3.

3.4. First results for the MOX fuel irradiation and the BACO code performance

The absence of fabrication induced failures in the irradiated rods is an important achievement in our MOX fuel development programme.

The presence of microcracks inside the cladding in the doped rods, the coincidence between the predicted and measured pellet-cladding gap values, the temperature calculated and the microstructure observed, the grain sizes distribution, indicated a good BACO code evaluation.

The defective zone of the A.1.3 rod had the biggest mechanical demands. The calculated hoop stress with BACO 2.20 was enough to indicate that PCI-SCC was the mechanism likely to produce the failure. The maximum hoop stress and pellet-cladding radial contact pressure appears in the axial section corresponding to the failure. That prediction was confirmed with the post-irradiation examinations.

4. BACO CODE SIMULATION OF THE MOX FUEL BEHAVIOUR DURING THE BU15 EXPERIMENT

The reactor's assumed power history which is used here for the fuel rod performance analysis was explained in 3.3.1) and is sketched in Figure 10. We consider five segments for calculation. Bottom axial section suffers the high power during irradiation. However, the biggest demand happens at the final power ramp for the top section due to its inversion at the PSF. We include a BACO code analysis at Reference [3] using an old version (2.20) of the code. There is no substantive difference in the usual calculation.

The calculated temperature corresponds to the post-irradiation. The maximum temperature calculated was 1600°C at the bottom of the fuel rod (See Figure 11).

Compression is generally predicted for the "hoop stress" calculated at the cladding (i.e., tangential stress at the inner surface of the cladding). However, stress reversal happens due to local power increment followed by stress relaxation, i.e. creep of the cladding. The maximum hoop stress calculated was up to 300 MP during the final ramp test at the top of the fuel (defective section). The maximum calculated radial contact pressure was up to 50 MPa. (See the fill curve in Figure 13).

There is an increment of pressure in the rod with irradiation due to the fission gas release and local variations are induced by power changes. The calculated gas pressure at EOL was 1.6 Mpa (See Figure 15).

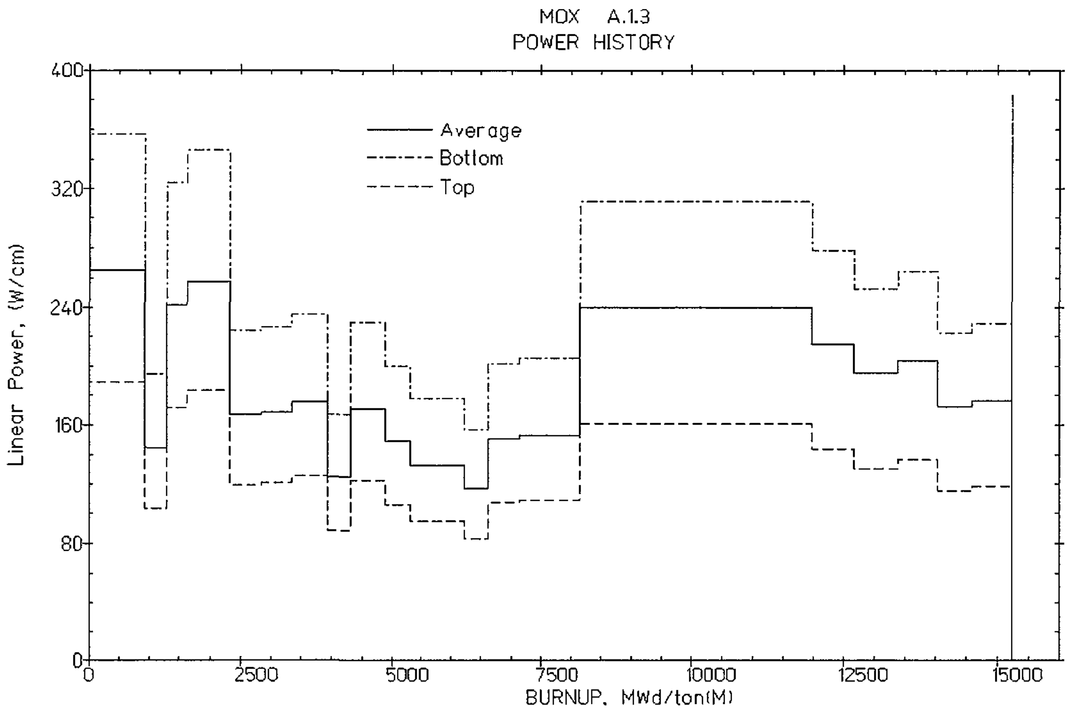

Fig. 10. Linear heat generation rate in function of burnup for the A.1.3 fuel rod. The upper curve corresponds to the top of the fuel, the lower curve corresponds to the defective zone of the rod.

4.3. Parametric analysis of the MOX fuel rod

The purpose of this exercise is to consider how the combination of assumed extreme rod dimension conditions — within reasonable tolerance for its fabrication — can affect performance. We define two extreme situations:

A rod with the largest gap between pellet and cladding compatible with the as-fabricated tolerances, and

A rod with the smallest gap.

The first situation should give rise to the maximum temperature in the fuel, and the second to maximum stress between pellet and cladding.

For the same power history of Figure 10, Figure 11 includes the pellet centre temperature at the maximum gap situation at the bottom of the fuel rod. The largest temperature attained in this case is 1675°C (while the minimum gap situation is 1600°C). Figure 13 includes the BACO calculation of hoop stress with a minimum gap situation. Here we do not see a big change; the wide range between both situations at the middle of life is due to the hard contact that occurs. The calculated curves show a narrow band due to the strict QA in lab conditions.

We obtain a stable solution, with the three parameters mentioned, which proves that the BACO code is a good tool to be used for fuel rod design.

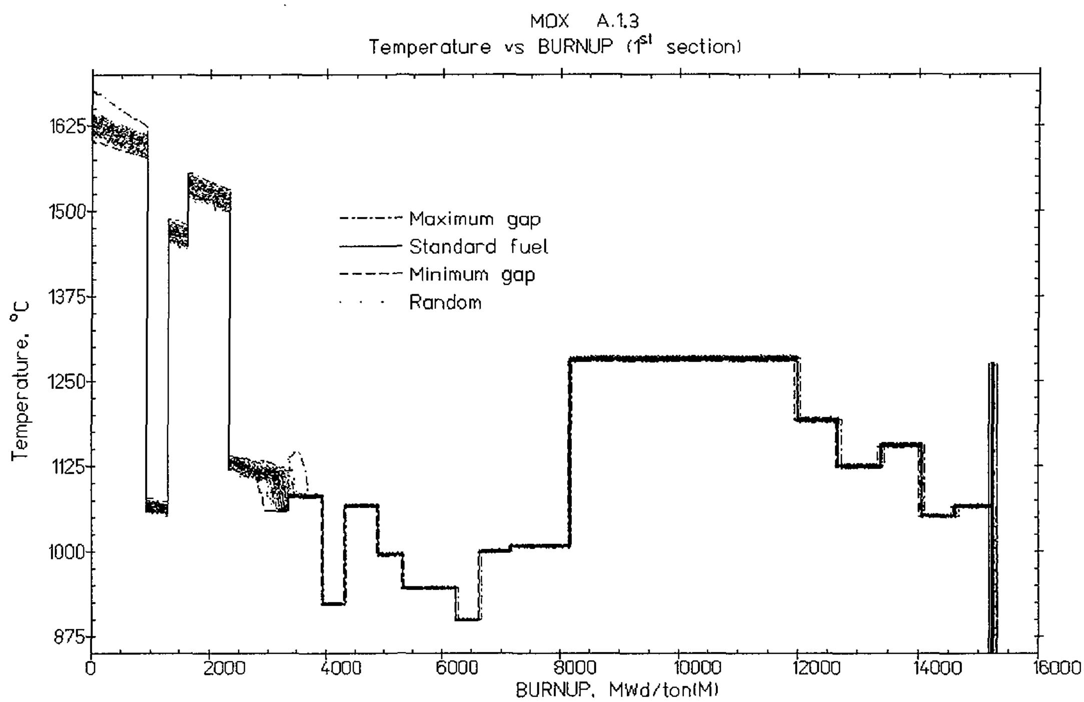

FIG. 11. Pellet centre temperature in the first segment of the fuel rod for the BU15 experiment (A.1.3 rod).

4.4. Sensitivity analysis of the BU15 experiment

As outlined in the Introduction, the flexibility of the BACO code and its speed in computer time allows performing systematic statistical analysis. Using allowed fabrication dimensional limits

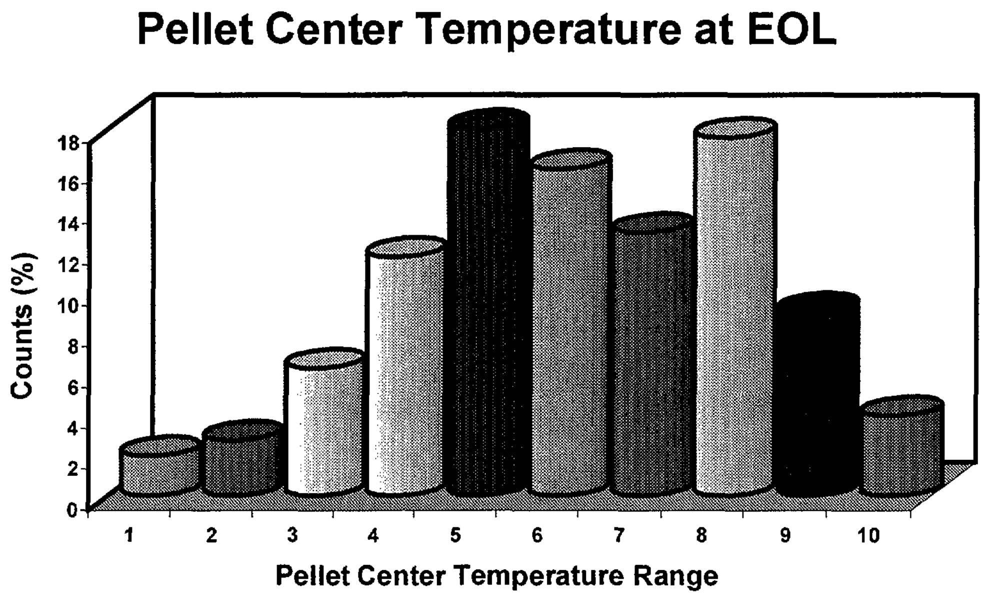

FIG. 12. Histogram of the pellet centre temperature at End of Life (EOL). The columns are between the maximum and minimum calculated values (1275-1278°C).

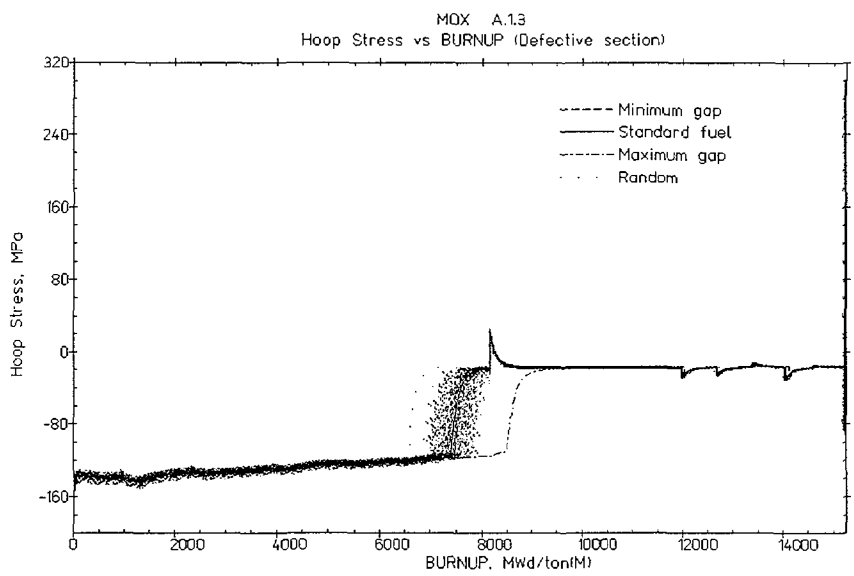

FIG. 13. Hoop stress at the defective segment of the fuel rod for the BU15 experiment (A.1.3 rod).

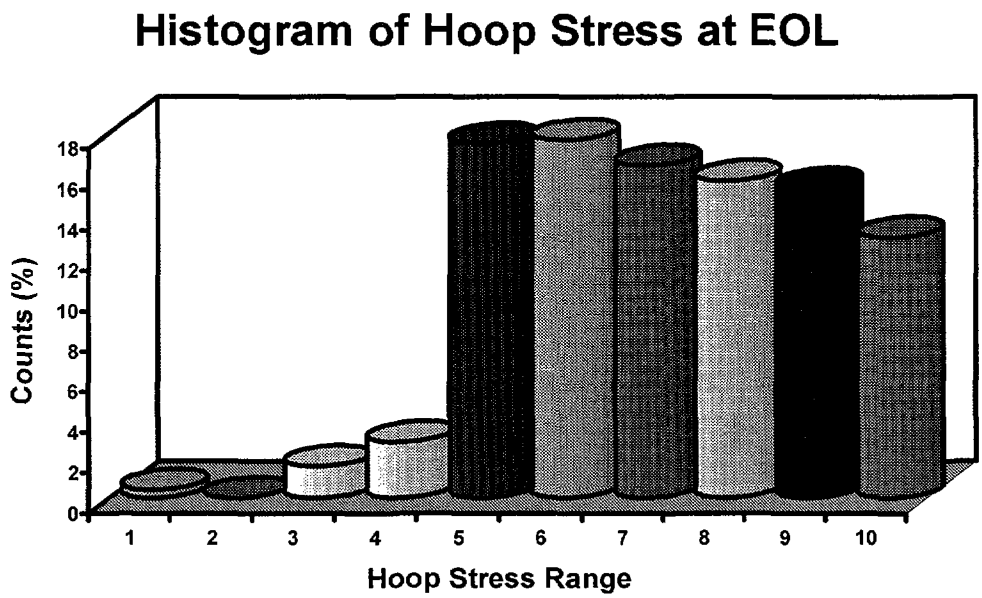

FIG. 14. Histogram of the hoop stress at End of Life (EOL). The columns are between the maximum and minimum calculated values (292-302 MPa).

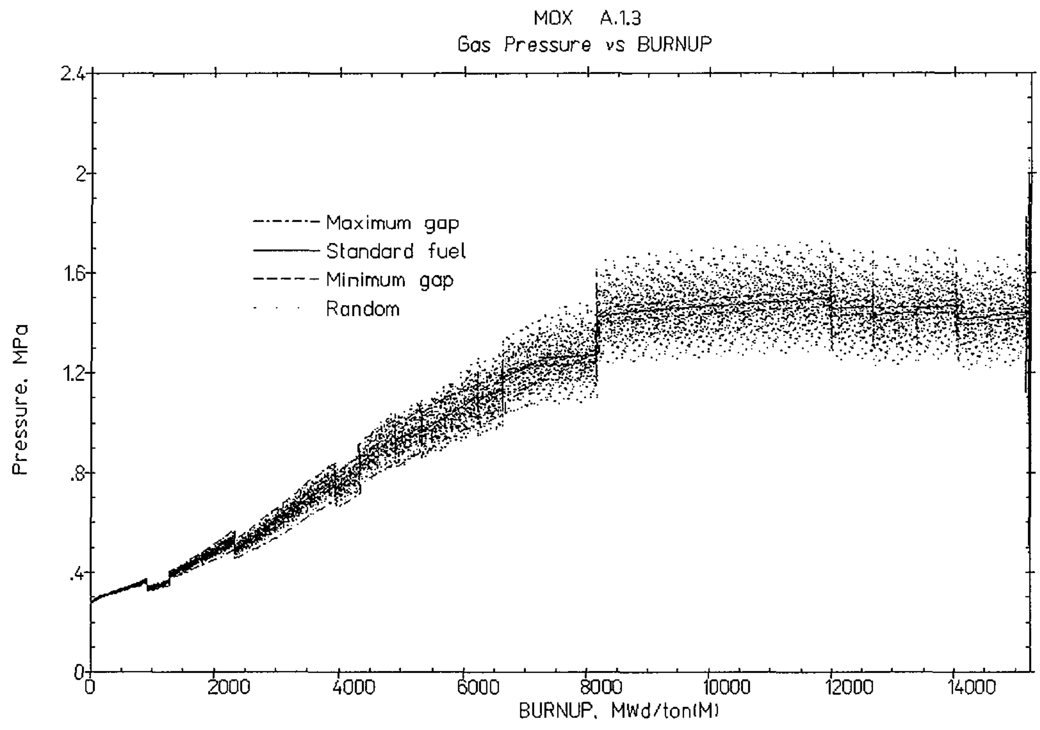

FIG. 15. Gas pressure of the free gases in the fuel rod for the BU15 experiment (A.1.3 rod).

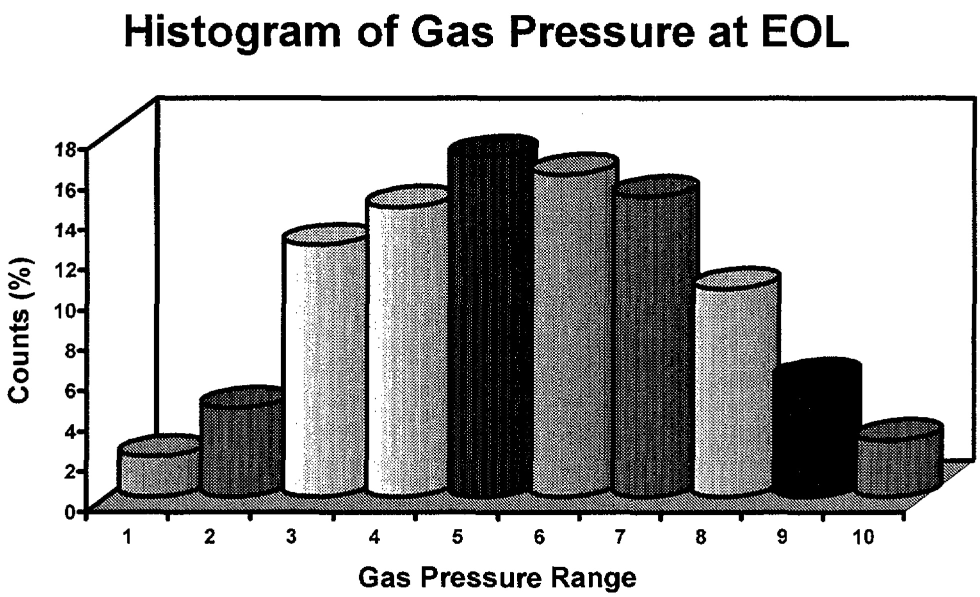

FIG. 16. Histogram of the gas pressure of the free gases in the rod at End of Life (EOL). The columns are between the maximum and minimum calculated values (1.60-2.07 MPa).

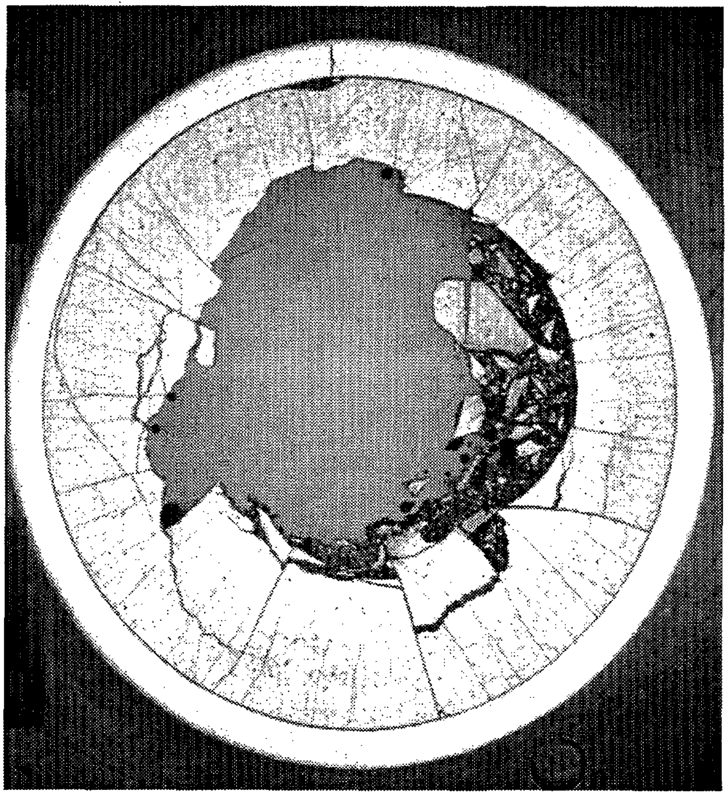

FIG. 17. Micrograph of a cross section near the defect in rod A.1.3. showing the hole. (Coversheet of Journal of Nuclear Material 229 (1996), [25]).

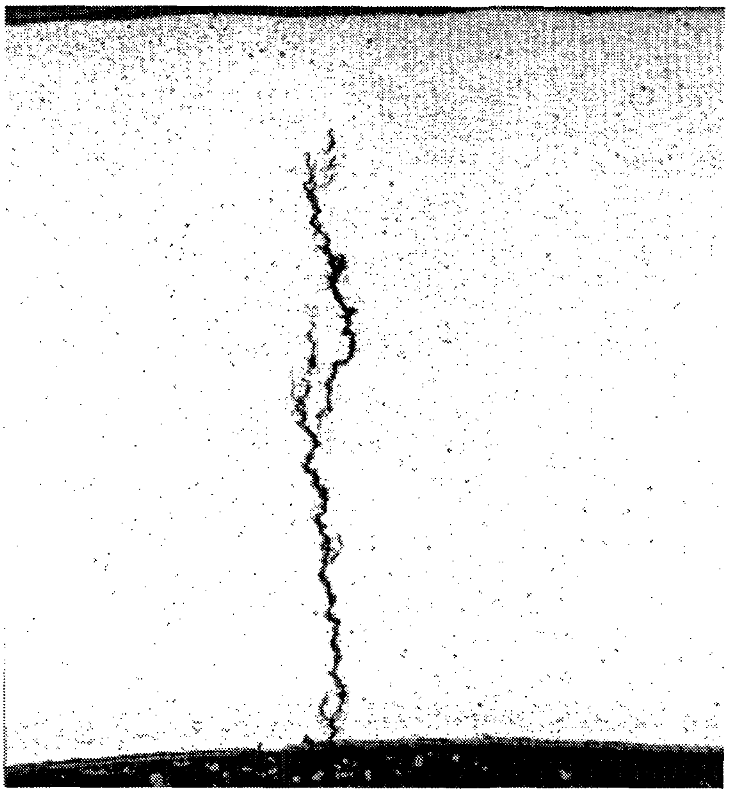

FIG. 18. Micrograph of the defect in fuel rod A.1.3. Cross section over the hole in the fuel.

and a statistical distribution of values within those; several runs (a minimum of 1000) are performed with different sets of initial values for the rod dimensions. We study the predicted variations in:

Pellet centre temperature,
Cladding hoop stress, and
Gas pressure predictions.

The rod input data were randomly selected within assumed deviations for pellet diameter and height, inner and outer diameter of the cladding and pellet density. The random selection of input values was done assuming a Gauss distribution of values between limits. See Table II and Figures 8 and 9.

Figures 11, 13 and 15 represent the BACO code sensitivity analysis of some performance parameters in the MOX fuel rod A.1.3. We plot in the curves:

Standard parameters of input data of the rod,
The parameters of the maximum gap situation,
The parameters of the minimum gap situation, and
The points of the random selection.

Figure 11 is the BACO code calculation of the pellet centre temperature for the same viewer history of Figure 1. All the random points calculated are between the extreme values in as-fabricated tolerances, taken at approximate realistic values. There is convergence of dots at EOL (End of Life) due to pellet-clad contact. Figure 12 shows a histogram of the pellet centre temperature at EOL, after the final ramp.

Figure 13 shows the dispersion in the cladding hoop stress with the same inputs as the previous plot. The points show a great dispersion at the middle of life due to the pellet-cladding contact situation. There are points out of the extreme limits of the previous analysis. The calculation shows that the hoop stresses converge during the irradiation that is clearly demonstrated by the small dispersion at EOL previous to the final ramp (see Figure 14).

Figure 15 is the free gas pressure at the fuel rod. The gas pressure calculation takes account of the thermal calculation, dimensional calculation (stresses), fission gas release, etc. That is the coupling of all the fuel rod parameters (input data and behaviour modelling). There is a small dispersion at BOL. The calculated values of pressure diverge during irradiation. Finally, after 4000 MWd/ton(M), there are values both smaller and larger than those predicted at the extreme conditions of the "gap" size situation. Figure 16 is a histogram of the rod gas pressure of the free gases at EOL. The main value agrees with the one calculated for the standard fuel parameters.

A similar detailed work is included in Reference [13] where a CANDU fuel analysis is done. At that situation the dispersion of parameters is relevant.

5. CONCLUSIONS

The schedule sketched in this paper begins with a validated code for the simulation of the fuel rod behaviour under irradiation, almost for internal use in the institution (2.3). The help obtained from international projects such as the CRP FUMEX are relevant at this point (2.3.4). The enhancement of fuel modelling about relocation was sketched with the classical blind test of EPRI (2.3.2). The application of simple rules was included in the CANDU fuelograms analyses (2.3.2). An example of institutional benchmarking of the BACO code was presented with the NRX irradiation (2.3.1).

A MOX fuel rod failure due to PCI-SCC was presented. The BACO code was a computing tool during all the stages of the experiments. The original scope of the MOX irradiation was the correct research, developing and manufacturing of MOX fuels in the α Facility. An additional developing was the induction of fuel failures due to SCC mechanism and the simulation of burnup extension with synthetic products (CsI and Iodine).

We are showing that a simple running code is not enough in order to simulate the behaviour of a fuel rod. Parametric analysis of extreme cases must be done. But the analysis sketched shows that the study of extreme cases is not enough. The smallest dispersion found in the selected parameters (temperature, hoop stress and gas pressure) is due to the QA procedures in the laboratories. Nevertheless, it is easy to see that the probability distributions of the fuel rod parameters must be known and statistical analysis must be included in order to follow the correct influence of the manufacturing QA procedure of fuel elements. A complete fuel element design must consider the dispersion in rod dimensions due to fabrication. Changes in the design of rod in fabrication parameters can be tested. This exercise shows, on the one hand, the sensitivity of the predictions concerning such parameters and, on the other, the potentiality of the BACO code for a probability study.

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