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Fuel modelling at extended burnup

Report of the Co-ordinated Research Programme on Fuel Modelling at Extended Burnup — FUMEX 1993–1996

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FOREWORD

It is fundamental to the future of nuclear power that reactors can be run safely and economically to compete with other forms of power generation. As a consequence, it is essential to develop the understanding of fuel performance and to embody that knowledge in codes to provide best estimate predictions of fuel behaviour. This in turn leads to a better understanding of fuel performance, a reduction in operating margins, flexibility in fuel management and improved operating economics.

Reliable prediction of fuel behaviour constitutes a basic demand for safety based calculations, for design purposes and for fuel performance assessments. Owing to the large number of interacting physical, chemical and thermomechanical phenomena occurring in the fuel rod during irradiation, it is necessary to perform calculations using computer codes. The ultimate goal is a description of fuel behaviour in both normal and abnormal conditions. From this knowledge, operating rules can be derived to prevent fuel failures and the release of fission products to the environment, also, in extreme cases, to prevent escalation of fuel and core damage and the consequential hazards.

The IAEA has therefore embarked on a series of programmes addressing different aspects of fuel behaviour modelling with the following objectives:

- to assess the maturity and prediction capabilities of fuel performance codes, and support interaction and information exchange between countries with code development and application needs (FUMEX);
- to build a database of well defined experiments suitable for code validation in association with OECD/NEA;
- to transfer a mature fuel modelling code to developing countries, to support teams in these countries in their efforts to adapt the code to the requirements of particular reactors, and give guidance on applying the code to reactor operation and safety assessments (RER 4/012);
- to provide guidelines for code quality assurance, code licensing and code application to fuel licensing.

This report describes the results of the FUMEX programme. This programme was initiated in early 1993 and completed in 1996. It followed a similar programme on fuel modelling called D-COM which was conducted between 1982 and 1984.

FUMEX was made possible as a result of the support and dedication of many organizations and individuals. The IAEA would like to thank the International Working Group on Fuel Performance and Technology (IWGFPT) for suggesting and supporting the programme, the OECD Halden Reactor Project for providing the experimental data and the participants for performing the calculations and submission of summaries and meeting contributions. During the course of FUMEX, the IAEA was advised by experts who also prepared the intermediate working material and the final report, these were mainly J.A. Turnbull (United Kingdom), W. Wiesenak (Halden Project), Y. Guerin (CEA), L. Alvarez and P. Chantoin, Scientific Secretary of this programme.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

The FUMEX co-ordinated research programme was initiated by the IAEA following a recommendation of the IWGFPT. It was conducted over the period 1993 to 1996. Fifteen countries took part. The FUMEX programme continued the work of the former CRP on "The development of computer models for fuel element behaviour in water reactors" (D-COM), which started in 1982 and was terminated in 1984.

The elements of the CRP were defined as follows:

- A blind prediction to be carried out by the participants on data provided by the Halden Project, Norway, in the form of irradiation histories, in-pile measurements and PIE of six experiments involving 10 fuel rods. Only after all the predictions were submitted were the measurements released.

- A comparison of calculations carried out after code improvement on the 10 rods of the FUMEX blind exercise.

- The definition of eight simplified cases, to assess code response to changes of single parameters such as internal gas composition, burnup, power steps, and a statistical analysis of two of the simplified cases.

- Follow-up of code status, progress in modelling and modification made at research co-ordination meetings (RCMs), also providing a forum for discussion and interaction among participants.

In early 1993 the specifications of six experiments performed at the Halden Project (Norway) were distributed to the participants. The first research co-ordination meeting took place in Halden, 28 June-1 July 1993. During this meeting a description of the 19 codes was given and the preliminary results were released.

The second RCM took place on 15 and 16 September 1994 in Windermere (UK). Here, the outcome of code predictions were discussed along with the future actions to be taken by the participants in code development and improvements. There was a general agreement that each participant should rerun the original FUMEX study, conduct a new study on simplified cases and a limited sensitivity study based on agreed uncertainties of power and dimensions to investigate the sensitivity of predictions.

The third RCM was held in Mumbai (India), 1-5 April 1996. The meeting focused on elementary model improvement, the impact of the FUMEX programme and the recommendations from the participating countries. In this meeting the role of quality assurance in developing and maintaining fuel performance codes was also introduced.

This report provides a description of the experiments chosen, an overview of the codes used by participants in the exercise, and the improvements implemented as a consequence of FUMEX. A commentary is given regarding the various aspects of fuel behaviour tested and a detailed quantitative comparison is made between experimental data and code predictions. The report concludes with a discussion of the main findings of the exercise, the identified improvements and shortcomings in codes and modelling, and outstanding technical issues that require further attention.

2. THE D-COM BLIND EXERCISE

The list of participants in the D-COM blind exercise is given in Table I. The detailed consultants report presenting the state-of-the-art in modelling the fuel rod behaviour and including a comprehensive review of fuel rod computer codes at that time is given in Refs [1, 2].

As part of this program a code exercise was conducted [3], where the objective was to investigate the capability of fuel performance codes to predict fission gas release. The test cases to be calculated by the codes consisted of three mini pins irradiated together (test HP 096) in the Danish DR 3 test reactor to a burnup of 32000 MW·d/tU. Two of the pins were finally bumped together with average heat ratings of 33.7 and 36.2 kW/m respectively at the end of bump. The blind code predictions were presented at the OECD-NEA-CSNI/IAEA Specialists' Meeting on Water Reactor Fuel Safety and Fission Product Release in Off-Normal and Accident Conditions, Risö National Laboratory, 1983 [4]. However, the results were not included in the proceedings of the meeting, but some are given in Ref. [5].

The main conclusions from the D-COM exercise were as follows:

(a) Temperature

Temperature predictions showed a large spread.

(b) Fission gas release

Fission gas release during the base irradiation was in fair agreement with experimental values. The fission gas release during the transient (bump test) was under-predicted by most of the codes.

(c) Mechanical behaviour

Since the exercise concentrated on the thermal behaviour and gas release, many participants did not provide dimensional data. Of those codes which submitted mechanical data most codes predicted the cladding creep down reasonably well, mechanical data during the ramp were scarce and showed considerable spread.

The D-COM blind code exercise was considered by participants as being very valuable in promoting discussions among modellers. A better knowledge of the centre line temperature during base irradiation was identified as an area of further development. It was also stated in the conclusions that basic phenomena such as gaseous swelling, transient gas release and grain growth should be better known during transients.

The subsequent experimental programmes both at Halden and Risö addressed these requests. Within these projects it was demonstrated that the fuel thermal conductivity degrades with burnup and can be modelled by an additional phonon contribution. The effect of this degradation is a higher fuel temperature which partially explains the general underprediction of fission gas release in the transient of the D-COM blind prediction.

It is of interest to note that some modelling groups that participated in the D-COM exercise also participated in the FUMEX blind exercise. This list is shown in Table II.

3. THE FUMEX BLIND EXERCISE

Following the D-COM exercise, the IAEA initiated a second code comparison exercise in 1993 addressing fuel thermal performance and fission gas release at high burnup as well as aspects of pellet-cladding mechanical interaction. There were a total of six cases, FUMEX 1-6 including 10 rods, which represented actual irradiations in the OECD Halden Heavy Water Reactor in Norway.

3.1. DESCRIPTION OF THE CODES USED IN THE FUMEX EXERCISE

Within the FUMEX exercise, blind predictions were submitted from 15 countries employing 19 codes or code variants (Table II). A list of Chief Scientific Investigators is given in Appendix I. Details of the codes are given in Appendix II. Their similarity in construction allows the following general observations.

All codes use an axi-symmetric fuel rod representation and consist of three main parts:

(i) Thermal analysis including gap conductance models which account for different pin pressures, gas compositions and gap sizes; standard correlations for the thermal conductivity of fuel and cladding are used. Standard numerical techniques such as finite difference (FD) and finite element (FE) methods are applied.

(ii) Mechanical analysis including cracking and relocation of fuel pellets; in a few cases a simplified mechanical treatment of the fuel is adopted. However, most codes are based on an axi-symmetric, modified plane strain assumption. Two codes offer the capability of a two dimensional treatment. FD and FE methods are used.

(iii) A variety of physical models or empirical correlations are used for densification, swelling, fission gas release, grain growth, etc.

The number of executable statements ranges from 2000 up to 30 000 and all the code descriptions claim that the codes represent state-of-the-art modelling. Two codes were specifically designed and validated for HWRs with a collapsible cladding. As to be expected, these codes showed some deficiencies in predicting an open gap situation, and modifications were necessary when applied to the Halden irradiated rods.

3.2. EXPERIMENTAL DATA USED FOR THE COMPARISON EXERCISE

The FUMEX irradiations were all provided by the OECD Halden Reactor Project. They represent a selection of experiments from the Halden Project fuel testing programme, which for a number of years has focussed on consequences of extended burnup on fuel operation. Rod characteristics utilized in FUMEX are given in Table III and simplified power histories in Figure 1. The six cases can be summarized briefly as follows:

FUMEX 1 — This data set represents the irradiation of production line PWR type fuel under benign conditions. Temperatures remained low but increased slightly with burnup.

FUMEX 2 — This was a small diameter rod designed to achieve rapid accumulation of burnup. Temperatures were estimated to remain low. The internal pin pressure was measured in-pile and an assessment of FGR was also provided by PIE.

FUMEX 3 This case consisted of 3 short rods equipped with centreline thermocouples each with a different gap and fill gas composition. After steady state irradiation to ~30 MW·d/kg UO_2, they were given a severe increase in power (power ramp).

FUMEX 4 Two rods filled with 3 bar He and 1 bar He/Xe mixture were irradiated to ~33 MW·d/kg UO_2. Both rods experienced a period of increased power part way through the irradiation.

FUMEX 5 The test case comprised a single rod base irradiated at low power to 16 MW·d/kg UO_2 with a power ramp and a hold period at the end of life. The main purpose of this case was to assess pellet clad mechanical interaction (PCMI) and fission gas release (FGR) under ramp conditions.

FUMEX 6 Two rods were base irradiated at low power. The rods were refabricated to include pressure transducers. Rod internal pressure was monitored during power ramps, one fast, one slow.

Further details are given in Appendix III.

3.2.1. Irradiation conditions in the Halden Reactor

The Halden Reactor is a heavy water moderated and cooled boiling water reactor. The nominal operation conditions are 240°C coolant temperature and a corresponding saturation pressure of 34 bar. These conditions imply decreased uncertainties for some effects with an influence on experimental results and data evaluation, namely:

- cladding creep-down is very small,
- cladding oxidation can be practically neglected,
- boiling conditions can be assumed for the entire length of the rod and consequently the clad surface temperature is known with high accuracy.

The reactor is operated with three major shut-downs per year for loading/unloading of driver fuel and experiments.

3.2.2. In-core instrumentation and experimental techniques

In-core instrumentation combined with suitable experimental techniques and test designs is the key to meaningful results for model development and validation. While PIE ascertains the state existing at the end of irradiation, in-pile instrumentation gives information on how phenomena developed during in-core service. The ten instrumented rods of the FUMEX cases thus provide a good basis for studying key parameters of fuel modelling. A summary of parameters recorded in each FUMEX rod is given in Table IV.

A general overview of instrumentation and experimental techniques applied in the Halden Project fuel testing programmes can be found in Refs [19-21]. Those employed for producing FUMEX data are repeated below.

Instrumentation

Fuel centre temperatures were measured with *refractory metal thermocouples*, the rod internal pressure was determined with *bellows pressure transducers*, and *elongation sensors* were used to obtain the length increase of the cladding. In FUMEX 4, the rods contained all three types of instrument, providing comprehensive information on the state of the fuel.

The *diameter gauge* is a two or three point contact feeler that can be moved along the length of a fuel rod during operation. Diameter changes can be detected with a micrometer resolution as demonstrated in FUMEX 5.

Re-instrumentation of irradiated fuel rods

Re-instrumentation of irradiated fuel rods is a method of shortening experiment execution times and costs. Since the instrumentation is exposed to irradiation for a shorter duration, the failure probability is decreased. FUMEX 6 is an example of re-instrumentation with a pressure transducer. The data obtained from this experiment gave a good indication of on-set and kinetics of fission gas release in response to a power increase.

Design features for simulation of burnup effects

Increasing burnup in general incurs increasing uncertainties in data interpretation, e.g. fuel temperature changes can be effected by a combination of causes like fission gas release, changes in gap size and conductivity degradation. Test designs with controlled and known influential parameters therefore facilitate the assessment of separate effects.

FUMEX 3 and FUMEX 4 are examples where fission gas release was simulated by addition of xenon to the fill gas. Xenon fill gas in combination with a small as-fabricated diametral gap (50-100 μm) simulates a high burnup situation with a large amount of released fission gas and a closed gap due to fuel swelling and clad creep-down. Gap conductance models can be validated with measured data from tests with such a design without the uncertainties caused by high burnup.

3.2.3. Error estimation of power and temperature data

At the first start-up of an experiment, assembly power is determined calorimetrically, resulting in a relation between total power and average neutron flux measured by neutron detectors. The calibration error is about 3% which has to be combined with a similar uncertainty for the distribution of total power to individual rods and local positions. The evaluation of start-up data of a large number of comparable HBWR experiments has indeed shown that the observed spread agrees with these considerations and is about 4%. During the course of irradiation, small local changes of the neutron flux distribution, which cannot be resolved with the arrangement of neutron detectors, will add another uncertainty. The combination of all sources leads to an error estimate for power data of about 5% for the time after the initial calibration. It is customary, however, to assign this uncertainty to temperature rather than power. In a representative selection of the FUMEX evaluation sheets, a 5% error on temperature above coolant temperature is indicated with a corresponding error bar; see Figures 2, 12 and 14.

4. RESULTS OF THE COMPARISON EXERCISE

Although the main benefit of this exercise is for individual code developers to compare their predictions with the experimental data, there is merit in an inter-comparison of code predictions, even where experimental data are absent. This allows the developers to benchmark the capabilities of their codes against others and also provides a general overview of fuel modelling as a predictive science and therefore highlights areas for further attention and possibly further experimental data.

Throughout this section, each case is introduced by a brief overview of the experiment with emphasis on the specific areas of modelling addressed. Comparisons are made at specific periods during the irradiation. These are given along with the figure where the predictions are reproduced. Each code has been given a number as identified in Table II and the predictions are presented in the figures as a function of code number plotted as the abscissa. Where experimental data are available, these are given with their estimated accuracy.

4.1. FUMEX 1

This dataset represents the irradiation of production line PWR type fuel under benign conditions. Temperatures remain low, but the effect of thermal conductivity degradation of the fuel pellet led to a slight increase with burnup. At mid-life, the test rig was moved to a position with higher flux. Some PCMI was observed following this event.

The measured parameters for this experiment were: fuel centreline temperature, total rod average fission gas release during PIE and cladding elongation.

Fuel temperatures at 5 MW·d/kg UO_2 and 15 kW/m; Figure 2

At this burnup, shortly after the start of irradiation and subsequent to the major part of fuel densification, most of the codes under predicted the temperatures. However, the majority of the predictions were within the limits of the experimental uncertainties estimated to be about 50°C.

Fuel temperatures at 20 MW·d/kgUO_2 and 15 kW/m; Figure 3

At this burnup the picture remains the same with the majority of codes under predicting the temperature with values now lying just outside the experimental uncertainties. According to the measured data, the temperature rise between 5 and 20 MW·d/kg UO_2 was approximately 30°C. Half the codes predicted this increase correctly whilst the others predicted decreasing temperatures.

Fission gas release at end-of-life; Figure 4

The through life measured temperature for this rod is given in Figure 5. On discharge, the fractional fission gas released from the fuel was measured to be 1.8%. This is a particularly difficult region for accurate predictions of gas release as it lies near the threshold for grain boundary saturation and small changes in materials and irradiation parameters lead to large variations in gas release. The difficulty is demonstrated by the under prediction of the majority of codes, whilst only 2 codes provided good predictions.

Clad elongation

Where predictions were presented, most codes provided reasonable estimates of clad elongations for free standing cladding due to thermal expansion. However, in contrast with the experimental data, where from measurement, a strong fuel-clad mechanical interaction was evident above 15 kW/m, no such predictions were made by any of the codes.

4.2. FUMEX 2

The experiment was designed for rapid burnup accumulation with the objective of studying fission gas release and fuel thermal conductivity degradation at high burnup. For most of the irradiation, fuel centre temperatures (estimated from a sibling rod) stayed below the empirical threshold for 1% fission gas release established on the basis of many results from Halden reactor tests.

Measured parameters in this experiment were: rod internal pressure and total rod average fission gas release from PIE measurements.

Fuel temperatures at 5 & 50 MW·d/kg UO_2 (EOL) and 15 kW/m; Figure 6

This is purely an inter-code comparison as there were no experimental data for fuel temperatures. All early-in-life predictions of temperatures were within 600 to 750°C. The spread of predictions increased at high burnup, 50 MW·d/kg UO_2, particularly since some codes predicted increasing temperatures others predict decreasing temperatures as a function of burnup. One code stood out as predicting very much lower temperatures than the others at high burnup.

Fuel temperatures at 5 MW·d/kg UO_2 and 40 kW/m; Figure 7

In this inter-code comparison, at low burnup but after fuel densification was mostly complete, the code predictions followed the same pattern as before but with slightly greater scatter reflecting the higher power at which the comparison is made.

Fission gas release; Figure 8

In this case, the experimental release was measured to be 3% and in contrast to FUMEX 1, most codes over predicted the release. As mentioned before, the region 1-3% is particularly difficult to model. However, despite this reservation, within the generally accepted accuracy of a factor of two, the predictions were quite acceptable but somewhat conservative.

Rod internal pressure; Figure 9

Figure 10 shows the internal pressure measurements during irradiation where it can be seen that there is a gradual increase throughout life. The predictions followed the same trend as the fission gas release in slightly over predicting the data. The scatter is somewhat amplified as the pressure predictions also reflect variations in the calculated free volume and temperature distribution in the rod.

4.3. FUMEX 3

This case consisted of three rods with the following parameter variations:

Rod 1	:	fill gas He, gap size 100 μm
Rod 2	:	fill gas Xe, gap size 100 μm
Rod 3	:	fill gas Xe, gap size 50 μm

The xenon fill gas together with a small initial gap in rods 2 and 3 simulated conditions with severe fission gas release at high burnup. The different gap sizes provide a test of gap conductance models and their response to different dimensions.

The curve of temperature versus power show the typical characteristics of helium filled rods, (linear behaviour) and xenon filled rods with a non linear response, Figure 11. This is because for xenon filled rods, the thermal resistance is dominated by that of the gap. As higher powers are reached, the gap is progressively closed by the thermal expansion of the pellet resulting in a reduced resistance across the pellet-clad gap.

The development of the fuel temperatures with burnup reflects the influence of gap closure by fuel swelling and fuel conductivity degradation. In the xenon filled rods, the improvement of gap conductance outweighs conductivity degradation, leading to an overall temperature reduction at constant power with increasing burnup. The helium filled rod is characterized by temperatures increasing with burnup; in this case, the fuel conductivity degradation has a stronger influence than gap closure.

The test finished with a strong power increase at the end-of-life. The temperature response of the helium filled rod indicated high fission gas release.

The measured parameters were fuel centreline temperatures in each of the three rods.

4.3.1. FUMEX 3.1

Fuel temperatures for 30 kW/m at 5 MW·d/kg UO_2 and just before ramp at about 36 MW·d/kg UO_2; Figure 12

Calculations of fuel temperature at 5 MW·d/kg UO_2 were in good agreement with the data, with the scatter within the experimental uncertainty. The experimental evidence was for an increase in temperature of just under 300°C from 5 MW·d/kg UO_2 up to the start of the ramp, see Figure 13. Several codes correctly reflected this increase in their predictions. Codes that did not predict increasing fuel temperatures with burnup consequently severely under predicted the temperatures at high burnup.

Temperature at top of ramp; Figure 14

The power experienced by this fuel rod at the top of the ramp was very high, 60 kW/m, and the measured fuel centreline temperature was marginally in excess of 2000°C. The majority of the codes over predicted this by about 300°C.

Fission gas release; Figure 15

Data on fission gas release are not available for this experiment. Comparing code predictions, there is no clear trend and the predictions before and after the power ramp and the incremental release during the ramp vary significantly from code to code. This must reflect the diversity of models and mechanisms employed in the codes. This becomes more evident when discussing the evolution of pressure in FUMEX cases 5 and 6.

4.3.2. FUMEX 3.2

Fuel temperatures for 25 kW/m at 5 MW·d/kg UO_2 and just before ramp at about 32 MW·d/kg UO_2; Figure 16

This case employed a xenon filled rod which, at constant power, showed a progressive reduction of fuel temperature with burnup. At the power chosen for the comparison, the reduction was about 300°C from 5 MW·d/kg UO_2 to the burnup at the start of the ramp. This was in marked contrast to the helium filled rod employed in FUMEX 3.1, where, over a similar burnup interval, the temperature *increased* by 300°C. At low burnup many of the predictions were in agreement with the measured temperature of ≈1350°C but at the start of the ramp the scatter was somewhat increased. It must be said that few codes provided simultaneously good prediction at *both* high and low burnup.

Fuel temperatures at the top of ramp; Figure 17

There is a general trend of over prediction similar to that found in FUMEX 3.1.

Fission gas release; Figure 18

The comments for this case are identical to those given for FUMEX 3.1.

4.3.3. FUMEX 3.3

Fuel temperatures for 25 kw/m at 5 MW·d/kg UO_2 and just before ramp at about 27 MW·d/kg UO_2; Figure 19

Like FUMEX 3.2, this case is also a xenon filled rod but with a small grain size, 3.4 μm compared to 20 μm and a smaller fuel to cladding gap, 50 μm compared to 100 μm in case 3.2.

Most codes provided lower estimates for temperature at 5 MW·d/kg UO_2 than for FUMEX 3.2 because of the smaller as-fabricated gap. A few codes provided higher estimates for FUMEX 3.3 than for 3.2; in these cases, the higher temperatures resulted from the calculation of large gaps generated by high densification on account of the small grain size fuel.

The experimental data showed a decrease in temperature by 100 to 150°C between 5 MW·d/kg UO_2 and the time of the ramp. With few exceptions this was reproduced by the codes and the magnitude of this change was well predicted by some of the codes.

FUMEX 3.2 and 3.3 are very sensitive tests of gap conductance models and prediction of gap size, as the poor conductivity of the xenon fill gas gives a very low gap conductance. This accounts for the high scatter of predictions between the codes.

Fuel temperatures at the top of the ramp; Figure 20

Apart from the prediction of 3 codes, agreement with data was good with a relatively small scatter compared with the experimental uncertainties. This is probably due to the codes predicting small or in some cases a closed fuel to cladding gap and a hence a reduced influence on uncertainties in calculated gap conductance compared with predictions for the other two rods in this case.

Fission gas release; Figure 21

The scatter observed in fuel temperature predictions is amplified in the scatter of fission gas release predictions. However, there is little correlation between error in temperature prediction and that for release. In some cases a high prediction of release is accompanied by a low prediction of temperature. This is evidence of considerable variation in the predictive capabilities of the individual gas release models even with identical input parameters. Unfortunately this case has no measured value for release to validate the predictions.

4.4. FUMEX 4

The data consisted of two rods which differed in initial fill gas composition and initial pressure: 3 bar helium for rod A and 1 bar 92% He / 8% Xe in rod B. This simulated fission gas release with controlled conditions, ie. fresh fuel and accurately known dimensions and material properties, which is reflected in somewhat higher temperatures for rod B.

A power increase at the middle-of-life led to appreciable fission gas release with a measurable effect on both temperature and pressure. This period is a good example of short term release kinetics and the transition of the temperature/power relationship from the linear response for helium fill to non-linear behaviour when the gap is contaminated with xenon. Figure 22 shows the through life temperatures at constant powers whilst Figure 23 shows the evolution of rod internal pressure for both rods A and B of this case.

Despite an as-fabricated gap of 220 μm, cladding elongation measurements showed PCMI and subsequent relaxation for most stages of the irradiation, with PCMI increasing with burnup.

Measured parameters were fuel centreline temperature, rod internal pressure and cladding elongation.

4.4.1. FUMEX 4A

Prediction of temperatures at start-up; Figure 24

The comparison for this case was made very early in life, before the fuel to clad gap had been affected by fuel densification. In general the predictions were very close to the observed temperatures.

Fuel temperatures prior to ramp and at end-of-life (EOL); Figure 24

The observed increase in temperature between the start of ramp and end-of-life was because fission gas released during the ramp contaminated the original helium fill gas. Whereas some codes predicted no change in temperature, others correctly reproduced this feature both qualitatively and in magnitude.

Fission gas release; Figure 25

From pressure measurements, the fractional fission gas release after the ramp for both FUMEX 4A and 4B was in the range 20 to 25%. Predicted values before the ramp were all within 0-10%. After the ramp the predictions were more scattered but were typically 20 ± 10%. By the end of life the predictions increased to 30-40% but with a similar scatter of predictions.

Rod internal pressure; Figure 26

The scatter in this comparison is very large and no consistent picture emerges. It is suspected that there were errors made in the input data for some of the codes.

4.4.2. FUMEX 4B

Fuel temperatures at 30 kW/m; Figure 27

During start-up, the agreement between predictions and data was only fair for this 92% He/8% Xe filled rod. This discrepancy which is larger than that in the helium filled rod of FUMEX 4A is probably due to the xenon content of the fill gas adding more sensitivity to predictions of the gap conductance models.

The increase in temperature between start-up and the beginning of the ramp was generally well predicted by the codes. Whereas some codes gave good predictions for end-of-life temperatures, others invariably over predicted temperatures by some 200 to 300°C.

Fission gas release; Figure 28

The predictions were all similar to those for FUMEX 4A with slightly higher gas release predicted before the ramp in accord with the higher average fuel temperatures.

Rod internal pressure; Figure 29

The variation in predicted fission gas release is also seen and somewhat magnified in the rod internal pressure predictions. It is suspected that errors have been made in the input data for some of the codes.

Rod elongation; Figures 30 and 31

The experimental data for rod elongation showed evidence of strong fuel-clad mechanical interaction, stronger than for the case of FUMEX 1. Of the codes that provided predictions of elongation as a function of power in the specified ramps, only three codes: predicted such an interaction. Another code gave predictions of only slight interaction. Predictions for these codes are reproduced in Figure 31. Even in these successful cases, the detailed evolution of length during the ramps was not reproduced correctly. This is not unexpected as the stochastic nature of the 'stick and slip' of the pellet column and the cladding makes modelling extremely difficult.

4.5. FUMEX 5

The main purpose of this experiment was to assess PCMI and fission gas release under power ramping conditions. The test case was composed of a base irradiation at low power and a ramp with a subsequent hold period at the end-of-life. The rod pressure increased during the hold period indicating fission gas release whose kinetics were diffusion controlled with a characteristic dependence on the square root of the hold time. In accordance with a tightly closed fuel-clad gap, communication of released fission gas to the plenum was restricted, leading to a step increase in measured pressure during power reduction.

The measured parameters for this experiment were: rod pressure and the axial variation in rod diameter.

Rod pressure and derived fission gas release
before and after a power ramp at 16 MW·d/kg UO_2; Figures 32-36

Before the ramp the fission gas release was very small. Most codes correctly predicted this and hence agreement between measured and predicted pressure was good.

After the ramp there is a large scatter in calculated fission gas release with a tendency for over prediction. As a consequence many codes over estimated the rod internal pressure. The comparison between measurements and predictions are given in Figures 32 and 35 for rod internal pressure and fission gas release respectively.

The observed evolution of pressure and hence gas release during the ramp as well as predictions for all codes for which results are available are shown in Figures 33 and 36 respectively. The shape of the curve for the experimental data is indicative of release by a diffusion controlled mechanism. It is to be noted that not all codes illustrate this behaviour.

Diameter changes in ramp

Figure 37 shows axial diameter profiles taken at various times as indicated after the power ramp. Predictions of diameter changes were requested at two axial positions during the course of the power ramp; these were at axial positions of 116.5 and 349.5 mm identified as LHR2 and LHR4 respectively.

In Figure 37, each trace has three calibration steps 50 μm apart at the extreme left hand side and details of each trace are made with reference to these fixed points.

Predictions of diameter change were requested during the hold period after the power ramp, and where predictions are available, these are shown in Figures 38 and 39 for positions LHR2 and LHR4 respectively. Note that the time of 61 days after the ramp in Figure 37 is equivalent to an increment of burnup of $\approx$2.0 MW·d/t UO_2 in the plots of the predictions.

Although there is much scatter in the predictions, the small increases are not dissimilar to the observed change in rod diameter. Apart from the predictions of one code, the scatter of predictions is very much as expected and reflect largely on the differences in radial temperature distribution in the fuel pellet. It is notable that many of the codes correctly predict the relaxation in diameter between the start and end of the over power period. A comparison between data and predictions is given in Figures 40 and 41 for sections LHR2 and LHR4 respectively.

4.6. FUMEX 6

This case consisted of two rods which experienced a power increase at the end-of-life. The primary interest was in fission gas release and the effect on it of rate of power increase;

6F was at 'normal' start-up speed whilst 6S was slow with the power increasing over several days.

Fission gas release was determined for the end of the base irradiation when the rods were re-instrumented with pressure gauges. During the subsequent irradiation at higher rating, the pressure data showed a threshold for the onset of fission gas release which was similar for both cases. In agreement with the data from FUMEX 5, a diffusion controlled release kinetics was observed with pressure increasing with the square root of time.

Measured quantities for this experiment were: rod internal pressure and clad elongation; the latter was not included in this exercise.

Rod internal pressure and fission gas release

At the end of the base irradiation the measured gas release was 16%. Most of the codes predicted significant quantities of release typically 10 $\pm$ 5%.

For the slow ramped rod the release was estimated from the pressure measurements to be in the region of 50%. There is a large scatter in predicted values but the majority of codes produced an under estimate. Comparison of predicted and measured values of pressure and fission gas release are given in Figures 42 and 43 respectively.

For the rod subjected to a fast over power transient, the gas release estimated from the measured pressure was slightly lower at around 45%. This reduced release was duly reflected in lower predicted values although the actual scatter remained high. Comparison of predicted and measured values of pressure and fission gas release are given in Figures 44 and 45 respectively.

Evolution of rod internal pressure and fission gas release (Figures 46-48)

For both cases, the rod pressure predictions reflected the scatter in calculated fission gas release. As with FUMEX 5, the experimental data showed release kinetics appropriate to a diffusion based process. Comparison of the data with predicted pressure and fission gas release evolution with time is given in Figures 47 and 48.

4.7. GENERAL COMMENTS

After this first part of the CRP it was clear that the exercise proved beneficial to all participants irrespective of the state of their code development. Most participants were satisfied with the thermal performance aspects of their codes with possible exception of cases involving high burnup and contaminated gaps. In all cases there was scope for improvement in calculating fission gas release and mechanical interaction effects.

Although in principle no difficulty was found in handling the many time steps into which the irradiation histories were divided, many participants found that the volume of data was larger than desired and as such tended to obscured some of the physical trends in behaviour. However it was recognized that any condensation of the data prejudged the importance of experimental parameters and this may not be appropriate for each individual code.

At this stage in the programme, the following points are worthy of note:

(i) The exercise has shown that modern codes can be run on state-of-the-art PCS without difficulty. This is a marked improvement over previous years where fuel performance codes were restricted to mainframe computers. This has been brought about principally through advances in PC technology but improvements in mathematical techniques and code organization have also played their part.

(ii) Despite the complexity and degree of difficulty of the experimental cases chosen for this comparison, in general, the codes could handle the volume of data and required mathematical convergence without difficulty. However, it would appear that in a small number of instances, some stability problems remained.

(iii) Thermal conductivity degradation with burnup is embodied in the majority of the codes that participated in the exercise and in general, they provide reasonable estimates of fuel temperatures, at least in steady state. This is a very important step, as many of the physical processes modelled are exponentially dependent on temperature, and a good estimate of temperature is mandatory before further models can be developed successfully to describe other phenomena. Some difficulties still remain in the modelling of temperatures during power ramps.

(iv) The exercise has shown that difficulties still remain with modelling fission gas release, and it is clear that the codes contain a variety of models, mechanistic and otherwise which, given identical conditions provide a wide spectrum of predictions. It is recognized that being a highly non-linear process, strongly influenced by temperature and feedback effects, accurate modelling is difficult over the whole range of release values 0 to 100%. In particular, the region around 1% is extremely difficult to predict accurately and this just happens to be *the* most important region above which gas release and rod internal pressure can run away. Although there is available the empirical Halden criterion relating fuel temperature to burnup at which 1% release can be exceeded, and this is easily incorporated within any code, there is a need for further data refining the kinetics of release in this region.

(v) It is apparent that the major lack of progress is in the area of mechanical interaction. Many codes cannot perform such calculations and even for codes which can, the predictions need further refining before details of the experimental data can be reproduced. However, it must also be recognized that there is not a universal need to calculate fuel-clad interactions, as only in a few countries are there any licensing restrictions requiring evaluation of clad strain and PCI failure. In which case it is a matter for individual code developers whether or not their code requires a detailed mechanical interaction model. Therefore the omission of mechanical interaction modelling should not be taken automatically as a failing of a code.

(vi) In discussion it became evident that many of the codes had been developed using a very limited set of well qualified data.

5. SIMPLIFIED CASES AND STATISTICAL ANALYSIS

At the second RCM, it was agreed by all participants that the FUMEX cases from the Halden Reactor provided a stringent test of code performance. However, by the complex nature of the irradiation histories, it was not possible to make an inter-comparison of model predictions. For this reason, a series of simplified cases were devised and circulated to all participants in a letter (Appendix IV). Since no experimental data were available for these cases, they were purely used to compare the predictive capability of the various codes and to test the stability of the calculation.

In addition, Appendix IV gives details of further experimental data supplied to participants giving fuel centreline temperatures at the beginning of life as a function of power for fuel rods containing different fill gases, helium, argon and xenon.

In order to investigate the sensitivity of predictions to changes in experimental variables, participants were requested to repeat the calculation on simplified cases allowing for:

$\pm 5\%$ uncertainty in power
± 5 μm in initial gap
$\pm 5\%$ in UO_2 thermal conductivity.

Table V shows the response to all cases. As the majority of participants performed the statistical analysis on the first two cases the comparison in section 5.3 is restricted to these cases only.

5.1. DESCRIPTION AND RESULTS OF SIMPLIFIED CASES

The following section gives a brief overview of each simplified case, its objectives and a review of the predictions submitted.

5.1.1. Case 1

This case consisted of a fuel rod running at a constant power of 20 kW/m to a final burnup of 50 MW·d/kg UO_2. The objective of the case was as follows:

To investigate code stability, since anything other than a smooth evolution of temperature and fission gas release (FGR) would be due to mathematical problems such as poor convergence.

Through the evolution of predicted temperature, to investigate how the codes balanced the improvement of gap conductance as the gap closes by fuel swelling and the decrease in fuel conductivity through the build-up of fission products.

Depending on predicted temperatures, it is expected that the Halden empirical fission gas release threshold would be exceeded towards the end-of-life. Predictions will show if such a threshold behaviour is calculated.

Mathematical stability

The majority of codes provided well behaved predictions. However, in a small number of cases there was evidence of poor convergence. This was highlighted during statistical analysis, where for a small number of codes, a reduction in power of 5% gave higher end-of-life FGR than either the nominal or the +5% power case. Such a behaviour is physically unacceptable and must result from mathematical problems within the code.

Fuel temperatures

A comparison of temperature predictions was made at 5, 30 and 50 MW·d/kg UO_2 as shown in Figure 49. Near the start-of-life but after fuel densification, ie. 5 MW·d/kg UO_2, the mean of all predictions was 873.2°C with a standard deviation σ of $\pm$62.9°C. This spread is rather high for the low power and simple form of the history. Much of the scatter may be due to different assumptions for densification as well as pellet cracking and relocation effects. It should be noted that predictions for cases 3 to 8 considered below, avoid the influence of densification and compare only the treatment of gap conductance.

At end-of-life the mean of the predictions was 980.4 °C with $\sigma = \pm 110.5$ C. The increase in temperature illustrates how the majority of the codes predict that fuel conductivity degradation exceeds the improvement in gap conductance; the increase in standard deviation is due to the variation in FGR predictions and in the quantitative treatment of fuel swelling and thermal conductivity degradation. A comparison of temperature change with burnup at around 20 MW·d/kg UO_2, Figure 50, showed that most codes predicted a rate of change within a band $\pm$1.5°C per MW·d/kg UO_2.

Fission gas release (FGR)

Fission gas release predictions at the end-of-life were typically <3%, with many codes predicting a release of less than 1%, ie. below the Halden empirical threshold, Figure 51. This was to be expected from the low power of the case. Where predictions were greater than 1%, the burnup at which the 1% threshold was exceeded was close to the end-of-life burnup and depended on the calculation of centreline temperature, Figure 52. Three codes had unexpected high values of release which were considered incorrect. It would be useful to have information on the models employed and justification for the predictions.

Rod internal pressure

The beginning of life predictions showed an unexpected high degree of scatter which requires clarification, Figure 53. Apart from code predictions of high fission gas release, end-of-life values were all close and reflect the generally low fission gas release.

5.1.2. Case 2

This was a modification of Case 1 where the power was held constant at 20 kW/m to a burnup of 30 MW·d/kg UO_2 when there was a ramp to 40 kW/m and this power was held to an end-of-life burnup of 50 MW·d/kg UO_2. The case was aimed at investigating:

- The increment of temperature caused by the increase in power.
- The fission gas release at the end-of-life.
- The kinetics of fission gas release after increasing the power.

Temperatures

A comparison of predicted temperatures was made at 30 MW·d/kg UO_2, just before and just after the ramp and at the end-of-life, Figure 54. Prior to the ramp, the Bulgarian code predicted substantial fission gas release and hence high temperatures. These predictions were considered to be incorrect and were omitted from the comparison. For similar reasons, the

FRAPCON2 code used by China was omitted from the comparison at the end-of-life. The mean and standard deviation of the remaining predicted temperatures are as follows:

30 MW·d/kg UO_2 before ramp	911 ± 78°C
30 MW·d/kg UO_2 after ramp	1620 ± 73°C
ΔT due to ramp	704 ± 77°C
50 MW·d/kg UO_2, EOL	1845 ± 193°C

At the time of the power ramp, it is interesting that the standard deviation remained constant irrespective of temperature and power. The agreement between codes is rather satisfactory.

The end-of-life temperatures are high because of fission gas contamination of a very small or in all probability, a closed gap. It is clear that all codes correctly reflect thermal feedback of released fission gas.

Fission gas release

A comparison of FGR predictions is shown in Figure 55. The majority of codes predict a low release $\leq 1\%$ before the ramp, an increase to $\approx 20\%$ at 31 MW·d/kg UO_2 following the ramp and predictions in the range ≈ 28 to 50 % at end-of-life. Two codes provided much higher values of release whilst one code predicted an end-of-life value less than 10%. As for Case 1, a knowledge of the predictive model and their justification would be helpful. By omitting these three cases, the end-of-life mean and standard deviation for FGR were 35 ± 8.2%. The small scatter in predictions is very encouraging. The relationship between FGR at end-of-life and the increment of temperature between 30 and 50 MW·d/kg UO_2 is shown in Figure 56. Such a plot can only be empirical, but it shows the strong systematic relationship between FGR and temperature to be expected for a thermally activated process.

This case is not unlike FUMEX 6 where experimental data showed a time$^{1/2}$ dependence of release following the power increase characteristic of a diffusion controlled process. Inspection of the predictions for this simplified case showed a variety of release kinetics. Some codes correctly reproduced the square-root-of-time type evolution but some did not. It is concluded therefore that the agreement for predictions at end-of-life was not a true reflection of the ability of the codes to provide accurate predictions at high powers, particularly at short times as would be expected in real transient situations.

Pressure

The end-of-life pressure predictions are shown in Figure 57. These depend on assumptions made by the code users as to the initial internal volume of the fuel rod. The intention was for a gap volume of 1.5 cc in addition to the plenum volume of 2.5 cc.

5.1.3. Cases 3-7

These cases required predictions of fuel temperatures at start-up for helium filled rods of different gap sizes. There was no influence of densification and the cases tested the ability of the codes to reproduce the correct dependence of gap conductance on gap size. Figures 58 and 59 show a comparison of temperature predictions for different gap sizes at 20 and 40 kW/m respectively.

Figure 60 shows the observed trends from several experiments in the Halden Reactor Programme. Here, thermocouple measured temperatures are shown at 20 and 30 kW/m as a function of gap size for both helium and xenon filled rods. The equivalent centre temperature for solid pellets are higher by 30-40 °C at 20 kW/m rising to 50-70 °C and 100-110°C at 30 and 40 kW/m respectively.

Experimentally it is observed for helium filled rods that changing the gap size from 50 to 230 μm at 20 kW/m produces an increase in temperature of ≈180 °C. Apart from one code, all gave predictions of temperature which agree with the observations both in magnitude and the increment with changing gap size. Differences between code predictions can be attributed mostly to the empirical treatment of pellet cracking and relocation.

The variation of predictions between codes is greater at 40 kW/m than at 20 kW/m, despite smaller calculated gaps. This is because of the larger temperature drop across the gap and the greater sensitivity to the calculated gap size.

5.1.4. Cases 4 and 8

Case 4 and Case 8 are for rods with 100 μm fuel-to-clad gaps filled with helium and xenon respectively. The thermal conductivity of xenon is much lower than that for helium and as a consequence results in higher temperatures for identical powers and design parameters. Correct prediction of xenon filled rods is necessary in order to reproduce the effect of released fission gases (xenon and krypton) on fuel temperatures. In Figures 61 to 64, comparison of predictions are made at power levels 10 to 40 kW/m.

For a rod of gap size 100 μm, the experimental trends given in Figure 60 are tabulated below.

Fill gas composition	Temperatures °C at 20 kW/m	30 kW/m
Helium	635	862
Xenon	923	1105
$\Delta T = (T_{Xe} - T_{He})$	288	243

The temperature difference between helium and xenon is around 288 and 243°C at 20 and 30 kW/m respectively. The majority of codes over predict these differences by ≈150°C.

From the table it appears that the difference in temperatures between helium and xenon filled gaps decreases with increasing power. Figure 60 also shows that at small gaps the temperature difference completely disappears. At 40 kW/m it is to be expected that the fuel-to-clad gap in a 100 μm gap rod will be very small or closed, consequently the experimental data suggest that the difference in predicted temperatures (T_{Xe}-T_{He}) should be small. However, apart from four codes, the predicted differences for this exercise are rather large and are of the order of 300°C. Codes over predicting this difference, despite a very small or closed gap, probably contain a gap conductance model which is a sensitive function of surface roughness.

Three of the four codes providing the best predictions for this particular case are known to be insensitive to values of surface roughness. It is interesting to note that out-of-pile experiments measuring the heat conduction across surfaces which are in contact show a pronounced effect of surface roughness, whereas the data showing little or no dependence on roughness come from in-pile reactor experiments at medium to high burnup. It is possible therefore that the influence of surface roughness on the gap conductance of a fuel rod is a decreasing function of burnup.

5.2. GENERAL COMMENTS

The simple histories chosen for these cases and the requested sensitivity analysis indicated that a small number of codes suffered from mathematical convergence problems. This was particularly apparent in sensitivity studies where a small reduction in power resulted in more fission gas release than the nominal case; this behaviour is clearly incorrect.

The majority of codes provided similar values for the parameters for which predictions were requested, there were however a few codes that gave predictions which deviated significantly from the general trends. Although in Cases 1 and 2 there are no data for comparison, it is considered that predictions of any code which deviated significantly from the general trend were incorrect. For Cases 4 and 8 experimental data do exist and it is apparent that it is the *majority* that over predicted the difference in temperature between helium and xenon filled rods, whilst the *minority* provided the correct difference. For this reason, in the absence of experimental data, care must be taken in judging whether or not a code provides accurate predictions, and benchmarking a code against another is no guarantee of accuracy.

The closest results occurred for the comparison of thermal performance of helium filled rods on start-up, when compared with experimental data, the code predictions were good.

As burnup was accumulated, there was an increase in the scatter of fuel temperature predictions. This is because of different treatments of: densification, gap closure due to fuel swelling and degradation of fuel thermal conductivity. It is clear that code developers would benefit from a better experimental definition of these phenomena.

The agreement between predictions and the small degree of scatter for the end-of-life FGR of Case 2 suggests accurate predictions at this level of release. However, the poor agreement over the form of release kinetics is cause for some concern, particularly if the codes are used to calculate FGR during short term transients. Further experimental data on release kinetics would be useful.

5.3. STATISTICAL ANALYSIS OF CASES 1 AND 2

For this comparison a distinction must be made between an **uncertainty** and a **probabilistic** analysis. In the linear case the standard deviation of an uncertainty analysis σ_u is given by the most unfavourable combinations of the individual uncertainties:

$$\sigma_u = \sigma_1 + \sigma_2 + \sigma_3 + ...$$

In a first approximation the thermal analysis can be considered as a linear problem. Since the uncertainty of the gap size is very low ($\pm 5\mu$), uncertainties of the linear rating $\sigma_{q'}$ and the thermal conductivity, σ_λ, each $\pm 5\%$ dominate the overall uncertainty. Thus,the following uncertainties result:

Uncertainty analysis:

Case 1: centre line temperature at EOL $\approx \pm 75$-$80°C$
Case 2: centre line temperature at EOL $\approx \pm 170°C$.

If the linear rating q′ and the thermal conductivity λ are treated as random variables, characterised by a mean and a distribution, the analysis becomes more complicated. Again, if a linear approximation is assumed and if both random variables are independent and Gaussian, the standard deviation is given by

$$\sigma_p = \sqrt{\sigma_{q'}^2 + \sigma_{\lambda}^2}$$

Thus an estimate of the standard deviation of a probabilistic analysis is:

Probabilistic analysis:

Case 1: centre line temperature at EOL $\approx \pm 57°C$
Case 2: centre line temperature at EOL $\approx \pm 120°C$.

Results of the uncertainty analysis

For both cases, the codes agree well in the sensitivity of the temperature predictions (Tables VI and VII and Figs 68 and 69) and give the expected variations. From the disagreement of the predicted fission gas release for case 1, one can already expect that the uncertainty analysis does not give a consistent picture. This is shown in Table VIII and Fig. 70. The results cannot be further commented. For case 2, the codes predict a similar, high fission gas release. The sensitivity is similar and rather consistent (Table IX and Fig. 71).

Results of the probabilistic analysis

As Tables X to XIII show only a few codes offer probabilistic capabilities. The TRANSURANUS code offers two different methods for probabilistic analyses (Monte Carlo technique and the Numerical Noise Analysis). Both techniques agree well but differ significantly in the numerical effort. At a first glance the results of the codes seem to differ significantly. However, a closer look reveals that different input data for the standard deviation were chosen: For all TRANSURANUS calculation the standard deviation of $\pm 5\%$ was taken for all random variables, whereas in the other cases $\pm 5\%$ was taken as $\pm 3\sigma$.

6. MODIFICATIONS MADE TO CODES AND RERUN OF FUMEX CASES

Most participants were satisfied with the thermal performance aspects of their codes with possible exception of cases involving high burnup and contaminated gaps. In all cases there was scope for improvement in calculating fission gas release and mechanical interaction effects, and consequently the majority of participants embarked on further developments of their codes. This was followed by a rerun of the FUMEX cases.

This section summarizes the modifications made to the codes as a result of the FUMEX exercise and the revised predictions of the original cases.

6.1. MODIFICATIONS MADE TO CODES

Table XIV provides an overview of the modifications made to the codes in the light of the FUMEX exercise, while the following sections explain the general background providing the justification for the changes. It is evident that most of the modifications deal with aspects of thermal performance and fission gas release, while only a few relate to mechanical behaviour, statistical analysis and programming aspects.

6.1.1. Thermal performance

The most important contribution to the advancement of fuel behaviour analysis has been the ability to measure fuel centreline temperatures at the start of life and throughout irradiation to high burnup. By carefully designed experiments, it has been possible to distinguish contributions to the temperature distribution made by the conductance of the fuel to clad gap and the thermal conductivity of the UO_2 pellet. The effect of gap size and fill gas composition is now well established.

Conductivity degradation

Only recently has it been recognized that the fuel conductivity decreases significantly with burnup due to the accumulation of fission products. Quantification of this process is very important as it has significant implications on high burnup fuel performance, in particular the amount of fission products released and hence rod internal pressure in transients occurring towards the end of life.

At the beginning of the FUMEX exercise, not all of the participating codes took into account fuel conductivity degradation. Improvements of temperature calculations therefore invariably encompassed the inclusion of this effect.

Gap conductance

Although the principal parameters involved in the pellet-clad heat transfer are known, their quantification still represents a source of uncertainty for the fuel temperature calculation. This is due to the complex inter-relations with other phenomena (fission gas release, fuel and cladding dimensional changes) and the stochastic nature of some processes involved (e.g. fuel cracking and relocation). The FUMEX cases have provided motivation for a number of code developers to modify the gap conductance calculation, affecting both low and high burnup data. The changes comprise fuel fragment relocation, tuning of swelling and densification models, and an improved treatment of the conductivity of gas mixtures.

All in all, the features now embodied in most of the codes in general provide reasonable estimates of fuel temperatures. This can also be seen from the "before-after" graphs of temperature calculations in Section 6.2: Rerun of the FUMEX Cases.

6.1.2. Fission gas release

The exercise has shown that difficulties still remain with modelling fission gas release. Code improvements of fission gas release calculation introduced as a consequence of the FUMEX programme encompassed a variety of modifications including completely new models, the adoption of well-established models from other codes, the identification of the best model among several available in a code, the tuning of a model, and the change of model details such as the diffusion coefficient. As can be expected, this led to overall improved FGR predictions, but the influence of better temperature calculations cannot be dissociated from these results.

6.1.3. Mechanical interaction

It is apparent that the major lack of progress is in the area of mechanical interaction. Many codes cannot perform such calculations and even for codes which can, the predictions need further refining before details of the experimental data can be reproduced. This is especially true for situations where codes do not calculate interaction at all while the experimental data clearly give evidence of PCMI (e.g. FUMEX 1). This also reflects the lack of a proper definition of the pellet-cladding gap that would allow a unified thermal-mechanical treatment. It was in fact demonstrated with ENIGMA(NE) that the improvement of mechanical modelling can have a positive impact on other code predictions, i.e. temperature calculation results.

6.1.4. Statistical analysis

Only a few codes have built-in features for statistical analysis, but all of them can be used for such a task by providing input variations manually. For two codes, improvements of statistical analysis capabilities were reported. The variations produced by the different codes and methods were in fair agreement with each other, and an urgent need for modifications or improvements could not be identified.

6.1.5. Code structure, execution control and solution algorithms

Modern codes can now be run on state-of-the-art PCs without difficulty. This has been brought about principally through advances in PC technology but improvements in mathematical techniques and code organization have also played their part. The utilization of parallel computing is an interesting example of how modern hardware can have an influence on problem solutions.

Despite the complexity and degree of difficulty of the experimental cases chosen for this comparison, in general, the codes could handle the volume of data and required mathematical convergence without difficulty. However, it would appear that in a small number of instances, some stability problems remained.

Reported improvements were related to time step control, extended capabilities to treat detailed irradiation histories, better algorithms for calculation of transient temperatures, elimination of coding errors, and the transition to Fortran 90. Participants have also indicated work on advanced graphical systems, which are especially useful for displaying results related to 2-D and 3-D calculations.

6.2. RERUN OF THE FUMEX CASES

Rerun of FUMEX cases has been done by many participants after modification of their code. Table XV summarizes the available information on these recalculations with brief comments.

Figures 72 to 82 give in the form of graphs, for all participants who have provided the data, the calculated blind predictions and revised predictions, compared to experimental values of temperature and FGR. A general improvement of the agreement between calculated and experimental values is clearly apparent.

In the codes which were under predicting temperature at high burnup, an improvement was obtained by taking into account the degradation of thermal conductivity with burnup.

In the blind prediction, the majority of codes were over predicting temperatures at the top of ramps. An improvement was obtained through modification of clad-gap heat transfer coefficient and in some cases of gas release.

The improvement of calculated temperature had a direct and positive impact on calculated FGR. In some codes, additional improvements were obtained through modification of FGR models.

After modifications, several codes were able to calculate more cases than for the blind prediction. In some codes, most modifications were directly connected to the FUMEX exercise with some parameters of the models tuned on the FUMEX cases. In other ones, modifications were based on a large data set of which the FUMEX cases were only a small subset.

The main effort has been focused on thermal analysis and as a result most recalculations give improved agreement between calculated and measured temperatures: predictive capability of FGR has been improved, however there is still room for more progress to be made.

It is noted that the FUMEX participants made little efforts to improve upon PCMI prediction capabilities.

7. ISSUES FOR FURTHER DEVELOPMENT

At the second RCM the participants were requested to provide comments on the FUMEX exercise under the following general headings:

- the strengths and weaknesses of their code as perceived through comparison of the predictions with the data,
- what benefit they had received from their participation and what code developments were to be initiated as a result of the FUMEX exercise,
- their perception of the cases chosen for comparison,
- what future activities they would welcome under the current programme.

Since the answers to these questions represent a true reflection of the interaction of the participants within the exercise, their replies have been included in Appendix V.

The final RCM reviewed the progress made throughout the FUMEX programme. Concluding summaries of the sessions are given in Appendix VI. Discussions amongst participants identified the following areas for further attention:

Fuel thermal performance

Although there are similarities between the methods adopted by the codes, the principal topics where improvements could be made are:

- the dependence of surface roughness on burnup
- the formulation of a reversible relocation
- the dependence of contact conductance on surface roughness and interfacial pressure
- the treatment of densification/swelling
- the influence of the high burnup 'rim' structure
- most appropriate formulation of UO_2 fuel thermal conductivity in terms of temperature, burnup, irradiation damage, additives, density (including gaseous swelling), O/M , etc.

Fission gas release

Despite the similarity in approach, the codes produced a diversity of predictions. This indicates a clear need for further information on gas release to improve models. This information should include:

- evidence for and against grain boundary sweeping
- data on intragranular bubbles, their size and concentration as a function of temperature and burnup
- availability of a comprehensive database for code validation
- data on additional phenomena - thermal resolution/dislocation sweeping, etc.
- influence of 'rim' formation at high burnup.

At high burnup there is the additional complication of 'rim' formation. Further investigations are needed to understand the formation and properties of this structure.

Mechanical behaviour

Compared to the thermal analysis, the treatment of mechanical behaviour has to deal with a more complex system, including the cladding and the axial dimensional changes. Deficiencies in code predictions of PCMI have been pointed out before (see 4.4.2). To improve the codes it is necessary to avoid over-simplifications; improvements should include:

- the treatment of structurally weak cracked pellets
- unified thermal and mechanical modelling
- inclusion of axial sliding as possible condition for the case of non-zero contact forces.

Mathematical methods

The following areas need further consideration:

- code structure
- automatic time step features
- numerical stability (robustness)
- convergence
- run-time

Specific issues

Regarding specific issues related to fuel modelling, the following points were suggested for consideration:

- Evaluation of maximum clad stress during power ramps in order to improve PCI failure prediction. 2-D or 3-D mechanical computation is probably necessary.

- A more mechanistic description of gas behaviour in MOX fuel. The specific microstructure and chemistry of this fuel is probably to be taken into account.

- Modelling of gadolinia bearing fuel.

- Statistical analysis is needed at the validation stage and for use of codes in safety analysis.

Quality assurance

The role of QA is of fundamental importance in developing and maintaining fuel performance codes, particularly if they are used for nuclear safety or design calculations.

The principal topics which require further work are:

- a document describing the implementation of QA for software
- Setting up a well defined database of experimental results to be used for code validation.

8. ONGOING IAEA ACTIVITIES RELATED TO FUEL MODELLING

It is clear that in many countries fuel modelling is at an early stage of development. This has been recognized by the IAEA which is already pursuing a number of activities related to the FUMEX programme, aimed at providing assistance in this area. Three such projects are described below.

OECD/NEA/IAEA Fuel Performance Database

This is a joint activity with the NEA designed to build a database of fuel rod behaviour in the public domain for the development and validation of fuel performance codes. This was initiated independent of the FUMEX and was greatly welcomed by the participants. The aim of the database is to include well qualified information on irradiated fuel rods for all reactor types generated in power reactors and international research programmes.

RER 4/012

The aim of this project is to aid eastern European countries to develop a code to support operation of their WWER reactors. Within the programme, the Transuranium Institute (TUI) TRANSURANUS code has been supplied to participant countries along with elements of the database described above. The transfer of the code and database is accompanied by expert support and instruction. It is hoped to extend this programme to other countries which have shown a keen interest.

New technical co-operation project

The objective of the project is to provide countries with the necessary knowledge to carry out the licensing procedure for fuel and computer codes used for fuel modelling, including the necessary knowledge of fuel fabrication.

It is proposed to look at the problem from three view points:

(a) The safety body in charge of the license delivery and also responsible for specifying the safety criteria.

(b) The utility which is in charge of requesting the fuel load or reload licensing.

(c) The code developer in charge of code qualification and licensing. The codes are used by utilities, the design organization, the code developer himself or the safety body.

9. CONCLUSIONS AND FUTURE WORK

The response to the FUMEX programme was very encouraging with a high degree of participation from Member Countries. All agreed that it was a worthwhile exercise and that the cases chosen were stringent tests of model and code performance. The exercise was useful in demonstrating the strong points of the codes as well as highlighting deficiencies where improvements were necessary. As a consequence most of the codes underwent some development during the programme, see Section 6. It was also apparent that many of the codes had been developed on only a limited database and that the FUMEX cases provided a valuable addition. The following points are worthy of note:

(i) It is now universally recognized that the fuel conductivity decreases significantly with burnup. As a result of the FUMEX exercise, all codes include a treatment of this phenomena. It is in the area of thermal performance that the greatest improvements have been made.

(ii) The exercise has shown that difficulties still remain with modelling fission gas release. However, through refining existing models and the introduction of new models there was a general improvement in predictive capabilities.

(iii) It is apparent that the major lack of progress is in the area of mechanical interaction. This is considered to be an important omission with adverse consequences on many aspects of fuel modelling.

(iv) The exercise has shown that modern codes can be run on state-of-the-art PCS without difficulty. Despite the complexity and degree of difficulty of the experimental cases chosen for this comparison, in general, the codes could handle the volume of data and required mathematical convergence without difficulty.

(v) QA was recognized as an essential part of the code development process.

(vi) To address the different aspects of fuel behaviour which have come to light during this CRP, the IAEA is undertaking several actions. In addition to co-operation in the WWER fuel seminar (Varna 1994) and the publication of a report reviewing the information on

WWER fuel, the IAEA has embarked on a series of programmes addressing different aspects of fuel behaviour modelling with the following objectives:

- to build a database of well-defined experiments suitable for code validation in association with OECD/NEA;

- to transfer a mature fuel modelling code to developing countries (TRANSURANUS developed by the Transuranium Institute, Karlsruhe), (RER 4/012) to support teams in these countries in their efforts to adapt the code to requirements of specific reactors;

- to provide guidelines for code quality assurance, code licensing, and code application to reactor operation, fuel licensing and safety assessments (new technical co-operation project to be started in 1997).

It has been learned from the FUMEX programme that the most important topics for follow-up activities are as follows:

- thermal performance
- fission gas release
- mechanical interaction
- mathematical methods.

The first topic to be addressed as a pilot study is Thermal Performance. It is thought that the proceedings of this study would provide a synthesis of the discussions resulting in clearly defined guidelines for model development and a catalogue of important experimental data necessary for development and validation.

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TABLES I–XV

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Table I Participants in the D-COM Blind Problem

Country	Organization	Code
Denmark	RISÖ	Experiment
Argentina	CNEA	BACO
Belgium	BN	COMETHE III-L
Canada	AECL	ELESIM2.MOD10
Czechoslovakia	Rez	PIN/RELA
F.R. Germany/CEC	TU-Darmstadt/ITU	URANUS
Finland	VTT	FRAPCON-2
France	CEA-Grenoble	CREOLE
France	EdF	CYRANO-2
France	CEN-Saclay	RESTA
India	BARC	PROFESS
Japan	CRIEPI	FEMAXI-III
Sweden	Studsvik	GAPCON-SV
United Kingdom	BNFL	HOTROD
United Kingdom	UKAEA	MINIPAD-E
USA	Exxon	RAMPX2

Table II Participants in the FUMEX Exercise

Code number	Country	Organization	Code [key reference] [version used in the blind exercise]
	Norway/OECD	Halden	Experiment
1	Argentina	CNEA	BACO [6]
2	Bulgaria	INRNE	PIN micro [7]
3	Canada	AECL	ELESIM.MOD11 [8]
4	Finland	VTT	ENIGMA 5.8f [9]
5	France	EdF	TRANSURANUS-EdF 1.01 [10]
6	France	CEA/DRN	METEOR-TRANSURANUS [10]
7	CEC	ITU	TRANSURANUS [10]
8	India	BARC	PROFESS [11]
9	India	BARC	FAIR [12]
10	India	NPC	FUDA [13]
11	Japan	NNFD	TRUST 1b [14]
12	Japan	CRIEPI	EIMUS [15]
13	China	CIAE	FRAPCON-2 [16]
14	Romania	INR	ROFEM-1B [17]
15	Swiss	PSI	TRANSURANUS-PSI [10]
16	Czech Republic	NRI Rez	PIN/W [7]
17	United Kingdom	BNFL	ENIGMA 5.2 [9]
18	United Kingdom	NE	ENIGMA 5.8 D [9]
19	Russia	IIM	START 3 [18]

NB: Turkey joined the Co-ordinated Research Programme at the time of the 3rd. RCM in Bombay. Turkey is using a version of FRAPCON-2.

Table III Main features of FUMEX rods

	Diametral gap (μm)	Rod	gas pressure (bar)	Enrich (%wt)	Grain size (μm)	Fuel density (%TD)	Fuel diameter (mm)	Fuel stack length (mm)
FUMEX 1	130	He	10	3.5	10	94.1	8.09	810
FUMEX 2	130	He	10	13.0	7-10	94.3	5.92	443
FUMEX 3 rod 1	100	He	1	10.0	3.4	95.0	10.70	140
FUMEX 3 rod 2	100	Xe	1	6.0	20	95.0	10.70	140
FUMEX 3 rod 3	50	Xe	1	10.0	3.4	95.0	10.75	140
FUMEX 4 rod A	220	He	3	9.9	12.0	95.0	10.68	781
FUMEX 4 rod B	220	He 92% Xe 8%	1	9.9	12.0	95.0	10.68	781
FUMEX 5	210	He	1	3.93	14.5	95.0	10.60	457
FUMEX 6	260	He	1	9.88	16.0	94.7	10.54	466

Table IV Measured parameters in FUMEX rods

	Fuel centre line temperature	Inside rod pressure	clad elongation	diameter change	FGR by PIE
FUMEX 1	X	X	X		X
FUMEX 2		X			X
FUMEX 3 rod 1	X				
FUMEX 3 rod 2	X				
FUMEX 3 rod 3	X				
FUMEX 4 rod A	X	X	X		
FUMEX 4 rod B	X	X	X		
FUMEX 5		X		X	
FUMEX 6		X	X		X (end base irrad.)

TABLE V Recalculation of the 6 FUMEX cases and of the Simplified cases
(Cases calculated by each laboratory)

Code		ORIGINAL FUMEX CASES							SIMPLIFIED CASES							
		F 1	F 2	F 3	F 4	F 5	F 6	statist Treat.	S 1	S 2	S 3	S 4	S 5	S 6	S 7	S 8
1.	Argentina							P	x*	x	x	x	x	x	x	x
2.	Bulgaria	x	x	x				W	x*	x	x	x	x	x	x	x
3.	Canada															
4.	Finland [2]							S+W	x*	x*	x*	x*	x*	x*	x*	x*
5.	EdF [1]				x		x	S	x*	x*	x	x	x	x	x	x
6.	CEA	x	x	x	x	x	x									
7.	CEC [3]	x	x	x	x	x	x	P	x*	x*	x*	x*	x*	x*	x*	x*
8.	Ind	x	x	x	x	x	x	S	x*	x*	x*	x*	x*	x*	x*	x*
9.	Ind	x	x	x	x	x	x	P	x*	x*	x*	x*	x*	x*	x*	x*
10.	Ind															
11.	Jpn															
12.	Jpn															
13.	China	x	x	x	x	x	x	-	x	x	x	x	x	x	x	x
14.	Rom [5]		x	x		x		P	x	x*	x	x	x	x	x	x
15.	Switz	x		x				P	x*	x*	x*	x*	x*	x*	x*	x*
16.	CZR	x	x	x				-	x	x	x	x	x	x	x	x
17.	UK [6]	x	x	x	x	x	x	S	x*	x*	x	x	x	x	x	x
18.	UK [7]			x (3.2)				S	x*	x*	x*	x*	x*	x*	x*	x*
19.	Russ [4]							S	x*	x*	x	x	x	x	x	x

* Statistical analysis performed on this case
P Probabilistic analysis
S Single effect analysis
W Worst case analysis
EdF no code modification for the recalculation of FUMEX 4 & 6
VTT no code development
CEC modification of the swelling model, all cases recalculated
unsuccessful attempt to modify the athermal FGR model
RUSS Sensitivity analysis S cases 1-8
ROM Sensitivity analysis S case 2
BNFL Enigma B 6.1 not run blind, predictions now available
Nucl. Elect. S cases and FUMEX 3.2 run with the 5.9 version

TABLE VI FUMEX S1: Fuel Temperatures at EOL

Uncertainty Analysis

Country	Code Name	No.	Average	Maximum	Minimum	Remarks
ARG	BACO	1	812	840	784	no variation of λ
BULG	PIN-micro	2				evaluation doubtful
CAN	ELESIM	3				no results available
FIN	VTT/ENIGMA	4	1061	1157	965.5	
FRA/EdF	TRANSURANUS-EdF	5	1126	1266	994	
FRA/CEA	TRANSURANUS-MET	6				no results available
CEC	TRANSURANUS	7	859	950.5 (a)	780 (a)	
IND1	PROFESS	8	1025	1079	873	
IND2	FAIR	9	855	946	782	
IND3	FUDA-MOD1	10				no results available
JPN1	TRUST-V1b	11				no results available
JPN2	EIMUS	12	1116	1274	1019	
CHIN	FRAPCON2	13				no results available
ROM	ROFEM-1b	14	913	1001.5	841	
SWIT	TRANSURANUS-PSI	15	926	1093	840	
CZE	PIN-W	16				no results available
UK/BNFL	ENIGMA	17	1083	1190	991	
UK/NE	ENIGMA	18	968	1043	899	
RUS	START	19				no results available

a) at the same time, not the same burnup.

TABLE VII FUMEX S2: Fuel Temperatures at EOL

Uncertainty Analysis

Country	Code Name	No.	Average	Maximum	Minimum	Remarks
ARG	BACO	1				no variation of λ
BULG	PIN-micro	2				evaluation doubtful
CAN	ELESIM	3				no results available
FIN	VTT/ENIGMA	4	1840	2000	1674	
FRA/EdF	TRANSURANUS-EdF	5	1890	2076	1711	
FRA/CEA	TRANSURANUS-MET	6				no results available
CEC	TRANSURANUS	7	1976	2194 (a)	1753 (a)	
IND1	PROFESS	8	1530	1733	1373	
IND2	FAIR	9	1930	2116	1709	
IND3	FUDA-MOD1	10				no results available
JPN1	TRUST-V1b	11				no results available
JPN2	EIMUS	12	2089	2251	1923	
CHIN	FRAPCON2	13				no results available
ROM	ROFEM-1b	14	1744	1896	1581	
SWIT	TRANSURANUS-PSI	15	1959	2128	1800	
CZE	PIN-W	16				no results available
UK/BNFL	ENIGMA	17	1876	2241	1714	
UK/NE	ENIGMA	18	1829	1985	1671	
RUS	START	19				no results available

a) at the same time, not the same burnup.

TABLE VIII FUMEX S1: Fractional Fission Gas Release (%) at EOL

Uncertainty Analysis

Country	Code Name	No.	Average	Maximum	Minimum	Remarks
ARG	BACO	1	2.65	2.78	2.54	no variation of λ
BULG	PIN-micro	2				evaluation doubtful
CAN	ELESIM	3				no results available
FIN	VTT/ENIGMA	4	0.46	2.1	0.1	
FRA/EdF	TRANSURANUS-EdF	5	12	21.8	6.2	
FRA/CEA	TRANSURANUS-MET	6				no results available
CEC	TRANSURANUS	7	<0.05	<0.05	<0.050	
IND1	PROFESS	8	3.25	3.85	1.03	
IND2	FAIR	9				
IND3	FUDA-MOD1	10				no results available
JPN1	TRUST-V1b	11				no results available
JPN2	EIMUS	12	1.9	12.2	0.45	
CHIN	FRAPCON2	13				no results available
ROM	ROFEM-1b	14	0.52	0.92	0.38	
SWIT	TRANSURANUS-PSI	15	0.23	1.86	0.02	
CZE	PIN-W	16				no results available
UK/BNFL	ENIGMA	17	1.3	6.5	0.5	
UK/NE	ENIGMA	18	0.29	0.57	0.19	
RUS	START	19				no results available

TABLE IX FUMEX S2: Fractional Fission Gas Release (%) at EOL

Uncertainty Analysis

Country	Code Name	No.	Average	Maximum	Minimum	Remarks
ARG	BACO	1				no variation of λ
BULG	PIN-micro	2				evaluation doubtful
CAN	ELESIM	3				no results available
FIN	VTT/ENIGMA	4	27.7	31.2	22.6	
FRA/EdF	TRANSURANUS-EdF	5	25.2	29.5	21.8	
FRA/CEA	TRANSURANUS-MET	6				no results available
CEC	TRANSURANUS	7	40	48.6	30.2	
IND1	PROFESS	8	9.1	14.6	6.6	
IND2	FAIR	9	36.8	43.7	27.5	
IND3	FUDA-MOD1	10				no results available
JPN1	TRUST-V1b	11				no results available
JPN2	EIMUS	12	45.3	48.5	39.7	
CHIN	FRAPCON2	13				no results available
ROM	ROFEM-1b	14	28.2	35	22.9	
SWIT	TRANSURANUS-PSI	15	47.6	52.5	42.6	
CZE	PIN-W	16				no results available
UK/BNFL	ENIGMA	17	38.8	44.6	31.9	
UK/NE	ENIGMA	18	36.6	42.7	31.1	
RUS	START	19				no results available

TABLE X FUMEX S1: Fuel Temperatures at EOL

Probabilistic Analysis

Country	Code Name	Number	Average	Stddev	Remarks
ARG	BACO	1	812	±15	Monte Carlo
CEC	TRANSURANUS	7	856	±58 (a)	Monte Carlo, 500 runs
CEC	TRANSURANUS	7	855	±57 (a)	Num. Noise Analysis, 1 run
IND2	FAIR	9	864	±31	
SWIT	TRANSURANUS-PSI	15	926	±57	Monte Carlo, 1000 runs

a) at the same time, not the same burnup.

TABLE XI FUMEX S2: Fuel Temperatures at EOL

Probabilistic Analysis

Country	Code Name	Number	Average	Stddev	Remarks
CEC	TRANSURANUS	7	1977	±162 (a)	Monte Carlo, 500 runs
CEC	TRANSURANUS	7	1972	±161 (a)	Num. Noise Analysis, 1 run
IND2	FAIR	9	1921	±79.5	
ROM	ROFEM-1b	14	1734	±35	
SWIT	TRANSURANUS-PSI	15	1959	±95	Monte Carlo, 1000 runs

a) at the same time, not the same burnup.

TABLE XII FUMEX S1: Fractional Fission Gas Release (%) at EOL

Probabilistic Analysis

Country	Code Name	Number	Average	Stddev	Remarks
ARG	BACO	1	2.65	±0.06	Monte Carlo
CEC	TRANSURANUS	7	<0.05	±0.01	Monte Carlo, 500 runs
SWIT	TRANSURANUS-PSI	15	0.23	±0.03	Monte Carlo, 1000 runs

TABLE XIII FUMEX S2: Fractional Fission Gas Release (%) at EOL

Probabilistic Analysis

Country	Code Name	Number	Average	Stddev	Remarks
CEC	TRANSURANUS	7	40	±6.7 (a)	Monte Carlo, 500 runs
IND2	FAIR	9	36.7	±2.8	
ROM	ROFEM-1b	14	28.4	±1.2	
SWIT	TRANSURANUS-PSI	15	47.6	±2.8	Monte Carlo, 1000 runs

a) at the same **time**, not the same burnup.

TABLE XIV **Overview of Code Modifications after Blind Calculations**

	Conductivity degradation	Gap conductance	Fission gas release	Mechanical behaviour	Statistical analysis	Code control and coding
BACO		gas mixture conductivity densification modif Macdonald h_{gap} model			implemented	power history treatment, Fortran 90
ELESIM	SIMFUEL, Matpro	relocation/offset model	analytical model allowing for varying Temperature & Power	new pellet deformation model		
METEOR-TRANSURANUS		measured surface roughness, contact pressure, internal zirconia layer				
PROFESS		cracking and relocation reactivated	parameters burnup dependent			
FAIR	SIMFUEL	KWU cracking and relocation				
FUDA	now included	improved	URGAS model			
TRUST		relocation, relocation recovery	extensive revision			time step control, coding errors eliminated
FRAPCON-2 China	SIMFUEL		selected Macdonald/Weisman as best model	AXISYM model modif and checked	improved	
TRANSURANUS PSI		KWU relocation model				
PIN-W	Kjaer-Pettersen model	relocation, tuned swelling, tuned densification, Matpro cladding creep	tuned Macdonald/Weisman model (parameters power and burnup dependent)	failure model included		
ENIGMA/NE			transient I-131 release	improved axial force balance		improved time step control, improved T-calc algorithm, formal quality control
PIN-MICRO (PINB1)	SIMFUEL, rim, cracked fuel	tuned relocation, modif contact conductance	diffusion coefficient	cladding creep, fuel swelling and creep for T>1050°C		

No modifications were reported/identified for codes not listed in the table

TABLE XV Rerun of the Fumex Cases

	Country	Code	Version (Blind)	Version (recalc)	Results of recalculation provided	Comments
1.	Argentina	BACO	2.2	2.3	X	Corr. of overpred. of T
2.	Bulgaria	PIN/MICRO	NEA.D.B.	B1.Fumex		graphs T and FGR=f(τ)
3.	Canada	ELESIM	MOD 11			
4.	Finland	ENIGMA	5.8 F			
5.	France/EDF	TRANS/EDF	1			only rerun of case 4 because of wrong input data
6.	France/CEA	METEOR/TU	0.3	1.3	X	Corr. of overpred. of T
7.	CEC	TRANSUR.	V1M1J93			rerun of all cases, small changes
8.	India	PROFESS	1.0		X	more calculated cases
9.	India	FAIR	1.0		X	Corr. of overpred. of T
10.	India	FUDA	MOD 1			
11.	Japan	TRUST	1.0	1c3.b	X	no systematic improvement investigation still going on
12.	Japan	EIMUS	4			
13.	China	FRAPCON-2	VO		X	corr. of T and FGR under prediction
14.	Romania	ROFEM	1B		X	only changes of input data
5.	Switzerland	TRANS/PSI	V1M1Y92		X	little changes
16.	Czech Rep.	PIN/W	1993		X	corr. of T under prediction
17.	UK/BNFL	ENIGMA	5.2	B 6.1	X	corr. of T over prediction Fumex not used in the improvement
18.	UK/NE	ENIGMA	5.8d	5.9	X	small changes
19.	Russian Fed.	START-1	3			

FIGURES 1–82

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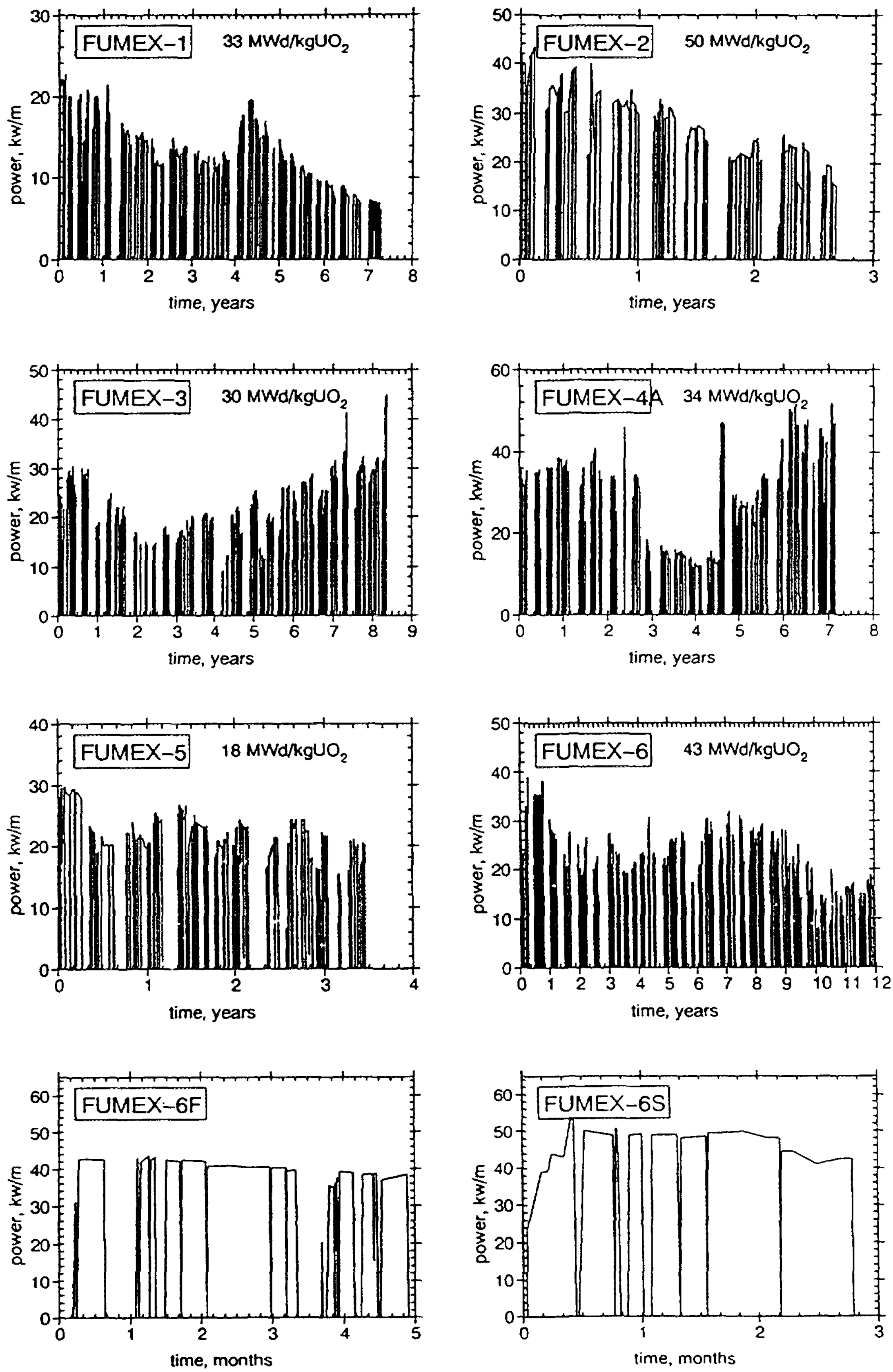


FIG. 1. Average linear heat ratings for the FUMEX test cases.

Country	Code Name	Number	Temp. predictions (Deg.C)
ARG	BACO	1	600
BUL	PIN-MICRO	2	680
CAN	ELESIM	3	560
FIN	VTT/ENIGMA	4	640
FRA/EDF	TRANSURANUS/EDF	5	650
FRA/CEA	METEOR/TRANSURANUS	6	665
CEC	TRANSURANUS	7	650
IND1	PROFESS	8	650
IND2	FAIR	9	665
IND3	FUDA-MOD1	10	590
JPN1	TRUST-VIb	11	676
JPN2	EIMUS	12	700
CHIN	FRAPCON-2	13	540
ROM	ROFEM-1B	14	1000
SWIT	TRANSURANUS	15	638
CZE	PIN-W	16	530
UK1	ENIGMA-5.2	17	660
UK2	ENIGMA-5.8D	18	640
RUS	START	19	760
	EXPERIMENTAL RESULT		705

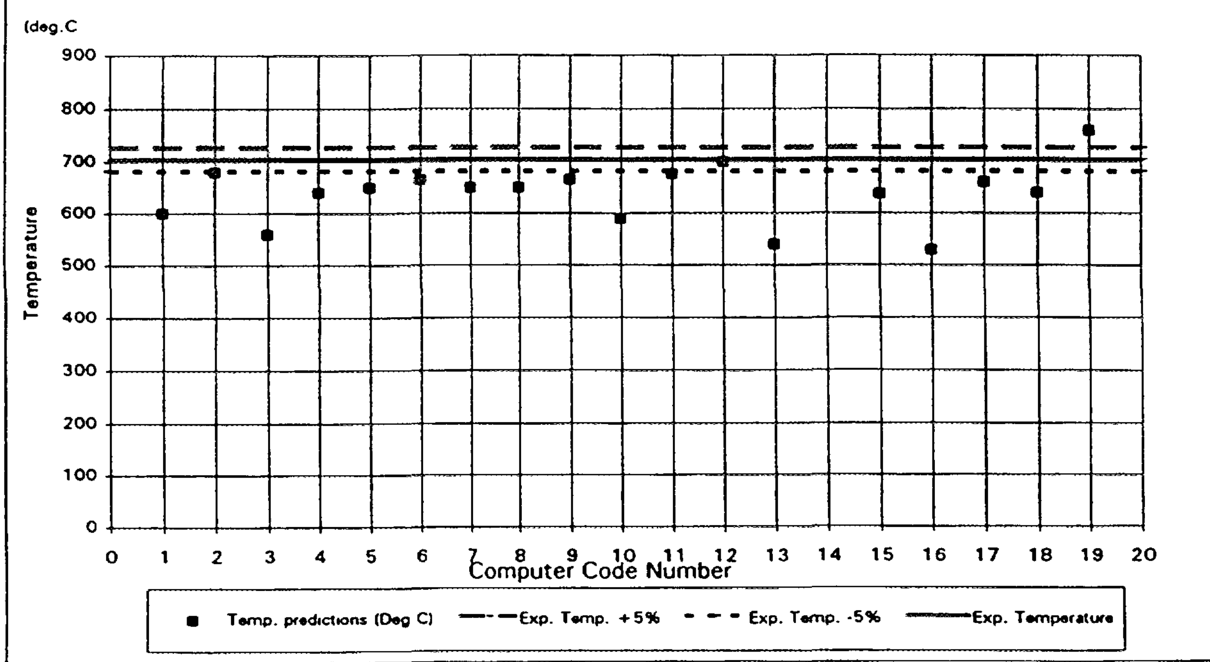


FIG. 2. Comparison of code predictions and data for FUMEX 1 fuel centre temperature at 5 MW·d/kgUO$_2$ and 15 kW/m.

Country	Code Name	Number	Temperature (Deg C)
ARG	BACO	1	600
BUL	PIN-MICRO	2	650
CAN	ELESIM	3	580
FIN	VTT/ENIGMA	4	680
FRA/EDF	TRANSURANUS/EDF	5	670
FRA/CEA	METEOR/TRANSURANUS	6	660
CEC	TRANSURANUS	7	610
IND1	PROFESS	8	610
IND2	FAIR	9	665
IND3	FUDA-MOD1	10	610
JPN1	TRUST-V1b	11	700
JPN2	EIMUS	12	800
CHIN	FRAPCON2	13	508
ROM	ROFEM-1B	14	na
SWIT	TRANSURANUS	15	600
CZE	PIN-W	16	535
UK1	ENIGMA-5 2	17	685
UK2	ENIGMA-5 8D	18	750
RUS	START	19	760
	EXPERIMENTAL RESULT		740

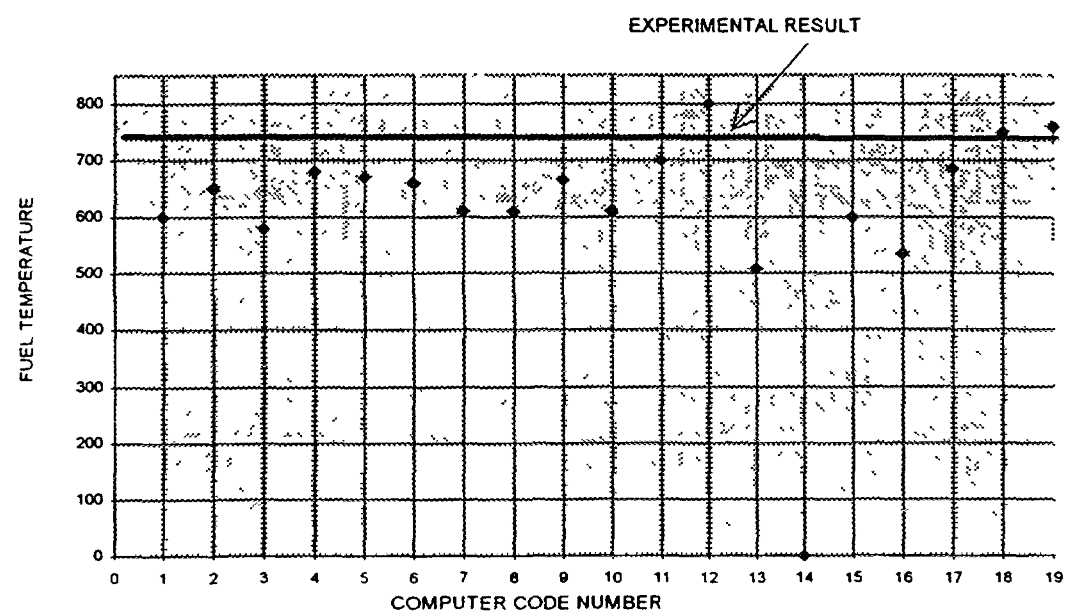


FIG. 3. Comparison of code predictions and data for FUMEX 1 fuel centre temperature at 20 MW·d/kgUO_2 and 15 kW/m.

Country	Code Name	Number	F.G.R (%)
ARG	BACO	1	2.18
BUL	PIN-MICRO	2	0.87
CAN	ELESIM	3	0.3
FIN	VTT/ENIGMA	4	0.05
FRA/EDF	TRANSURANUS/EDF	5	1.66
FRA/CEA	METEOR/TRANSURANUS	6	0 13
CEC	TRANSURANUS	7	0.19
IND1	PROFESS	8	0.6
IND2	FAIR	9	0.2
IND3	FUDA-MOD1	10	1.4
JPN1	TRUST-V1b	11	0.6
JPN2	EIMUS	12	1.06
CHIN	FRAPCON2	13	0.1
ROM	ROFEM-1B	14	100
SWIT	TRANSURANUS	15	0.2
CZE	PIN-W	16	0.55
UK1	ENIGMA-5.2	17	0.05
UK2	ENIGMA-5.8D	18	0.07
RUS	START	19	0.35
	EXPERIMENTAL RESULT	20	

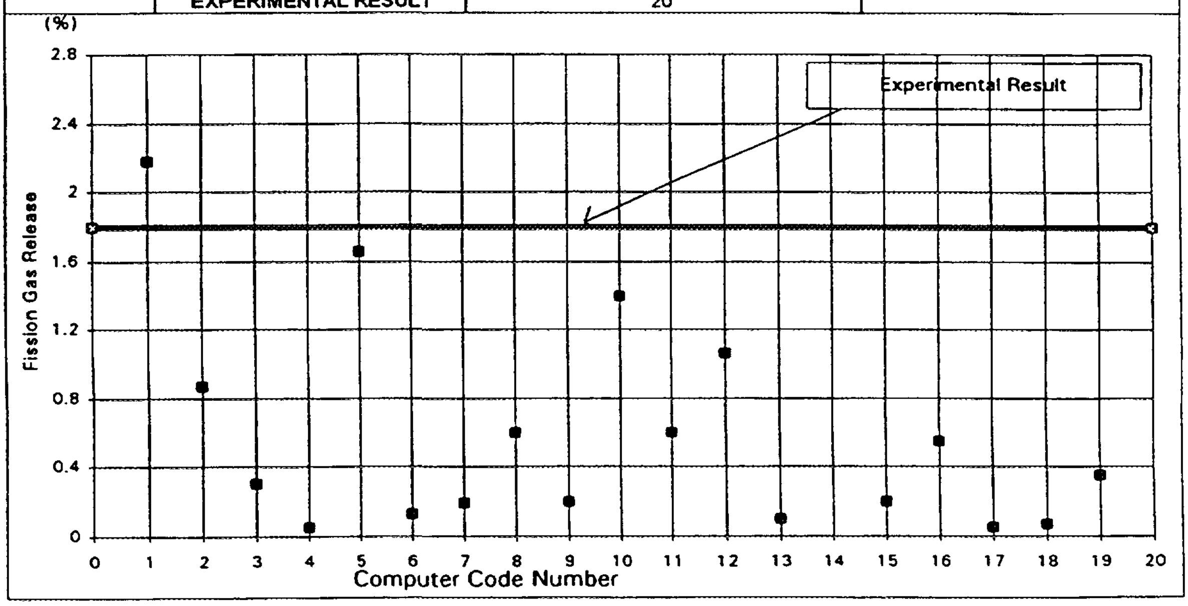


FIG. 4. Comparison of code predictions and data for FUMEX 1 fission gas release at end-of-life.

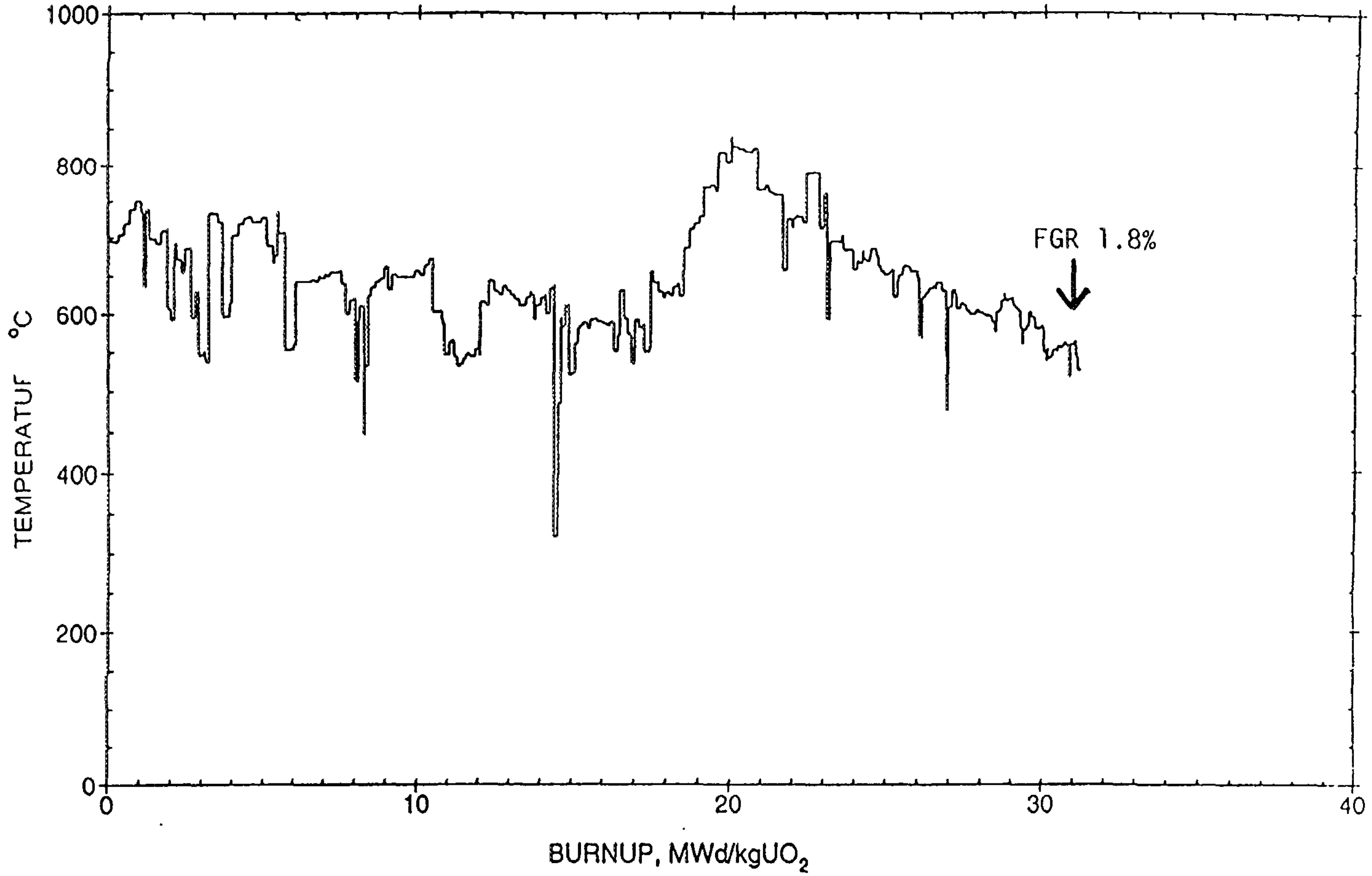


FIG. 5. Fuel centreline temperature for FUMEX 1 as a function of burnup.

Country	Code Name	Number	E O L	5 MWd/kg
ARG	BACO	1	610	605
BUL	PIN-MICRO	2	620	740
CAN	ELESIM	3	[illegible]70	570
FIN	VTT/ENIGMA	4	850	660
FRA/EDF	TRANSURANUS/EDF	5	780	675
FRA/CEA	METEOR/TRANSURANUS	6	3000	3000
CEC	TRANSURANUS	7	700	720
IND1	PROFESS	8	1000	3000
IND2	FAIR	9	707	723
IND3	FUDA-MOD1	10	676	720
JPN1	TRUST-V1b	11	785	616
JPN2	EIMUS	12	3000	3000
CHIN	FRAPCON2	13	577	600
ROM	ROFEM-1B	14	768	724
SWIT	TRANSURANUS	15	661	680
CZE	PIN-W	16	540	605
UK1	ENIGMA-5 2	17	845	725
UK2	ENIGMA-5 8D	18	885	695
RUS	START	19	480	735
	EXPERIMENTAL RESULT	20		

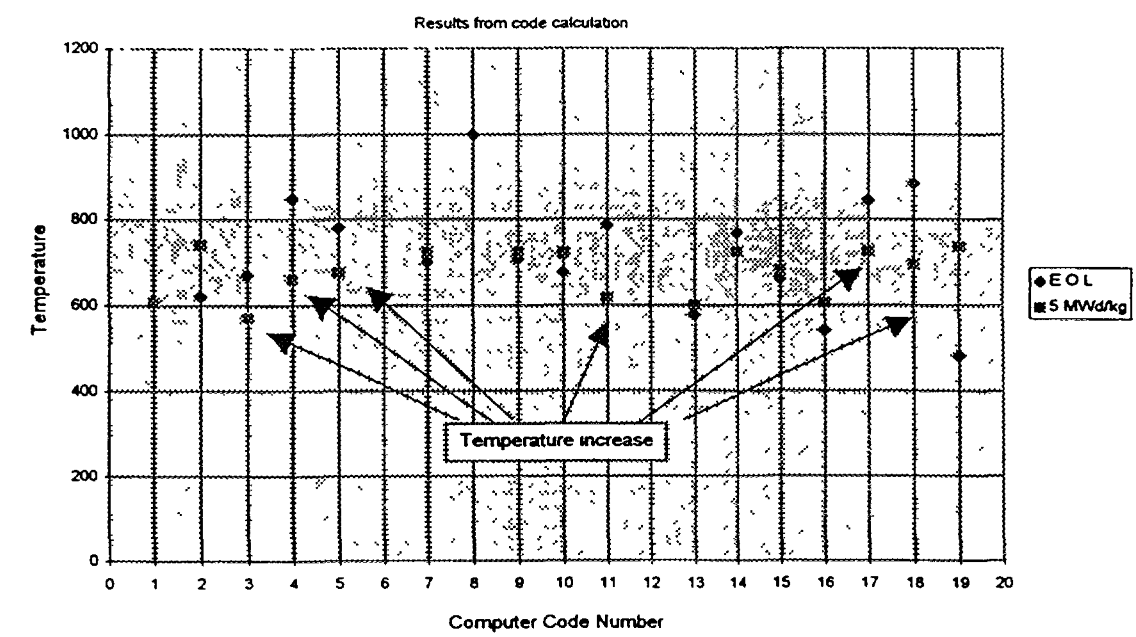


FIG. 6. Comparison of code predictions and data for FUMEX 2: fuel centre temperatures at 15 kW/m, 5 MW·d/kgUO$_2$ and end-of-life.

Country	Code Name	Number	Temperature
ARG	BACO	1	1590
BUL	PIN-MICRO	2	1820
CAN	ELESIM	3	1330
FIN	VTT/ENIGMA	4	1520
FRA/EDF	TRANSURANUS/EDF	5	1510
FRA/CEA	METEOR/TRANSURANUS	6	1590
CEC	TRANSURANUS	7	1640
IND1	PROFESS	8	1760
IND2	FAIR	9	1584
IND3	FUDA-MOD1	10	1753
JPN1	TRUST-V1b	11	1461
JPN2	EIMUS	12	
CHIN	FRAPCON2	13	1412
ROM	ROFEM-1B	14	1656
SWIT	TRANSURANUS	15	1554
CZE	PIN-W	16	1210
UK1	ENIGMA-5 2	17	1640
UK2	ENIGMA-5 8D	18	1625
RUS	START	19	1445

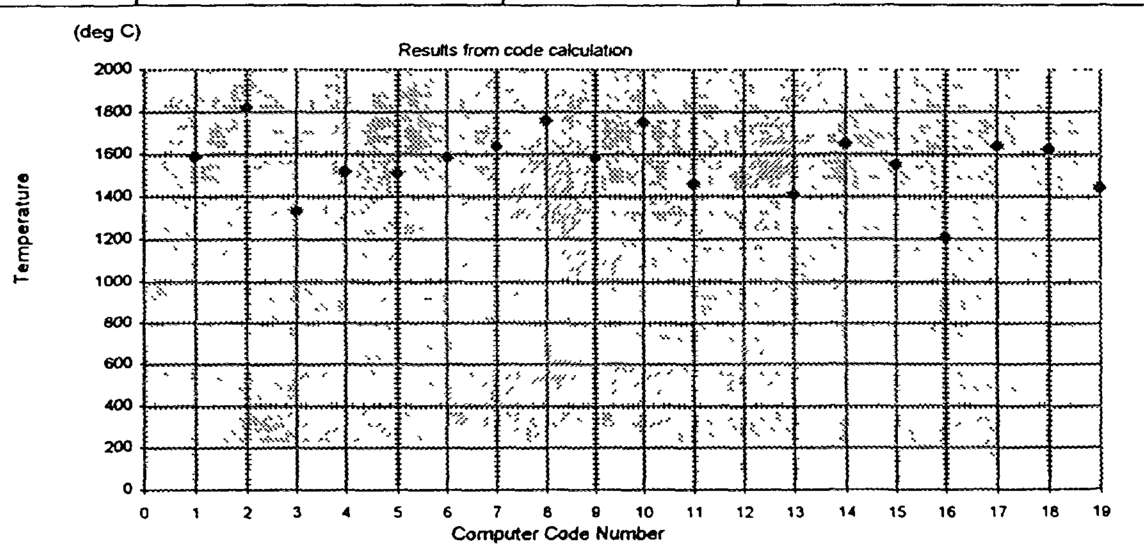


FIG. 7. Comparison of code predictions for FUMEX 2: fuel centre temperatures at 5 MW·d/kgUO$_2$ and 40 kW/m.

Country	Code Name	Number	F G R (%)
ARG	BACO	1	3 4
BUL	PIN-MICRO	2	6 3
CAN	ELESIM	3	2
FIN	VTT/ENIGMA	4	2 9
FRA/EDF	TRANSURANUS/EDF	5	6 4
FRA/CEA	METEOR/TRANSURANUS	6	14
CEC	TRANSURANUS	7	4
IND1	PROFESS	8	9 6
IND2	FAIR	9	4 5
IND3	FUDA-MOD1	10	6 9
JPN1	TRUST-V1b	11	10 8
JPN2	EIMUS	12	28 8
CHIN	FRAPCON2	13	3 1
ROM	ROFEM-1B	14	6 2
SWIT	TRANSURANUS	15	2 6
CZE	PIN-W	16	1 2
UK1	ENIGMA-5 2	17	3 3
UK2	ENIGMA-5 8D	18	4 7
RUS	START	19	8 5
	EXPERIMENTAL RESULT		3

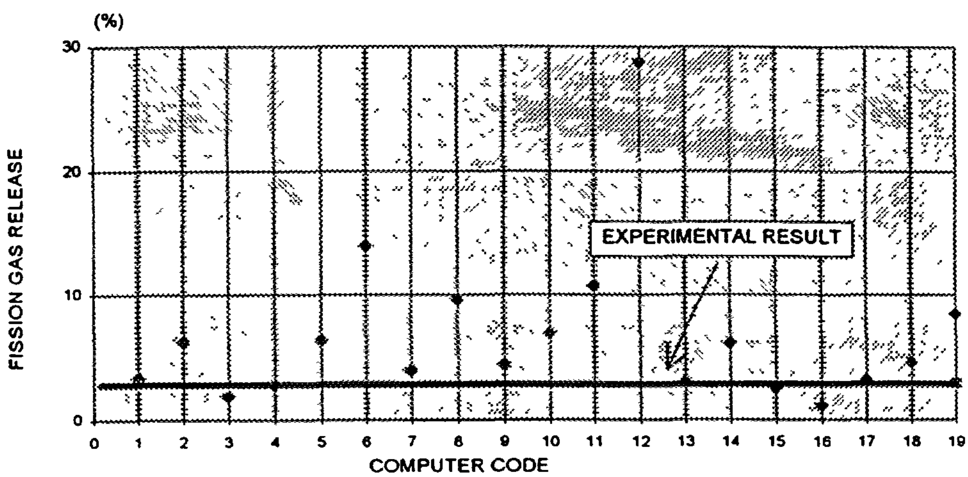


FIG. 8. Comparison of code predictions and data for FUMEX 2: fission gas release at the end-of-life.

Country	Code Name	Number	Pressure (bar)
ARG	BACO	1	26
BUL	PIN-MICRO	2	28 8
CAN	ELESIM	3	25 6
FIN	VTT/ENIGMA	4	25 3
FRA/EDF	TRANSURANUS/EDF	5	28 5
FRA/CEA	METEOR/TRANSURANUS	6	39 6
CEC	TRANSURANUS	7	24 4
IND1	PROFESS	8	30 6
IND2	FAIR	9	26 8
IND3	FUDA-MOD1	10	46
JPN1	TRUST-V1b	11	35 6
JPN2	EIMUS	12	50
CHIN	FRAPCON2	13	23 7
ROM	ROFEM 1B	14	29 3
SWIT	TRANSURANUS	15	24 7
CZE	PIN W	16	20 1
UK1	ENIGMA-5.2	17	24.3
UK2	ENIGMA-5.8D	18	27
RUS	START	19	30 7
	EXPERIMENTAL RESULT	20	

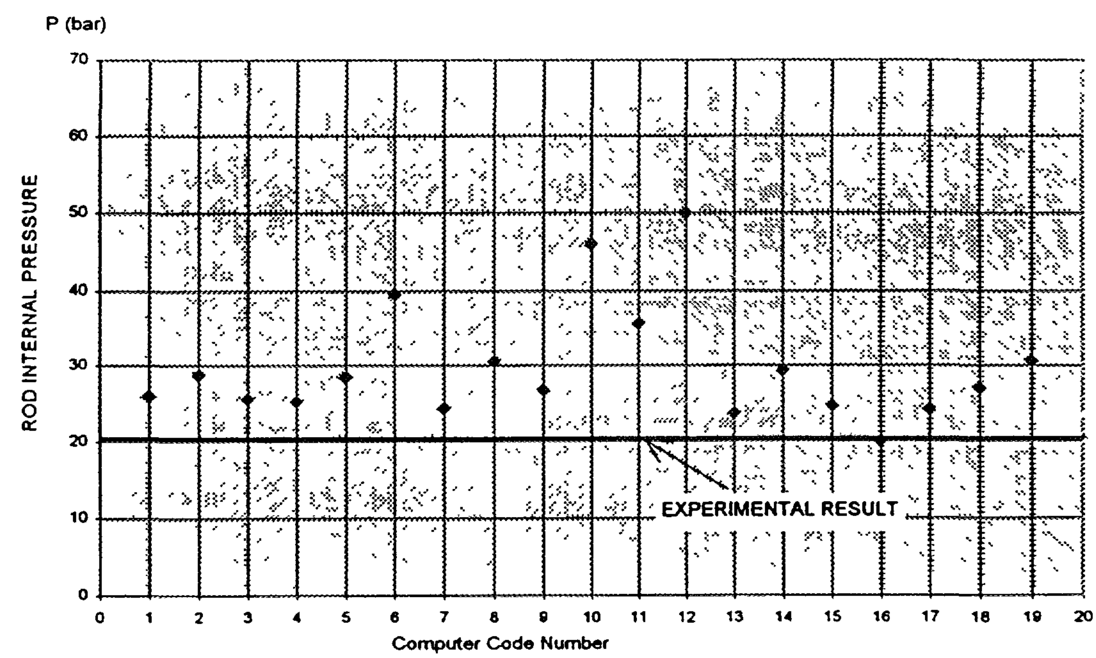


FIG. 9. Comparison of code predictions and data for FUMEX 2: internal rod pressure at power and end-of-life.

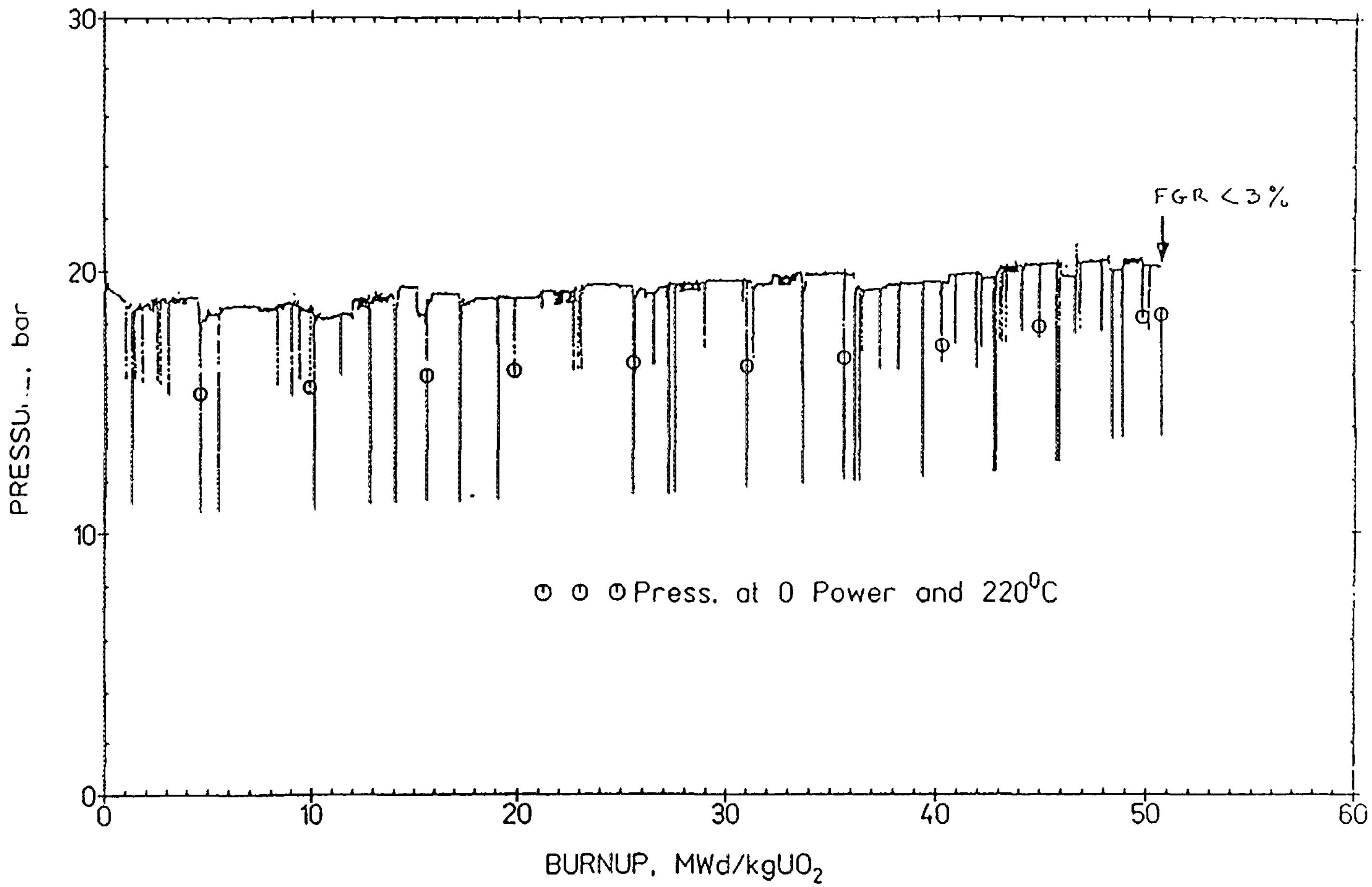
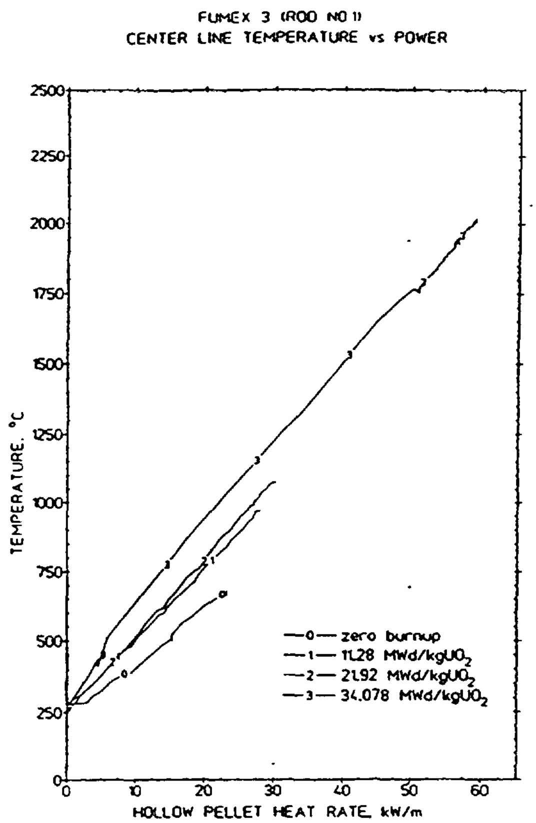
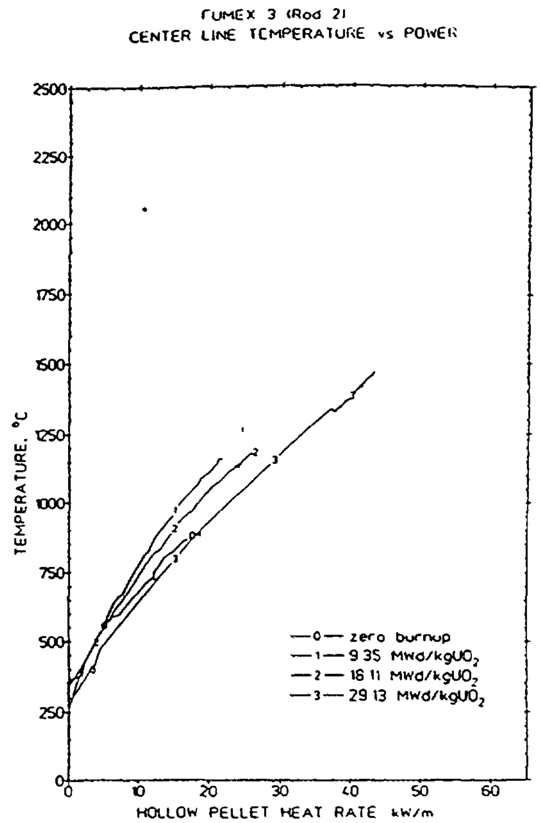


FIG. 10. Rod internal pressure for FUMEX 2 as a function of burnup.





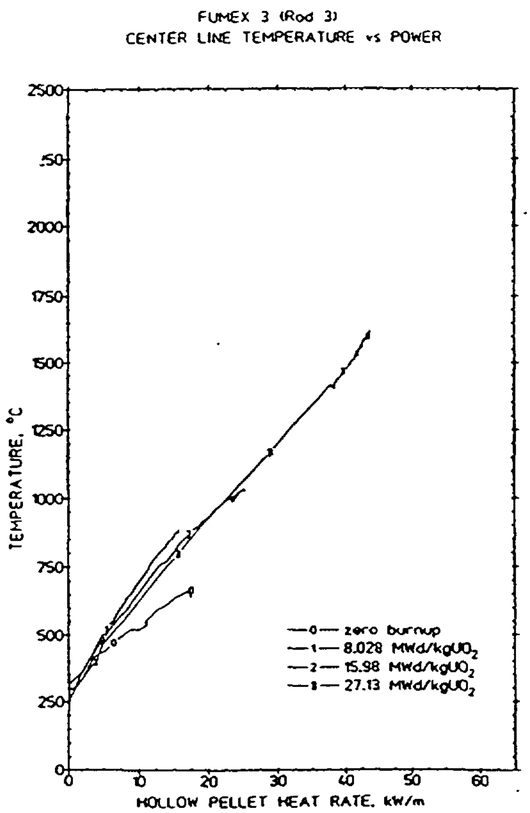


FIG. 11. Fuel centreline temperature for the three rods of FUMEX 3 as a function of power: rod 1, helium filled, 100 μm gap; rod 2, xenon filled, 100 μm gap; rod 3, xenon filled, 50 μm gap.

Country	Code Name	Number	5MWd/kg	Before Ramp
		0		
ARG	BACO	1	904	
BUL	PIN-MICRO	2	1040	1134
CAN	ELESIM	3	890	981
FIN	VTT/ENIGMA	4	1027	1192
FRA/EDF	TRANSURANUS/EDF	5	962	1230
FRA/CEA	METEOR/TTRANSURANUS	6	1088	1405
CEC	TRANSURANUS	7	1038	1060
IND1	PROFESS	8		
IND2	FAIR	9	1127	1173
IND3	FUDA-MOD1	10	947	836
JPN1	TRUST-V1b	11	1019	1346
JPN2	EIMUS	12	1059	1280
CHIN	FRAPCON2	13	865	875
ROM	ROFEM 1B	14	1134	1245
SWIT	TRANSURANUS	15	954	961
CZE	PIN-W	16	775	755
UK1	ENIGMA-5 2	17	1070	1269
UK2	ENIGMA-5 8D	18	1022	1155
RUS	START	19		
		20		

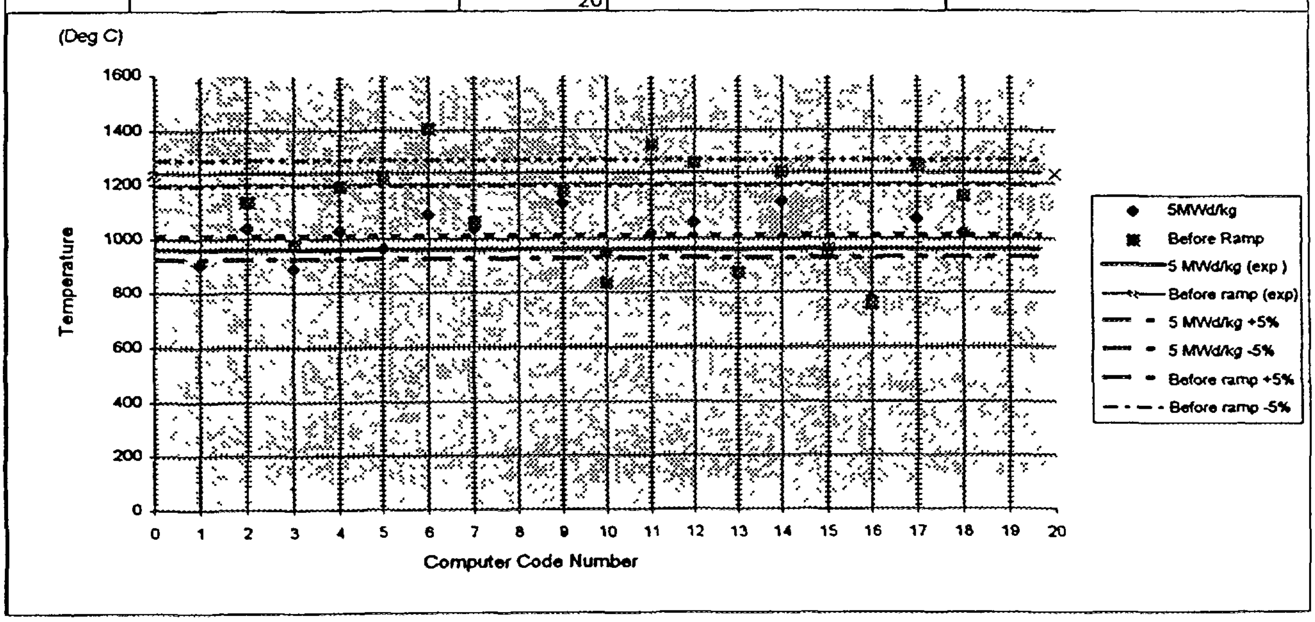


FIG. 12. Comparison of code predictions and data for FUMEX 3 rod 1: fuel centre temperatures at 30 kW/m, 5 MW·d/kgUO$_2$ and before the power ramp.

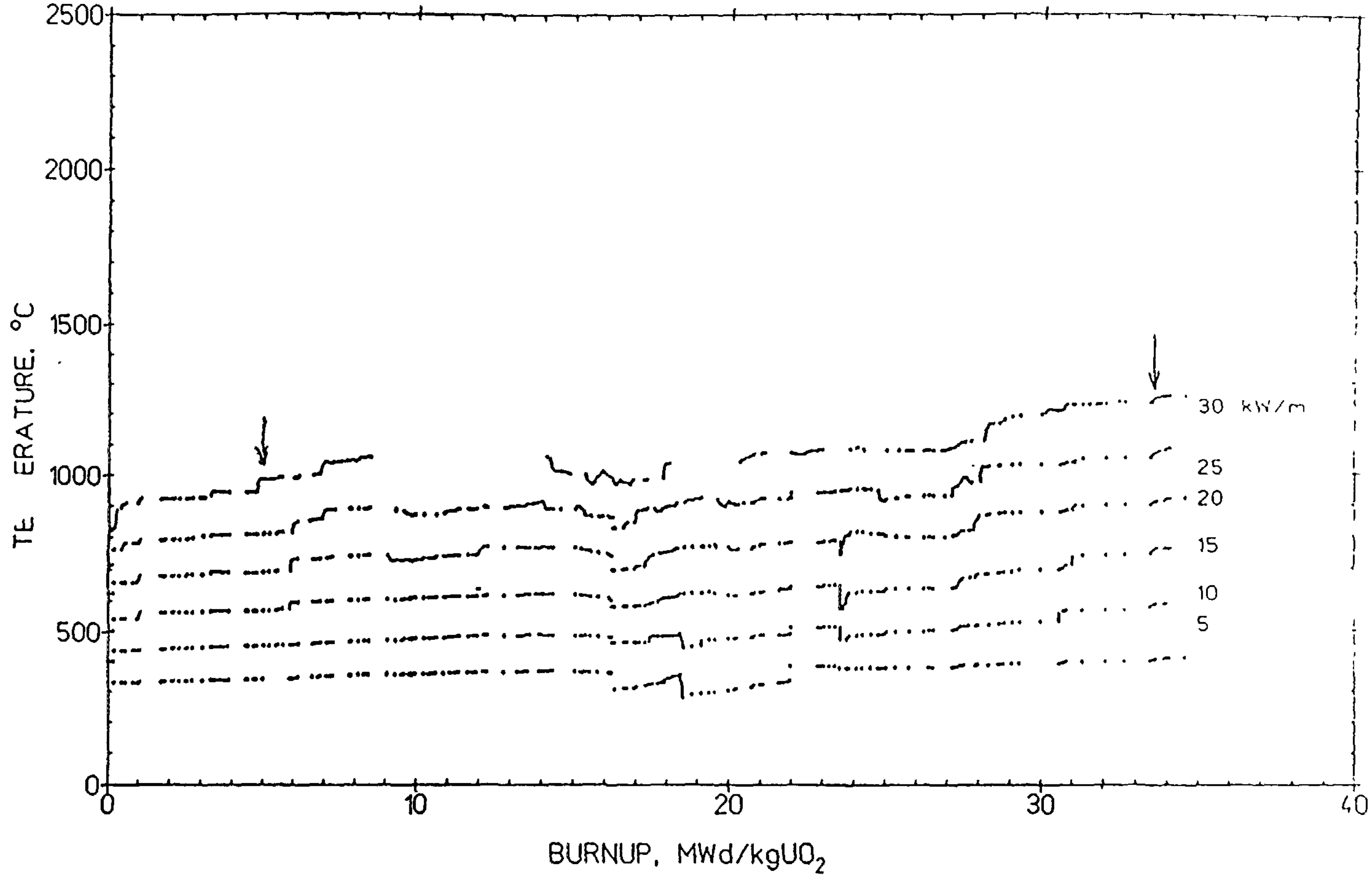


FIG. 13. Fuel centreline temperatures at constant power levels for FUMEX 3 rod 1 as a function of burnup.

Country	Code Name	Number	Before ramp	Top of Ramp
		0		
ARG	BACO	1		2422
BUL	PIN MICRO	2	1134	1928
CAN	ELESIM	3	981	2133
FIN	VTT/ENIGMA	4	1192	2328
FRA/EDF	TRANSURANUS/EDF	5	1230	2161
FRA/CEA	METEOR/TRANSURANUS	6	1405	2347
CEC	TRANSURANUS	7	1060	2327
IND1	PROFESS	8		
IND2	FAIR	9	1173	2310
IND3	FUDA MOD1	10	836	1704
JPN1	TRUST-V1b	11	1346	2347
JPN2	EIMUS	12	1280	2339
CHIN	FRAPCON2	13	875	1606
ROM	ROFEM 1B	14	1245	2435
SWIT	TRANSURANUS	15	961	2109
CZE	PIN W	16	755	1646
UK1	ENIGMA 5 2	17	1269	2424
UK2	ENIGMA 5 8D	18	1155	2386
RUS	START	19		

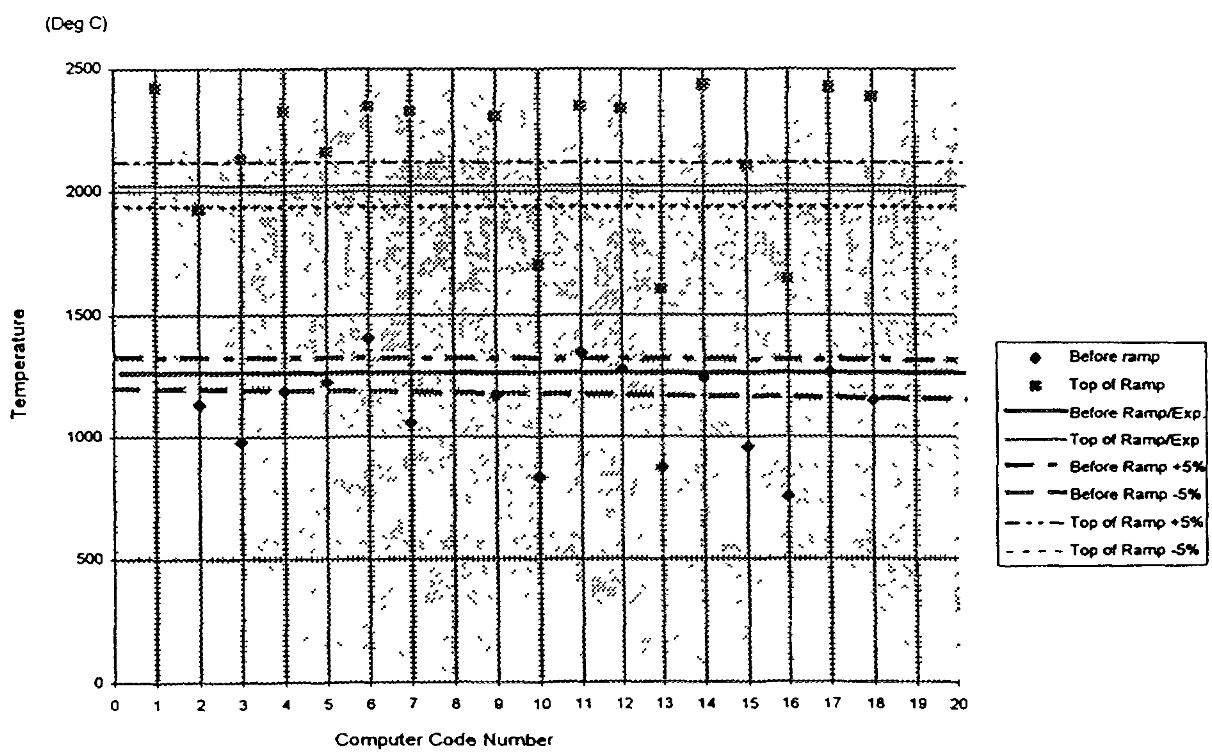


FIG. 14. Comparison of code predictions and data for FUMEX 3 rod 1 fuel centreline temperatures just before and at the top of the power ramp.

Country	Code Name	Number	Before Ramp	After Ramp
ARG	BACO	1	6 7	37 2
BUL	PIN-MICRO	2	5 3	14 8
CAN	ELESIM	3	1	11 1
FIN	VTT/ENIGMA	4	21 5	48 1
FRA/EDF	TRANSURANUS/EDF	5	21 6	24 8
FRA/CEA	METEOR/TRANSURANUS	6	38 1	61 2
CEC	TRANSURANUS	7	2 3	52 3
IND1	PROFESS	8		
IND2	FAIR	9	14 5	69 9
IND3	FUDA-MOD1	10	1 8	37 9
JPN1	TRUST-V1b	11	40 5	54 3
JPN2	EIMUS	12	21 9	50
CHIN	FRAPCON2	13	20	36 1
ROM	ROFEM-1B	14	16 9	48 2
SWIT	TRANSURANUS	15	0	39 6
CZE	PIN-W	16	0 7	38 6
UK1	ENIGMA-5.2	17	24 9	61 1
UK2	ENIGMA-5.8D	18	24 4	53 8
RUS	START	19		

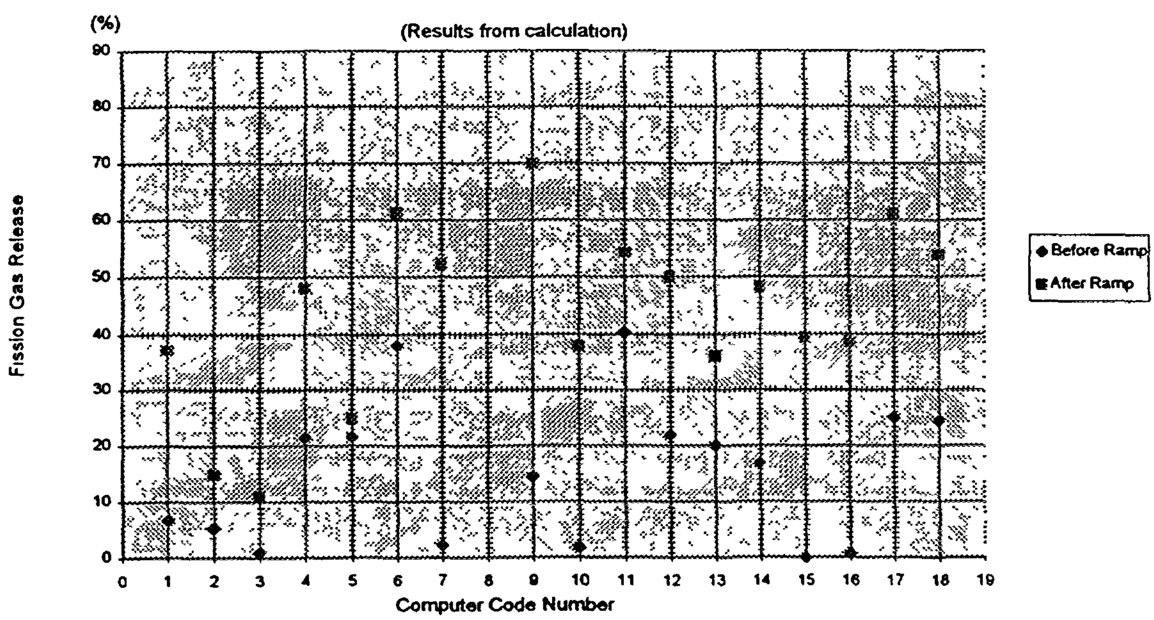


FIG. 15. *Comparison of code predictions and data for FUMEX 3 rod 1 fission gas release before and after the power ramp.*

Country	Code Name	Number	T at 5 MWd/kg	T Before Ramp
		0		
ARG	BACO	1	1310	1352
BUL	PIN MICRO	2	1488	1365
CAN	ELESIM	3	930	865
FIN	VTT/ENIGMA	4	1170	1057
FRA/EDF	TRANSURANUS/EDF	5	1190	1137
FRA/CEA	METEOR/TRANSURANUS	6	1413	1254
CEC	TRANSURANUS	7	1432	1040
IND1	PROFESS	8		
IND2	FAIR	9	1400	1125
IND3	FUDA-MOD1	10	1212	1230
JPN1	TRUST V1b	11	1430	1211
JPN2	EIMUS*	12	1224	1133
CHIN	FRAPCON2	13	1057	961
ROM	ROFEM-1B	14	1501	1198
SWIT	TRANSURANUS	15	1336	980
CZE	PIN-W	16	1144	1109
UK1	ENIGMA-5 2	17	1080	1065
UK2	ENIGMA 5 8D	18	1079	1083
RUS	START	19		
	EXP. RESULT	20	1345	1040

* Results communicated after release of the experimental data.

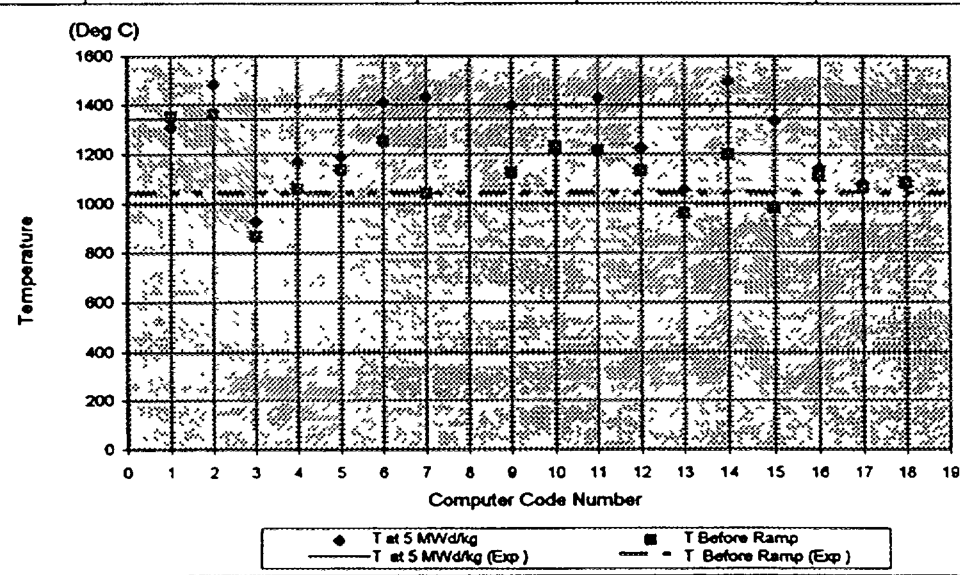


FIG. 16. Comparison of code predictions and data for FUMEX 3 rod 2: fuel centre temperatures at 25 kW/m, 5 MW·d/kgUO$_2$ just before the power ramp.

COUNTRY	Code Name	Numb	T Bef Ramp	T Top of Ramp
		0		
ARG	BACO	1	1352	1953
BUL	PIN-MICRO	2	1365	1832
CAN	ELESIM	3	865	1715
FIN	VTT/ENIGMA	4	1057	1876
FRA/EDF	TRANSURANUS/EDF	5	1137	1667
FRA/CEA	METEOR/TRANSURANUS	6	1327	2136
CEC	TRANSURANUS	7	1040	1718
IND1	PROFESS	8		
IND2	FAIR	9	1125	1736
IND3	FUDA-MOD1	10	1230	2108
JPN1	TRUST-V1b	11	1211	1870
JPN2	EIMUS*	12	1133	1867
CHIN	FRAPCON2	13	961	1327
ROM	ROFEM 1B	14	1198	1894
SWIT	TRANSURANUS	15	980	1559
CZE	PIN-W	16	1109	1711
UK1	ENIGMA-5 2	17	1065	1894
UK2	ENIGMA 5 8D	18	1083	1911
RUS	START	19		
	EXP. RESULT	20	1040	1470

* Results communicated after the release of the experimental data

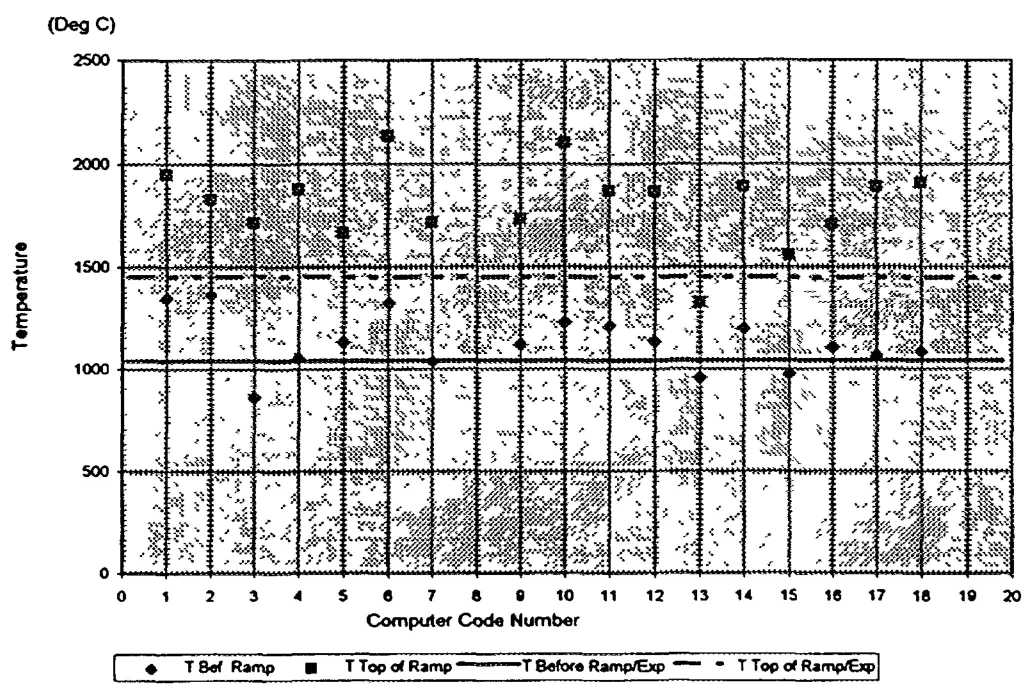


FIG. 17. Comparison of code predictions and data for FUMEX 3 rod 2: fuel centre temperatures just before the ramp at a power of 25 kW/m, and at the top of the ramp.

Country	Code Name	Number	FGR Bef Ramp	FGR Aft Ramp
ARG	BACO	1	26 8	30 8
BUL	PIN-MICRO	2	24 3	25
CAN	ELESIM	3	0 6	30 1
FIN	VTT/ENIGMA	4	8 4	28 6
FRA/EDF	TRANSURANUS/EDF	5	5 3	5 6
FRA/CEA	METEOR/TRANSURANUS	6	44	50 5
CEC	TRANSURANUS	7	4 7	20
IND1	PROFESS	8		
IND2	FAIR	9	3 2	10 7
IND3	FUDA-MOD1	10	38 3	46
JPN1	TRUST-V1b	11	5 4	7 7
JPN2	EIMUS*	12	15 7	32 6
CHIN	FRAPCON2	13	6 2	6 81
ROM	ROFEM-1B	14	10 1	18 1
SWIT	TRANSURANUS	15	2 2	10 4
CZE	PIN-W	16	21 7	28
UK1	ENIGMA-5 2	17	8 3	27 5
UK2	ENIGMA-5 8D	18	10	32 3
RUS	START	19		

* Results communicated after release of the experimental data

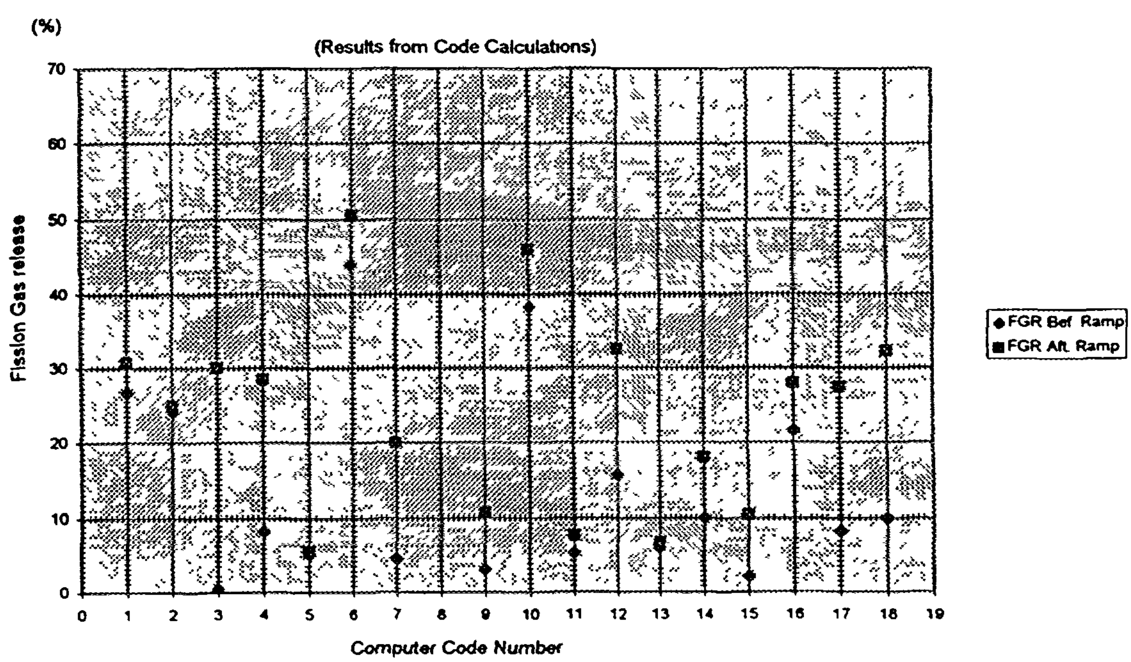


FIG. 18. Comparison of code predictions and data for FUMEX 3 rod 2 fission gas release before and after the power ramp.

Country	Code Name	Number	5MWd/kg	Before Ramp
		0		
ARG	BACO	1	1430	1400
BUL	PIN-MICRO	2	1240	1192
CAN	ELESIM	3	850	865
FIN	VTT/ENIGMA	4	1169	980
FRA/EDF	TRANSURANUS/EDF	5	1100	1040
FRA/CEA	METEOR/TRANSURANUS	6	1288	1307
CEC	TRANSURANUS	7	1057	1000
IND1	PROFESS	8		
IND2	FAIR	9	1136	1115
IND3	FUDA-MOD1	10	1174	1230
JPN2	TRUST-V1b	11	1336	1173
JPN1	EIMUS*	12	1012	1056
CHIN	FRAPCON2	13	1019	971
ROM	ROFEM-1B	14	1214	1020
SWIT	TRANSURANUS	15	1030	951
CZE	PIN-W	16	1091	1100
UK1	ENIGMA-5 2	17	1155	985
UK2	ENIGMA-5 8D	18	1183	1004
RUS	START	19		
	EXP RESULT	20	1080	940

* Results communicated after release of the experimental data

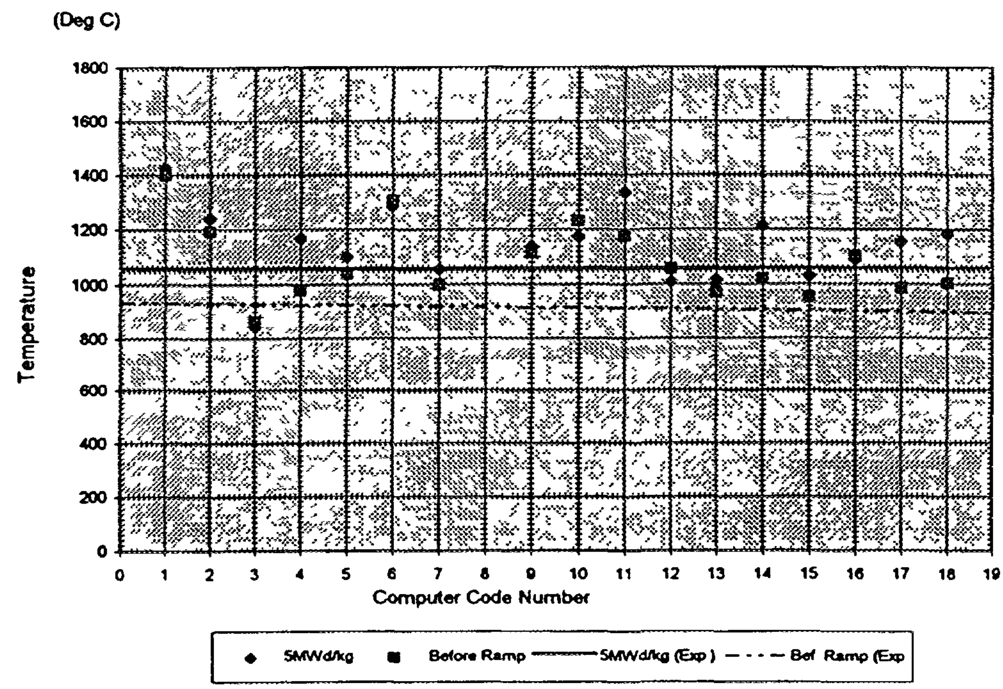


FIG. 19. Comparison of predictions and data for FUMEX 3 rod 3: fuel centre temperatures at 25 kW/m, 5 MW·d/kgUO$_2$ and before the power ramp.

Country	Code Name	Number	Bef Ramp	Top of Ramp
		0		
ARG	BACO	1	1400	2254
BUL	PIN-MICRO	2	1192	1555
CAN	ELESIM	3	865	1743
FIN	VTT/ENIGMA	4	980	1787
FRA/EDF	TRANSURANUS/EDF	5	1057	1602
FRA/CEA	METEOR/TRANSURANUS	6	1307	2202
CEC	TRANSURANUS	7	1000	1685
IND1	PROFESS	9		
IND2	FAIR	9	1115	1720
IND3	FUDA-MOD1	10	1230	2135
JPN1	TRUST-V1b	11	1173	1851
JPN2	EIMUS *	12	1056	1792
CHIN	FRAPCON2	13	971	1351
ROM	ROFEM-1B	14	1020	1850
SWIT	TRANSURANUS	15	951	1551
CZE	PIN-W	16	1100	1470
UK1	ENIGMA-5 2	17	985	1787
UK2	ENIGMA-5 8D	18	1004	1803
RUS	START	19		
	EXP RESULT	20	940	1615
	* Results communicated after release of experimental data			

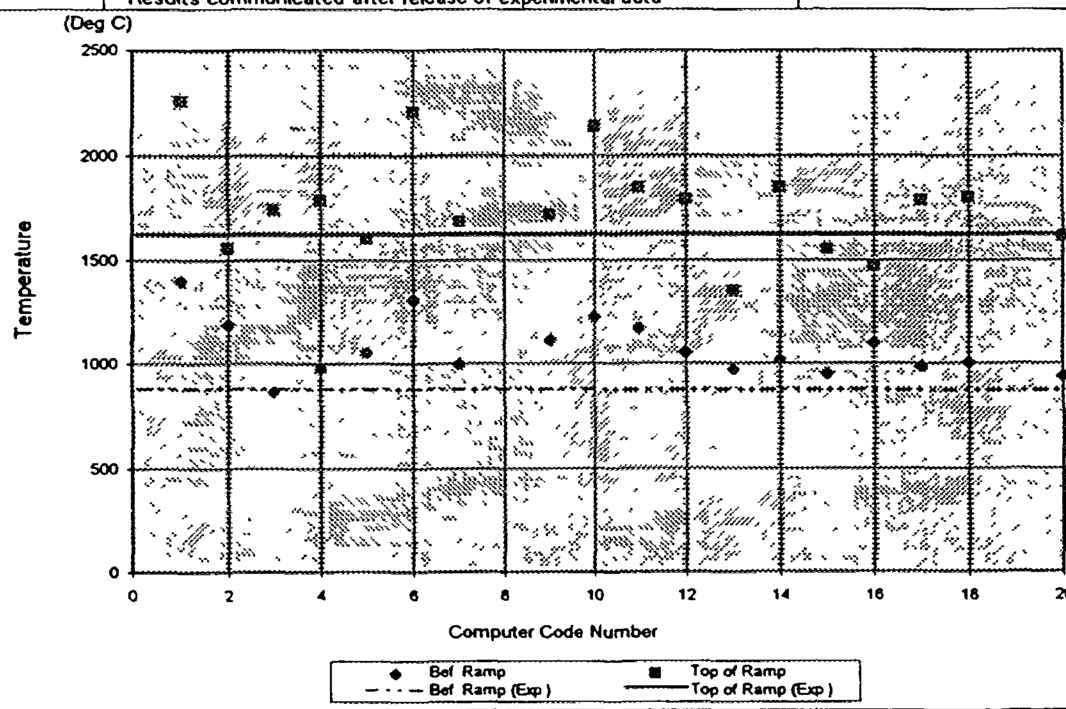


FIG. 20. Comparison of predictions and data for FUMEX 3 rod 3: fuel centre temperatures just before the ramp at a power of 25 kW/m, and at the top of the ramp.

Country	Code Name	Number	Before Ramp	After Ramp
ARG	BACO	1	16 2	19 3
BUL	PIN-MICRO	2	4 4	7 1
CAN	ELESIM	3	0 3	30 8
FIN	VTT/ENIGMA	4	16 1	31 4
FRA/EDF	TRANSURANUS/EDF	5	8 9	11 2
FRA/CEA	METEOR/TU	6	39 3	48 9
CEC	TRANSURANUS	7	4 2	24 2
IND1	PROFESS	8		
IND2	FAIR	9	22 1	45
IND3	FUDA-MOD1	10	27 4	34 8
JAP1	TRUST-V1b	11	26 8	32 2
JAP2	EIMUS *	12	14 5	34 9
CHIN	FRAPCON2	13	55 4	61 5
ROM	ROFEM 1B	14	9 2	20 7
SWIT	TRANSURANUS	15	0 7	13
CZE	PIN W	16	5 6	9 1
UK1	ENIGMA 5 2	17	12 8	37 7
UK2	ENIGMA-5 8D	18	15 4	35 4
RUS	START	19		

* Results communicated after release of experimental data

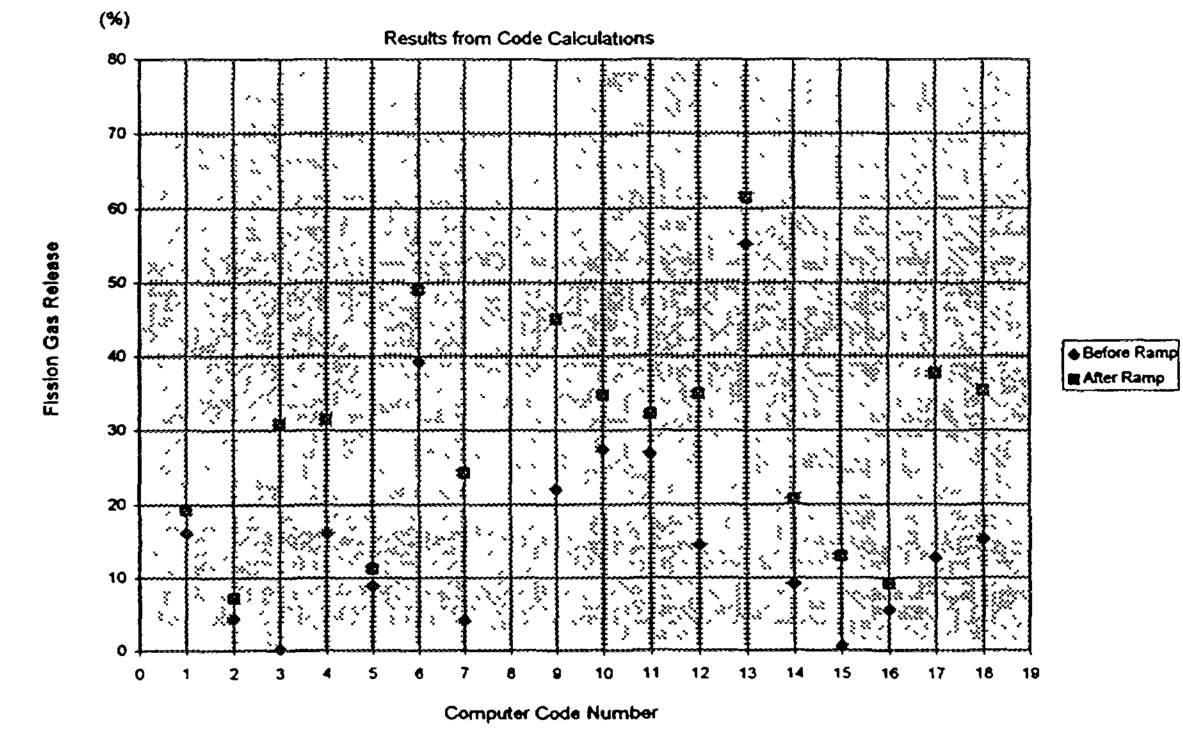
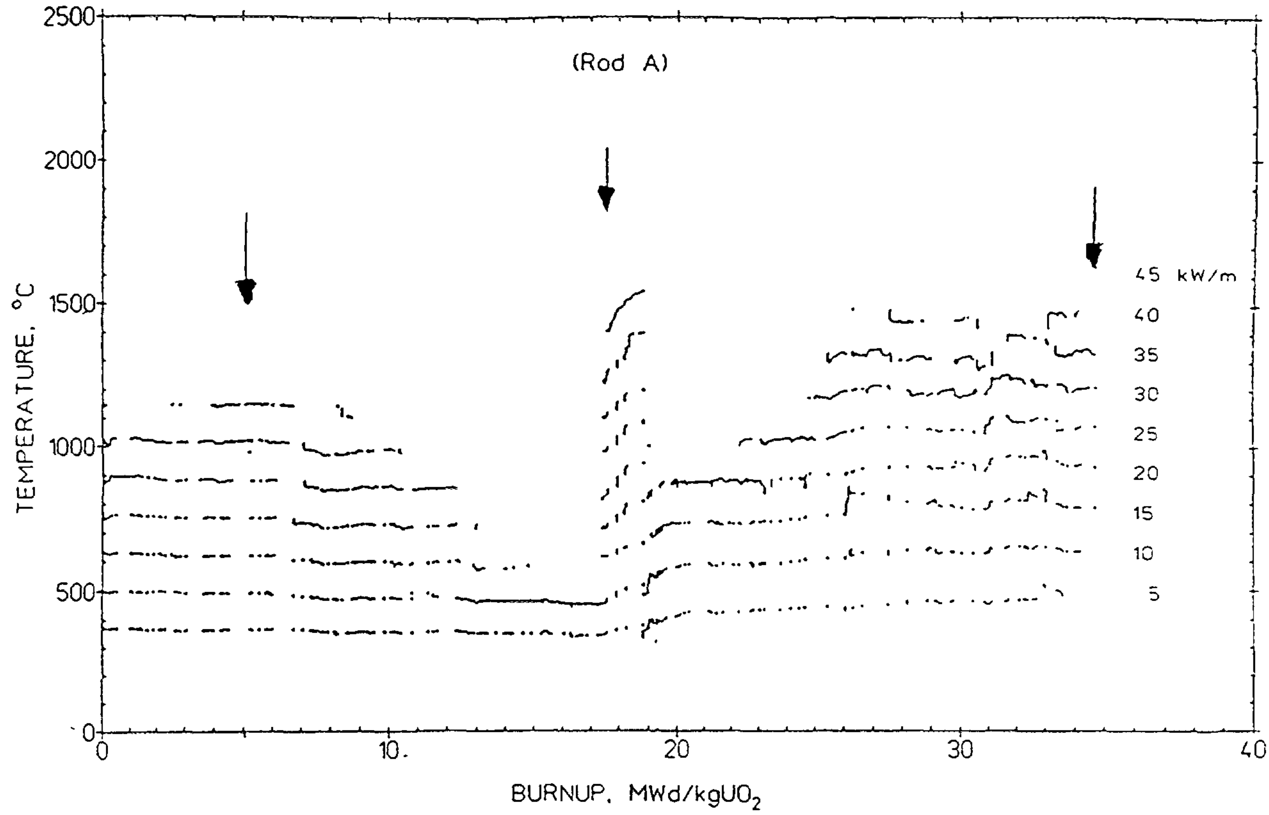


FIG. 21. Comparison of code predictions for FUMEX 3 rod 3 fission gas release before and after the power ramp.



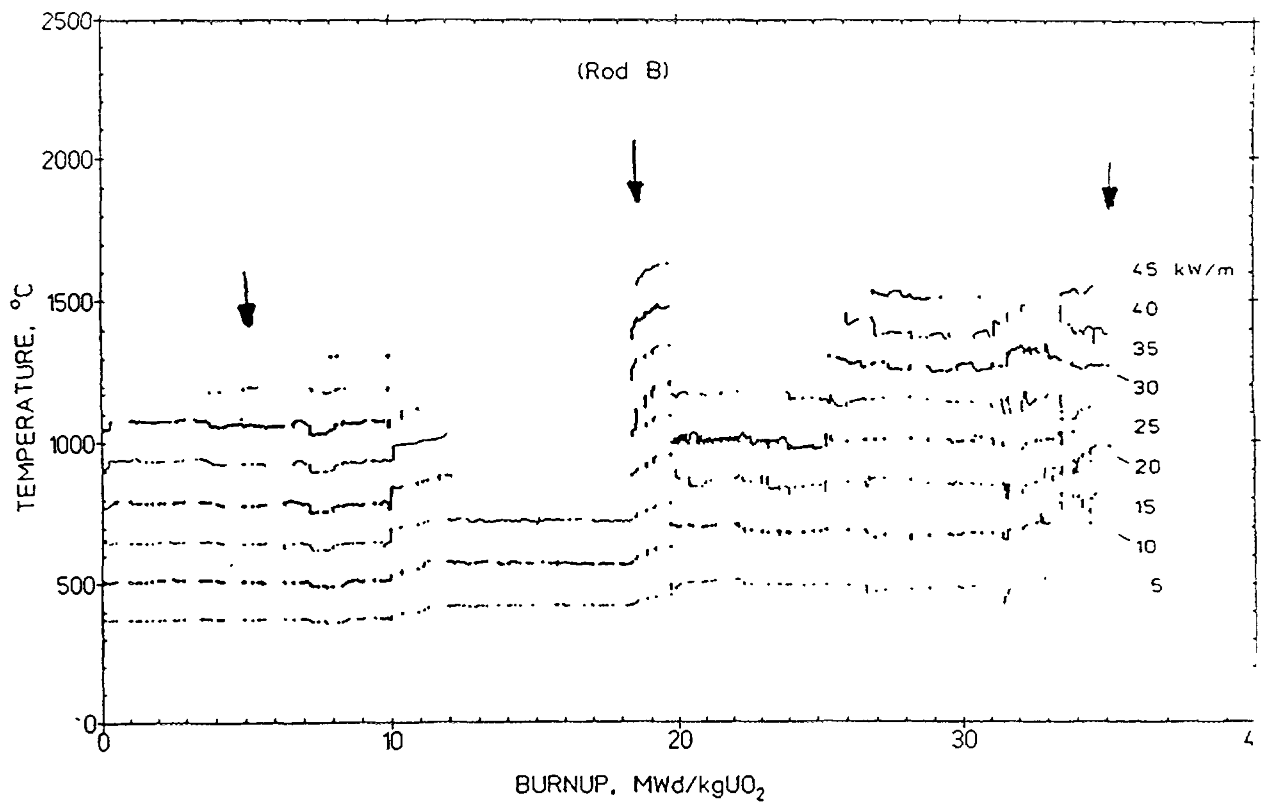


FIG. 22. Fuel centreline temperatures for FUMEX 4 rods as a function of burnup showing where the code comparisons were made: (a) rod A, (b) rod B.

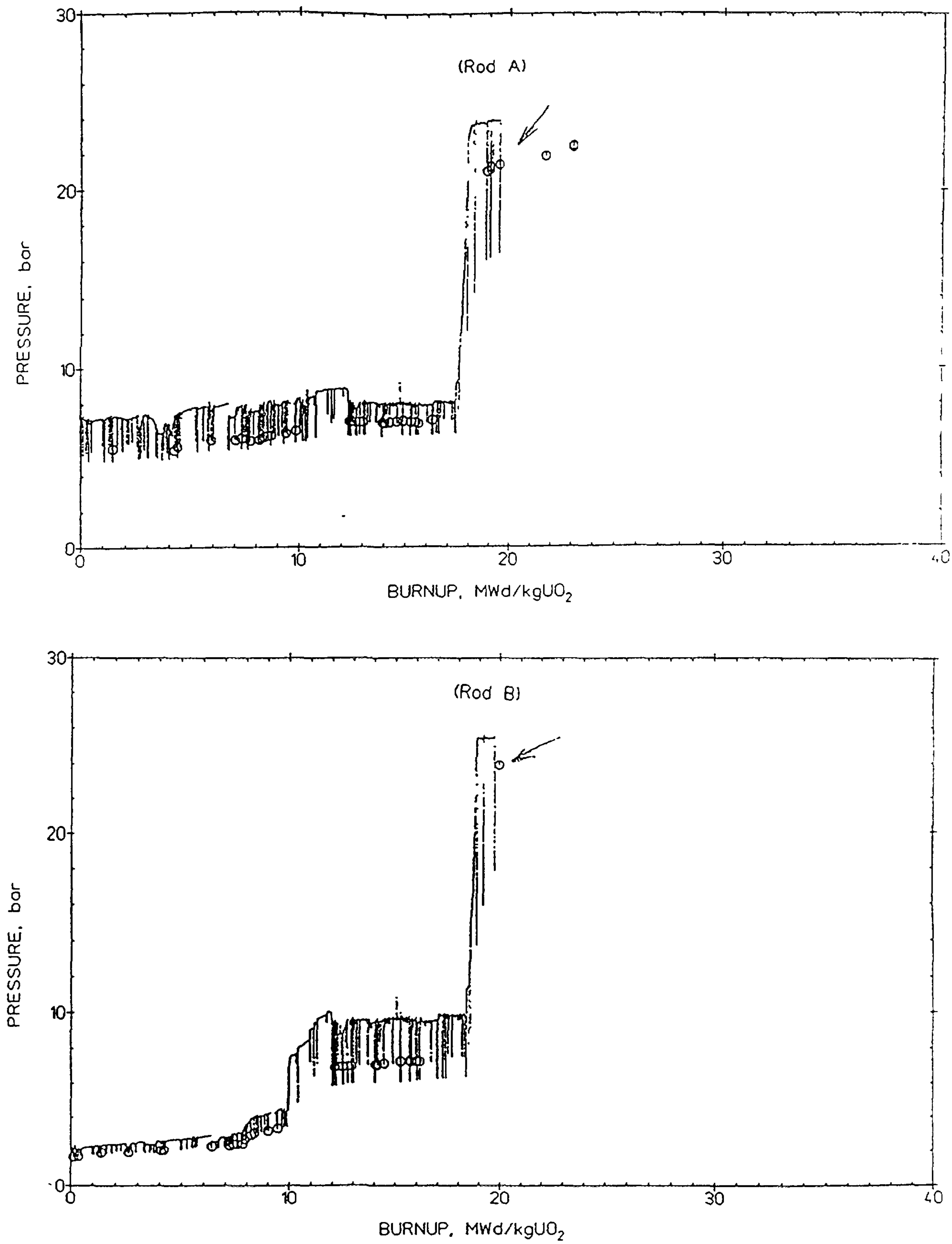


FIG. 23. Measured rod internal pressure for FUMEX 4 rods as a function of burnup. (a) rod A, (b) rod B.

Country	Code Name	Number	T Start up	T Ramp	T E O.L
		0			
ARG	BACO	1	1014	1150	1399
BUL	PIN-MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1059	1234	1353
FRA/EDF	TRANSURANUS/EDF	5	1103	1458	1654
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	1080	1281	1294
IND1	PROFESS	8	1197	1409	
IND2	FAIR	9	1138	1533	1557
IND3	FUDA-MOD1	10	1000	1005	1035
JPN1	TRUST-V1b	11	1398	1436	2246
JPN2	EIMUS	12	990	1331	1353
CHIN	FRAPCON2	13	1047	792	988
ROM	ROFEM 1B	14			
SWIT	TRANSURANUS	15	1002	1163	1165
CZE	PIN-W	16	876		1162
UK1	ENIGMA-5 2	17	1130	1330	1240
UK2	ENIGMA 5 8D	18	1071	1195	1355
RUS	START	19			
	EXP RESULT	20	1020	1125	1225

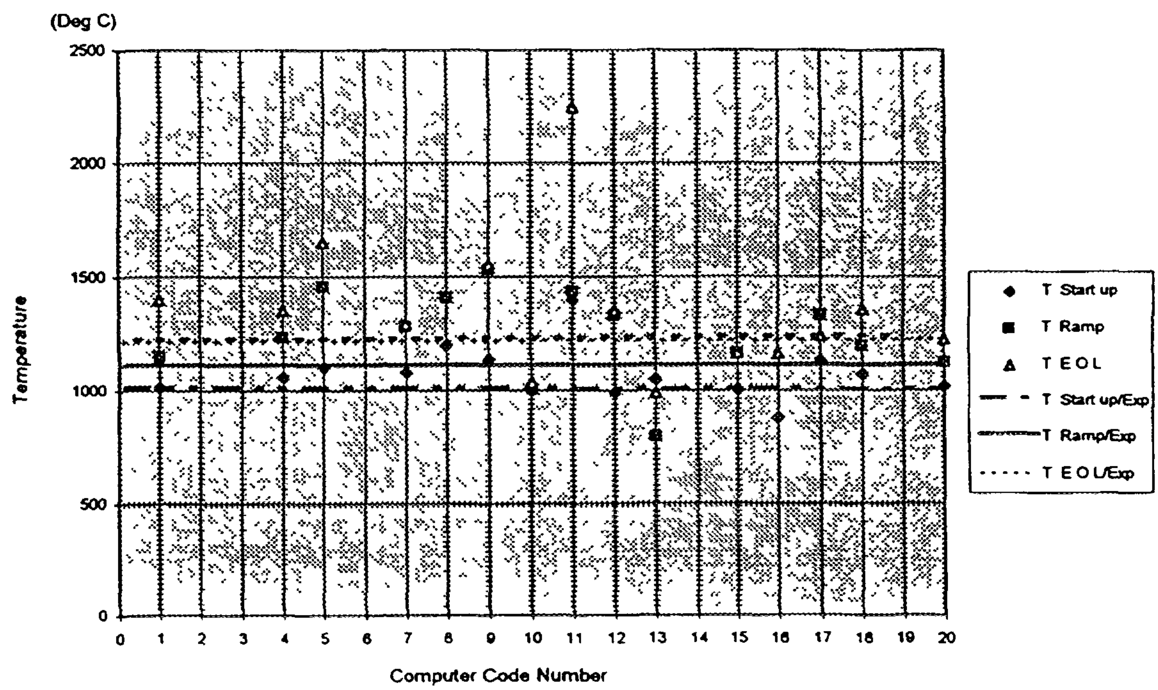


FIG. 24. Comparison of predictions and data for FUMEX 4 rod A fuel centreline temperatures at 30 kW/m: on start-up, during the power ramp and at end-of-life.

Country	Code Name	Number	Before Ramp	After Ramp	E.O.L
ARG	BACO	1	4 8	10 4	40 9
BUL	PIN MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1 5	19 8	36 2
FRA/EDF	TRANSURANUS/EDF	5	3	5 5	23 5
FRA/CEA	METEOR/TRANSURANUSU	6			
CEC	TRANSURANUS	7	4 4	26 1	47 7
IND1	PROFESS	8	8 4	22 6	
IND2	FAIR	9	10 1	15 9	40 7
IND3	FUDA-MOD1	10	0 3	15 9	26 5
JPN1	*TRUST-V1b*	*11*	*1 9*	33	45 7
JPN2	EIMUS	12	10 6	28 4	45 1
CHIN	FRAPCON2	13	23	3 1	15 4
ROM	ROFEM 1B	14			
SWIT	TRANSURANUS	15	3 2	12 2	32 5
CZE	PIN W	16	7	0 7	29 3
UK1	ENIGMA 5 2	17	4 6	26 1	53 8
UK2	ENIGMA-5 8D	18	0 7	20 7	43
RUS	START	19			

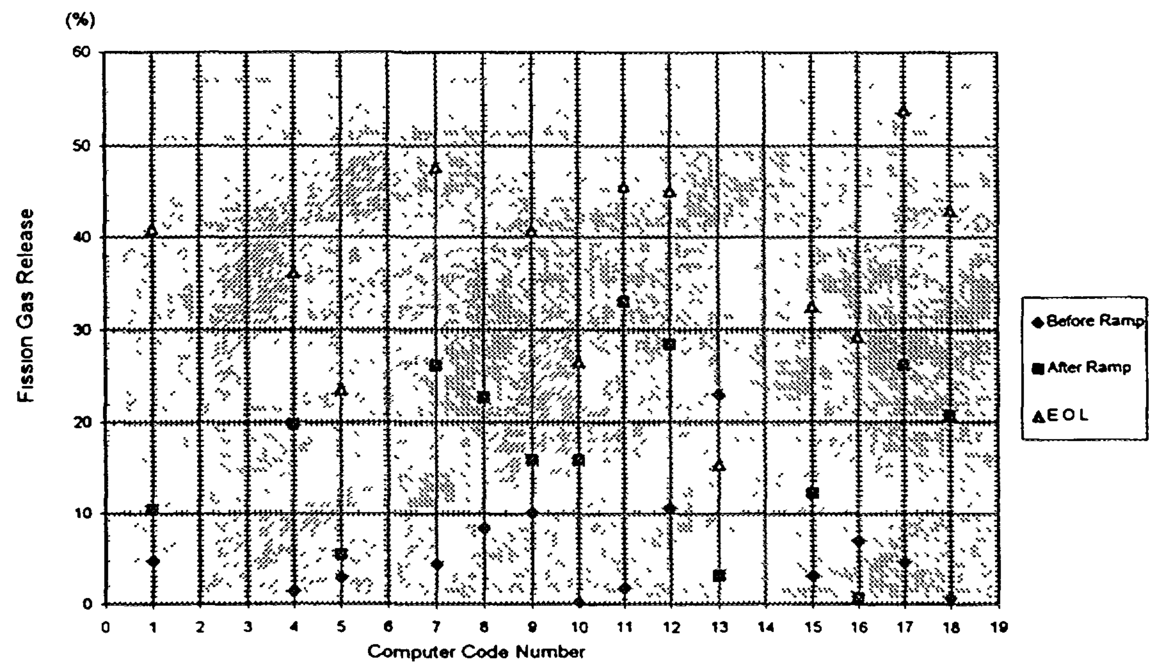


FIG. 25. Comparison of code predictions for FUMEX 4 rod A fission gas release before, after the power ramp and at end-of-life.

Country	Code Name	Numb.	Pressure (bar)
		0	
ARG	BACO	1	18.6
BUL	PIN-MICRO	2	
CAN	ELESIM	3	
FIN	VTT/ENIGMA	4	27.3
FRA/EDF	TRANSURANUS/EDF	5	10.4
FRA/CEA	METEOR/TRANSURANUS	6	
CEC	TRANSURANUS	7	30.4
IND1	PROFESS	8	
IND2	FAIR	9	22.7
IND3	FUDA-MOD1	10	20.5
JPN1	TRUST-V1b	11	40
JPN2	EIMUS	12	45.7
CHIN	FRAPCON2	13	9.2
ROM	ROFEM-1B	14	
SWIT	TRANSURANUS	15	21.25
CZE	PIN-W	16	4.8
UK1	ENIGMA-5.2	17	25
UK2	ENIGMA-5.8D	18	21.2
RUS	START	19	
	EXPERIMENTAL RESULT		21

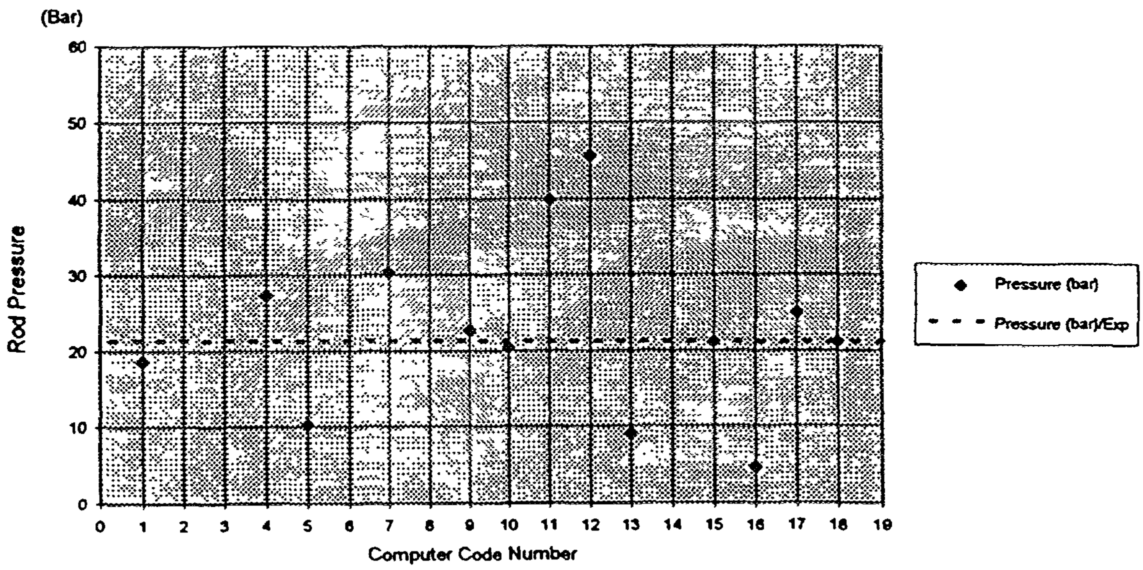


FIG. 26. Comparison of predictions and data for FUMEX 4 rod A rod internal pressure at power after the power ramp.

Country	Code Name	Number	Start-up	Top of Ramp	E.O.L
		0			
ARG	BACO	1	953	1200	1473
BUL	PIN-MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1125	1281	1350
FRA/EDF	TRANSURANUS/EDF	5	1165	1634	1634
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	1169	1512	1325
IND1	PROFESS	8			
IND2	FAIR	9	1244	1462	1427
IND3	FUDA-MOD1	10	1465	1465	1465
JPN1	TRUST-V1b	11	1522	1522	2203
JPN2	EIMUS	12	991	1593	1634
CHIN	FRAPCON2	13	1178	1283	1315
ROM	ROFEM-1B	14			
SWIT	TRANSURANUS	15	1052	1275	1213
CZE	PIN-W	16	953		
UK1	ENIGMA-5 2	17	1071	1435	1330
UK2	ENIGMA-5 8D	18	1158	1281	1355
RUS	START	19			
	EXP. RESULT	20	1065	1260	1290

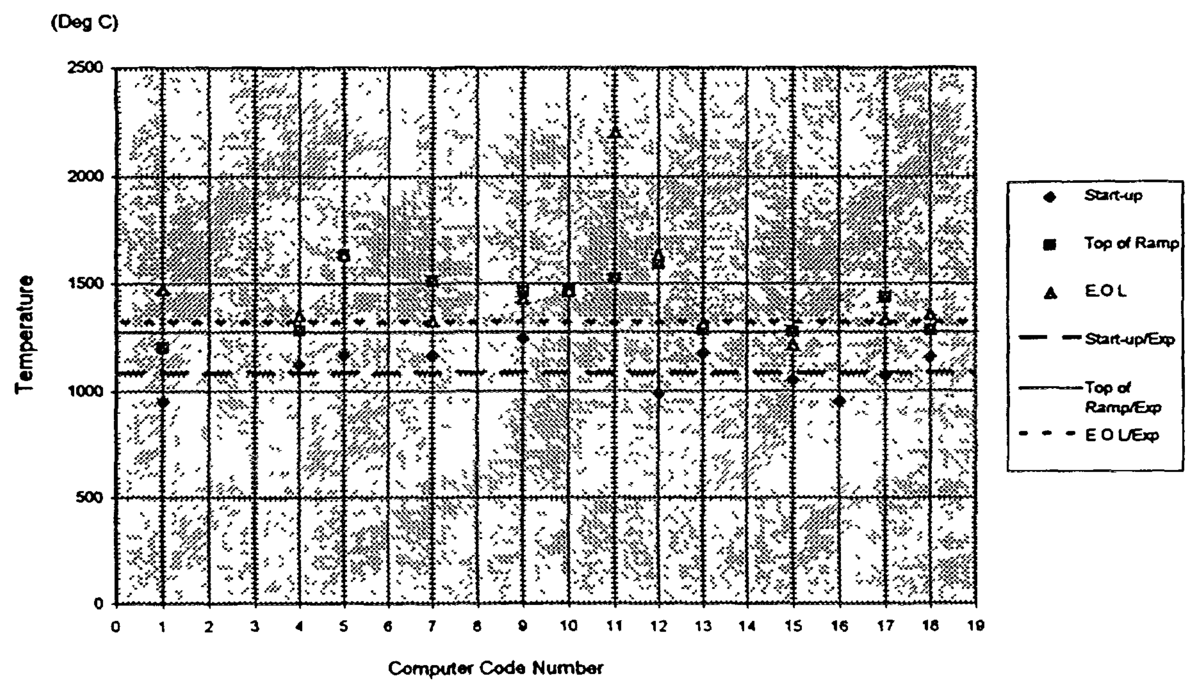


FIG. 27. Comparison of predictions and data for FUMEX 4 rod B fuel centreline temperatures at 30 kW/m for the top of the ramp and at end-of-life.

Country	Code Name	Number	Before Ramp	After Ramp	E.O.L
ARG	BACO	1	5 9	13 3	34 1
BUL	PIN-MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1.5	18 7	37 4
FRA/EDF	TRANSURANUS/EDF	5	5 5	7 8	27 5
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	24.6	36 2	54 6
IND1	PROFESS	8			
IND2	FAIR	9	6 1	10.9	43
IND3	FUDA-MOD1	10	15.8	38.9	47 5
JPN1	TRUST-V1b	11	6.9	28.5	41
JPN2	EIMUS	12	10	28.6	44.2
CHIN	FRAPCON2	13	11 5	18	40
ROM	ROFEM-1B	14			
SWIT	TRANSURANUS	15	5.3	13 6	34 7
CZE	PIN-W	16	1 2	2 7	39 3
UK1	ENIGMA-5 2	17	7 7	25 4	50
UK2	ENIGMA-5 8D	18	1.5	21 5	43
RUS	START	19			
	EXP RESULT	20			

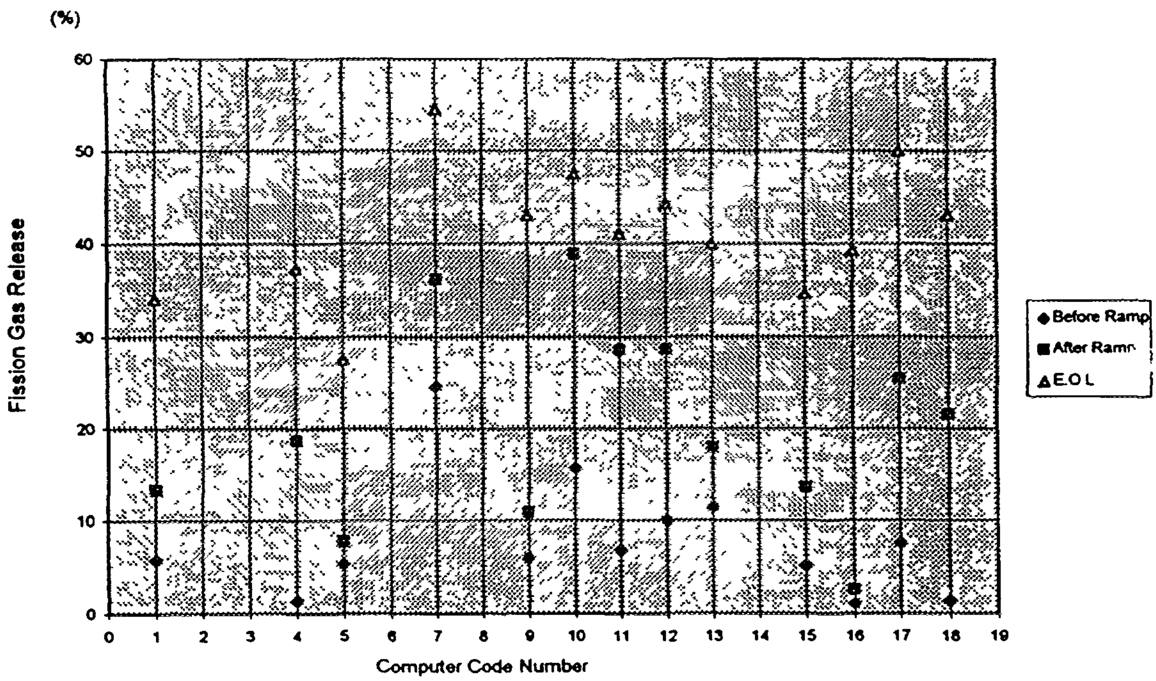


FIG. 28. Comparison of code predictions for FUMEX 4 rod B fission gas release before, after the power ramp and at end-of-life.

Country	Code Name	Numb	Rod Pressure (bar)
		0	
ARG	BACO	1	13 3
BUL	PIN-MICRO	2	
CAN	ELESIM	3	
FIN	VTT/ENIGMA	4	25 6
FRA/EDF	TRANSURANUS/EDF	5	9 3
FRA/CEA	METEOR/TRANSURANUS	6	
CEC	TRANSURANUS	7	37 5
IND1	PROFESS	8	
IND2	FAIR	9	16 7
IND3	FUDA-MOD1	10	41
JPN1	TRUST-V1b	11	34
JPN2	EIMUS	12	45 3
CHIN	FRAPCON2	13	15
ROM	ROFEM-1B	14	
SWIT	TRANSURANUS	15	19 3
CZE	PIN-W	16	3
UK1	ENIGMA-5.2	17	22
UK2	ENIGMA-5.8D	18	19
RUS	START	19	
	EXP. RESULT	20	23 9

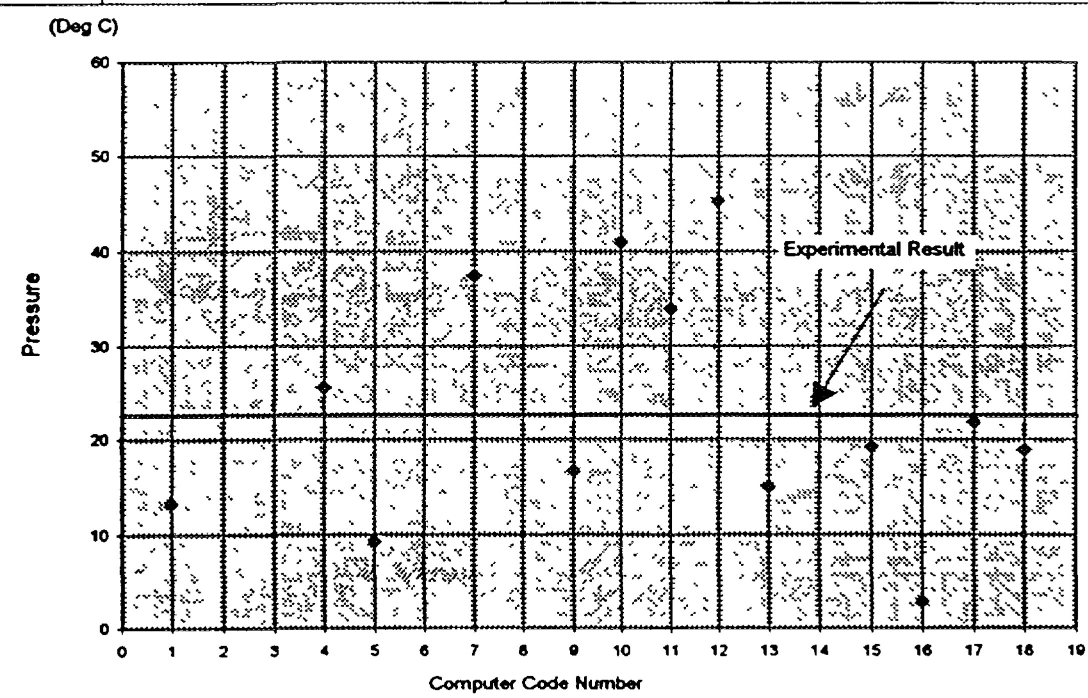


FIG. 29. Comparison of predictions and data for FUMEX 4 rod B rod internal pressure at hot standby after the power ramp.

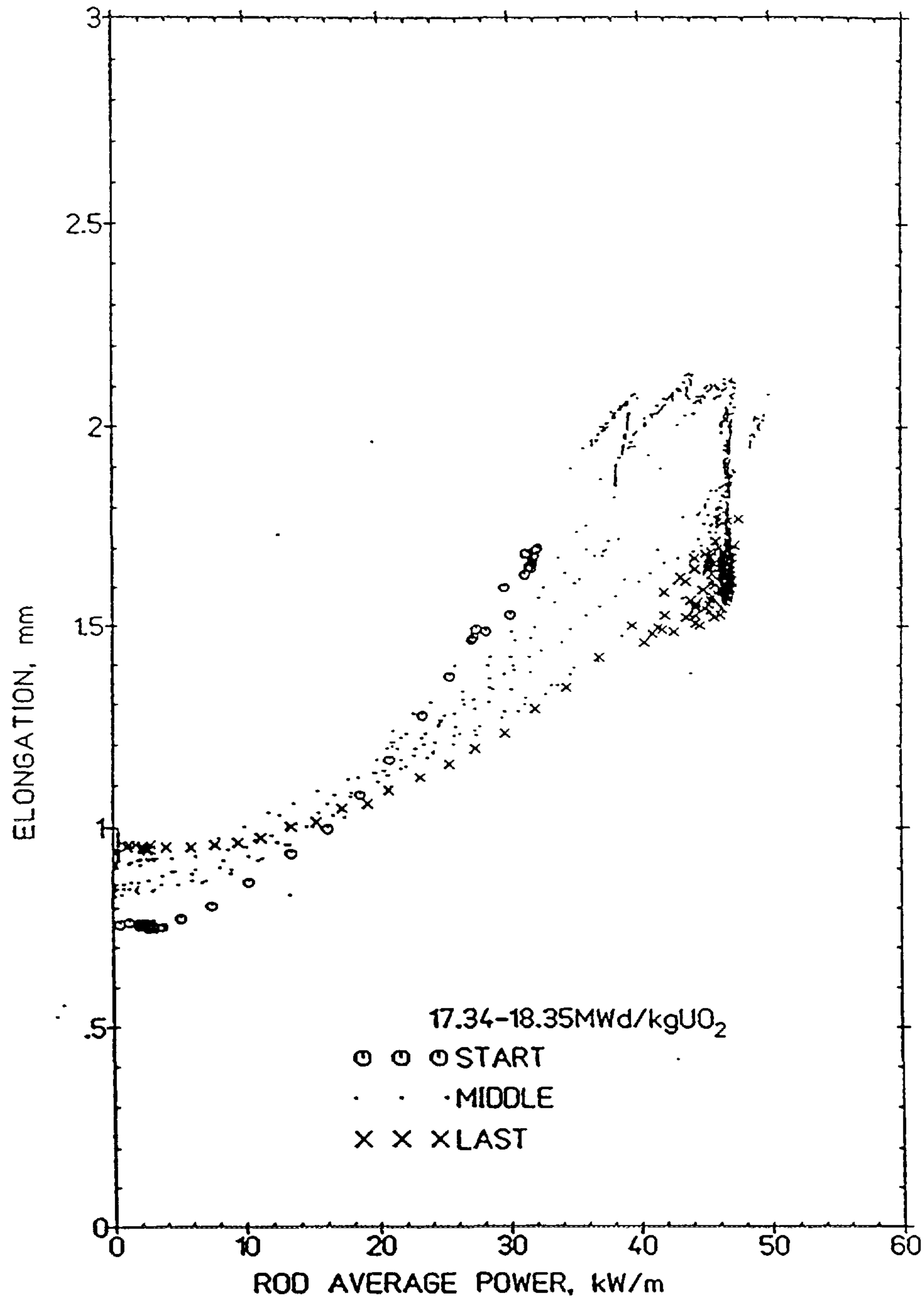


FIG. 30. FUMEX 4 rod A elongation measurements as a function of rod average power during ramping between 17.34 and 18.35 MW·d/kgUO$_2$.

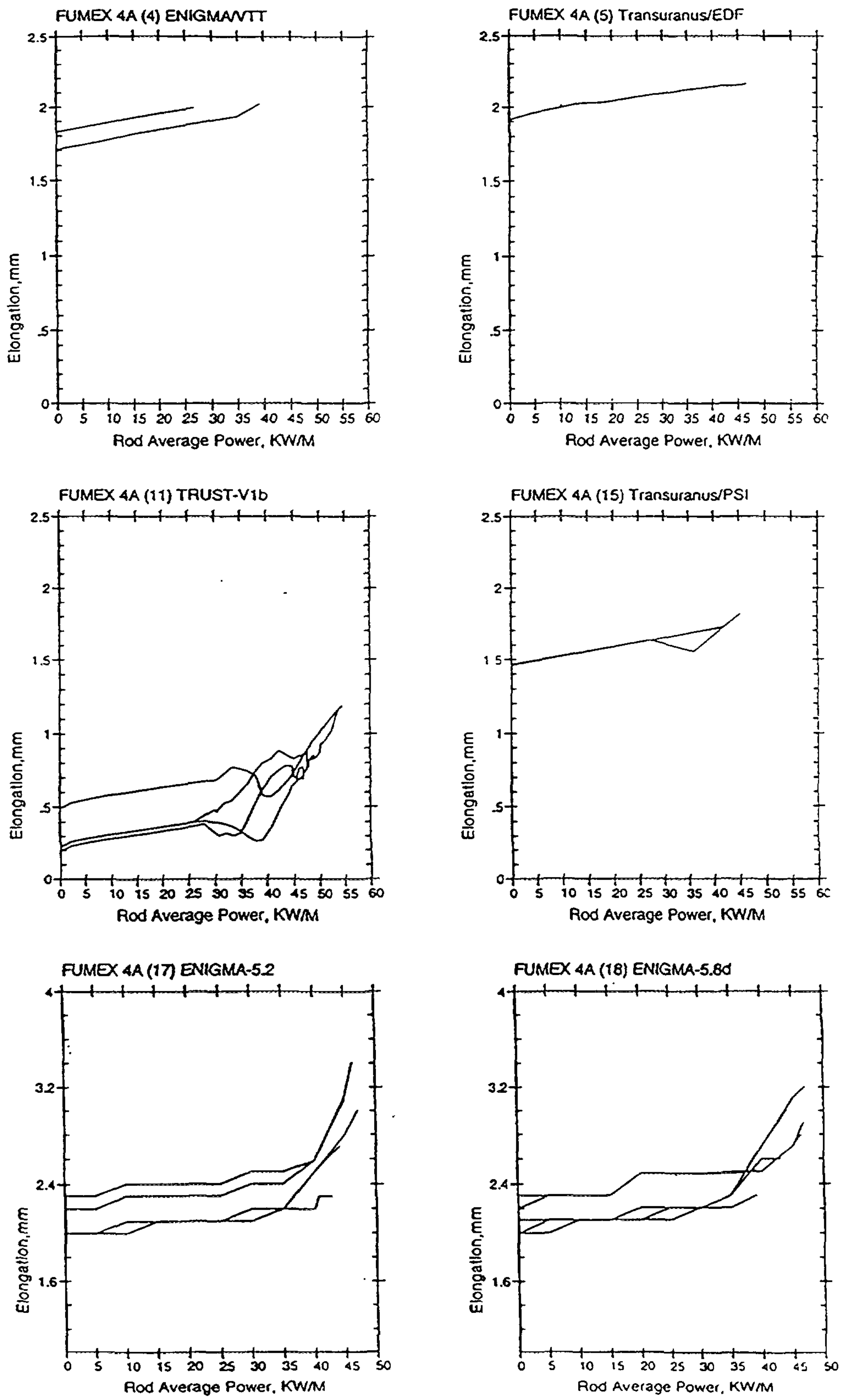


FIG. 31. Code predictions of rod elongation for FUMEX 4 rod A as a function of rod average power during ramping between 17.34 and 18.35 MW·d/kgUO$_2$.

Country	Code Name	Number	P at Start up	P at E O L
ARG	BACO	1	3 54	6 64
BUL	PIN-MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	2 74	14 3
FRA/EDF	TRANSURANUS/EDF	5	2 46	5 69
FRA/CEA	METEOR/TRANSURANUS	6	2 39	21 7
CEC	TRANSURANUS	7	2 33	27 4
IND1	PROFESS	8	18 6	27 2
IND2	FAIR	9	66 6	82 6
IND3	FUDA-MOD1	10	10 5	9
JPN1	TRUST-V1b	11	3 6	41 8
JPN2	EIMUS	12	11 8	16 5
CHIN	FRAPCON2	13	3 7	4 6
ROM	ROFEM-1B	14	13 8	39 7
SWIT	TRANSURANUS	15	2 8	3 9
CZE	PIN W	16		
UK1	ENIGMA-5 2	17	2 7	16
UK2	ENIGMA 5.8D	18	2 8	15 4
RUS	START	19	9 8	21 5
	EXP. RESULT		2 3	9 4

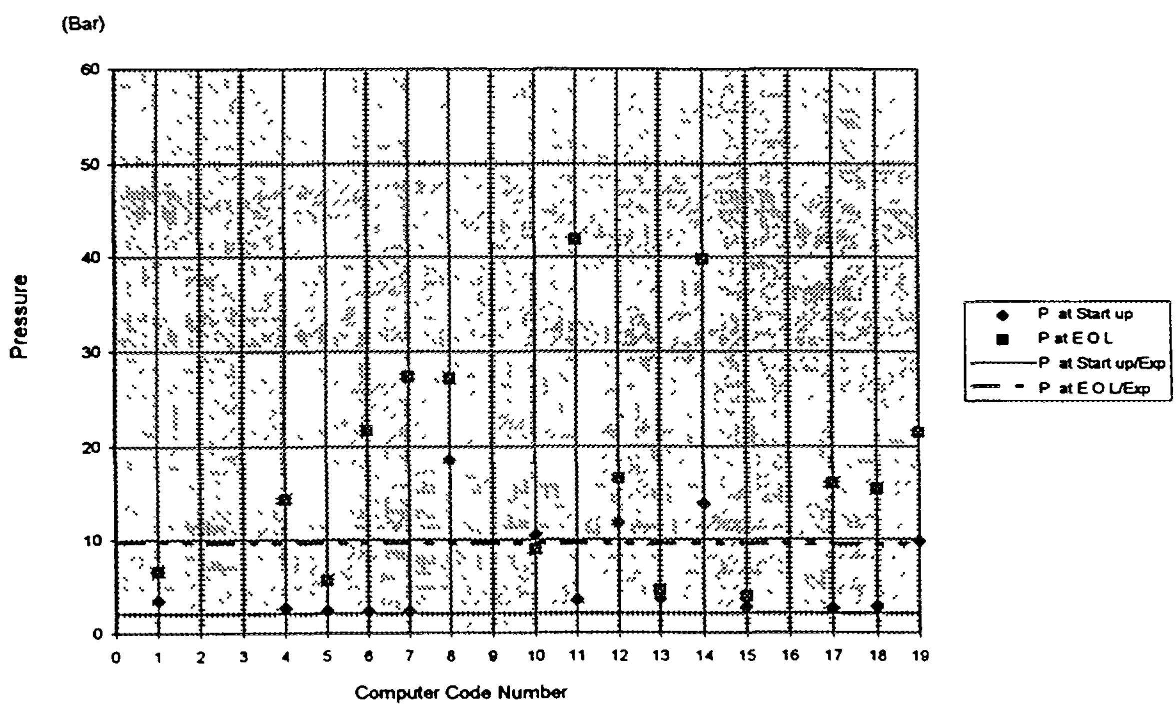


FIG. 32. Comparison of predictions and data for FUMEX 5 rod internal pressure at power for start-up and end-of-life.

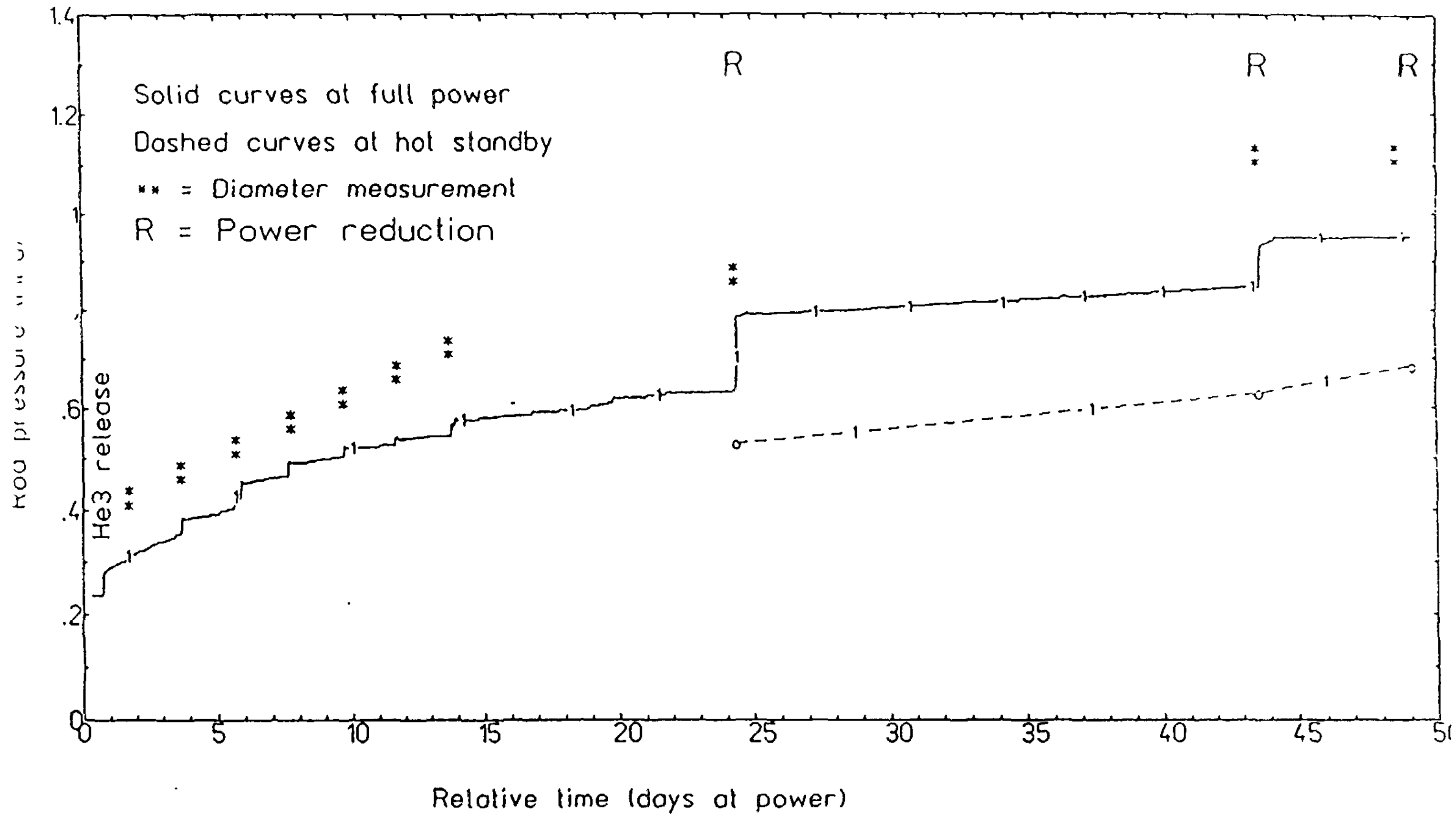


FIG. 33. Evolution of rod internal pressure as measured for FUMEX 5 during the period of high power at the end-of-life.

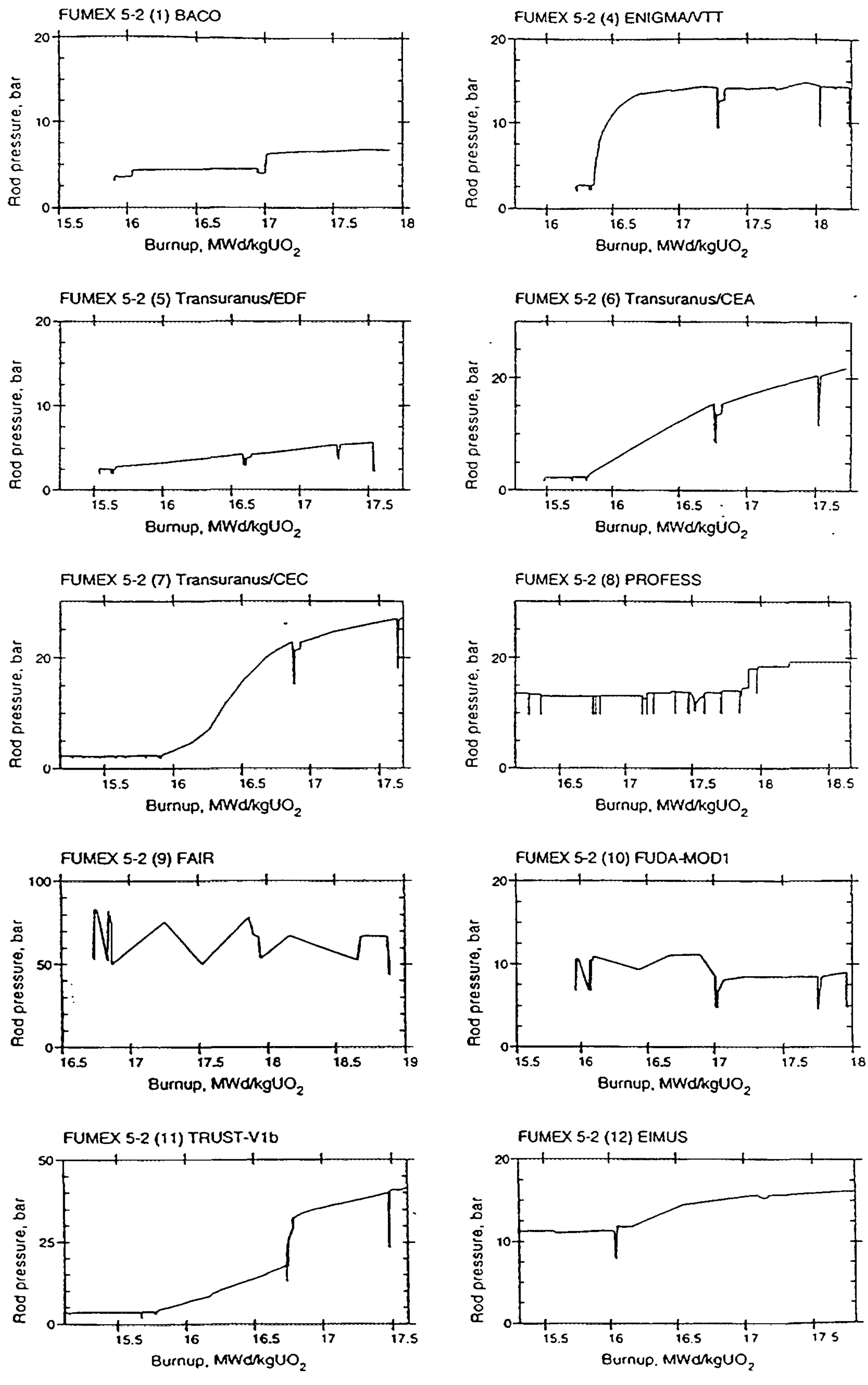
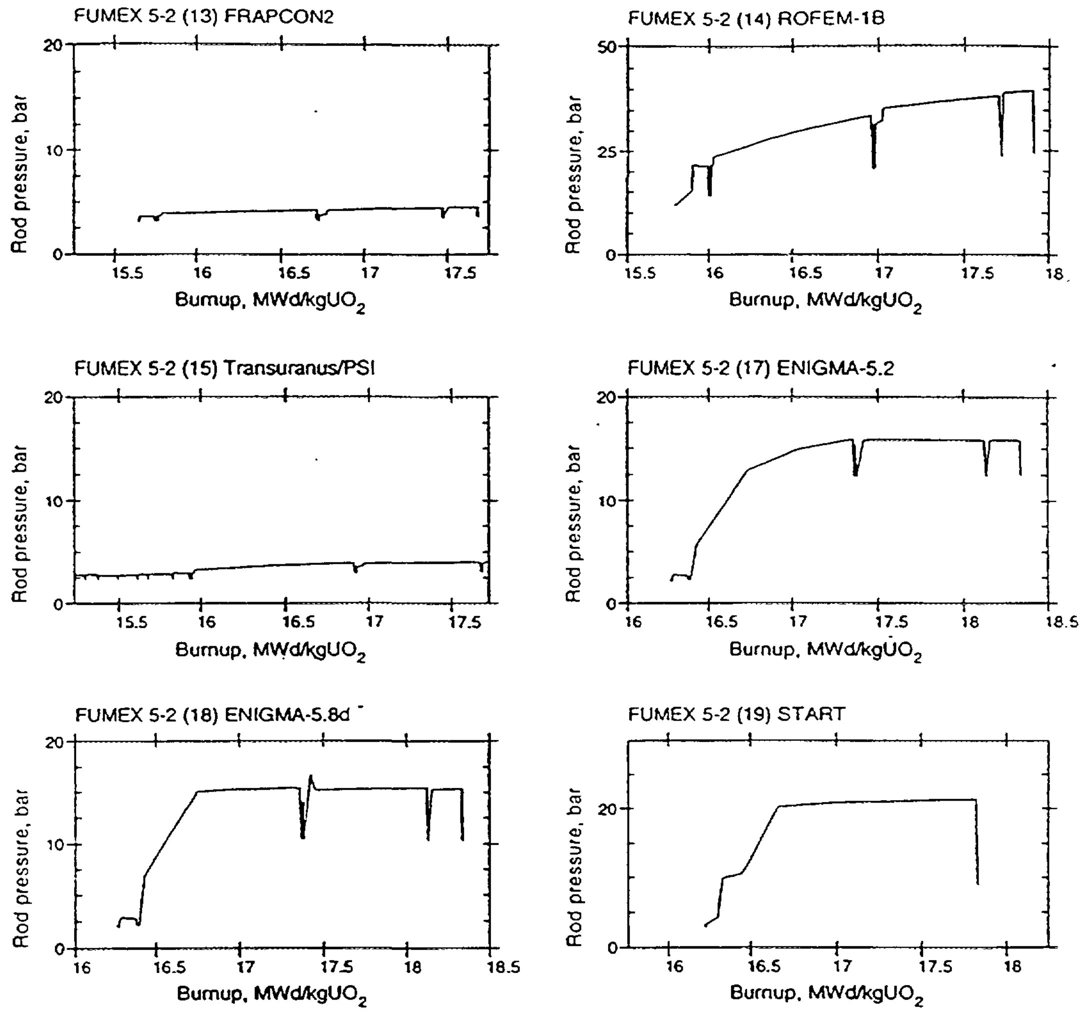


FIG. 34. Code predictions of the evolution of rod internal pressure for FUMEX 5 during the period of high power at the end-of-life.



FUMEX 5-2 (13) FRAPCON2
Rod pressure, bar
Burnup, MWd/kgUO2
FUMEX 5-2 (14) ROFEM-1B
Rod pressure, bar
Burnup, MWd/kgUO2
FUMEX 5-2 (15) Transuranus/PSI
Rod pressure, bar
Burnup, MWd/kgUO2
FUMEX 5-2 (17) ENIGMA-5.2
Rod pressure, bar
Burnup, MWd/kgUO2
FUMEX 5-2 (18) ENIGMA-5.8d
Rod pressure, bar
Burnup, MWd/kgUO2
FUMEX 5-2 (19) START
Rod pressure, bar
Burnup, MWd/kgUO2

FIG. 34. (cont.).

Country	Code Name	Number	Start of Ramp	E.O.L
		0		
ARG	BACO	1	1 2	2.9
BUL	PIN-MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	2	6
FRA/EDF	TRANSURANUS/EDF	5	0 1	2
FRA/CEA	METEOR/TRANSURANUS	6	0 07	10 5
CEC	TRANSURANUS	7	0 1	14 3
IND1	PROFESS	8	9 1	12 7
IND2	FAIR	9	43 1	46 8
IND3	FUDA-MOD1	10	0	0 05
JPN1	TRUST-V1b	11	0 7	19 6
JPN2	EIMUS	12	16 3	21 7
CHIN	FRAPCON2	13	0.7	1
ROM	ROFEM-1B	14	12 9	21 3
SWIT	TRANSURANUS	15	0 25	0 77
CZE	PIN W	16		
UK1	ENIGMA-5 2	17	1 3	8 5
UK2	ENIGMA-5.8D	18	0 2	9 4
RUS	START	19	1 7	11 3
	Experimental Result	20	0	5 8

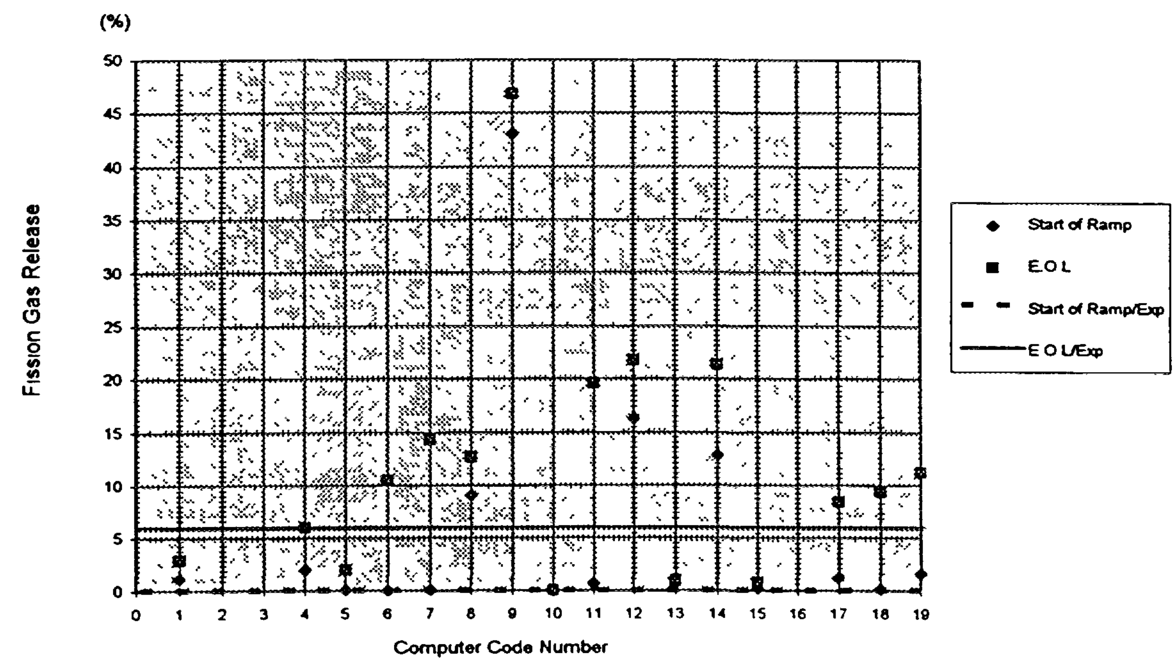


FIG. 35. Comparison of predictions and data for FUMEX 5: fission gas release before the period of high power and at end-of-life.

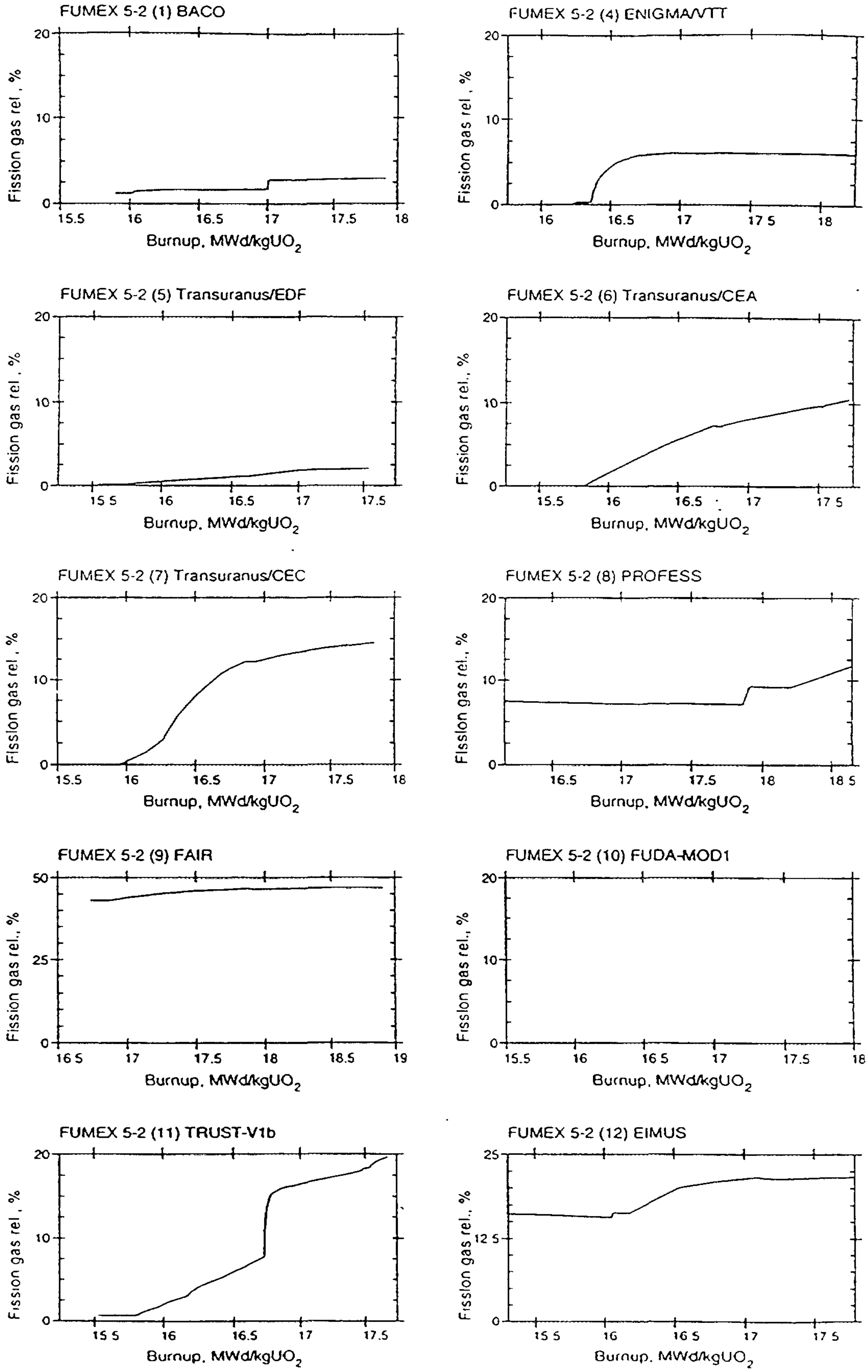


FIG. 36. Code predictions of the evolution of fission gas release for FUMEX 5 during the period of high power at the end-of-life.

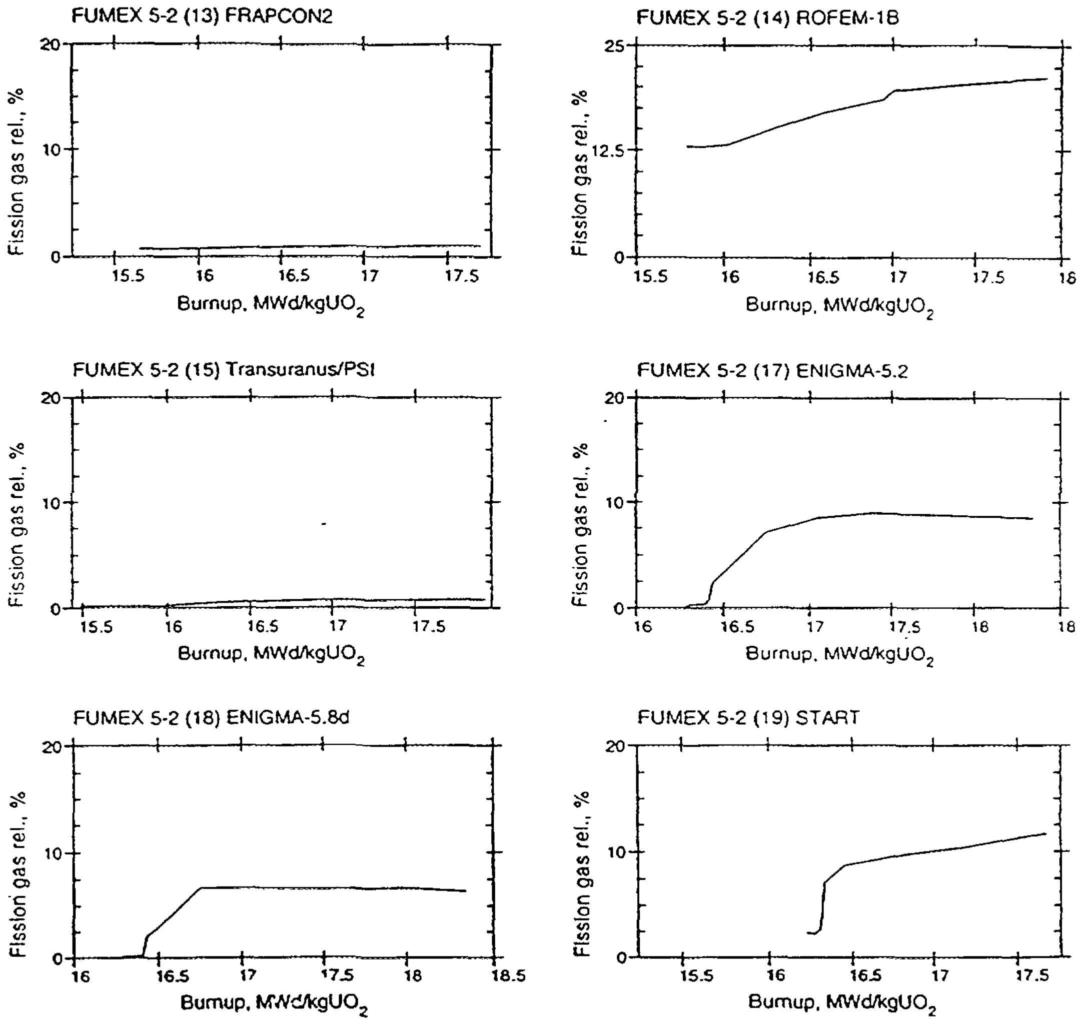

FUMEX 5-2 (13) FRAPCON2
FUMEX 5-2 (14) ROFEM-1B
FUMEX 5-2 (15) Transuranus/PSI
FUMEX 5-2 (17) ENIGMA-5.2
FUMEX 5-2 (18) ENIGMA-5.8d
FUMEX 5-2 (19) START
Fission gas rel., %
Burnup, MWd/kgUO2

FIG. 36. (cont.).

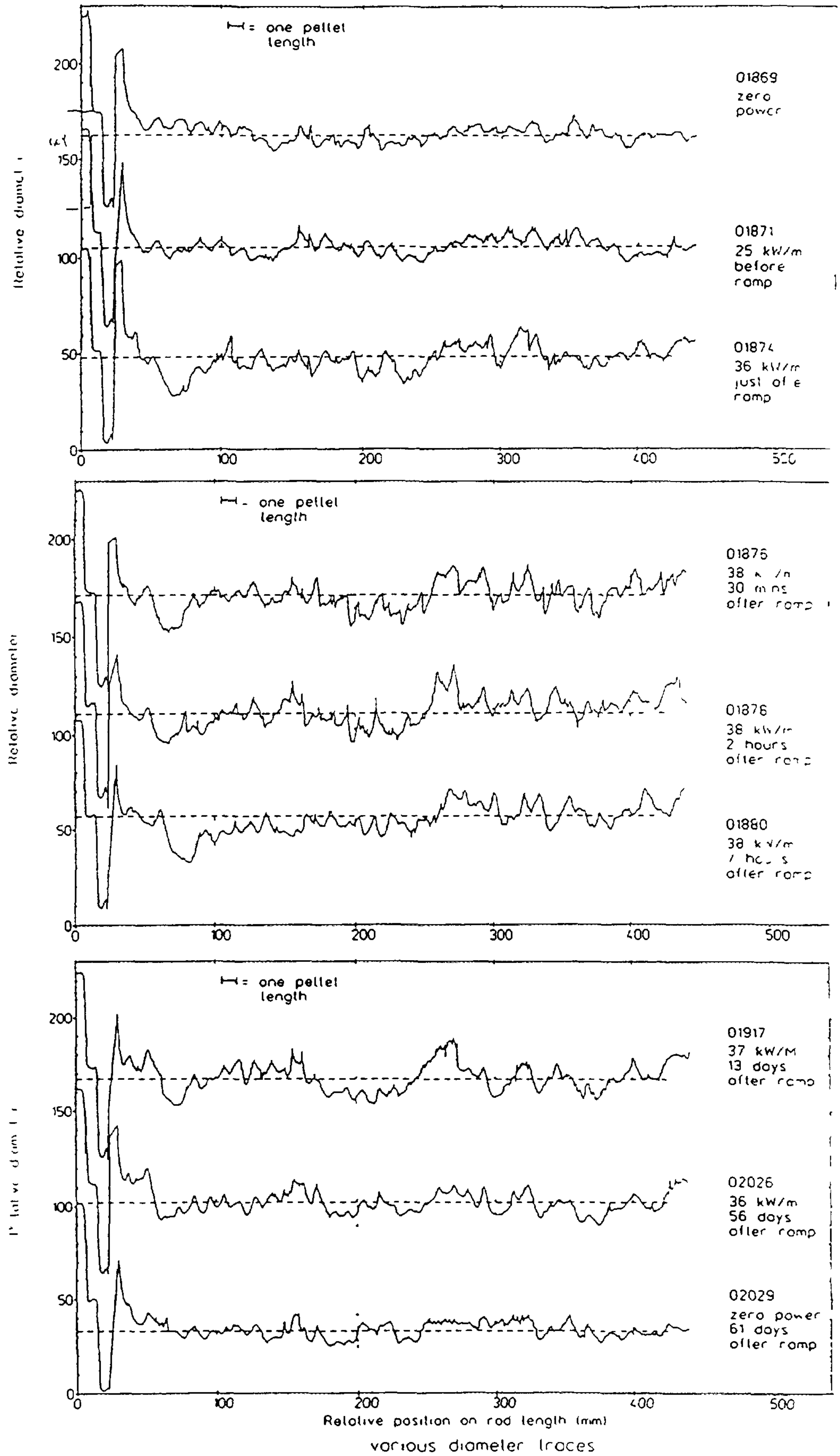


FIG. 37. Measurements of rod diameter along FUMEX 5 at different times before and during the period of high power at the end-of-life. Note the three calibration marks at 50 μm spacing at the extreme left hand side of each trace; diameters and diameter changes are measured relative to these fixed points.

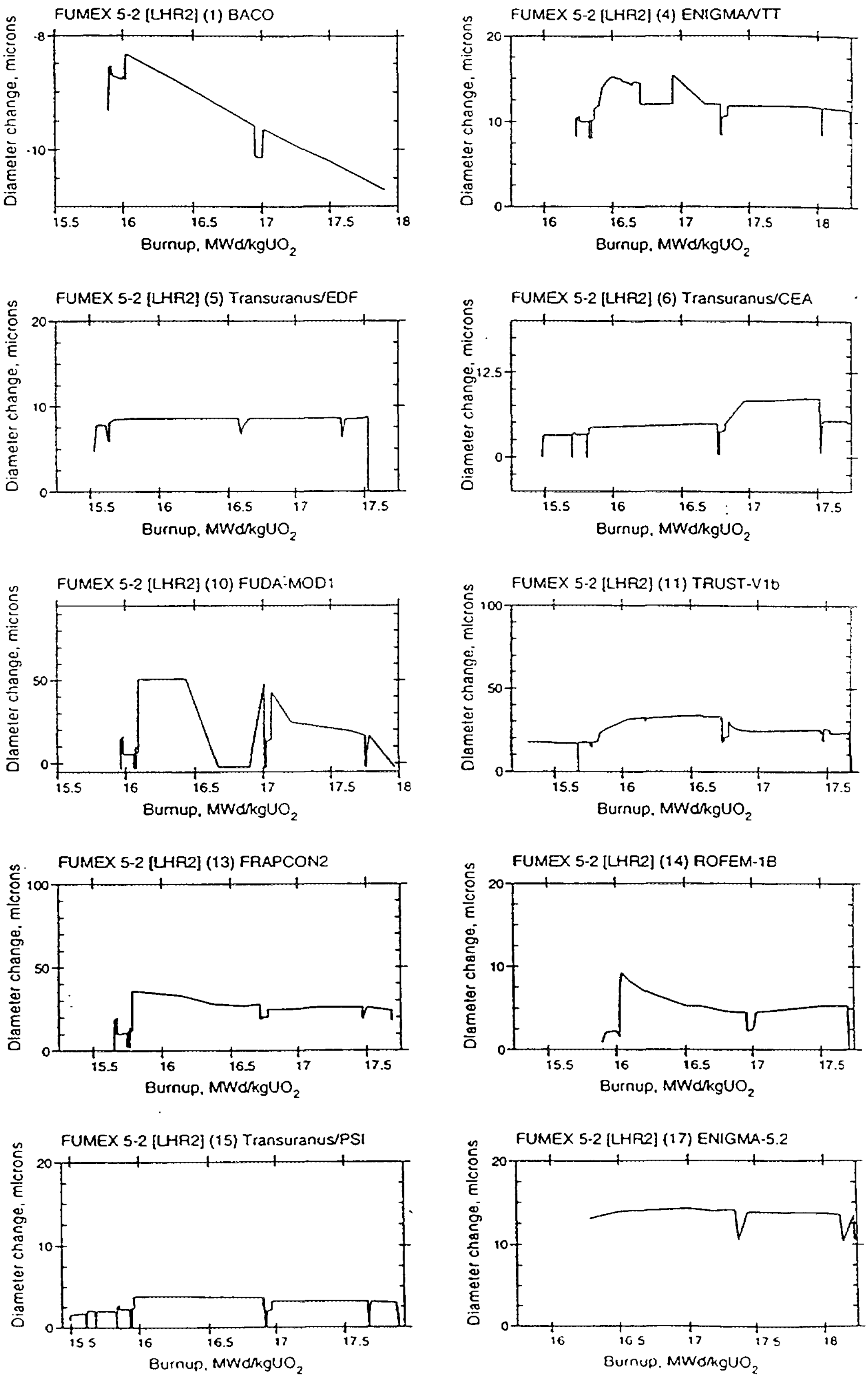
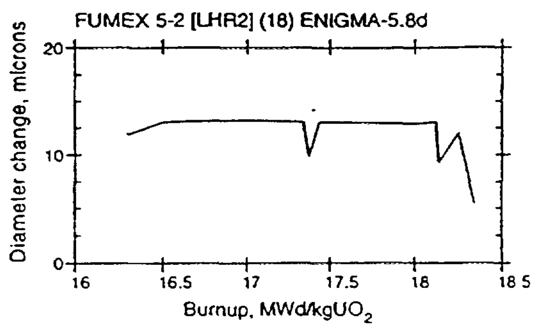
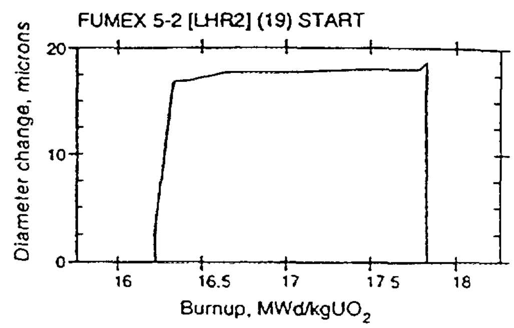


FIG. 38. Code predictions of diameter change relative to the start of ramp dimensions for axial position LHR2 on the rod irradiated as FUMEX 5.



FUMEX 5-2 [LHR2] (18) ENIGMA-5.8d
Diameter change, microns
20
10
0
16
16.5
17
17.5
18
18.5
Burnup, MWd/kgUO2



FUMEX 5-2 [LHR2] (19) START
Diameter change, microns
20
10
0
16
16.5
17
17.5
18
Burnup, MWd/kgUO2

FIG. 38. (cont.).

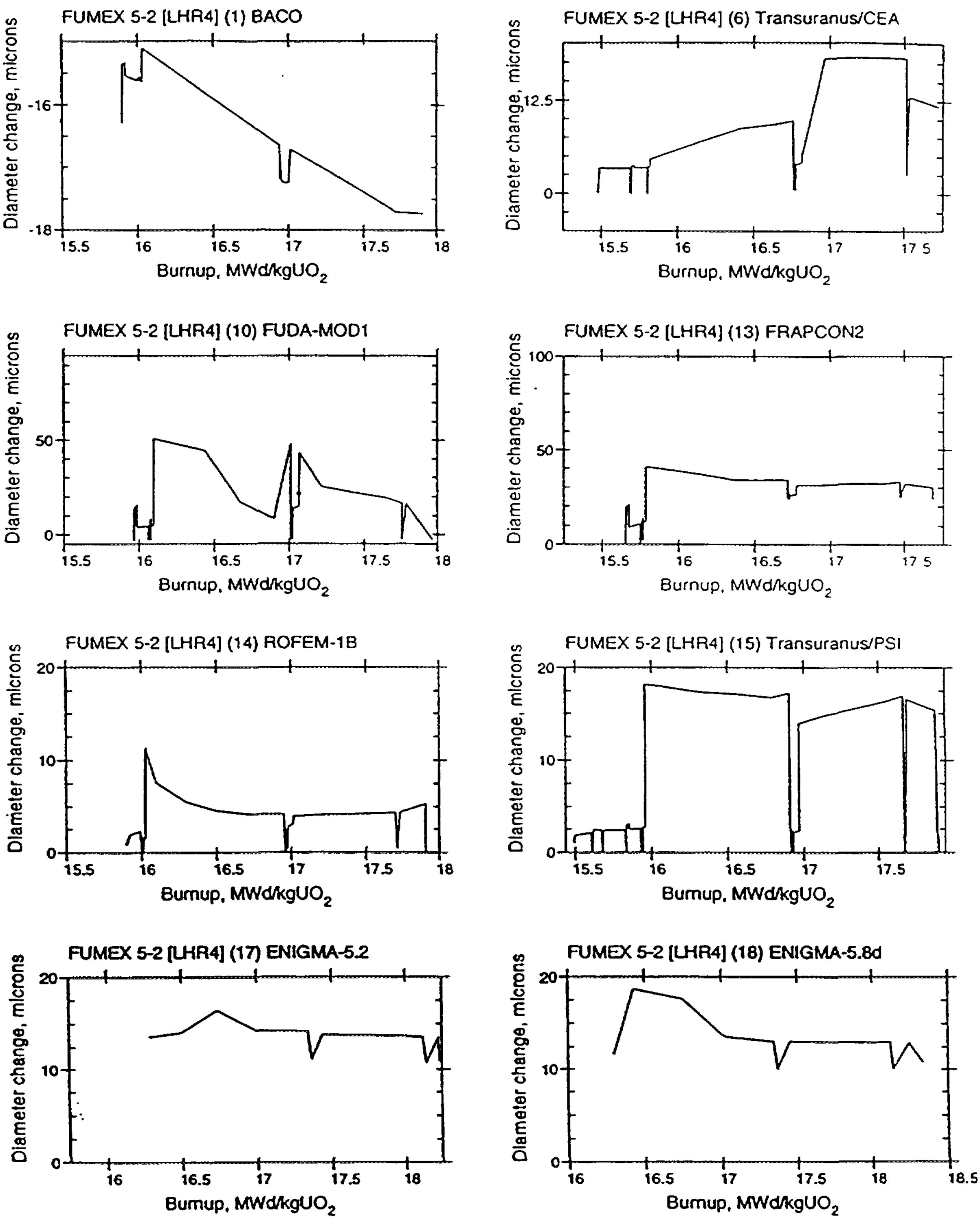


FIG. 39. Code predictions of diameter change relative to the start of ramp dimensions for axial position LHR4 on the rod irradiated as FUMEX 5.

Country	Code Name	Number	Start (Max Overp period)	E.O.L
		0		
ARG	BACO	1		
BUL	PIN-MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	6 4	2
FRA/EDF	TRANSURANUS/EDF	5	2 3	2 3
FRA/CEA	METEOR/TRANSURANUS	6	8 4	4 6
CEC	TRANSURANUS	7		
IND1	PROFESS	8		
IND2	FAIR	9		
IND3	FUDA-MOD1	10	47 7	17
JPN1	TRUST-V1b	11	13	5 3
JPN2	EIMUS	12		
CHIN	FRAPCON2	13	16 1	5 8
ROM	ROFEM 1B	14	9 7	5 4
SWIT	TRANSURANUS	15	3 8	3
CZE	PIN W	16		
UK1	ENIGMA-5 2	17	3	3
UK2	ENIGMA-5 8D	18	3	3 4
RUS	START	19	17 5	17.5
	Experimental Result	20	11	2

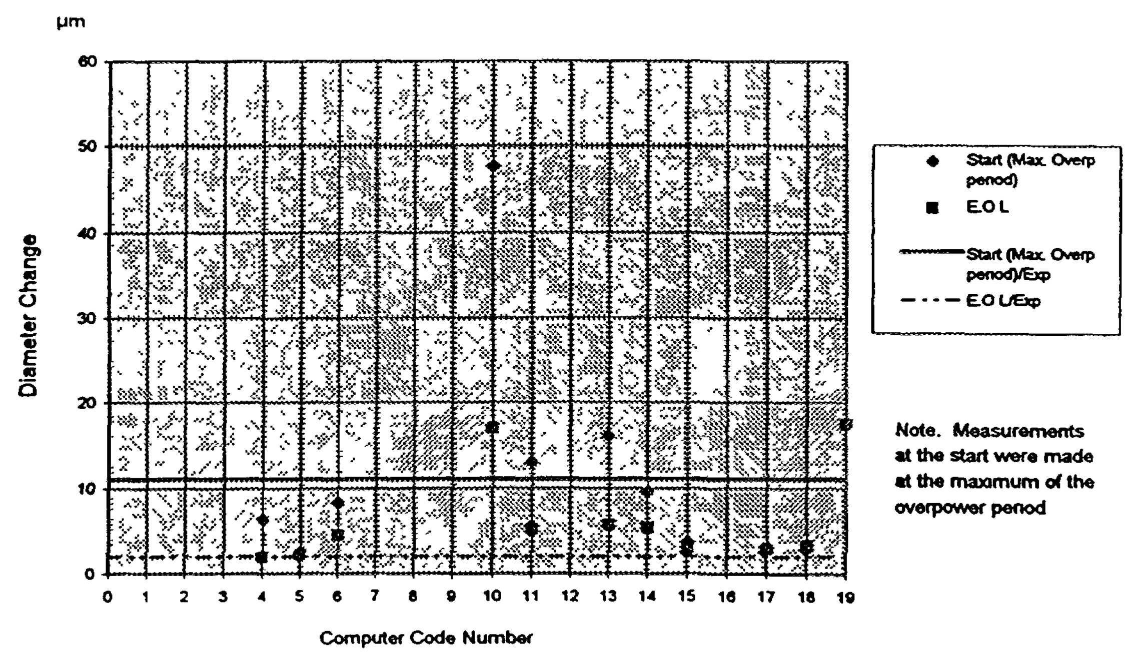


FIG. 40. Comparison of predictions and data for FUMEX 5 clad diameter change at position LHR2 immediately following increased power and at the end of the high power period.

Country	Code Name	Number	Start (Max Overp period)	E.O.L
		0		
ARG	BACO	1		
BUL	PIN MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	6 4	2
FRA/EDF	TRANSURANUS/EDF	5	2 3	2 3
FRA/CEA	METEOR/TRANSURANUS	6	9 5	17.9
CEC	TRANSURANUS	7		
IND1	PROFESS	8		
IND2	FAIR	9		
IND3	FUDA-MOD1	10	47 7	17
JPN1	TRUST-V1b	11	16	5 3
JPN2	EIMUS	12		
CHIN	FRAPCON2	13	20	13 1
ROM	ROFEM 1B	14	11	5 3
SWIT	TRANSURANUS	15	18 3	15 2
CZE	PIN W	16		
UK1	ENIGMA-5 2	17	5 3	3
UK2	ENIGMA-5 8D	18	7.6	3 4
RUS	START	19	17.5	17.5
	Experimental Result	20	11	2

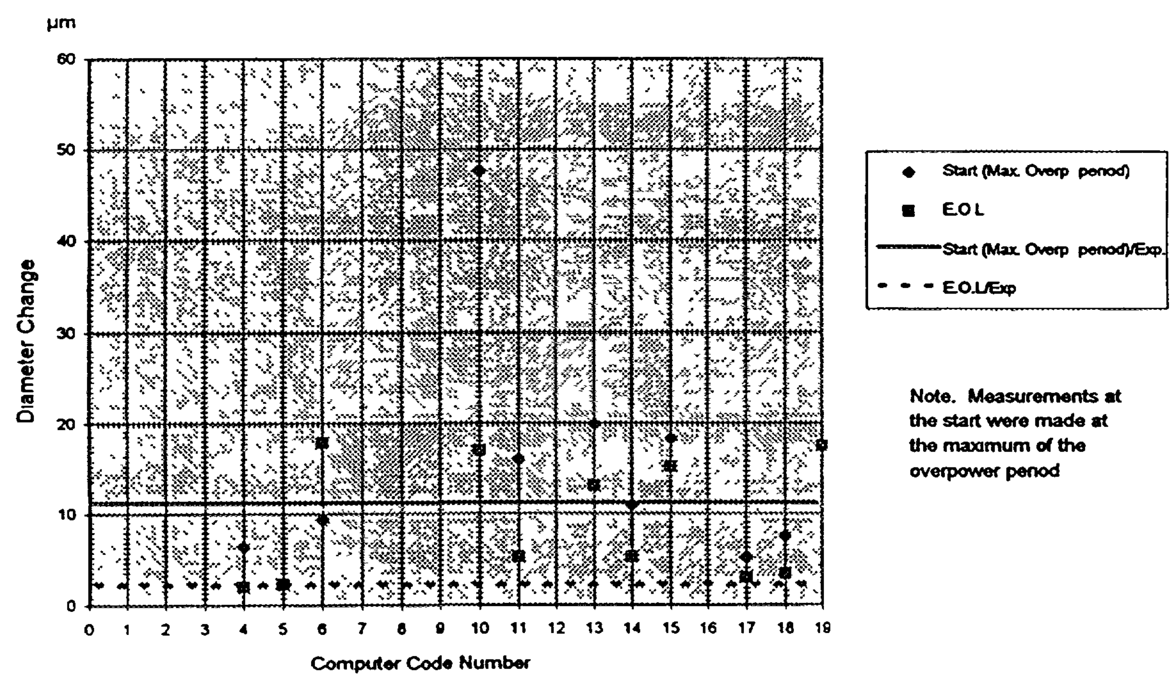


FIG. 41. Comparison of predictions and data for FUMEX 5 clad diameter change at position LHR4 immediately following increased power and at the end of the high power period.

Country	Code Name	Number	d Pressure End Base Irradiati	Rod Pressure - E.O.L
		0		
ARG	BACO	1	13 1	
BUL	PIN-MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	17	77 6
FRA/EDF	TRANSURANUS/EDF	5	12.2	
FRA/CEA	METEOR/TRANSURANUS	6		
CEC	TRANSURANUS	7	20 1	74 2
IND1	PROFESS	8		
IND2	FAIR	9	7 4	59 6
IND3	FUDA-MOD1	10	79 5	
JPN1	TRUST-V1b	11	25 4	106.7
JPN2	EIMUS	12	19 8	40 4
CHIN	FRAPCON2	13	7.6	65 4
ROM	ROFEM-1B	14		
SWIT	TRANSURANUS	15	10 8	69 5
CZE	PIN W	16	2	
UK1	ENIGMA-5 2	17	16 8	84 5
UK2	ENIGMA-5 8D	18	15	83
RUS	START	19		
	EXP. RESULT	20		92 3

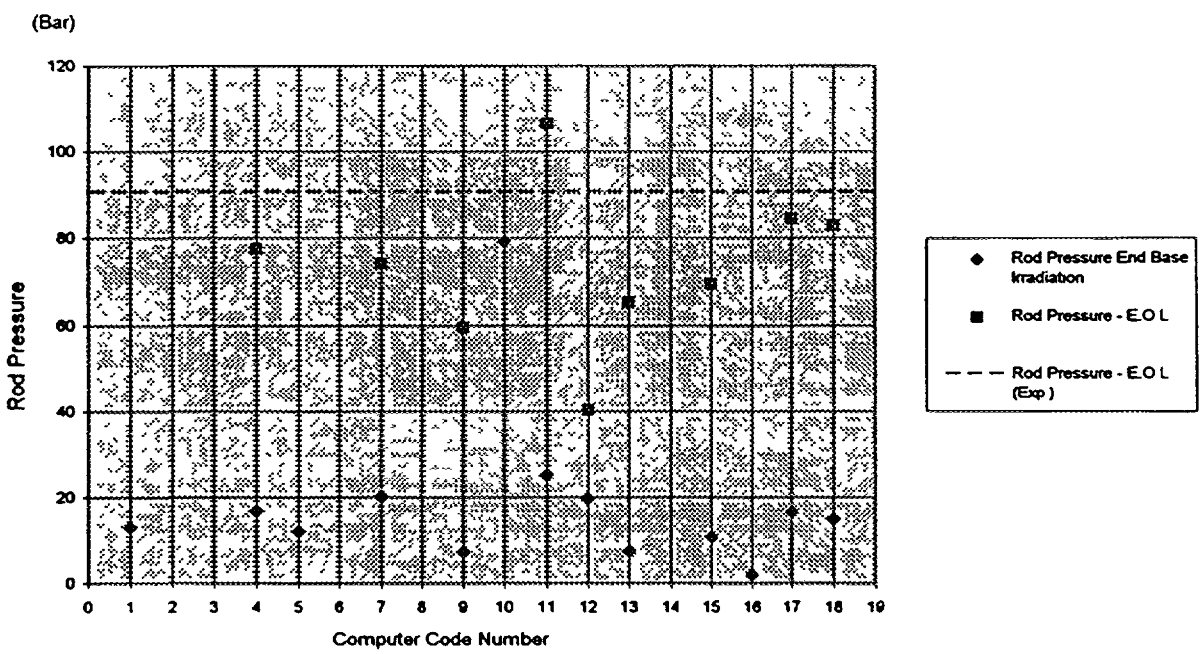


FIG. 42. Comparison of predictions and data for FUMEX 6S of rod internal pressure at the end of the base irradiation and at end-of-life for the rod that experienced the slow power ramp.

Country	Code Name	Number	F.G.R - End Base Irradiation	F.G R - E.O.L 6S
		0		
ARG	BACO	1	8 8	
BUL	PIN MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	10 2	38 5
FRA/EDF	TRANSURANUS/EDF	5	7 3	
FRA/CEA	METEOR/TRANSURANUS	6		
CEC	TRANSURANUS	7	13 8	34 4
IND1	PROFESS	8		
IND2	FAIR	9	3 75	14
IND3	FUDA-MOD1	10	37 8	
JPN1	TRUST-V1b	11	16 9	40.8
JPN2	EIMUS	12	20 02	40 4
CHIN	FRAPCON2	13	3 45	16 9
ROM	ROFEM 1B	14		
SWIT	TRANSURANUS	15	7	29 2
CZE	PIN-W	16	0 9	
UK1	ENIGMA 5 2	17	13 6	50 4
UK2	ENIGMA 5 8D	18	11 4	40 4
RUS	START	19		
	EXP RESULT	20	16 4	50

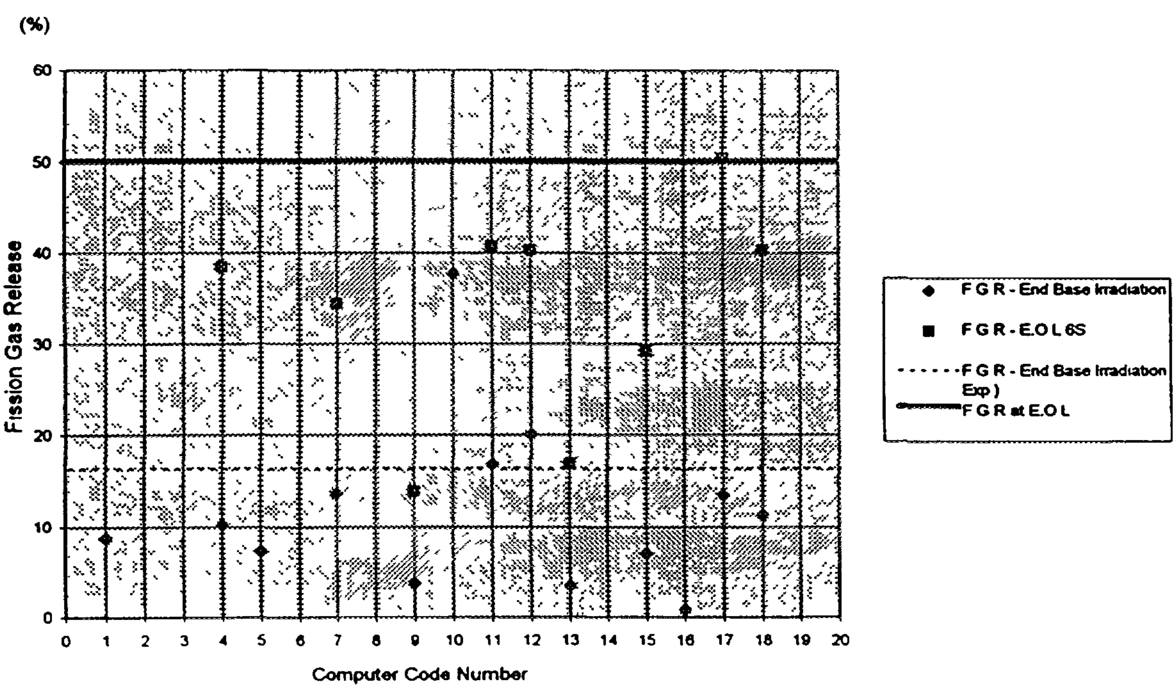


FIG. 43. Comparison of predictions and data for FUMEX 6S of fission gas release at the end of the base irradiation and at end-of-life for the rod that experienced the slow power ramp.

Country	Code Name	Number	d Pressure End Base Irradiati	Rod Pressure - E.O.L
		0		
ARG	BACO	1	13.1	
BUL	PIN-MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	17	71 2
FRA/EDF	TRANSURANUS/EDF	5	12 2	
FRA/CEA	METEOR/TRANSURANUS	6		
CEC	TRANSURANUS	7	20 1	67 1
IND1	PROFESS	8		
IND2	FAIR	9	7.4	56 2
IND3	FUDA-MOD1	10	79 5	
JPN1	TRUST-V1b	11	25.4	102
JPN2	EIMUS	12	19.8	40 4
CHIN	FRAPCON2	13	7 6	64 6
ROM	ROFEM 1B	14		
SWIT	TRANSURANUS	15	10 8	63
CZE	PIN-W	16	2	
UK1	ENIGMA-5 2	17	16 8	76 7
UK2	ENIGMA-5.8D	18	15	79 8
RUS	START	19		
	EXP. RESULT	20		84 6

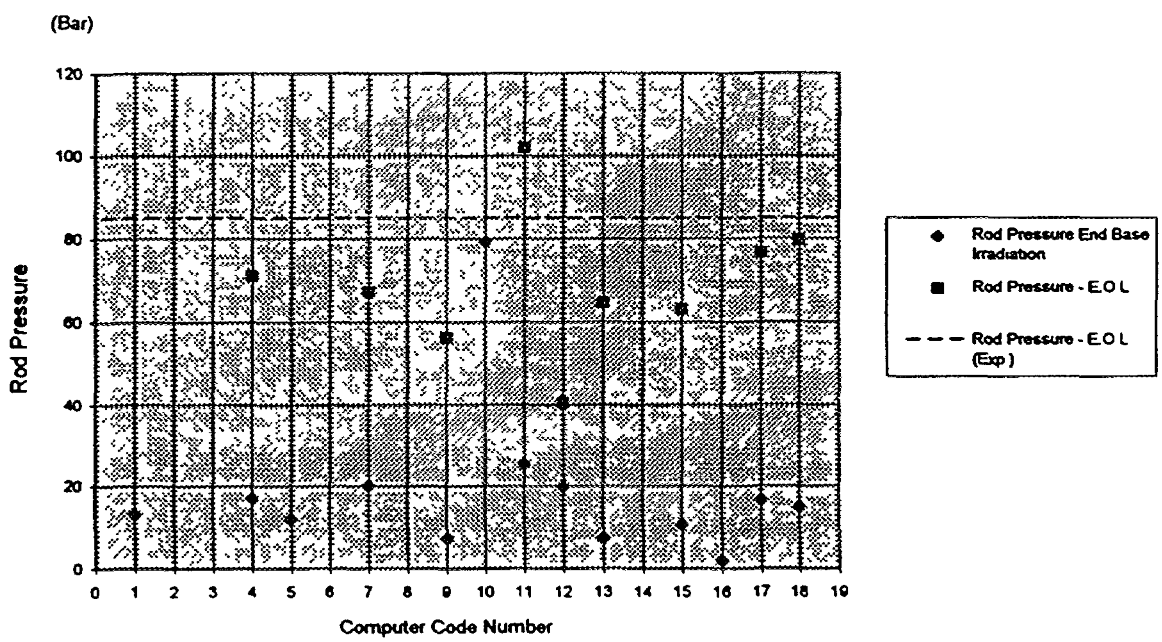


FIG. 44. Comparison of predictions and data for FUMEX 6F of rod internal pressure at the end of the base irradiation and at end-of-life for the rod that experienced the fast power ramp.

Country	Code Name	Number	F.G R End Base Irradiation	FGR End Fast Transient 6F
		0		
ARG	BACO	1	8 8	
BUL	PIN-MICRO	2		
CAN	ELESIM	3		
FIN	VTT/ENIGMA	4	10.2	29 3
FRA/EDF	TRANSURANUS/EDF	5	7 3	
FRA/CEA	METEOR/TRANSURANUS	6		
CEC	TRANSURANUS	7	13 8	24 8
IND1	PROFESS	8		
IND2	FAIR	9	3 75	8 9
IND3	FUDA-MOD1	10	37.8	
JPN1	TRUST-V1b	11	16.9	30 1
JPN2	EIMUS	12	20 02	
CHIN	FRAPCON2	13	3 45	
ROM	ROFEM-1B	14		
SWIT	TRANSURANUS	15	7	19 6
CZE	PIN-W	16	0 9	
UK1	ENIGMA-5 2	17	13 6	38 2
UK2	ENIGMA-5 8D	18	11 4	34 8
RUS	START	19		
	EXP. RESULT	20	16.4	45

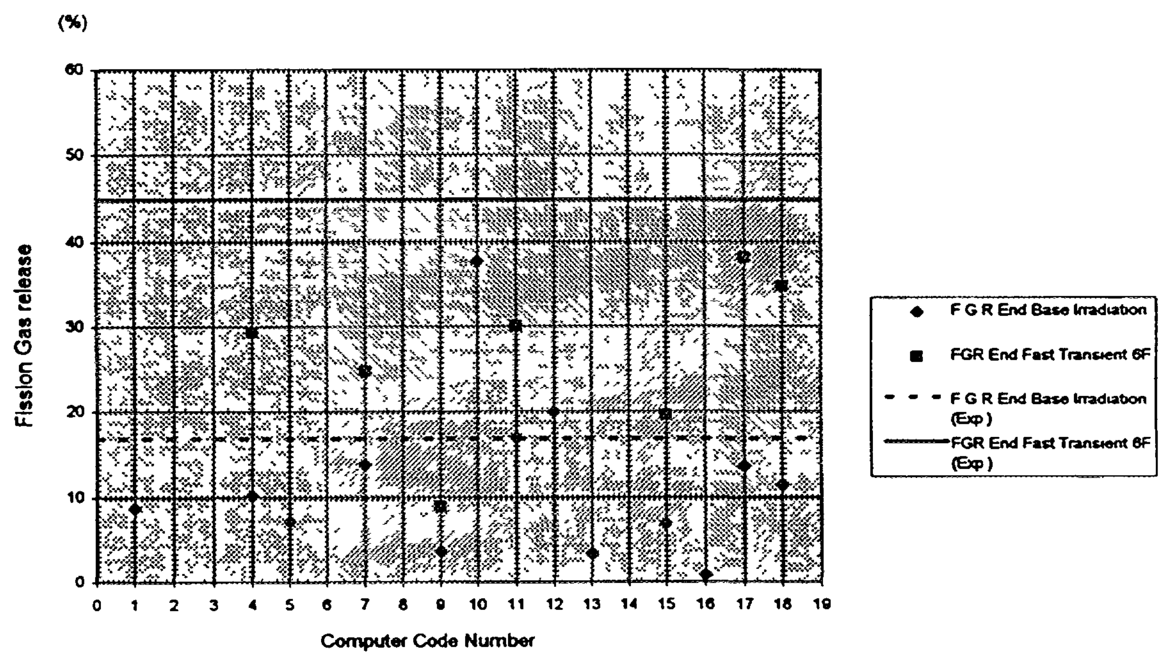


FIG. 45. Comparison of predictions and data for FUMEX 6F of fission gas release at the end of the base irradiation and at end-of-life for the rod that experienced the fast power ramp.

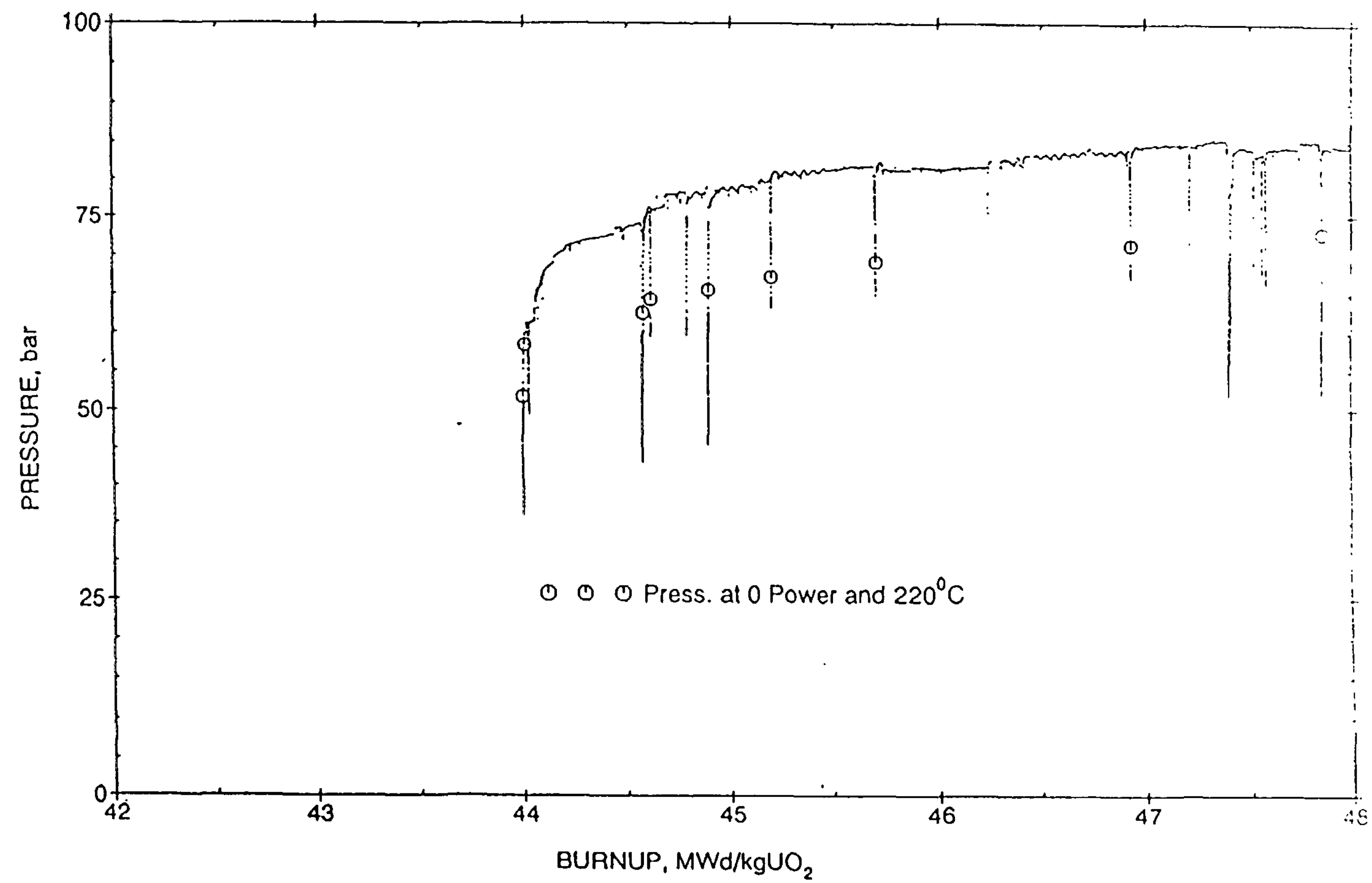


FIG. 46. Evolution of rod internal pressure as measured for FUMEX 6F during the period of high power at the end-of-life.

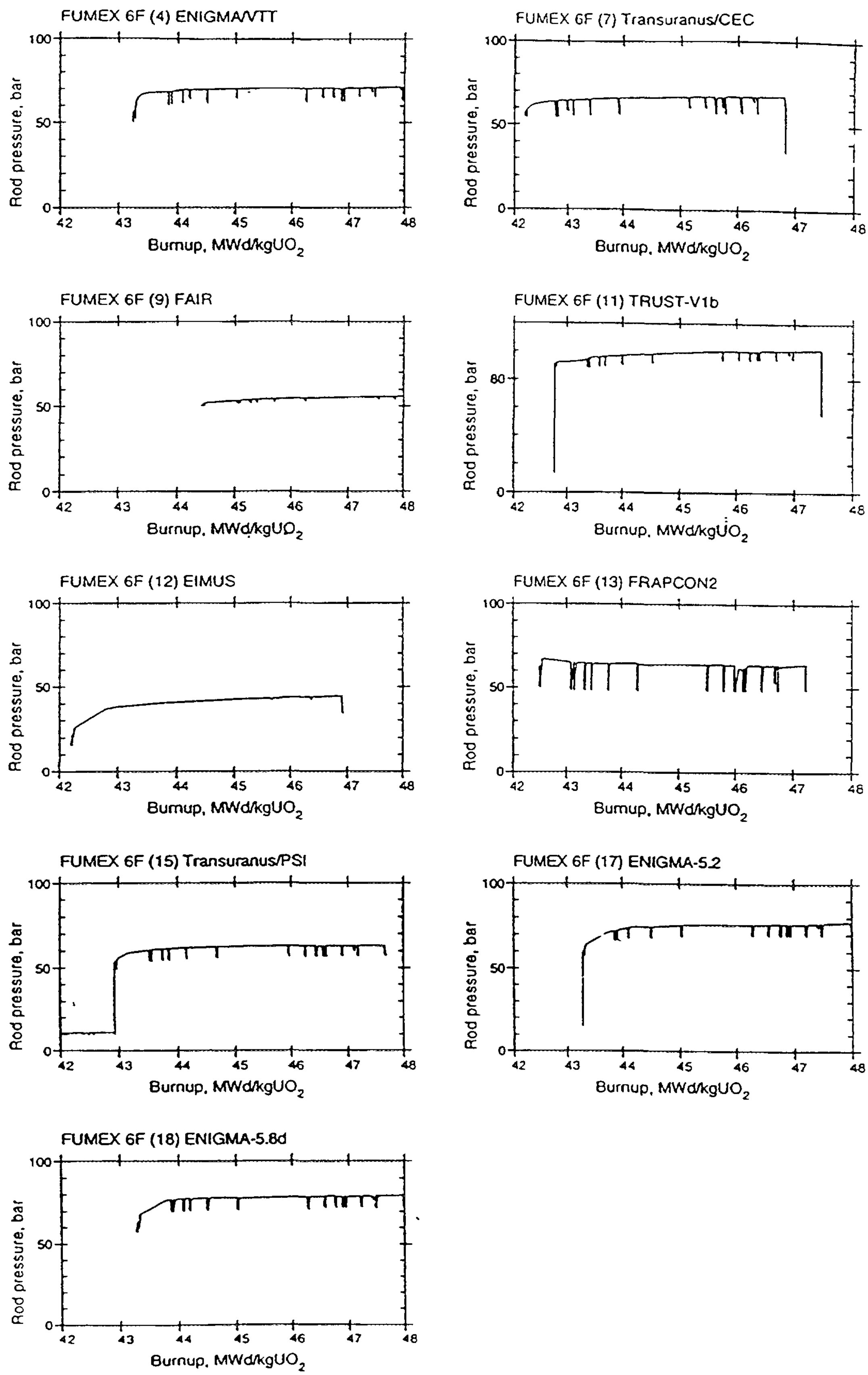


FIG. 47. Code predictions of the evolution of rod internal pressure as measured for FUMEX 6F during the period of high power at the end-of-life.

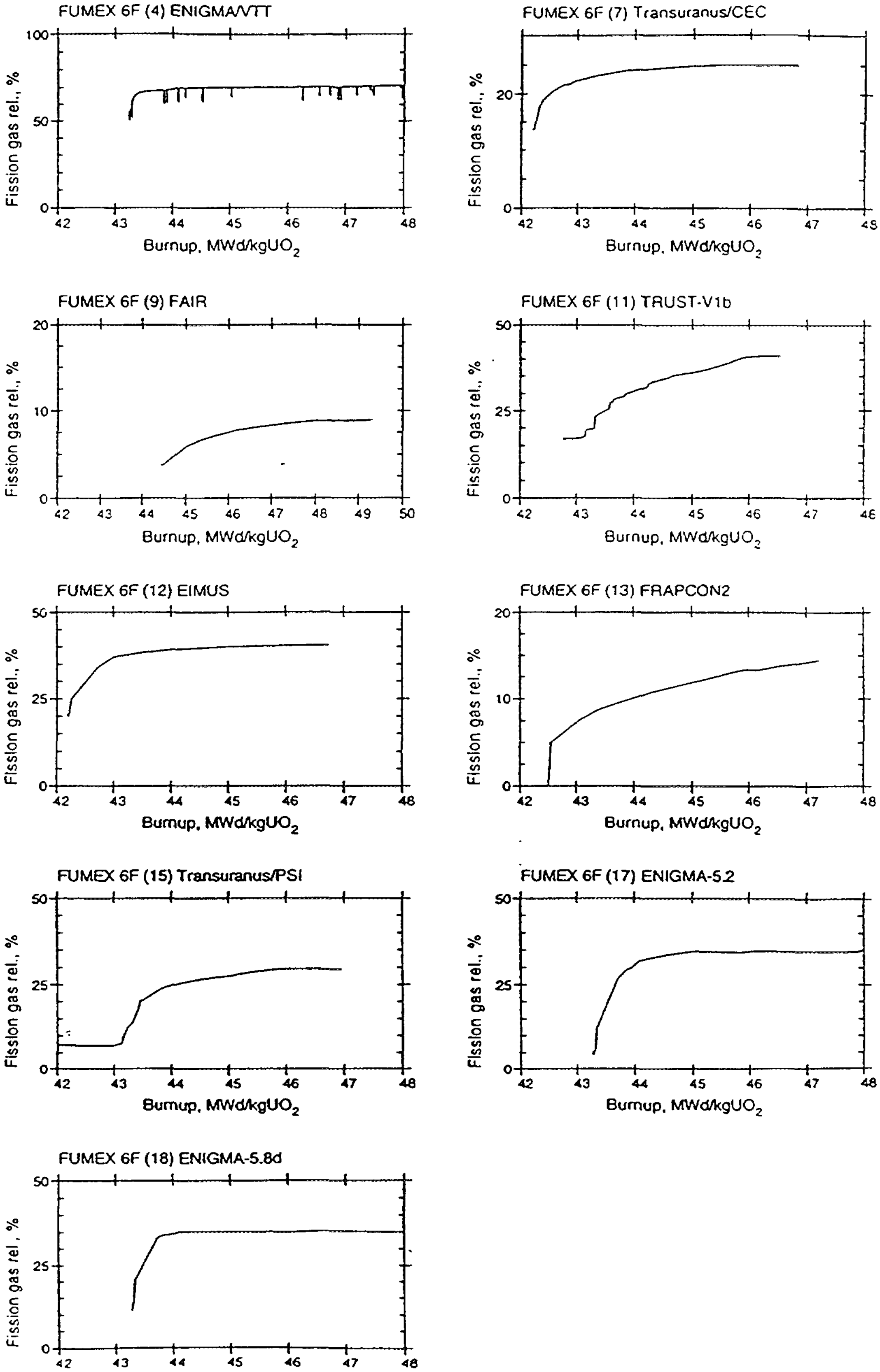


FIG. 48. Code predictions of the evolution of fission gas release for FUMEX 6F during the period of high power at the end-of-life.

Country	Code Name	Number	Tf at 5 MWd/kg	Tf at 30 MWd/kg	Tf at 50 MWd/k	Slope at 20 MWd/kg	Comments
ARG	BACO	1	725	739	816	0 56	Smooth PCMI at 38 MWd/kg
BULG	PIN-MICRO	2	938	1746	1061	10 15	Slope at 5-10 MWd/kg
CAN	ELESIM	3					
FIN	VTT/ENIGMA	4	861	977	1061	4 64	No PCI
FRA/EDF	TRANSURANUS/EDF	5	920	954	1126	0 24	FGR contaminating the gap
FRA/CEA	METEOR/TRANSURANUS	6					
CEC	TRANSURANUS	7	893	857	862	-1 4	PCMI 43 4 MWD/kg
IND1	PROFESS	8	970	960	1025	-0 4	PCMI 30 MWd/kg
IND2	FAIR	9	815	800	855	0 6	PCMI 37 5 MWd/kg
IND3	FUDA-MOD1	10					
JPN1	TRUST-V1b	11					
JPN2	EIMUS	12	891 2	1003 1	1108	4 27	
CHIN	FRAPCON2	13	910	910	1120	-11 7	PCMI 12 MWd/kg
ROM	ROFEM 1b	14	869	897	913	1 12	
SWIT	TRANSURANUS	15	912	916	926	0 95	No PCI
CZE	PIN-W	16	768	792	790	0 96	
UK/BNFL	ENIGMA	17	875	908	1083	1 32	
UK/NE	ENIGMA	18	834	917	970	3 3	PCMI >50 MWd/kg
RUS	START	19	917	960	990	0 86	PCMI sharp at 48 MWd/kg

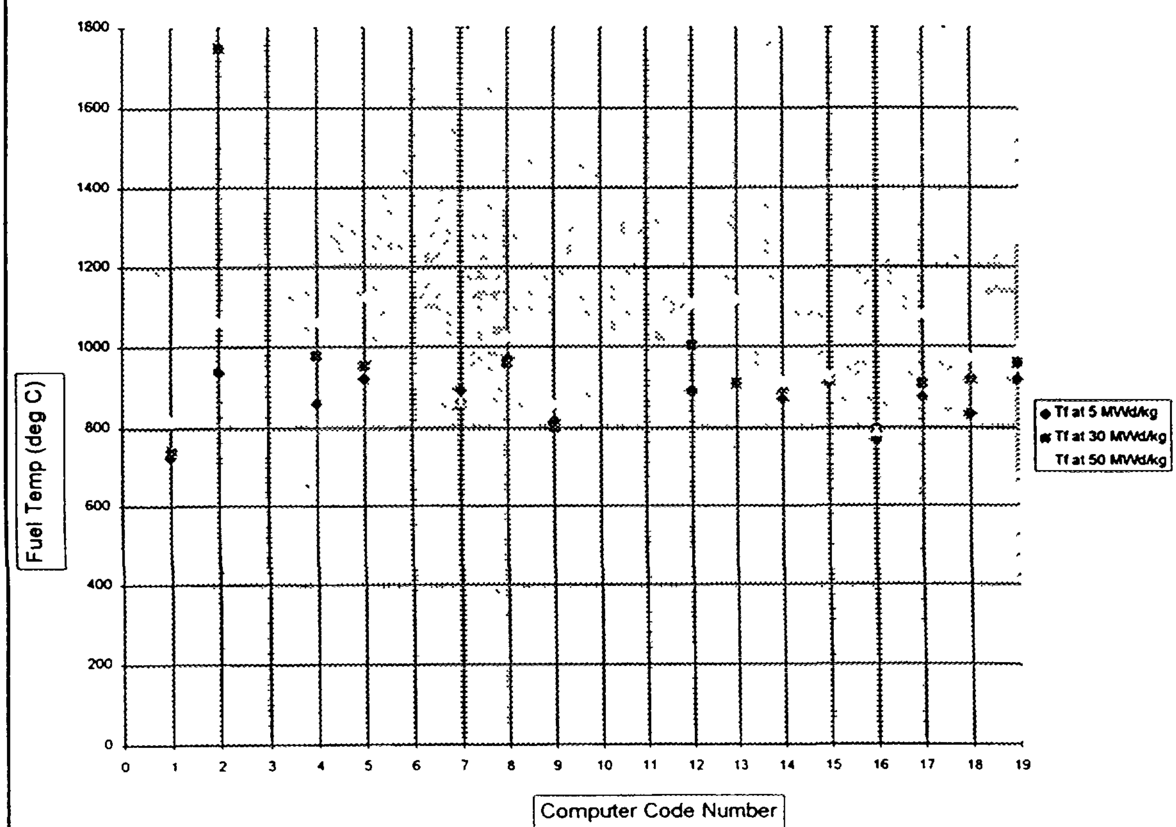


FIG. 49. FUMEX S1: Fuel centre temperature.

Country	Code Name	Number	Tf at 5 MWd/kg	Tf at 30 MWd/kg	Tf at 50 MWd/k	Slope at 20 MWd/kg	Comments
ARG	BACO	1	725	739	816	0.56	Smooth PCMI at 38 MWd/kg
BULG	PIN-MICRO	2	938	1746	1061	10.15	Slope at 5-10 MWd/kg
CAN	ELESIM	3					
FIN	VTT/ENIGMA	4	861	977	1061	4.64	No PCI
FRA/EDF	TRANSURANUS/EDF	5	920	954	1126	0.24	FGR contaminating the gap
FRA/CEA	METEOR/TRANSURANUS	6					
CEC	TRANSURANUS	7	893	857	862	-1.4	PCMI 43.4 MWD/kg
IND1	PROFESS	8	970	960	1025	-0.4	PCMI 30 MWd/kg
IND2	FAIR	9	815	800	855	-0.6	PCMI 37.5 MWd/kg
IND3	FUDA-MOD1	10					
JPN1	TRUST-V1b	11					
JPN2	EIMUS	12	880	980	1100	4	
CHIN	FRAPCON2	13	910	910	1120	-11.7	PCMI 12 MWd/kg
ROM	ROFEM-1b	14	869	897	913	1.12	
SWIT	TRANSURANUS	15	912	916	926	0.95	No PCI
CZE	PIN-W	16	768	792	790	0.96	
UK/BNFL	ENIGMA	17	875	908	1083	1.32	
UK/NE	ENIGMA	18	834	917	970	3.3	PCMI >50 MWd/kg.
RUS	START	19	917	960	990	0 86	PCMI sharp at 48 MWd/kg

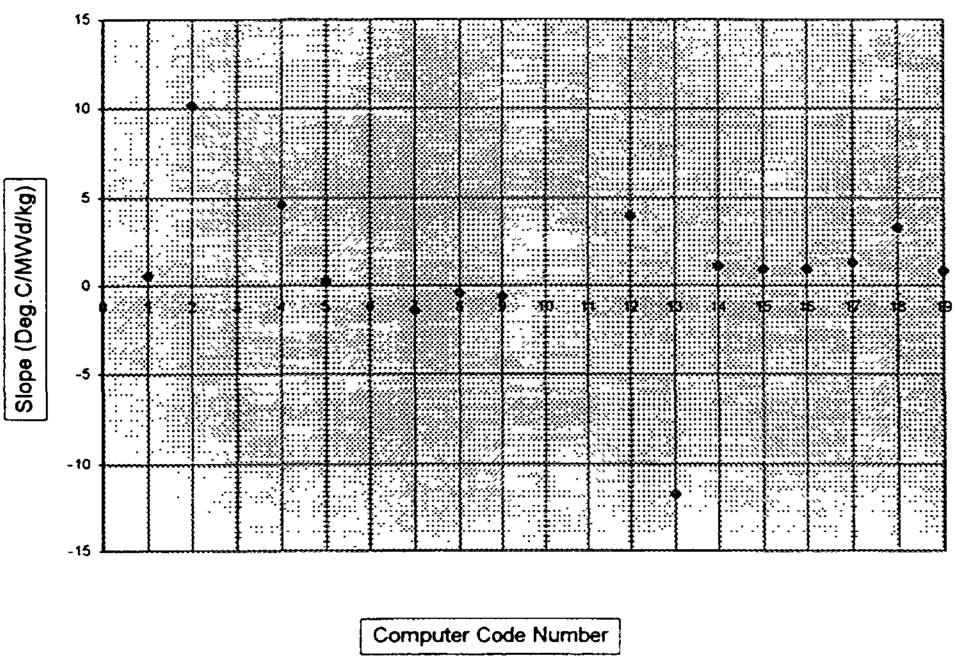


FIG. 50. FUMEX S1: Temperature evolution slope at 20 MW·d/kg.

Country	Code Name	Number	B.U for 1% FGR	FGR at 60 MWd/kg	Comments
ARG	BACO	1	31	2.7	No Threshold, smooth increase with BU
BULG	PIN-MICRO	2	20	56	Threshold 20 MWd/kg, 65% max at 36 G
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4		0.46	Threshold at 25 MWd/kg
FRA/EDF	TRANSURANUS/EDF	5	32.5	12	Threshold at 22 MWd/kg
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7		0	0 release calculated
IND1	PROFESS	8	0	3.25	Threshold at 30 MWd/kg
IND2	FAIR	9		0.25	Linear FGR , 0.05 % per 10 MWd/kg
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12		1.93	
CHIN	FRAPCON2	13	4	8.2	Slope change at 14 MWd/kg, rapid increase after 20 MWd/kg
ROM	ROFEM-1b	14		0.52	Initially 0.05 % per 10 MWd/kg with threshold at 41 MWd/kg
SWIT	TRANSURANUS	15		0.23	Threshold at 19 MWd/kg
CZE	PIN-W	16	10	3.2	Smooth increase with BU
UK/BNFL	ENIGMA	17	47	1.3	Initially linear, threshold at 34-36 MWd/kg
UK/NE	ENIGMA	18		0.29	Threshold 30MWd/kg
RUS	START	19	35	2.4	Threshold 25 MWd/kg

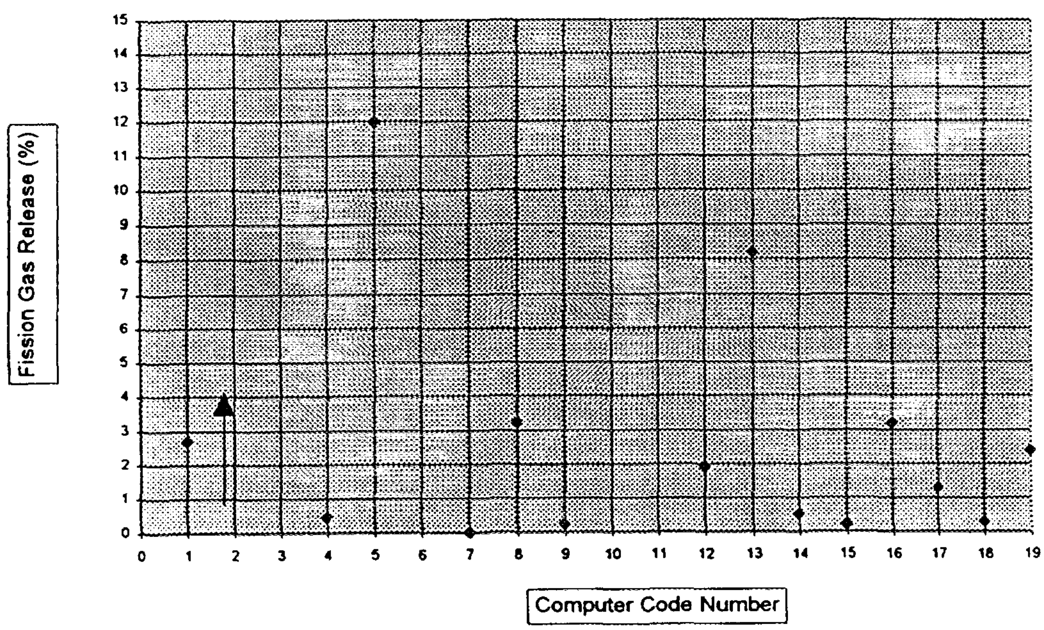


FIG. 51. FUMEX S1: Fission gas release at 50 MW·d/kg.

Country	Code Name	Number	B U for 1% FGR	FGR at 50 MWd/kg	Comments
ARG	BACO	1	31	2 7	No Threshold, smooth increase with BU
BULG	PIN-MICRO	2	20	56	Threshold 20 MWd/kg, 65% max at 36 G
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4		0 46	Threshold at 25 MWd/kg
FRA/EDF	TRANSURANUS/EDF	5	32 5	12	Threshold at 22 MWd/kg
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7		0	0 release calculated
IND1	PROFESS	8	0	3 25	Threshold at 30 MWd/kg
IND2	FAIR	9		0 25	Linear FGR 0 05 % per 10 MWd/kg
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	45 5	1 9	Accelerated release starts at 37 MWd/kg
CHIN	FRAPCON2	13	4	8 2	Slope change at 14 MWd/kg rapid increase after 20 MWd/kg
ROM	ROFEM-1b	14		0 52	Initially 0 05 % per 10 MWd/kg with threshold at 41 MWd/kg
SWIT	TRANSURANUS	15		0 23	Threshold at 19 MWd/kg
CZE	PIN-W	16	10	3 2	Smooth increase with BU
UK/BNFL	ENIGMA	17	47	1 3	Initially linear threshold at 34-36 MWd/kg
UK/NE	ENIGMA	18		0 29	Threshold 30MWd/kg
RUS	START	19	35	2 4	Threshold 25 MWd/kg

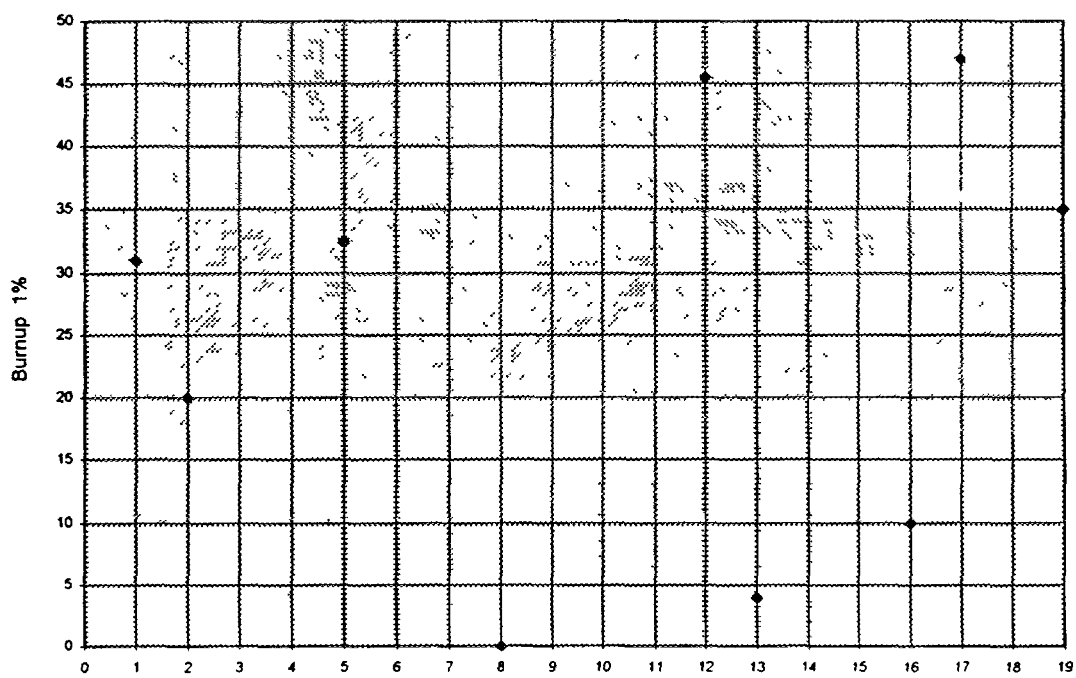


FIG. 52. FUMEX S1: Burnup for 1% fission gas release.

Country	Code Name	Number	at BOL	Minim.	EOL
ARG	BACO	1	0 98	0 905	1 75
BULG	PIN-MICRO	2	1 04	1 04	12 5
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	0 99	0 95	1 27
FRA/EDF	TRANSURANUS/EDF	5	1	0 9	3 5
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	1 05	0 95	1 4
IND1	PROFESS	8	0 99	0 93	1 84
IND2	FAIR	9	1 07	1 06	1 22
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	1 07	1 06	1 5
CHIN	FRAPCON2	13	1 05	0 98	2 95
ROM	ROFEM-1b	14	1 04	0 98	1 29
SWIT	TRANSURANUS	15	1	0 94	1 12
CZE	PIN-W	16	0 923	0 913	1 69
UK/BNFL	ENIGMA	17	0 87	0 87	1 39
UK/NE	ENIGMA	18	0 89	0 97	1 36
RUS	START	19	1 15	1 1	1 85

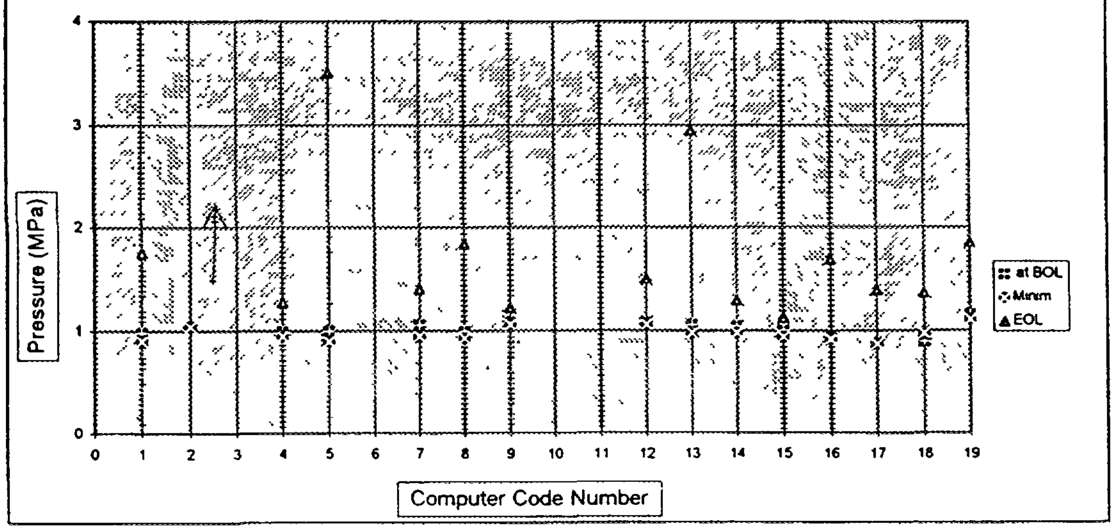


FIG. 53. FUMEX S1: Rod pressure.

Country	Code Name	Number	Tf 30G bef. ramp	Tf 30G after Ramp	Tf 60 G	Comments
ARG	BACO	1	704	1530	1369	
BULG	PIN-MICRO	2	1743	2395	2031	
CAN	ELESIM	3				
FIN	VTT/ENIGMA	4	965	1632	1840	
FRA/EDF	TRANSURANUS/EDF	5	950	1660	1900	
FRA/CEA	METEOR/TRANSURANUS	6				
CEC	TRANSURANUS	7	841	1659	1976	
IND1	PROFESS	8	970	1500	1530	
IND2	FAIR	9	807	1476	1930	
IND3	FUDA-MOD1	10				
JPN1	TRUST-V1b	11				
JPN2	EIMUS	12	1006	1710	2099	
CHIN	FRAPCON2	13	915	1577	2817	
ROM	ROFEM-1b	14	888	1684	1744	
SWIT	TRANSURANUS	15	915	1692	1959	
CZE	PIN-W	16	926	1590	1895	
UK/BNFL	ENIGMA	17	1000	1673	1876	
UK/NE	ENIGMA	18	919	1606	1827	
RUS	START	19	960	1700	2040	

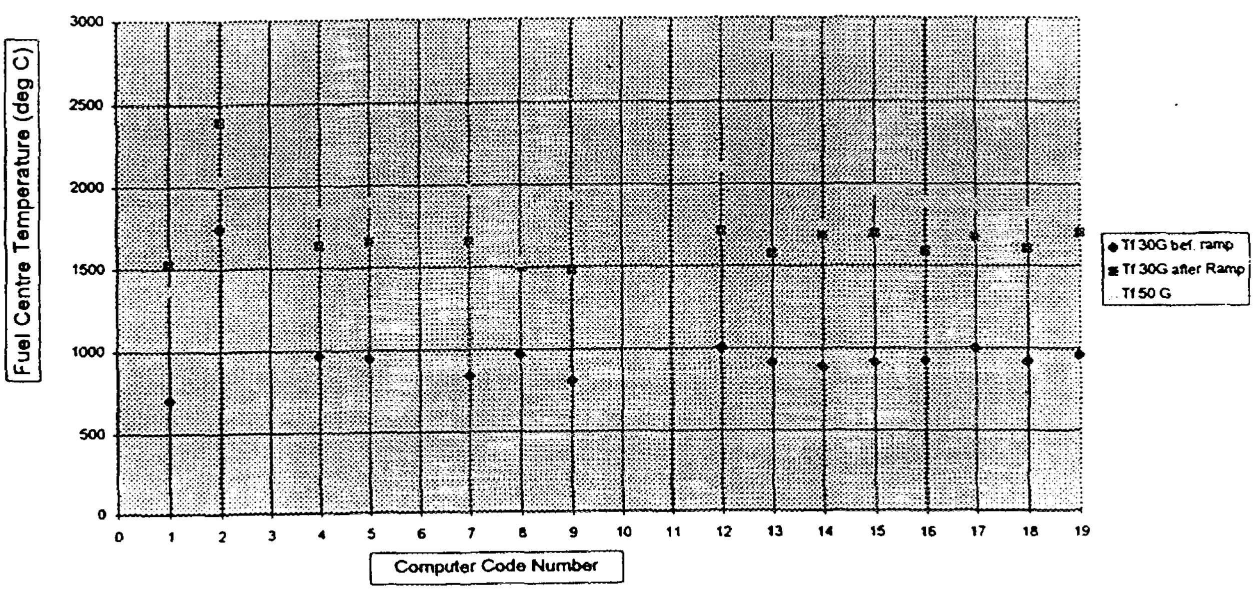


FIG. 54. FUMEX S2: Fuel temperature.

Country	Code Name	Number	FGR at 30 G before ramp	FGR at 31 G	FGR at 60 G	Comments
ARG	BACO	1	1	19	17	
BULG	PIN-MICRO	2	58.5	60	58.5	FGR well before ramp
CAN	ELESIM	3				
FIN	VTT/ENIGMA	4	0	13.8	28.1	
FRA/EDF	TRANSURANUS/EDF	5	0.6	2.2	26	
FRA/CEA	METEOR/TRANSURANUS	6				
CEC	TRANSURANUS	7	0	19	40	
IND1	PROFESS	8	1	7.2	9.1	
IND2	FAIR	9	0.1	12.1	36.96	
IND3	FUDA-MOD1	10				
JPN1	TRUST-V1b	11				
JPN2	EIMUS	12	0.25	24	45.4	
CHIN	FRAPCON2	13	2	4.5	77	
ROM	ROFEM-1b	14	0.2	18.6	28.5	
SWIT	TRANSURANUS	15	0.08	31	47	
CZE	PIN-W	16	2.3	17.2	41	
UK/BNFL	ENIGMA	17	0.4	19.3	37.5	
UK/NE	ENIGMA	18	0.2	19	36.6	
RUS	START	19	1	12	36	

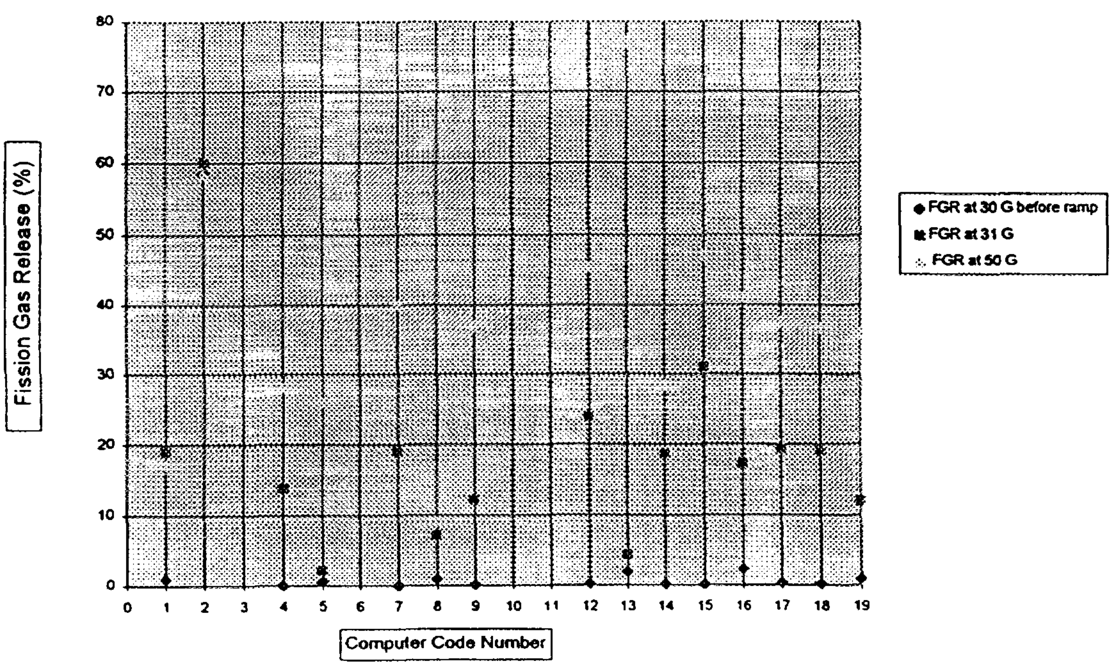


FIG. 55. FUMEX S2: Fission gas release (%).

Country	Code Name	Number	Tf 30G aft. Ramp	Tf 50 G	"dTf"	FGR at 50 G
ARG	BACO	1	1530	1369	-161	17
BULG	PIN-MICRO	2	2395	2031		
CAN	ELESIM	3			0	
FIN	VTT/ENIGMA	4	1632	1840	208	28 1
FRA/EDF	TRANSURANUS/EDF	5	1660	1900	240	26
FRA/CEA	METEOR/TRANSURANUS	6			0	
CEC	TRANSURANUS	7	1659	1976	317	40
IND1	PROFESS	8	1500	1530	30	9 1
IND2	FAIR	9	1476	1930	454	36 96
IND3	FUDA-MOD1	10			0	
JPN1	TRUST-V1b	11			0	
JPN2	EIMUS	12	1710	2099	389	45 4
CHIN	FRAPCON2	13	1577	2817	1240	77
ROM	ROFEM-1b	14	1684	1744	60	28 5
SWIT	TRANSURANUS	15	1692	1959	267	50
CZE	PIN-W	16	1590	1895	305	47
UK/BNFL	ENIGMA	17	1673	1876	203	37 5
UK/NE	ENIGMA	18	1606	1827	221	36 6
RUS	START	19	1700	2040	340	36

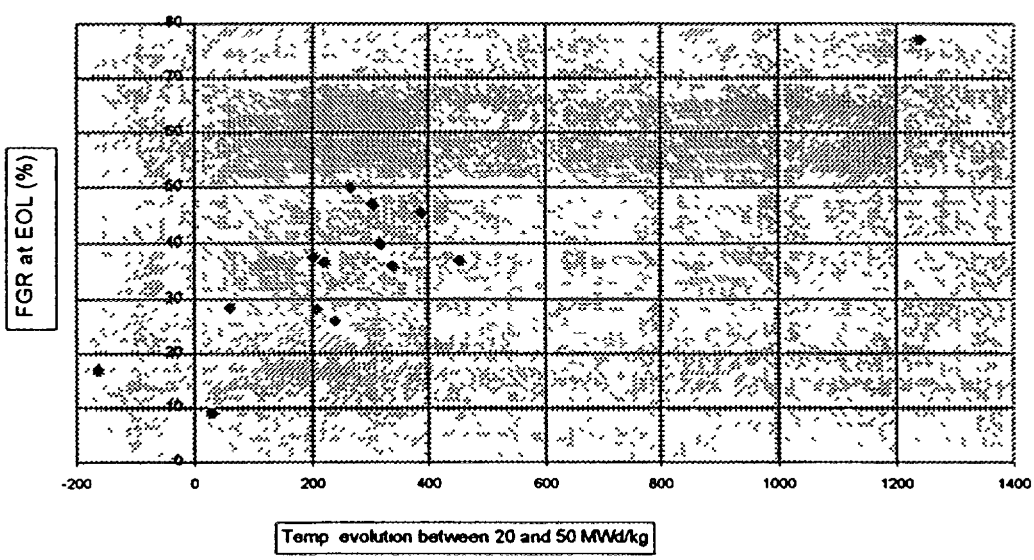


FIG. 56. FUMEX S2: FGR evolution versus temperature variation between 20 and 50 MW·d/kg.

Country	Code Name	Number	EOL
ARG	BACO	1	4 36
BULG	PIN-MICRO	2	15 3
CAN	ELESIM	3	
FIN	VTT/ENIGMA	4	6 95
FRA/EDF	TRANSURANUS/EDF	5	6 6
FRA/CEA	METEOR/TRANSURANUS	6	
CEC	TRANSURANUS	7	10 6
IND1	PROFESS	8	3 5
IND2	FAIR	9	9 85
IND3	FUDA-MOD1	10	
JPN1	TRUST-V1b	11	
JPN2	EIMUS	12	10 2
CHIN	FRAPCON2	13	20 2
ROM	ROFEM-1b	14	7 9
SWIT	TRANSURANUS	15	11
CZE	PIN-W	16	10 3
UK/BNFL	ENIGMA	17	10 6
UK/NE	ENIGMA	18	14 4
RUS	START	19	9.3

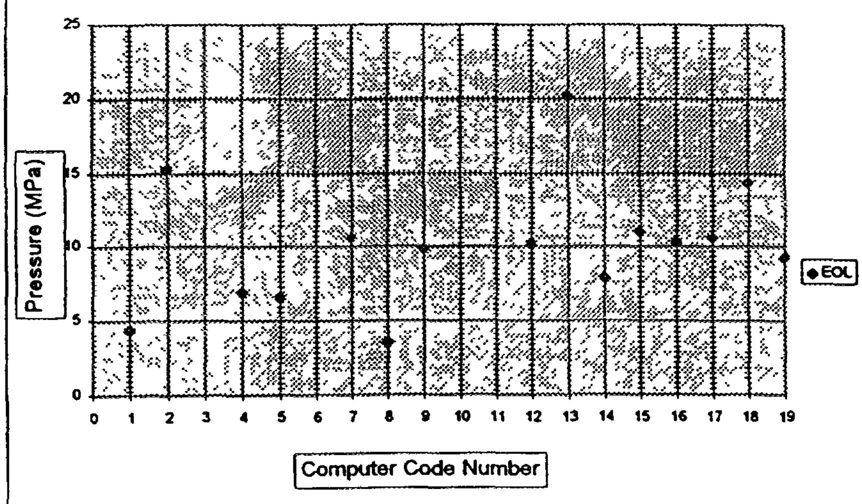


FIG. 57. FUMEX S2. Rod pressure.

Country	Code Name	Number	Gap 50	gap 100	gap 150	gap 200	gap 230
ARG	BACO	1	694	719	737	743	746
BULG	PIN-MICRO	2	646	715	785	846	892
CAN	ELESIM	3					
FIN	VTT/ENIGMA	4	658	703	747	792	822
FRA/EDF	TRANSURANUS/EDF	5	630	700	750	800	820
FRA/CEA	METEOR/TRANSURANUS	6					
CEC	TRANSURANUS	7	631	695	749	802	833
IND1	PROFESS	8	695	770	840	900	940
IND2	FAIR	9	649	679	717	755	777
IND3	FUDA-MOD1	10					
JPN1	TRUST-V1b	11					
JPN2	EIMUS	12	598	658	702	751	771
CHIN	FRAPCON2	13	720	740	790	850	895
ROM	ROFEM-1b	14	640	700	735	790	820
SWIT	TRANSURANUS	15	653	714	770	825	857
CZE	PIN-W	16	610	670	727	782	813
UK/BNFL	ENIGMA	17	658	725	800	830	840
UK/NE	ENIGMA	18	646	692	738	780	808
RUS	START	19	660	710	750	795	820

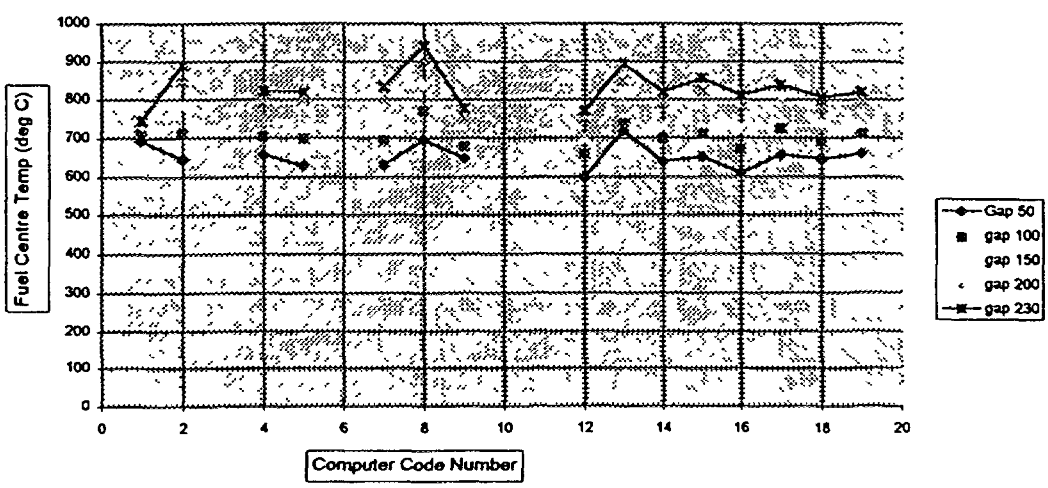


FIG. 58. FUMEX S3 to 7: Centre temperatures for different gap sizes at 20 kW/m.

Country	Code Name	Number	gap 50	gap 100	gap 150	gap 200	gap 230
ARG	BACO	1	1285	1327	1397	1425	1439
BULG	PIN-MICRO	2	1075	1196	1326	1439	1507
CAN	ELESIM	3					
FIN	VTT/ENIGMA	4	1188	1228	1332	1436	1495
FRA/EDF	TRANSURANUS/EDF	5	1230	1240	1300	1410	1470
FRA/CEA	METEOR/TRANSURANUS	6					
CEC	TRANSURANUS	7	1191	1247	1368	1482	1547
IND1	PROFESS	8	1260	1320	1460	1580	1645
IND2	FAIR	9	1208	1223	1253	1332	1374
IND3	FUDA-MOD1	10					
JPN1	TRUST-V1b	11					
JPN2	EIMUS	12	1089	1098	1180	1282	1338
CHIN	FRAPCON2	13	1315	1375	1455	1575	1645
ROM	ROFEM-1b	14	1155	1200	1290	1400	1455
SWIT	TRANSURANUS	15	1258	1312	1435	1552	1610
CZE	PIN-W	16	1085	1106	1205	1298	1351
UK/BNFL	ENIGMA	17	1208	1253	1413	1537	1585
UK/NE	ENIGMA	18	1186	1231	1343	1443	1500
RUS	START	19	1240	1250	1350	1450	1500

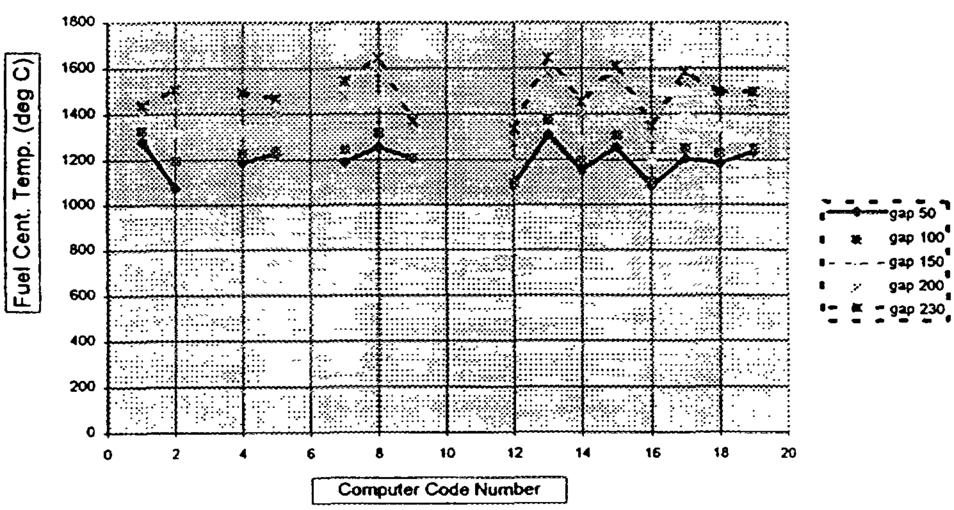


FIG. 59. FUMEX S3 to 7: Centre temperatures for different gap sizes at 40 kW/m.

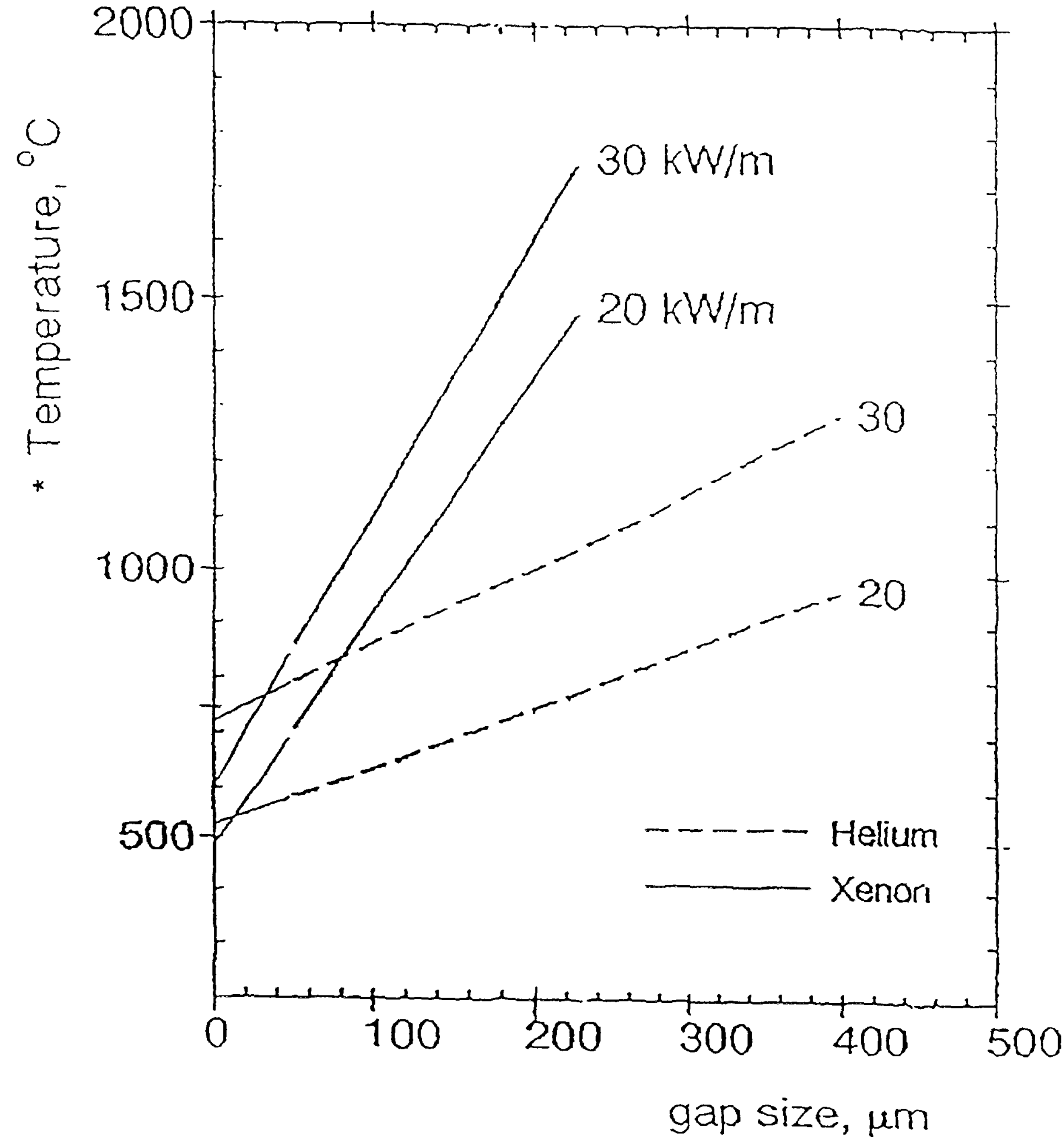


FIG. 60. Fuel temperature in helium and xenon filled rods.

Country	Code Name	Number	He	Xe	T Xe-T He
ARG	BACO	1	461	651	190
BULG	PIN-MICRO	2	476	923	447
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	457	681	224
FRA/EDF	TRANSURANUS/EDF	5	450	770	320
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	457	800	343
IND1	PROFESS	8	490	930	440
IND2	FAIR	9	453	785	332
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	436	660	224
CHIN	FRAPCON2	13	487	872	385
ROM	ROFEM-1b	14	456	839	383
SWIT	TRANSURANUS	15	470	812	342
CZE	PIN-W	16	443	911	468
UK/BNFL	ENIGMA	17	482	678	196
UK/NE	ENIGMA	18	464	705	241
RUS	START	19	460	690	230

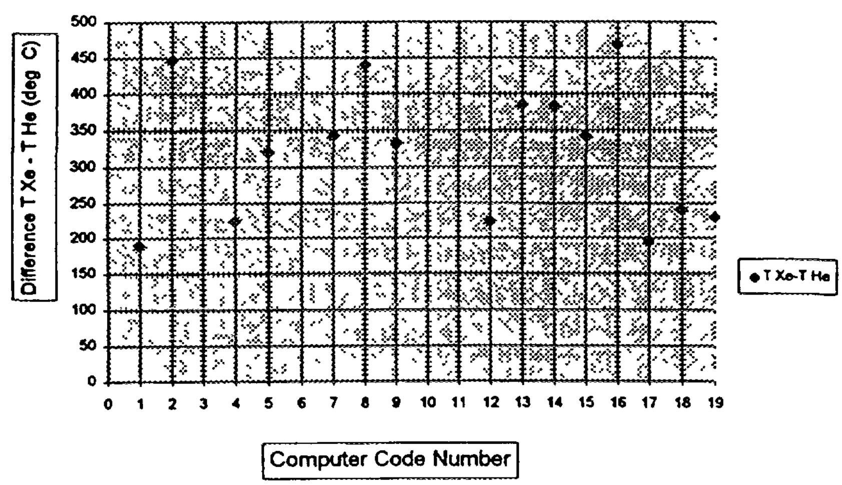


FIG. 61. FUMEX S4 and 8: Comparison between He and Xe filled rods at 10 kW/m.

Country	Code Name	Number	He	Xe	T Xe - T He
ARG	BACO	1	723	1047	324
BULG	PIN-MICRO	2	723	1277	554
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	676	928	252
FRA/EDF	TRANSURANUS/EDF	5	690	990	300
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	695	1060	365
IND1	PROFESS	8	780	1230	450
IND2	FAIR	9	679	996	317
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	651	851	200
CHIN	FRAPCON2	13	774	1200	426
ROM	ROFEM-1b	14	700	1045	345
SWIT	TRANSURANUS	15	716	1083	367
CZE	PIN-W	16	670	1156	486
UK/BNFL	ENIGMA	17	725	918	193
UK/NE	ENIGMA	18	694	957	263
RUS	START	19	700	1000	300

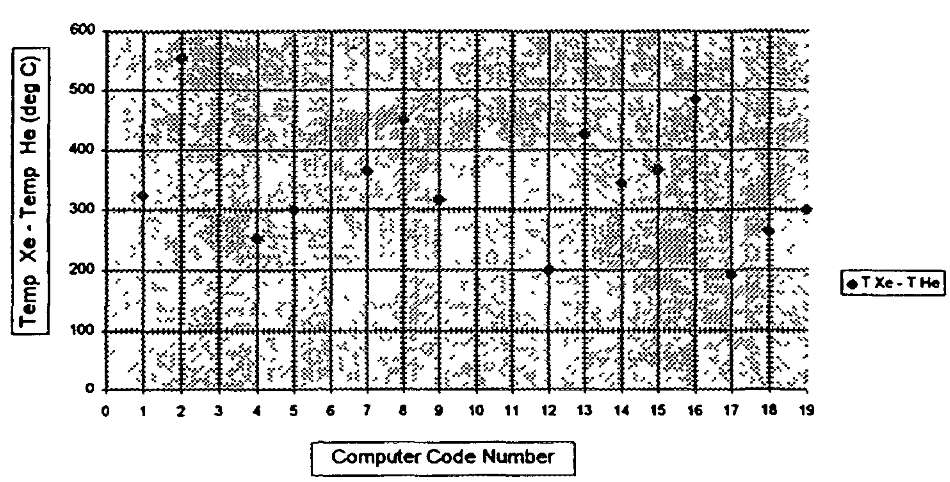


FIG. 62. FUMEX S4 and 8: Comparison between He and Xe filled rods at 20 kW/m.

Country	Code Name	Number	He	Xe	T Xe - T He
ARG	BACO	1	1008	1420	412
BULG	PIN-MICRO	2	992	1554	562
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	919	1128	209
FRA/EDF	TRANSURANUS/EDF	5	940	1280	340
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	960	1310	350
IND1	PROFESS	8	1045	1420	375
IND2	FAIR	9	921	1223	302
IND3	FUDA-MOD1*	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	866	984	118
CHIN	FRAPCON2	13	1091	1460	369
ROM	ROFEM-1b	14	944	1210	266
SWIT	TRANSURANUS	15	1000	1350	350
CZE	PIN-W	16	892	1495	603
UK/BNFL	ENIGMA	17	980	1148	168
UK/NE	ENIGMA	18	951	1169	218
RUS	START	19	970	1250	280

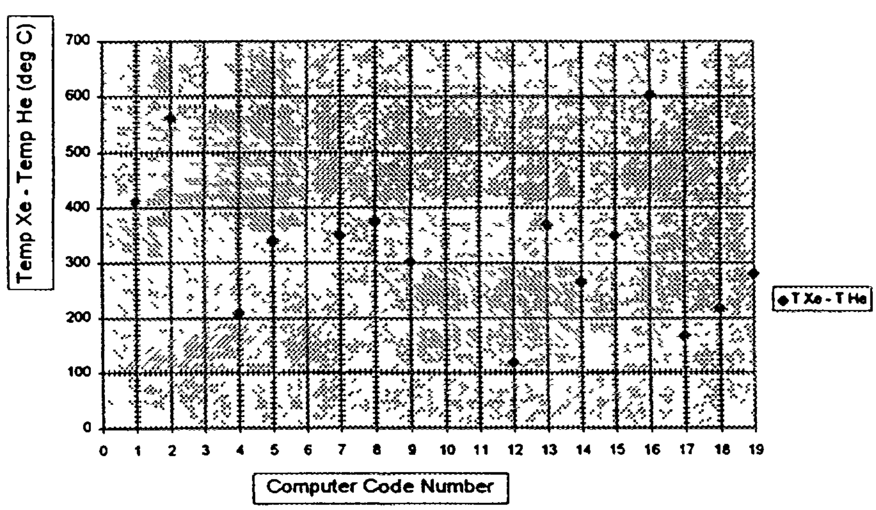


FIG. 63. FUMEX S4 and 8: Comparison between He and Xe filled rods at 30 kW/m.

Country	Code Name	Number	He	Xe	T Xe - T He
ARG	BACO	1	1317	1634	317
BULG	PIN-MICRO	2	1223	1769	546
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1176	1290	114
FRA/EDF	TRANSURANUS/EDF	5	1240	1670	430
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	1247	1660	413
IND1	PROFESS	8	1330	1610	280
IND2	FAIR	9	1215	1513	298
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	1098	1242	144
CHIN	FRAPCON2	13	1385	1709	324
ROM	ROFEM-1b	14	1200	1412	212
SWIT	TRANSURANUS	15	1312	1714	402
CZE	PIN-W	16	1106	1718	612
UK/BNFL	ENIGMA	17	1253	1386	133
UK/NE	ENIGMA	18	1231	1354	123
RUS	START	19	1250	1520	270

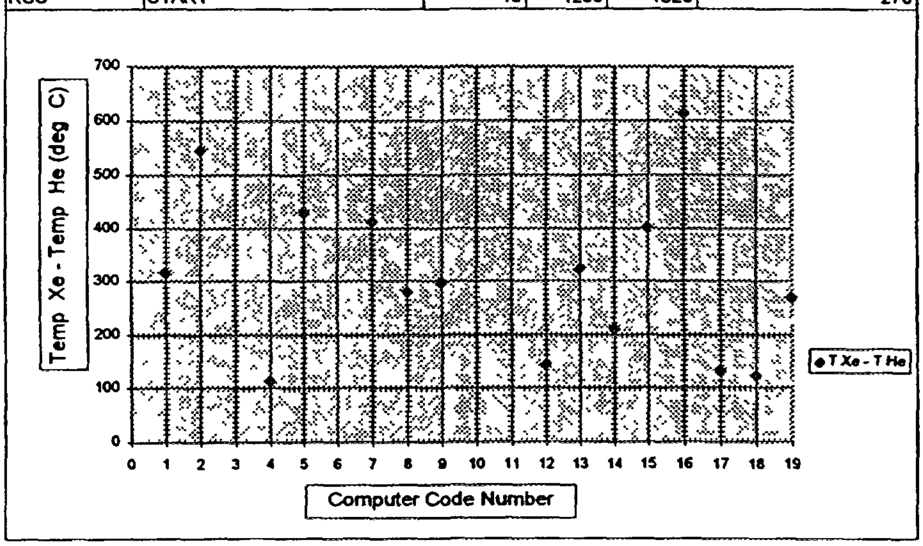


FIG. 64. FUMEX S4 and 8: Comparison between He and Xe filled rods at 40 kW/m.

Country	Code Name	Number	FGR at 50 MWd/kg	"+d(FGR)"	"-d(FGR)"
ARG	BACO	1	2.7	0.13	-0.11
BULG	PIN-MICRO	2	56		
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	0.46	1.6	-0.345
FRA/EDF	TRANSURANUS/EDF	5	12	6.7	-4.8
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	0	0	0
IND1	PROFESS	8	3.25		
IND2	FAIR	9	0.25		
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12			
CHIN	FRAPCON2	13	8.2		
ROM	ROFEM-1b	14	0.52		
SWIT	TRANSURANUS	15	1.2	5.4	-1.2
CZE	PIN-W	16	3.2		
UK/BNFL	ENIGMA	17	1.3	2	-0.893
UK/NE	ENIGMA	18	0.29	0.19	-0.12
RUS	START	19	2.4	3.72	-2.15

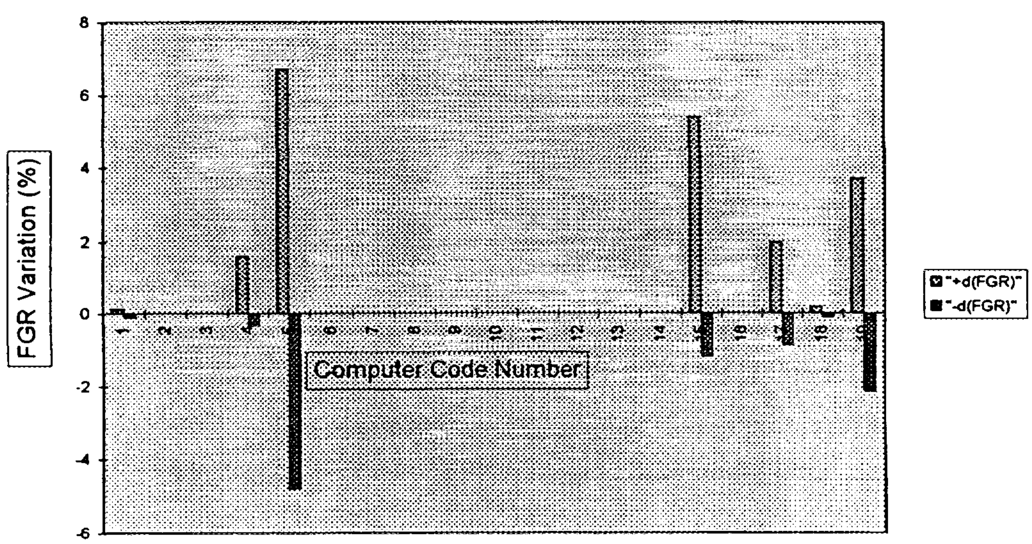


FIG. 65. FUMEX S1: FGR at EOL, statistical analysis.

Country	Code Name	Number	Tf 50 G	"+dTf"	"-dTf"
ARG	BACO	1	1369		
BULG	PIN-MICRO	2	2031		
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1840	160	-155
FRA/EDF	TRANSURANUS/EDF	5	1900	154	-154
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	2010	162	-162
IND1	PROFESS	8	1530	195 5	-169 9
IND2	FAIR	9	1930	79 5	-79 5
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12			
CHIN	FRAPCON2	13	2817		
ROM	ROFEM-1b	14	1744	88 7	-144 7
SWIT	TRANSURANUS	15	1959	95	-95
CZE	PIN-W	16	1895		
UK/BNFL	ENIGMA	17	1876	233	-325
UK/NE	ENIGMA	18	1827	145	-160
RUS	START	19	2040	188	-188

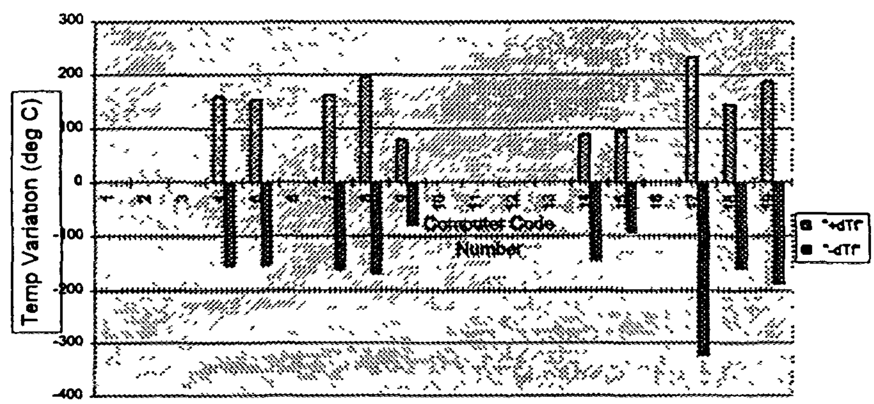


FIG. 66. FUMEX S2: Fuel temperature at EOL, statistical analysis.

Country	Code Name	Number	FGR at 60 G	"+d(FGR)"	"-d(FGR)"
ARG	BACO	1	17		
BULG	PIN-MICRO	2	58 5		
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	28 1	3 9	-5 3
FRA/EDF	TRANSURANUS/EDF	5	26	3 2	-2 8
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	42	7 4	-7 4
IND1	PROFESS	8	9 1	5 8	-3 8
IND2	FAIR	9	36 96	2 77	-2 77
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12			
CHIN	FRAPCON2	13	77		
ROM	ROFEM-1b	14	28 5	4 59	-3 95
SWIT	TRANSURANUS	15	47	3	-3
CZE	PIN-W	16	41		
UK/BNFL	ENIGMA	17	37.5	5 9	-8 7
UK/NE	ENIGMA	18	36 6	6 1	-5 3
RUS	START	19	36	3 3	-3 3

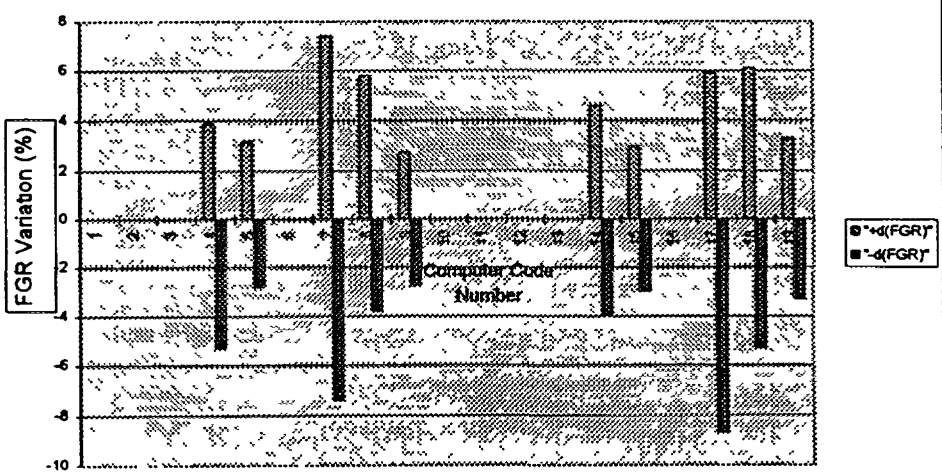


FIG. 67. FUMEX S2: FGR at EOL, statistical analysis.

Country	Code Name	Number	Fuel Temperature EOL	+ ΔT	- ΔT
ARG	BACO	1	812	840	784
BULG	PIN-MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1061	1157	965 5
FRA/EDF	TRANSURANUS/EDF	5	1126	1266	994
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	859	950 5	780
IND1	PROFESS	8	1025	1079	873
IND2	FAIR	9	855	946	782
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	1116	1274	1019
CHIN	FRAPCON2	13			
ROM	ROFEM 1b	14	913	1001 5	841
SWIT	TRANSURANUS	15	926	1093	840
CZE	PIN-W	16			
UK/BNFL	ENIGMA	17	1083	1190	991
UK/NE	ENIGMA	18	968	1043	899
RUS	START	19			

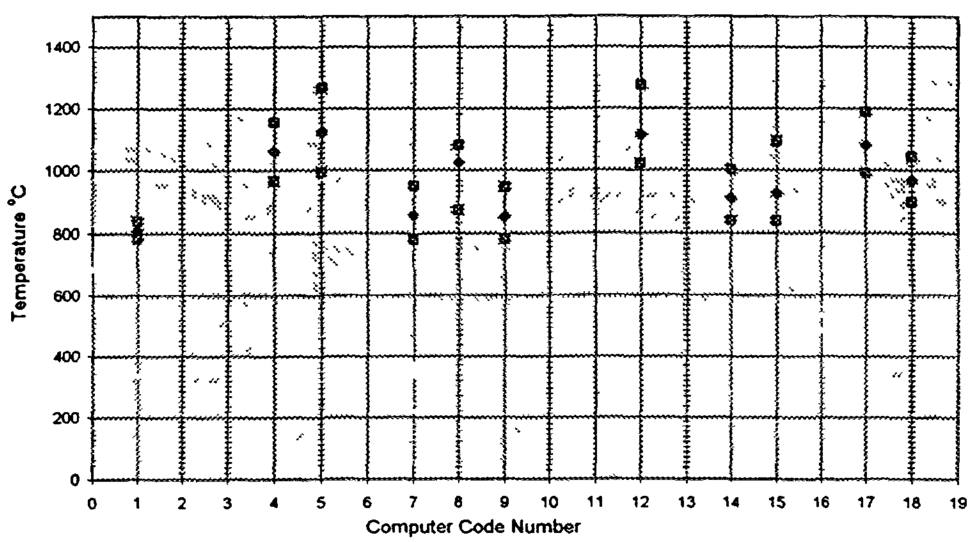


FIG. 68. FUMEX S1: Fuel temperature at EOL, uncertainty analysis.

Country	Code Name	Number	Fuel Temperature EOL	+ ΔT	- ΔT
ARG	BACO	1			
BULG	PIN-MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	1840	2000	1674
FRA/EDF	TRANSURANUS/EDF	5	1890	2076	1711
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	1976	2194	1753
IND1	PROFESS	8			
IND2	FAIR	9	1930	2116	1709
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	2089	2251	1923
CHIN	FRAPCON2	13			
ROM	ROFEM-1b	14	1744	1896	1581
SWIT	TRANSURANUS	15	1959	2128	1800
CZE	PIN-W	16			
UK/BNFL	ENIGMA	17	1876	2241	1714
UK/NE	ENIGMA	18	1829	1985	1671
RUS	START	19			

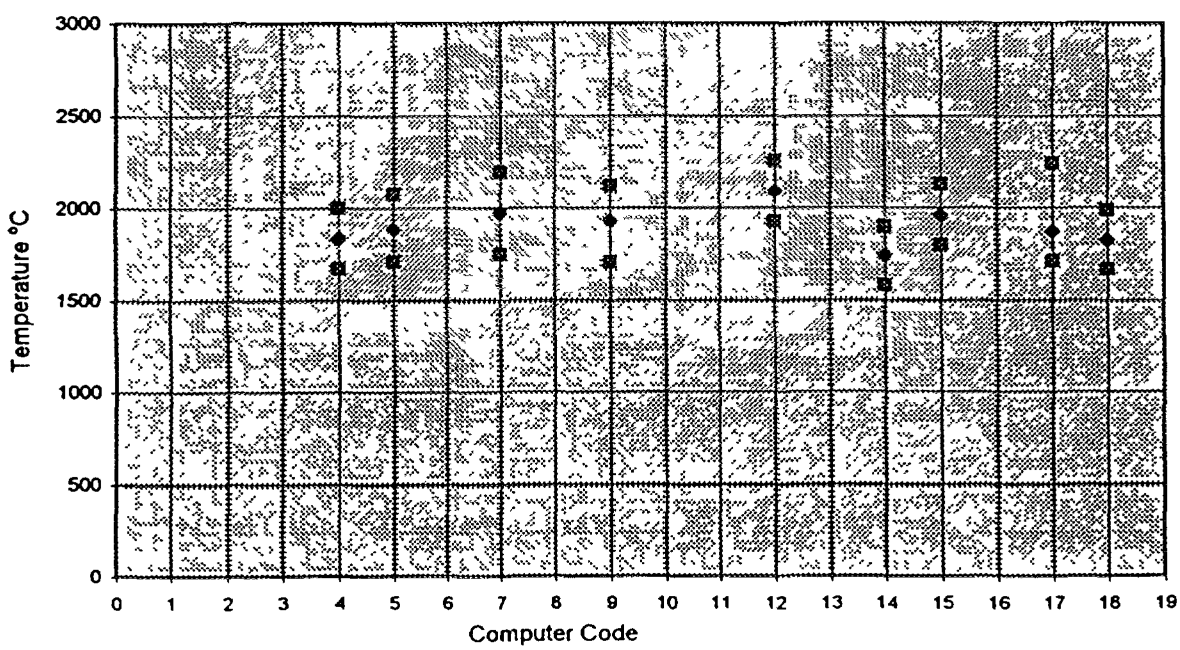


FIG. 69. FUMEX S2: Fuel temperature at EOL, uncertainty analysis.

Country	Code Name	Number	FGR EOL	+ Δ FGR	- Δ FGR
ARG	BACO	1	2.65	2.78	2.54
BULG	PIN-MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	0.46	2.1	0.1
FRA/EDF	TRANSURANUS/EDF	5	12	21.8	6.2
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	0.04	0.03	0.05
IND1	PROFESS	8	3.25	3.85	1.03
IND2	FAIR	9			
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	1.9	12.2	0.45
CHIN	FRAPCON2	13			
ROM	ROFEM-1b	14	0.52	0.92	0.38
SWIT	TRANSURANUS	15	0.23	1.86	0.02
CZE	PIN-W	16			
UK/BNFL	ENIGMA	17	1.3	6.5	0.5
UK/NE	ENIGMA	18	0.29	0.57	0.19
RUS	START	19			

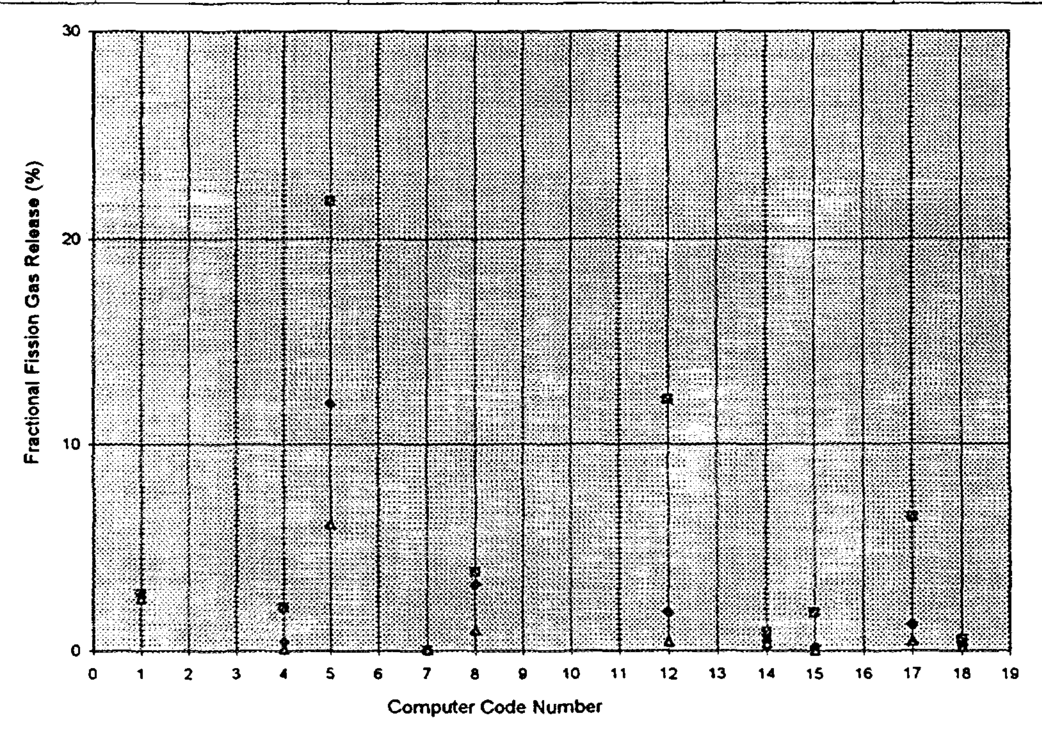


FIG. 70. FUMEX S1: Fractional fission gas release at EOL, uncertainty analysis.

Country	Code Name	Number	Fuel Temperature EOL	+ ΔT	- ΔT
ARG	BACO	1			
BULG	PIN-MICRO	2			
CAN	ELESIM	3			
FIN	VTT/ENIGMA	4	27 7	31 2	22 6
FRA/EDF	TRANSURANUS/EDF	5	25 2	29 5	21 8
FRA/CEA	METEOR/TRANSURANUS	6			
CEC	TRANSURANUS	7	40	48 6	30 2
IND1	PROFESS	8			
IND2	FAIR	9	36 8	43 7	27 5
IND3	FUDA-MOD1	10			
JPN1	TRUST-V1b	11			
JPN2	EIMUS	12	45 3	48 5	39 7
CHIN	FRAPCON2	13			
ROM	ROFEM-1b	14	28 2	35	22 9
SWIT	TRANSURANUS	15	47 6	52 5	42 6
CZE	PIN-W	16			
UK/BNFL	ENIGMA	17	38 8	44 6	31 9
UK/NE	ENIGMA	18	36 6	42 7	31 1
RUS	START	19			

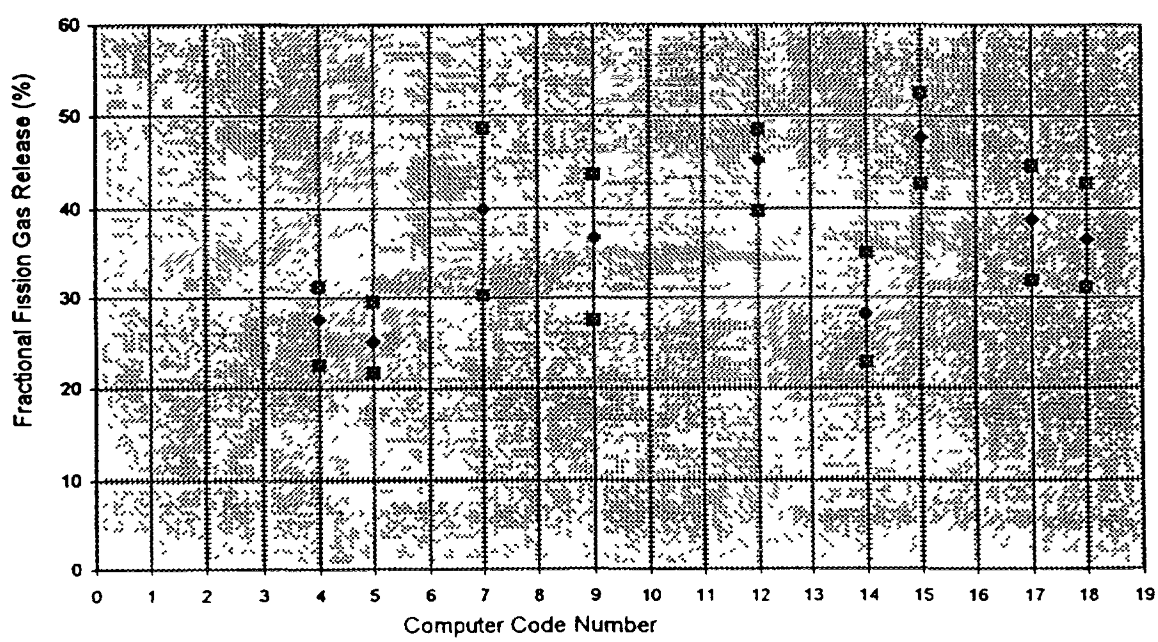


FIG. 71. FUMEX S1: Fractional fission gas release at EOL, uncertainty analysis.

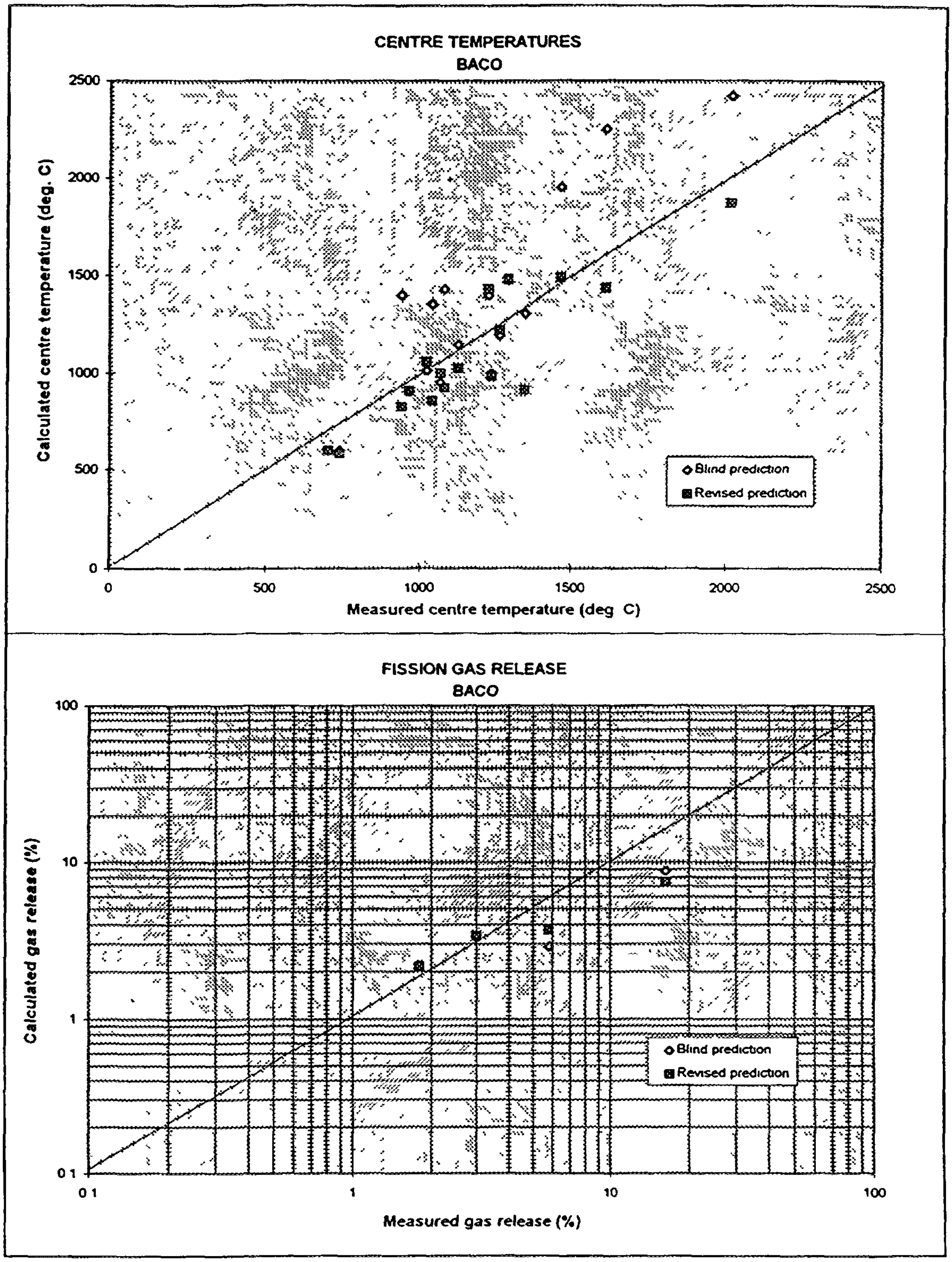
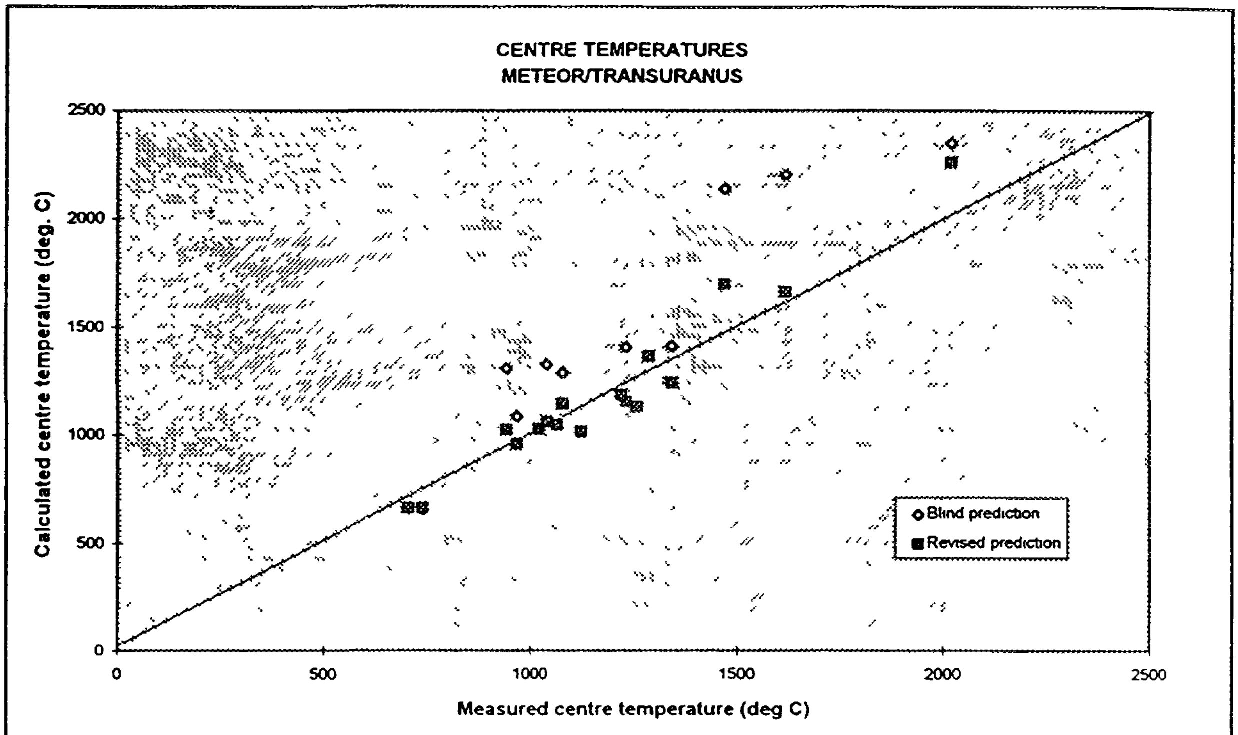


FIG. 72. Comparison calculation/experiment for the blind prediction and after code modification – BACO.



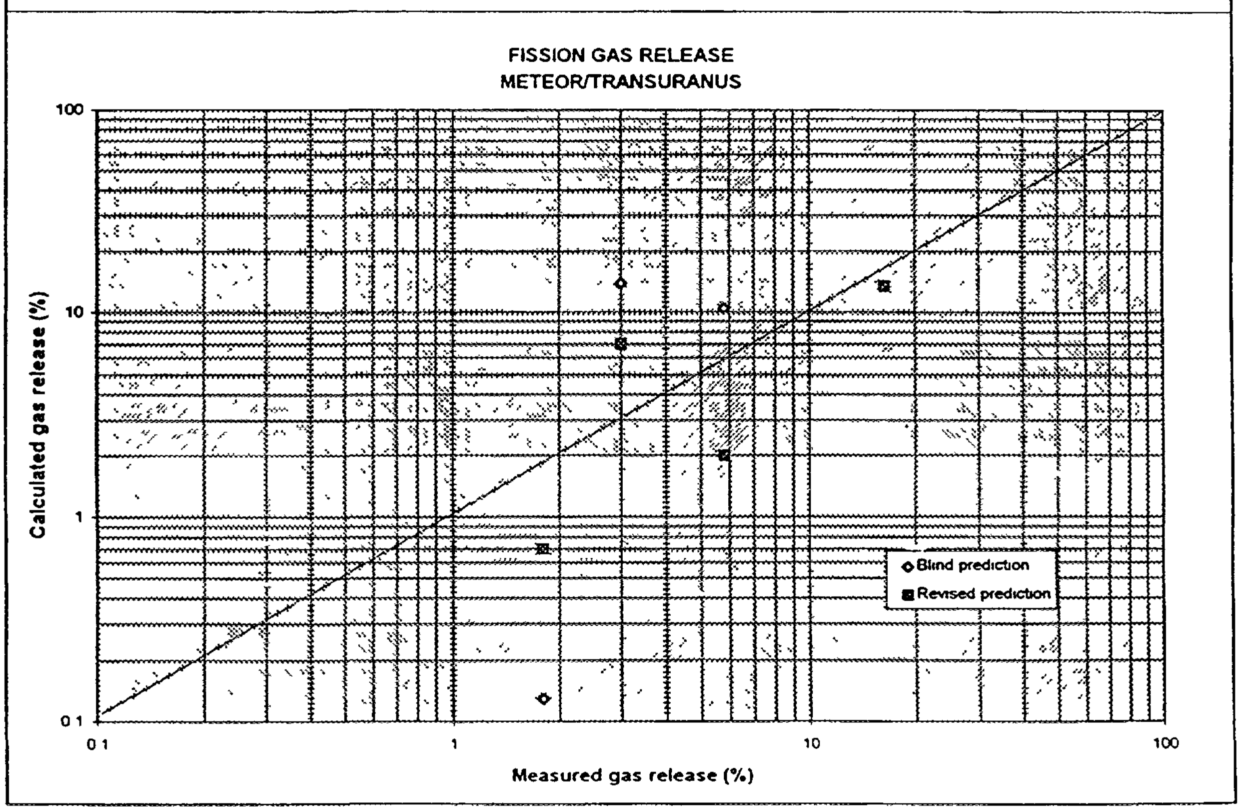
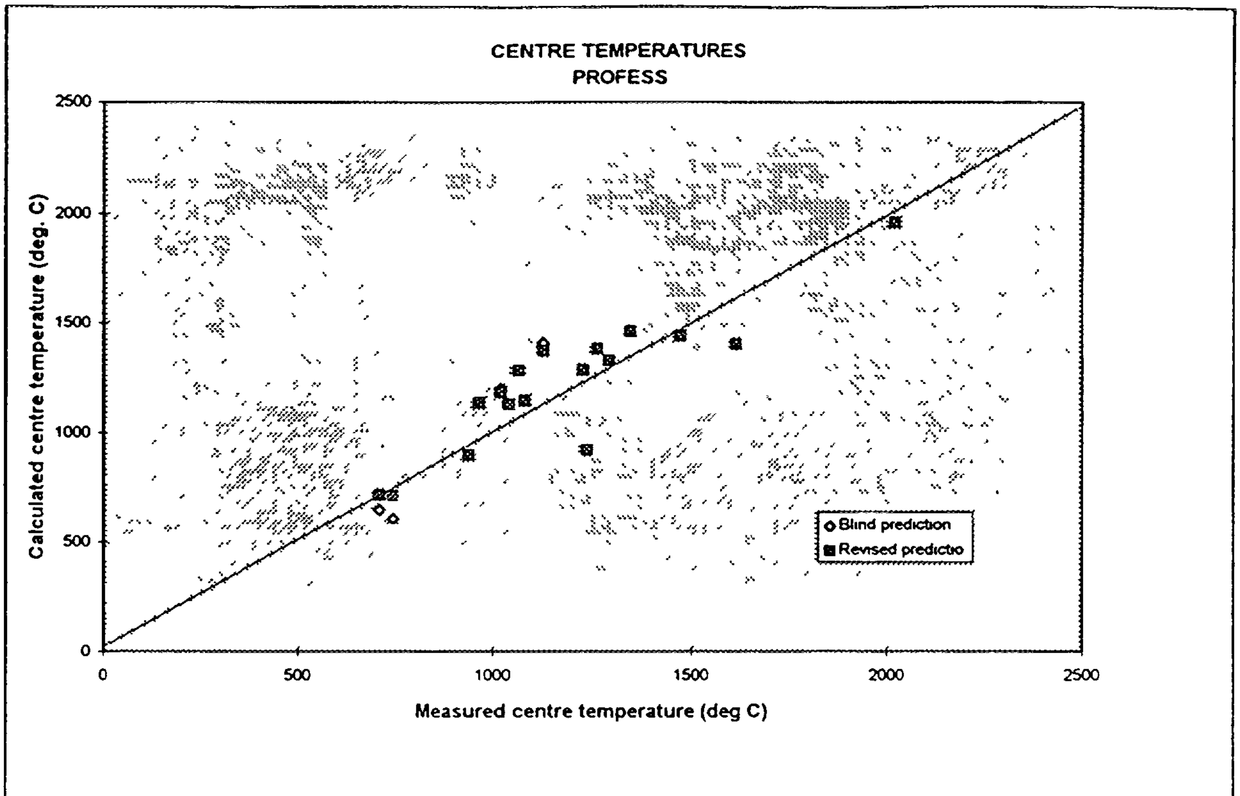


FIG. 73. Comparison calculation/experiment for the blind prediction and after code modification – METEOR/TRANSURANUS.



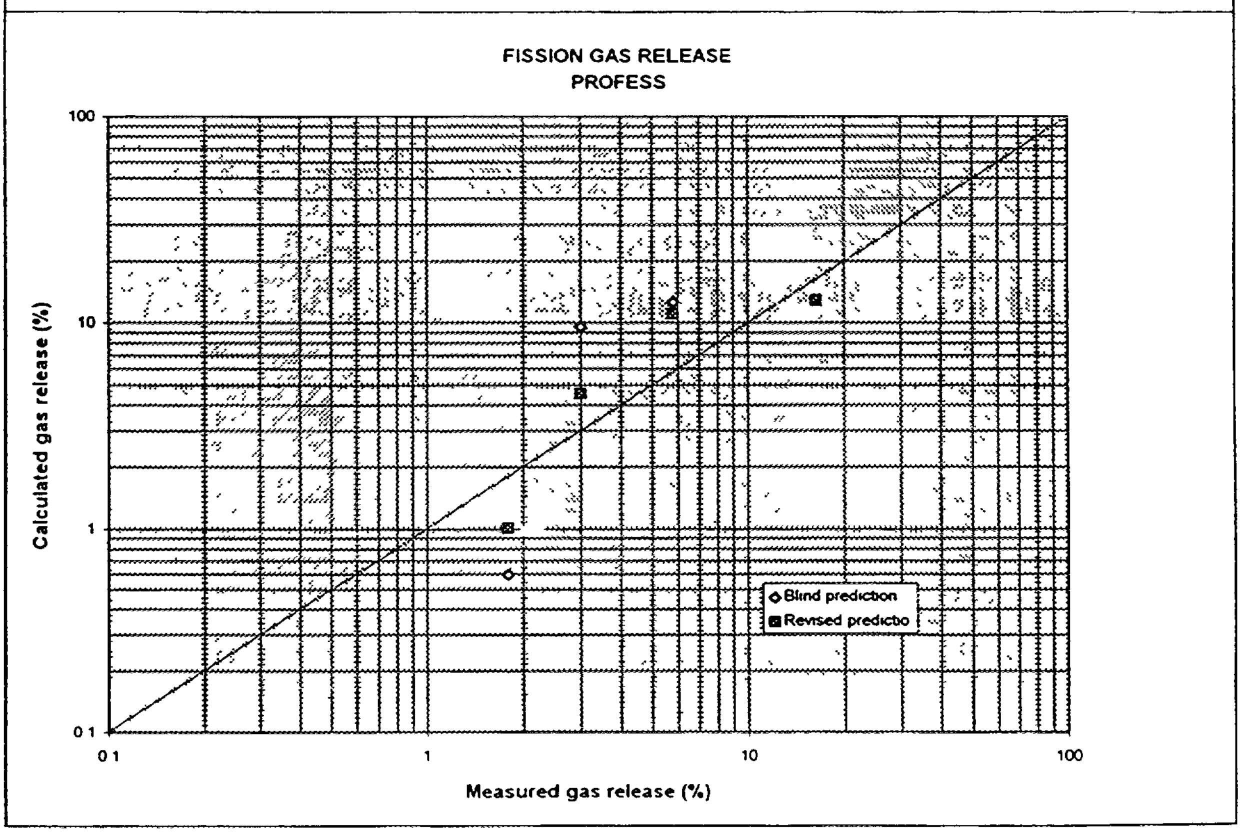
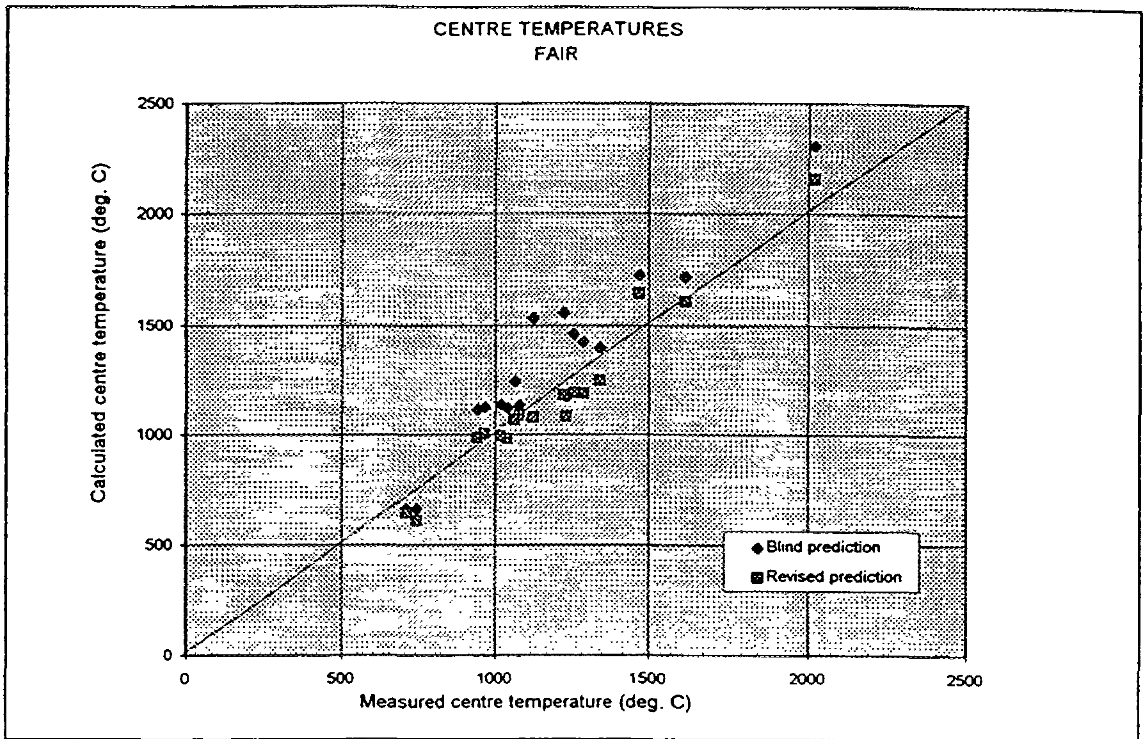


FIG. 74. Comparison calculation/experiment for the blind prediction and after code modification – PROFESS.



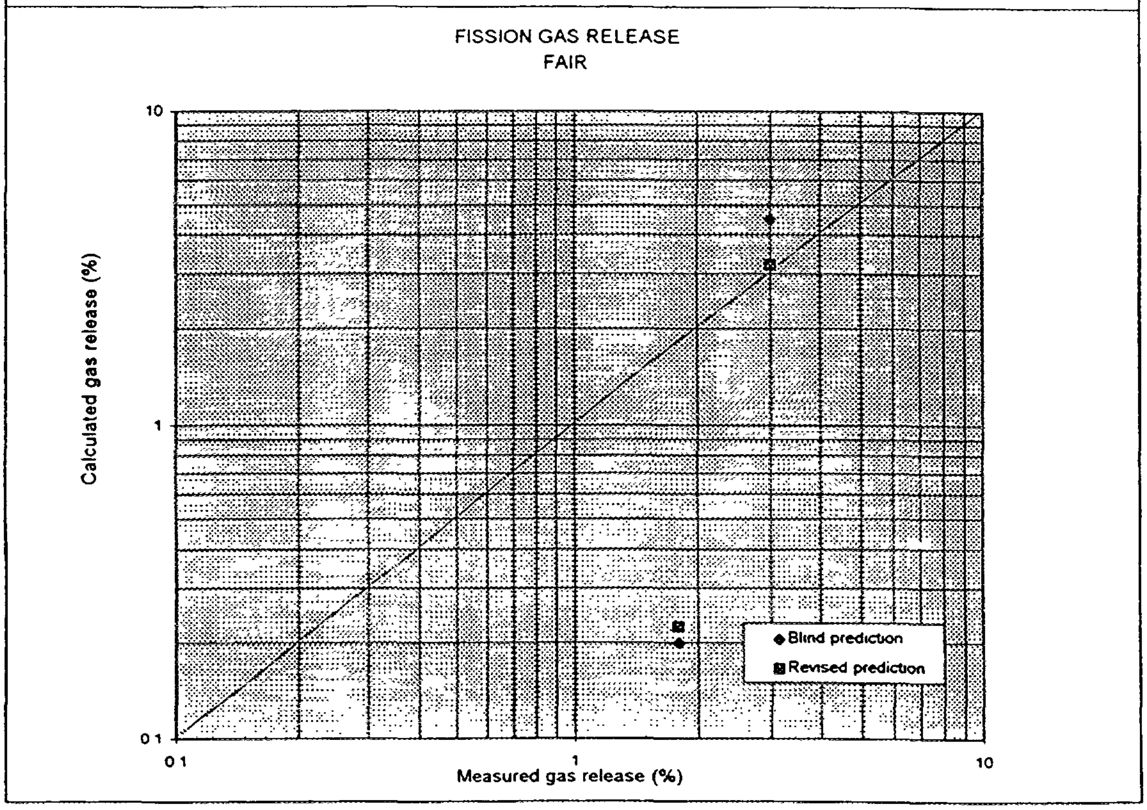


FIG. 75a. Comparison calculation/experiment for the blind prediction and after code modification – FAIR.

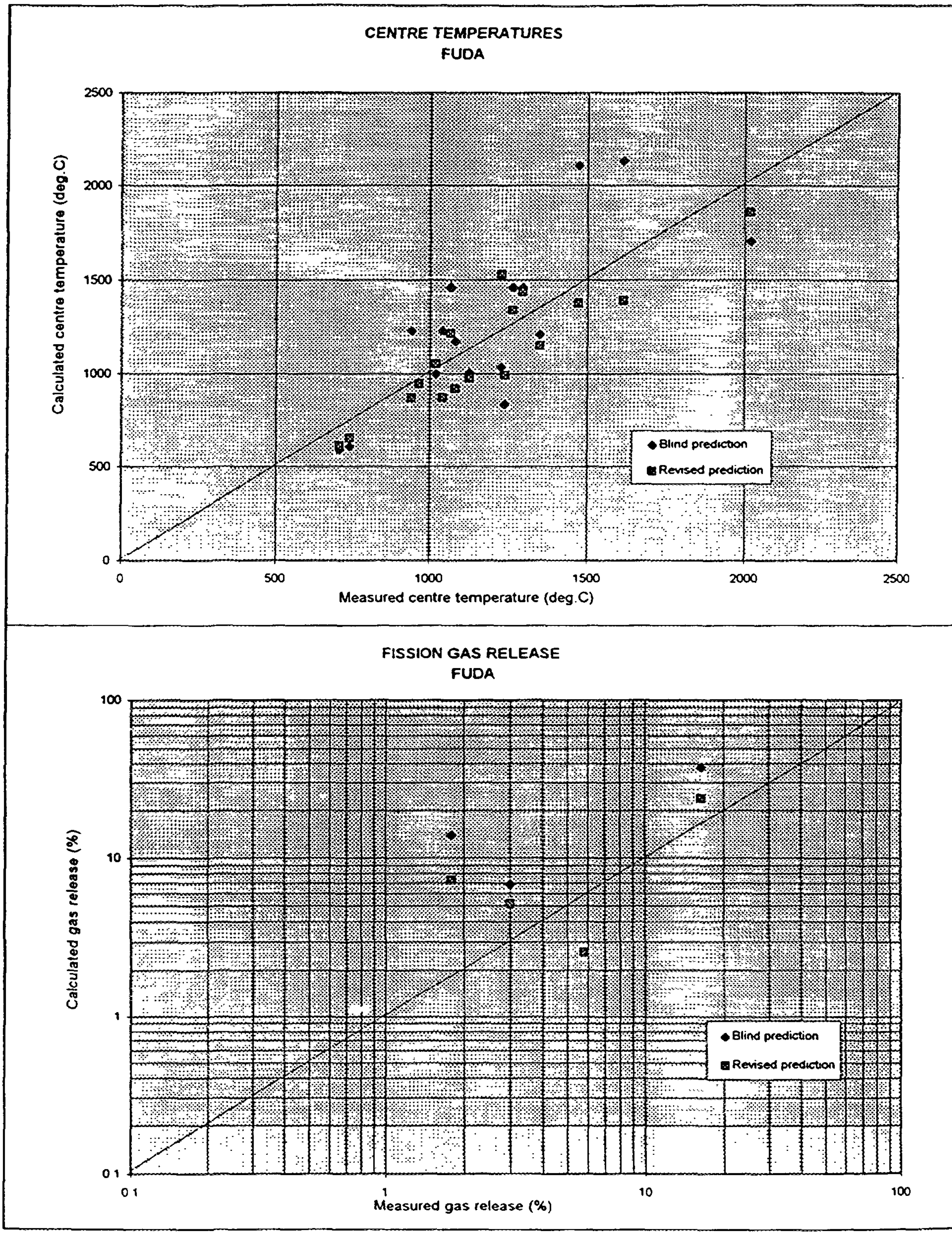


FIG. 75b. Comparison calculation/experiment for the blind prediction and after code modification - FUDA.

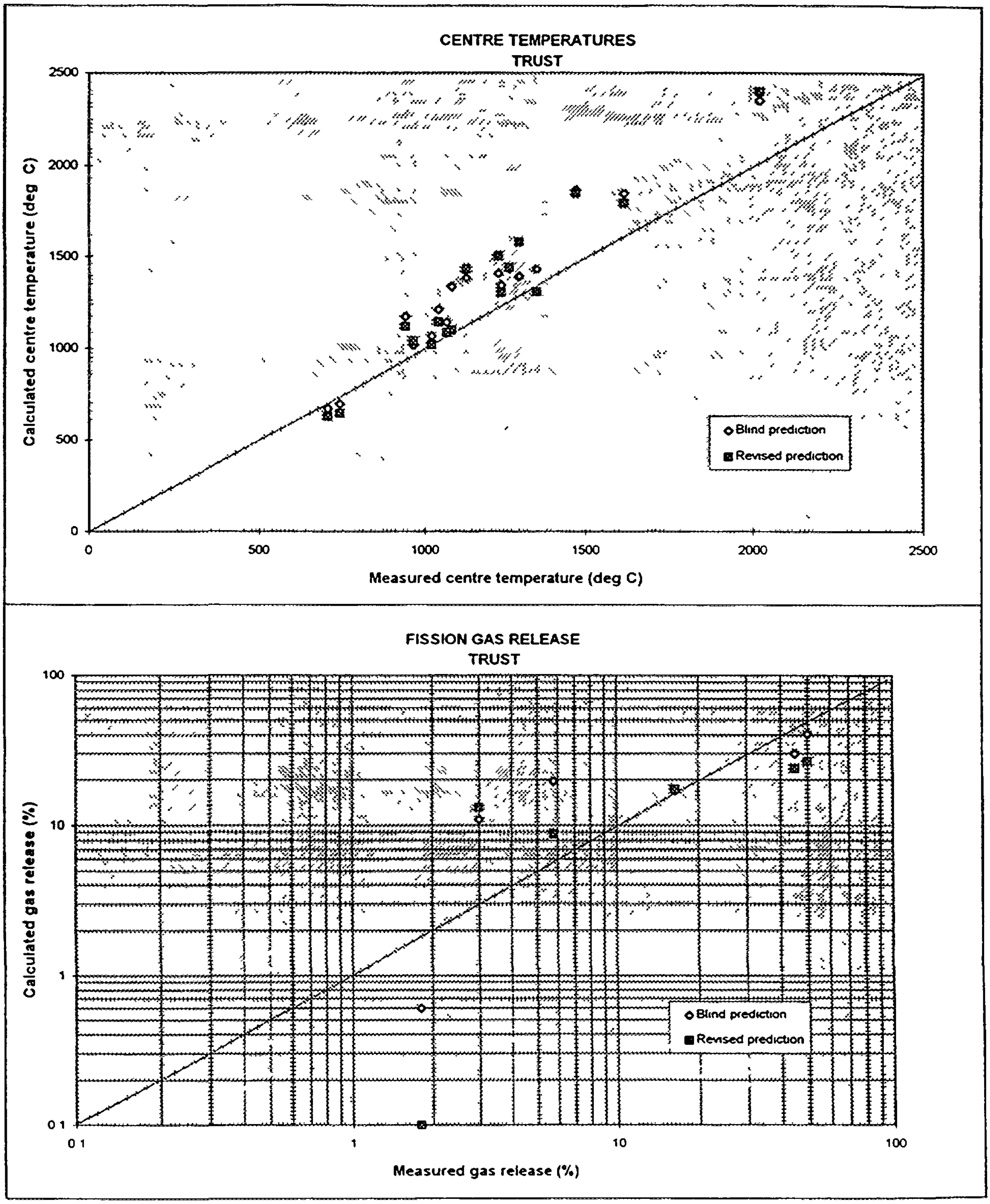


FIG. 76. Comparison calculation/experiment for the blind prediction and after code modification – TRUST.

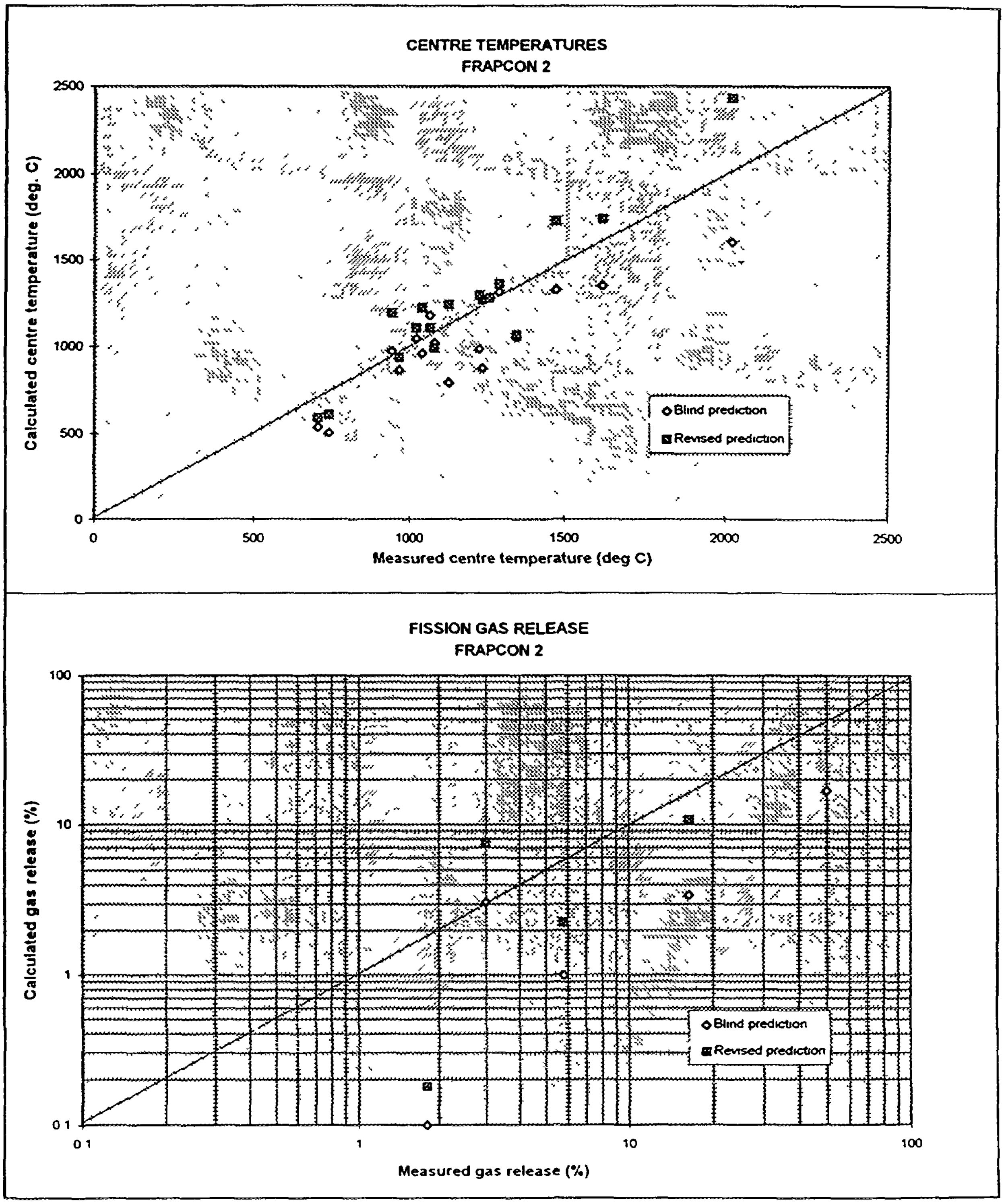


FIG. 77. Comparison calculation/experiment for the blind prediction and after code modification – FRAPCON 2.

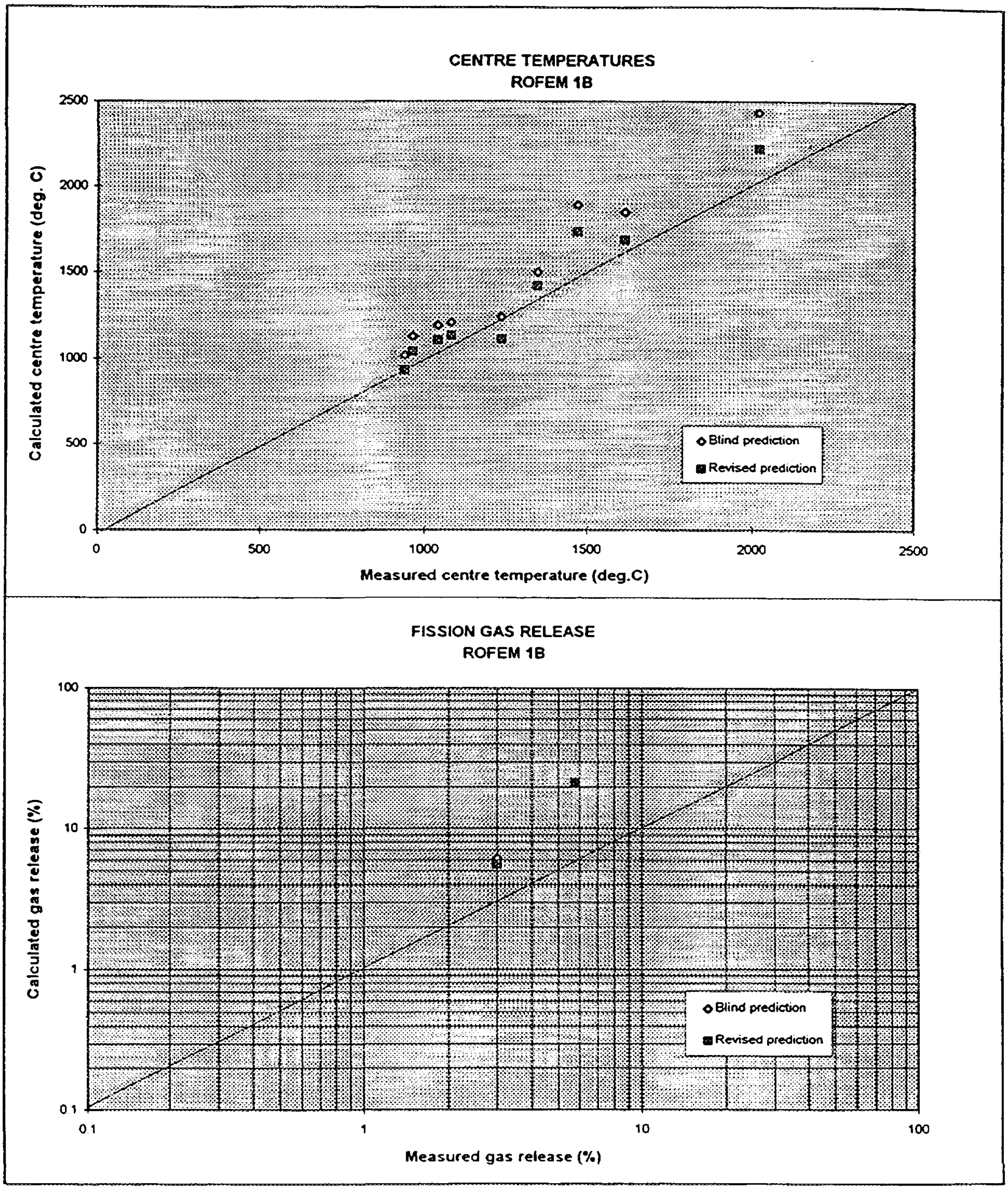


FIG. 78. Comparison calculation/experiment for the blind prediction and after code modification – ROFEM 1B.

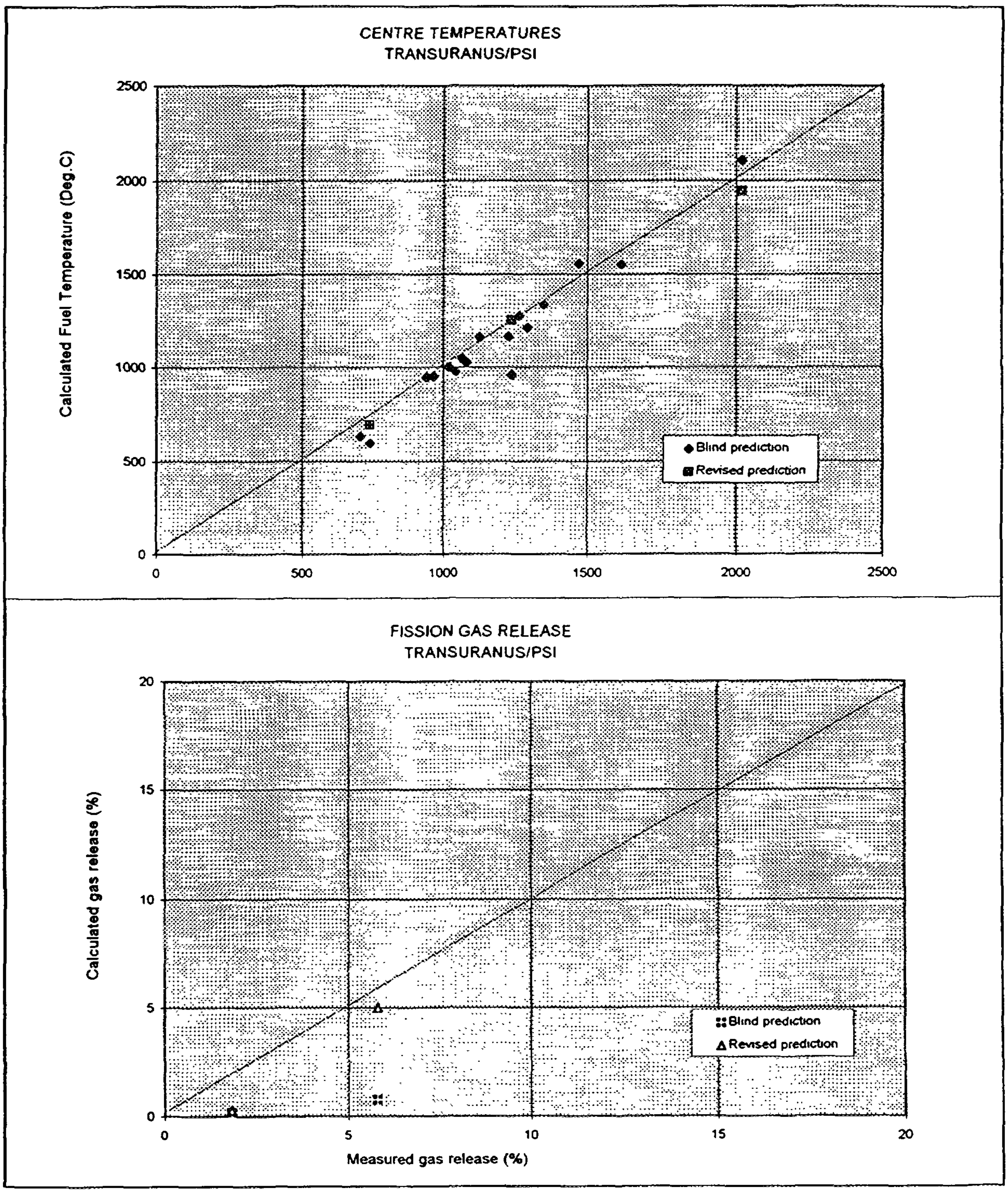


FIG. 79. Calculation against experiment for fuel centre temperature and fission gas release - TRANSURANUS/PSI.

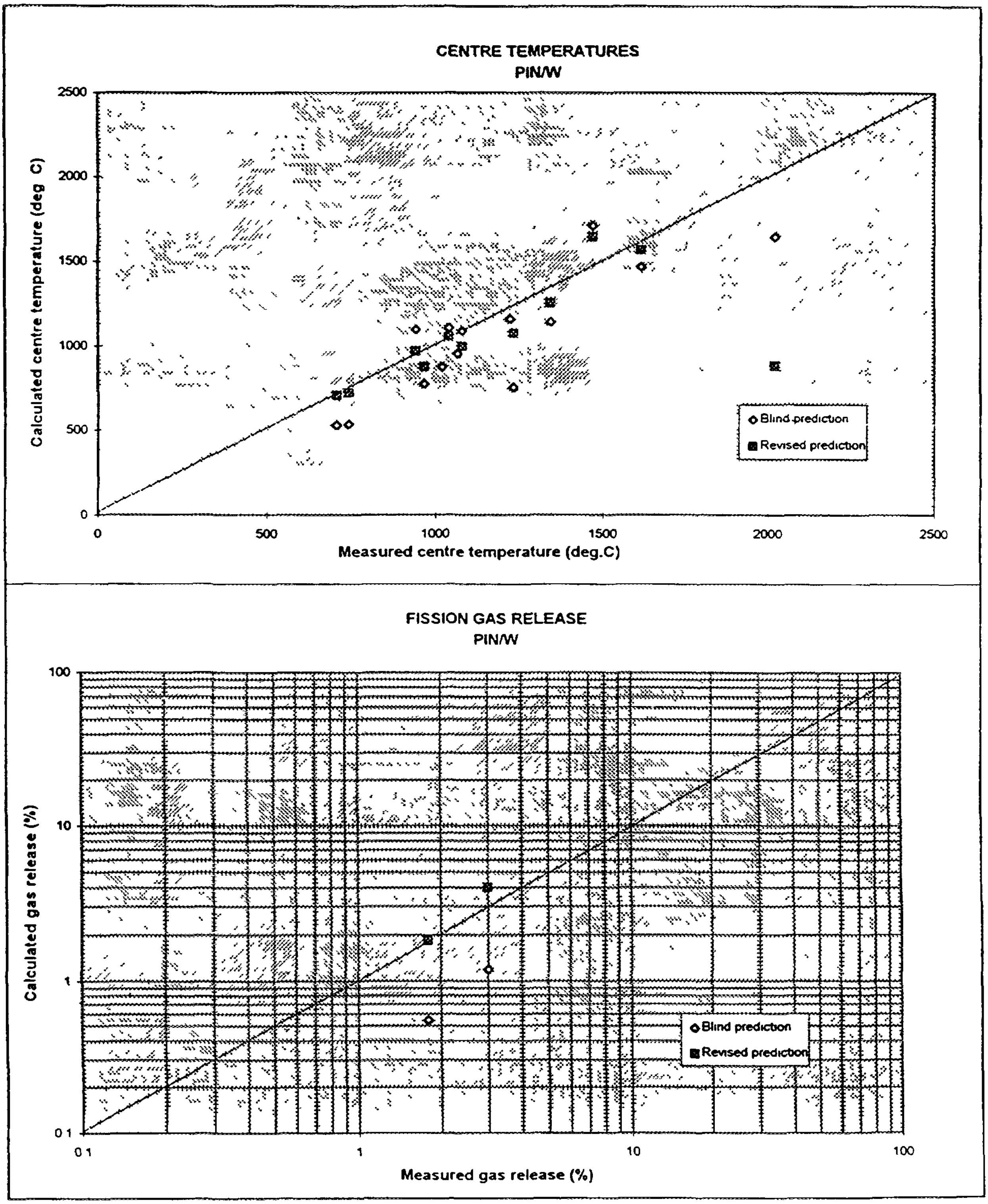


FIG. 80. Comparison calculation/experiment for the blind prediction and after code modification – PIN/W.

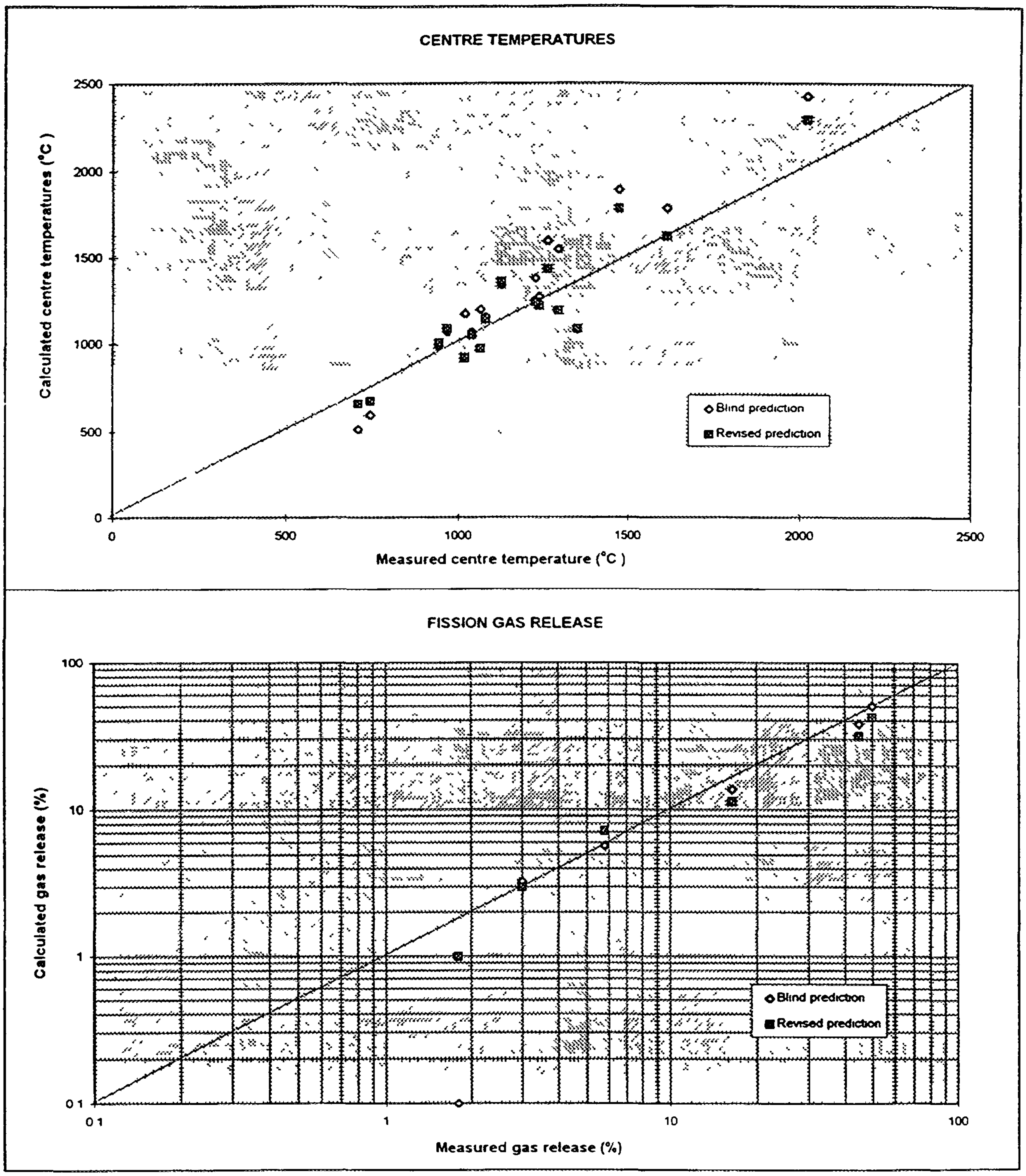


FIG. 81. Comparison calculation/experiment for the blind prediction and after code modification – ENIGMA/BNFL.

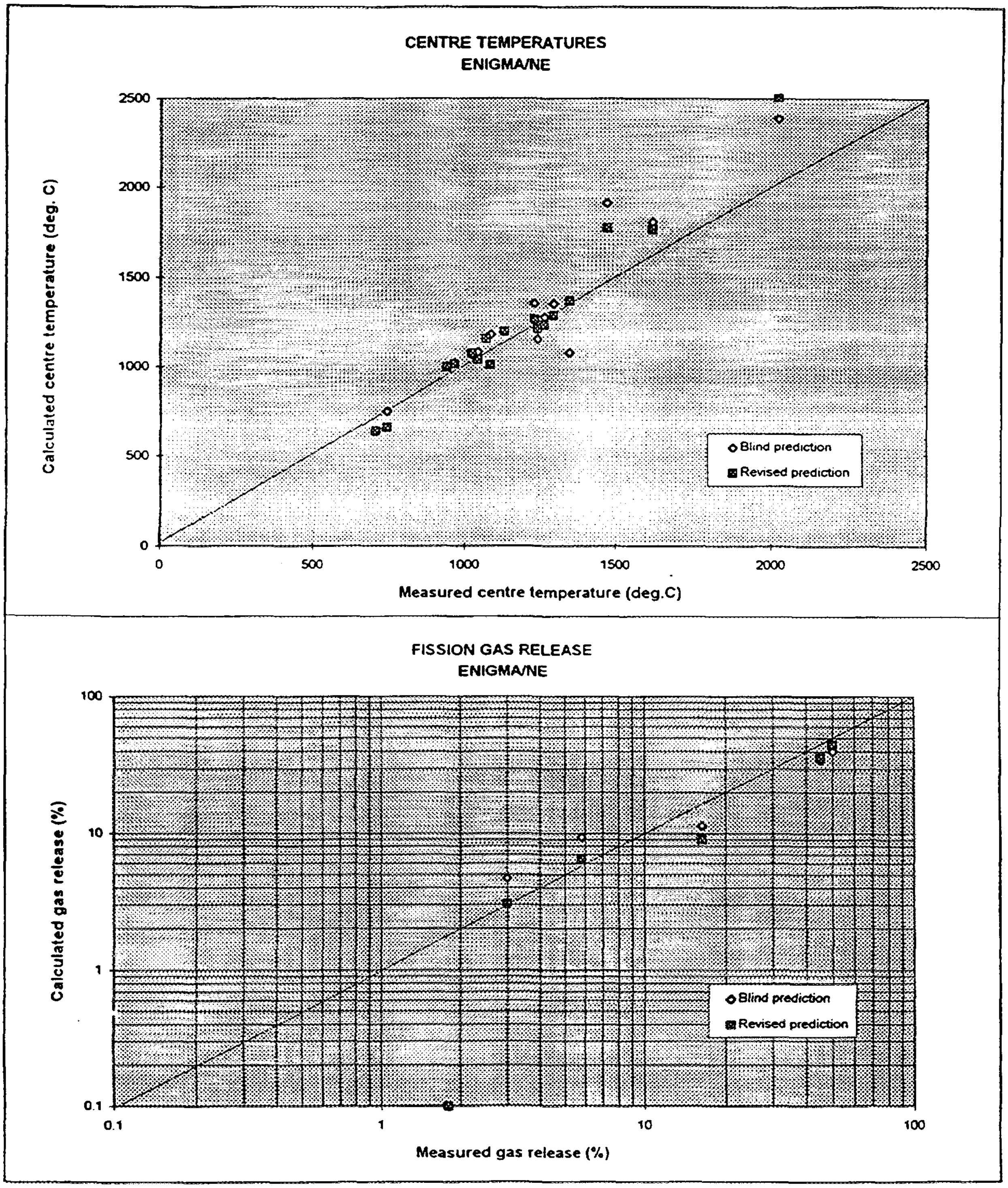


FIG. 82. Comparison calculation/experiment for the blind prediction and after code modification – ENIGMA/NE.

Appendix I

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Appendix II

DESCRIPTION OF CODES USED IN THE FUMEX CODE COMPARISON EXERCISE

The codes are presented in the order below as given in Table II

Country	Organization	Code
Argentina	CNEA	BACO
Bulgaria	INRNE	PIN micro
Canada	AECL	ELESIM.MOD11
Finland	VTT	ENIGMA 5.8f
France	EdF	TRANSURANUS-EdF 1.01
France	CEA/DRN	METEOR-TRANSURANUS
CEC	ITU	TRANSURANUS
India	BARC	PROFESS
India	BARC	FAIR
India	NPC	FUDA
Japan	NNFD	TRUST 1b
Japan	CRIEPI	EIMUS
China	CIAE	FRAPCON-2
Romania	INR	ROFEM-1B
Swiss	PSI	TRANSURANUS-PSI
Czech Republic	NRI Rez	PIN/W
United Kingdom	BNFL	ENIGMA 5.2
United Kingdom	NE	ENIGMA 5.8 D
Russia	IIM	START 3

CODE NAME : BACO

Date: 04-04-1993

Version : 2.2

Development Team :

Dr. Eduardo J. Savino
Lic. Armando C. Marino
Ing. Silvio Terliski
Ing. Horacio Nassini
Dra. Alicia Denis
Dra. Rosa Piotrkowski

Originally based on : BACO 1.0

Address :

Av. Libertador 8250
1429 Buenos Aires
Argentina

OBJECTIVES : Quasi two-dimensional thermo-mechanical description of a fuel pin.

MAIN ASSOCIATED OPTIONS : Steady state and transient analysis.

LANGUAGE : Fortran 77

Running on : IBM PC-486

Number of instructions : 8000

Typical Running time : 1-20 min.

Links with other tools : Data post-processing with a program set developed using **QBasic 4.5 (or Visual Basic under Windows 3.1), and worksheet Quattro Pro 4.0** (or the corresponding version for **Windows 3.1**).

Type of NUMERICAL treatments :

- Cylindrical symmetry.
- For the numerical modeling the hypothesis of axial symmetry and modified plane strain (constant axial strain) are adopted. The three-dimensional stress-strain problem to be reduced to a quasi-two-dimensional problem.
- Pellet and clad are divided into circular concentric rings.
- Behavior equations integrated with a finite difference scheme.
- Fuel pin irradiation history divided into subsequent finite time steps for the temporal integration.

MECHANICAL treatments :

It is assumed that during the time interval (t_0, $t_0 + \delta t$), the strain-stress increments can be expressed as the superposition of the strain-stress increments due to different existing deformation mechanisms.

Defined the strain-stress state at time t_0, at the corresponding time $t_0 + \delta t$ (with δt very small), it's possible calculate as follows:

$$\varepsilon = \varepsilon_0 + \delta\varepsilon$$

where :

ε_0 : are a stress-strain magnitude at t_0, and

$\delta\varepsilon$: the corresponding time step variation.

The equations to be integrated are, essentially, the compatibility equation of each ring, the equilibrium equation, and the Hooke's elastic plastic constitutive equations, subject to the appropriate boundary conditions. That constitutes a system of seven coupled differential equations. The finite differences approximation lead to a nonlinear system of algebraic equations, that system is linearized through a Taylor expansion. For a given time increment, the main stresses are calculated by direct matrix inversion of the equations.

THERMAL treatments :

The temperature distribution in pellet and cladding is obtained by solving Fourier's equation for steady state heat transmission at each time step, the proper geometry of the pin is updated.. The boundary condition is the temperature at the cladding external surface.

MAIN models :

UO$_2$ pellet :

- Thermal and physical Properties: MATPRO 09
- Linear thermal expansion : Roth-Halteman, NUMEC-2389-9
- Elastics constants, creep laws and fuel cracking stress: Matthews, AERE-M 2643
- Creep (alternative model) : Model developed by Liu-Bement
- Restructuring : Freeburn et al., EPRI NP-369
- Thermal conductivity and thermal radiation Properties: Model of Brandt-Neuer
- Swelling: Gittus's model.
- Densification : Own model or a model of Assmman-Stehle
- Cracks opening : Own model.
- Fission gas release: Empirical model.

Zircaloy cladding :

- Linear thermal expansion of Zry : Scott, WCAP-3269-41
- Elastic constants : Northwood et al.
- Plasticity laws : Casario-Santinelli, CNEA
- Creep: Ross-Ross and Hunt model.
- Growth under irradiation : Elbel and Gobel
- Thermal conductivity : Scott, WCAP-3269-41

- Gap pellet-clad thermal conductance : Ross-Stoute

DOMAIN OF USE :

- PHWR, may be extended for particular applications.

VALIDATION FIELD :

- Harriague, Savino - RES Mechanica 16(1985)193-230
- Preusser, Harriague, Savino - Report RTDA-97-83
- Harriague, Savino, Baigorria, de Grande - Proceedings SMiRT, C1/7, Chicago (1983)
- Forlerer, Coroli, Harriague, Savino - Meeting A.A.T.N., Bariloche, (1983)
- Harriague, Savino - International Seminar on Heavy Water Reactor Fuel Technology - IAEA/CNEA, Bariloche (1983)

REFERENCES :

- S. Harriague, G. Coroli, E. Savino - 5th SMiRT Conf. (1979), Paper D1/1
- S. Harriague, G. Coroli, E. Savino - BACO (BArra COmbustible), a computer code for simulating a reactor fuel rod performance" - *Nuc. Eng. & Des.* 56(1980)91-103
- S. Harriague, E. Savino, G. Coroli, F. Basombrío, G. Sánchez Sarmiento - "Theoretical fuel element modeling at CNEA" - *Nuc. Eng. & Des.* 56(1980)83-89
- S. Harriague ,D. Aguero, I. Lopez Pumarega, A. C. Marino - "Prediction of the influence of material properties on fuel rod behavior" - IAEA-TC-578/13
- S. Harriague ,D. Aguero, I. Lopez Pumarega ,A.C. Marino "Modelado de barras combustibles en CNEA - Estado actual y aplicaciones" - XIV Reunión científica de la Asociación Argentina de Tecnología Nuclear, (octubre/1986)

CODE NAME:
PIN-micro: A Computer Code for Modelling of LWR Fuel Rod Thermomechanical Behaviour

Version:

NEA Data Bank-version

Originally based on:

GAPCON-THERMAL-2, PIN-01

Date: 31.10.1991

Development Team:

Pazdera F., Valach M., Strijov P., (original authors)
Stefanova S., Manalova M., Vitkova M., Ivanov K., Haralampieva Z.

Address:

Nuclear Research Institute, Rez, Czech Republic
Institute for Nuclear Research and Nuclear Energy, Sofia, Bulgaria

OBJECTIVES:

Knowledge and experience on fuel rod behaviour modelling during normal operation and operational transients, important for:

MAIN ASSOCIATED OPTIONS:

- optimization of operational regimes
- increase of availability
- burnup increase
- use of new optimized type of fuel rods
- study of fuel rod failure causes

LANGUAGE:

MS Fortran 4.01 or later

Running on:

IBM PC compatible

Number of instructions:

~ 3000

Typical running time:

Demonstration test case for NEA Data Bank consumes on IBM PC 386/387 25MHz 10 min and 51 min on the IBM at 286/287 16 MHz.

Type of NUMERICAL treatments:

Fuel rod temperature response is solved by using one dimensional finite element method combined with weighted residuals method.

Thin shell approach for cladding stress-strain prediction.

Up to 20 radial segments, 10 axial nodes, 200 time steps.

Brief description of
MECHANICAL treatment

Simplified mechanistic approach

Fuel:	Thermal expansion Swelling (G-T-2, COMETHE-IIIG Densification (G-T-2 - IAE) Cracking and relocation Stack elongation Restructuring (G-T-2, MATPRO-9) Formation of central hole	Cladding:	Thin shell approach Thermal expansion Elastic strain Primary and secondary creep (IAE-models) Irradiation growth (anisotropic) (IAE) Elongation

Gap: Open gap
Contact pressure

THERMAL treatment:

One dimensional thermal conductivity solution

Radial fuel and cladding temperature distribution prediction (G-T-2)
Gap conductance/contact thermal resistance (G-T-2, FRAP-S)
Power distribution input
Radial neutron flux depression (radial power distribution)
Coolant boundary conditions

MAIN MODELS:

Other models:

Fission gas release (G-T-2 with MATPRO constants, MATPRO-9 with Czech constants)

DOMAIN OF USE:

Only steady-state solution

No cladding failure criterion

No model for axial fuel-cladding interaction

No specific models for higher burnup (will be developed)

VALIDATION FIELD:

(LWR fuel rod steady state performance (as GAPCON-THERMAL-2 version))

VVER fuel rod steady state performance up to ~30000 MWd/tU and higher

based on:

VVER specific in-pile and post-irradiation experiments: centerline temperature, internal gas pressure, fission gas release.

KEY REFERENCES:

1. PAZDERA F., VALACH M., STRIJOV P., Programs PIN-02 and PLOT-1., UJV-5384, 1980.
2. PAZDERA F., VALACH M.,: User's Guide for PIN: A Computer Program for the Calculation of the Thermal Behaviour of an Oxide Fuel Rod. UJV-6124T, 1982.
3. STRIJOV P., YAKOVLEV V., DUBROVNIN K., PAZDERA F., VALACH M., et al. An improved version of the PIN code and its verification. IAEA Technical Committee Meeting on Water Reactor Fuel Element Computer Modelling in Steady-State, Transient and Accident Conditions. Preston, England, 19-22 September 1988. IWGFPT/32.
4. STRIJOV P., et al., Research of VVER-440-type Fuel Rods in MR-reactor. IAEA International Symposium on Improvements in Water Reactor Fuel Technology and Utilization, Stockholm, Sweden, 15-18 September 1986.
5. STRIJOV P., DUBROVNIN K., YAKOVLEV V., PAZDERA F,: Computer and experimental VVER fuel rod modelling for extended burnup. IAEA Technical Committee Meeting on Fuel Performance at High Burnup for Water Reactors held in Studsvik, Sweden, 5 - 8 June 1990. IWGFPT/36.

CODE NAME: ELESIM

Date: 1993 June

Version: MOD 11 (Developmental)

Development Team
Fuel Modelling Group
Fuel Engineering Branch

Originally based on:
The first version of ELESIM was released in 1970. Subsequent versions have followed since then.

Address:
AECL Research
Chalk River Laboratories
Chalk River, Ontario
Canada K0J 1J0
Phone: (613) 584-3311
Facsimile: (613) 584-4200

OBJECTIVES: The code calculates parameters that describe the performance of CANDU-type fuel elements (collapsible, Zircaloy cladding with UO_2 pellets) during irradiation under normal operating conditions. These parameters are: fuel surface and centre temperature, sheath temperature, sheath elastic and plastic strain, stable fission gas release (percentage and volume), fuel to sheath gap, and fuel conditions at the end of life.

MAIN ASSOCIATED OPTIONS: The code outputs fission product inventory using ANS 5.4 if requested.

LANGUAGE: FORTRAN-77 Compatible

Running on:
CDC CYBER 990E
Digital Equipment VAX Computers

Number of Instructions:
Approximately 3300 lines of code.

Typical running time:
94 Burnup Interval Problem - 2.0 minutes on a VAXstation 4000 Model 60 (12.0 VUP rating - by comparison a VAX 11/780 has a 1.0 VUP rating)

Links with other tools: A high temperature transient CANDU fuel behaviour code called ELOCA requires initial conditions from ELESIM.

Type of NUMERICAL treatments: ELESIM uses a one-dimensional, axisymmetric model to simulate the thermal-mechanical behaviour of the fuel. Fuel calculations are performed at 100 concentric, equi-spaced, annuli. The NOTPAT model for fission-gas release is used which accounts for thermal-mechanical feed-back effects. Calculations are performed on an incremental basis assuming steady-state conditions prevail for a given element power.

Brief description of
MECHANICAL treatment:
ELESIM[1-5] uses a pellet stress-strain model that consists of plastic core and an elastic cracked region delineated by a "plasticity temperature". If a given annulus reaches the "plasticity temperature" it is assumed to be perfectly plastic. The plastic core is subjected only to gas pressure while the cracked region of the pellet is also subjected to the interface pressure between the fuel and the sheath. In addition to fuel thermal expansion, the program also accounts for fuel swelling due to intergranular fission gas bubbles and solid fission products. Fuel densification is treated empirically as a function of burnup and temperature. Calculation of the internal gas pressure is based on the voidage create/consumed by thermal expansion, fuel-to-sheath roughness, and radial and axial clearance.

Fuel expansion, swelling and densification, internal gas pressure, external coolant pressure, and creep are accounted for in the calculation of sheath strain. The sheath strain calculation is based on a generalized strain-rate equation that accounts for irradiation dose at temperatures of approximately 573 K.

THERMAL treatment:
A finite-difference method is used to calculate the one-dimensional, radial heat conduction equation starting with specified values of the coolant temperature and sheath-to-coolant heat transfer coefficient The local volumetric heat generation rate is calculated accounting for radial flux depression across the pellet. A semi-mechanistic model, based on CANDU fuel performance, is used to calculate the fuel-to-sheath heat transfer coefficient. Changes in fuel thermal conductivity with temperature, expansion/densification, and burnup are accounted for

MAIN MODELS:
The main models used in ELESIM are

Fission Product Release: A modification of the NOTPAT model[5] is used to calculate stable fission gas release. NOTPAT calculates the number of stable fission gas atoms diffused to, and released from the grain boundaries during a burnup interval, the fraction of grain boundary area swept by grain boundary movement, the number of atoms produced up to the end of the present interval, and the grain size at the end of the burnup interval. Intragranular diffusion of gas atoms is treated using an effective diffusion coefficient and idealizing the grain shape to be spherical. Diffusion of gas atoms existing in the fuel grains prior to the time-step, and new gas atoms created during the time-step are accounted for Gas atoms arriving at the grain boundary are assumed to enter intergranular, lenticular shaped bubbles Venting of the gas atoms to the voidage proceeds until the pressure inside the gas bubbles equals the gas pressure in the voidage A constant yield is assumed for fission gas, which is assumed to consist only of stable xenon and krypton isotopes

Thermal Conductivity: Thermal conductivity is calculated using the MATPRO-9[6] correlation To simulate low-temperature-irradiation degradation, a constant conductivity is assumed below 727 K. Recently, the MATPRO-9 correlation was modified to account for degradation due to burnup-related solid fission product buildup. The modification is based on SIMFUEL data.[7]

Temperature As described above under thermal treatment.

Fuel-To-Sheath Heat Transfer: The fuel-to-sheath heat transfer coefficient is calculated using a modified Ross-Stoute model.[8,9] It takes into account the temperature-jump distance in the gap and uses a square-root dependence on interfacial pressure for the solid-component of heat transfer. It also models the gas properties of the filling gas (He or Ar) and the fission gas released during irradiation.

Pellet Stress and Sheath Strain Models: As described above under mechanical treatment.

Activity Release: An approximation of the active isotope inventory and associated activities is made using the ANS-5.4 model.[10]

DOMAIN OF USE:
ELESIM has been extensively used for design and licensing of CANDU fuel. It is presently used to provide initial conditions for ELOCA, a code for modelling the behaviour of CANDU fuel during high temperature transients

VALIDATION FIELD:
The program has been validated against a CANDU fuel database that includes volume of fission gas release, UO_2 grain size and radius of grain growth, fuel volume, and sheath strain from post-irradiation examinations. It has also been validated against a limited number of in-reactor measurements of gas pressure, sheath strain, and central fuel temperature.

KEY REFERENCES:

1) M.J.F. Notley, "Calculation of Fission-Product Gas Pressure In Operating UO_2 Fuel Element", *Nucl. Appl.*, 3, 334, (1967).

2) M.J.F. Notley, "A Computer Program To Predict The Performance of UO_2 Fuel Elements Irradiated At High Power Outputs To A burnup of 10 000 MWd/MTU", *Nucl. Appl.*, 9, 195 (1970).

3) M.J.F. Notley, "ELESIM: A Computer Code For Predicting The Performance of Nuclear Fuel Elements", *Nucl. Appl.*, 44, 445 (1979).

4) M.J.F. Notley and I.J. Hastings, "A Microstructure-Dependent Model For Fission Product Gas Release and Swelling In UO_2 Fuel", *Nucl. Eng. Des.*, 56, 163 (1980).

5) V.I. Arimescu and W.R. Richmond, "Modeling CANDU-Type Fuel Behaviour During Extended Burnup Irradiations Using A Revised Version Of The ELESIM Code", Presented at the IAEA Technical Committee Meeting on Fission-Gas Release and Fuel-Rod Chemistry Related to Extended Burnup, Pembroke, Ontario, Canada, 1992 April 27 - May 01, AECL Report AECL-10622, 1992 May.

6) "MATPRO-Version 09. A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behaviour," TREE-NUREG-1005, U.S. Nuclear Regulatory Commission (1976).

7) P.G. Lucuta, et al., "Thermal conductivity of SIMFUEL", J. Nucl. Mater, 188, 198 (1992).

8) F.R. Campbell, L.R. Bourque, R. Deshaies, H. Sills, and M.J.F. Notley, "In-Reactor Measurement of the Fuel-To-Sheath Heat Transfer Coefficients Between UO_2 and Stainless Steel", AECL-5400, AECL Research (1977).

9) M.J.F. Notley, F.R. Campbell, I.J. Hastings, and H.E. Sills, "Fuel Modeling: Gap Conductivity, Gas Bubble Swelling and Fission Gas Release", ANS Topl. Mtg. Water Reactor Fuel Performance, St. Charles, Illinois (1977).

10) American National Standard, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel", American Nuclear Society, ANSI/ANS-5.4-1982.

COUNTRY: FINLAND

CODE NAME: **ENIGMA**

Version: 5.8f Oct 1992

Development team:

Kilgour WJ, White RJ, Turnbull JA; Nuclear Electric plc UK
(Bull WJ, Jackson PA, Palmer ID; British Nuclear Fuels plc)

OBJECTIVES:

Analysis of thermal and mechanical performance of an LWR fuel rod as function of burnup, including temperature distributions, fission gas and iodine release, elastic plastic and creep strains.

MAIN ASSOCIATED OPTIONS:

Options for reactor change, rod internal gas change, two axial gas communication models, and thermal transients

VVER specific materials properties

Analysis of thermal and mechanical performance of a GCR fuel rod.

LANGUAGE: FORTRAN 77

Running on: Hewlett Packard Apollo 700 series work station

No. of instructions: 8000; fully modular, 90 subroutines

Typical running time: 20 to 300 CPU s

Type of NUMERICAL treatments

Finite difference approximations for thermal and mechanical models using multi-zone representation. Iterative inner loop for thermo-mechanical solution, outer loop with explicit treatment of non-linear processses.

THERMAL AND MECHANICAL TREATMENT:

1 ½ -D treatment for coolant energy, rod pressure and gas transport.
1-D axisymmetric mechanical calculation under assumption of generalized plane strain in both pellet and cladding. Pellet solution perturbed to take account of axial extrusion and strains from wheatsheafing. Pellet treated as non-isotropic but homogenous material with directionally dependent elastic constants, functions of the state of cracking. Plasticity and creep obey the Levi-Mises equations for isotropic material; relation to uniaxial properties through the von Mises generalized stress. Stress equilibrium and strain-displacement relationships approximated by finite difference scheme.

MAIN MODELS:

The principal processes and properties modelled include:

Fuel thermal expansion
Fuel thermal conductivity (burnup dependent)
Transient thermal behaviour
Fuel creep (irradiaion, linear, power law)
Sintering of porosity
Solid fission product swelling
Fuel grain growth
Burnup dependent radial power distribution (RADAR module)
Fuel axial extrusion

Clad thermal expansion
Clad thremal conductivity
Clad primary and secondary creep
Clad plasticity
Clad axial growth
Clad external oxidation

Pellet wheatsheafing
Stress concentration over fuel cracks
Stable fission gas release and swelling (diffusion, bubble formation and resolution)
Iodine131-release
Axial gas mixing
Fuel-to-clad gap conductance
Clad damage accumulation
Clad fatigue

Coolant conductivity, specific heat, visvosity

DOMAIN OF USE:

Domain of applications in Finland:

Best-estimate type fuel thermal and mechanical performance evaluations for BWR and VVER type reactors, assessment of changes in design and operational data of various fuel types, obtaining initial data for transient and accident analyses. Support of licencing.

Analysing and qualification of data from test reactor fuel irradiations; planning of such irradiations.

VALIDATION FIELD:

International test programmes (Battelle HBEP, EPRI NFIR, BN Tribulation, Studsvik, Risø, Halden); commercial PWR irradiations

In Finland: additional BWR, VVER and test reactor irradiations

KEY REFERENCES:

WJ Kilgour, JA Turnbull, WJ White, AJ Bull, PA Jackson, ID Palmer, Capabilities and Validation of the ENIGMA Fuel Performance code. Proceedings of the ANS-ENS International Topical Meeting on LWR Fuel Performance. Avignon France April 21-24, 1991. pp. 919-928.

CODE NAME : TRANSURANUS - EDF(*)

Date : June 1993

Version : 1

Development team : EDF SEPTEN MS/SC

Originally based on :
TRANSURANUS Version V2M1J91 (March 1992)

Address :
12-14 Avenue Dutriévoz
F-69628 VILLEURBANNE
FRANCE

OBJECTIVES :

- Simulation of the whole behaviour of a fuel rod during its irradiation in a power reactor.
- Simulation of analytical experiments or rod irradiations in experimental reactors.
- Verification of fuel behaviour against different criteria, and proposal of plant operation specifications which are consistent with Pellet-Clad Mechanical Interaction phenomena.

MAIN ASSOCIATED OPTIONS :

- Thermomechanical treatment of fuel and cladding with axial and radial discretization (the fuel rod is considered as an axisymmetric structure).
- All physical phenomena (creep, thermal expansion, cladding irradiation growth, fuel swelling, pellet fracturation, densification, fission gas production and release, pellet-cladding interaction ...) are taken into account.

LANGUAGE : FORTRAN-77

Running on :
IBM, CRAY and work stations with UNIX environment

Number of instructions : about 30,000

Typical running time :
on CRAY machine : 30 axial slices, 5 PWR cycles (55000MWd/tM) : about 200 seconds

Links with other tools :
CRACO : fuel rod database

Type of numerical treatments :

Thermics : finite differences with implicit calculation method (Newton-Raphson for steady-state conditions, Crank-Nicholson for transient conditions), with a Regula-Falsi technique to solve the equations. Time-dependent terms of energy equations are taken into account during transients.
Mechanics : implicit calculations for elastic and creep strains, explicit calculations for all other strains. Utilization of a variable multi-zone concept for the semi-analytic solution of the local mechanical equations.
Physics : explicit incremental treatment for all phenomena (except fission gas release).
Coupling between thermics and mechanics by the means of a convergence test on either pellet-clad gap width or contact pressure.

Brief description of MECHANICAL treatment :

Fuel and clad stresses and strains are calculated in the three directions (axial, tangential and radial) which are supposed to be the principal directions for the stress and strain tensors, with the hypothesis of generalized plane strains.
Phenomena involved :
- in the cladding : elastic strains, creep, thermal expansion, irradiation growth, and optionnally, cladding weakening (thickness reduction) due to waterside corrosion.
- in the fuel : elastic strains, densification, solid and gaseous swelling, pellet fracturation (with downgraded mechanical properties), thermal expansion.
- between fuel and cladding during PCMI : perfect sticking is assumed in the axial direction ; the gap is set to zero ; reversible pellet fragment relocation due to cladding creepdown.
The mechanical properties of the materials (Young's modulus, Poisson's ratio, thermal expansion coefficient, ...) vary with temperature.
For cumulative creep strains in the cladding in varying conditions, the strain-hardening rule is assumed.

Brief description of THERMAL treatment :

Phenomena involved : heat generation and conduction in the fuel, heat transfer through the pellet-clad gap, cladding and waterside zirconia layer, heat transfer from cladding to coolant.
Axial heat transfers are neglected.
Models :
Fuel conductivity : Philipponneau's model (CEA) with modified burn-up dependence and correction factor for MOX fuel.
Gap conductance : URGAP2 model (Lassmann)
Cladding conductivity : MATPRO model.
Clad-coolant heat transfer : Dittus-Boelter correlation for liquid state, Jens-Lottes correlation for nucleate boiling. Optionnally, clad outer temperature may be prescribed.

MAIN MODELS :

Power radial depression : calculated curves depending on burn-up and fuel enrichment for LWRs, RADAR model with given values for fast leakage factor and resonance escape probability for experimental reactors (BR3, HALDEN, ...)
Fission gas release : URGAS model with both athermal and thermal FGR, the latter one with a burn-up dependent temperature threshold. EDF empirical correlations for diffusion coefficients.
Densification : three different models depending on fuel fabrication route (IDR, ADU or AUC), porosity distribution, grain size and resintering test results.
Pellet fragment relocation : reversible model depending on as-fabricated fuel-clad gap, linear power, with "accommodation" (reverse relocation) governed by contact pressure during PCMI.
Clad axial growth : EDF model with fast flux, fluence, irradiation temperature and material microstructure dependence.
Clad creepdown : EDF model with both thermal and irradiation creep, each of them with a primary creep and a secondary creep component.

DOMAIN OF USE :

PWRs and experimental reactors.
Normal and off-normal (power ramps) steady-state and transient operating conditions.
Various fuel designs : annealed or CWSR zircaloy cladding, UO2, mixed-oxide (MOX) or gadolinia (UO2-Gd2O3)(foreseen) fuel, solid or hollow pellets, ...

VALIDATION FIELD :

Interpretation of whole behaviour of irradiated fuel rods by comparison with experimental results : outer rod diameter, waterside corrosion thickness, fuel and cladding axial elongations, FGR measurements, internal end-of-life pressure and void volume, hydrostatic density, central temperature.

based on :

,O2 and MOX fuel rods irradiated in PWRs (FGA and SIEMENS fuel rods).
TRIBULATION programme (BN, W, FGA and BBR fuel rods).
OVER-RAMP and SUPER-RAMP programmes (W and KWU fuel rods).
TRANS-RAMP programme (FGA fuel rods).
PRIMO programme (MOX fuel irradiated in BR3).
CONTACT, GRIMOX (thermics in UO2 and MOX fuel).

KEY REFERENCES :

K. LASSMANN and H. BLANK :
"Modelling of fuel rod behaviour and recent advances in the TRANSURANUS code"
Nucl. Eng. Des. 106 (1988) p. 291.

K. LASSMANN :
"URANUS : a computer programme for the thermal and mechanical analysis of a fuel rod in a nuclear reactor"
Nucl. Eng. Des. 45-2 (1978) p. 325.

K. LASSMANN :
"TRANSURANUS handbook Version V2M1J91 (February 1992).

K. LASSMANN and F. HOHLEFELD :
"The revised URGAP model to describe the gap conductance between fuel and cladding"
Nucl. Eng. Des. 103 (1987) p. 215.

K. LASSMANN :
"A fast and simple iteration scheme for the temperature calculation in a fuel rod"
Nucl. Eng. Des. 103 (1987) p. 211.

P.T. ELTON and K. LASSMANN :
"Calculational methods for diffusional gas release"
Nucl. Eng. Des. 101 (1987) p. 259.

Date: 06/93

CODE NAME : METEOR/TRANSURANUS

Development team : CEA/DRN

Version: 0.3

Address:

CEN CADARACHE
DEC/SDC
bt 315
13108 St Paul lez Durance

Originally based on : TRANSURANUS

OBJECTIVES :

- Prediction of the behaviour of UO2 and MOX rods under standard and incidental .onditions in a French PWR.

- Precalculation and interpretation of analytical in-pile expériments

MAIN ASSOCIATED OPTIONS :

-Thermomechanical treatment of fuel and clad 1,5 D (2 or 3D calculations possible with TOUTATIS module)
-Coolant temperature may be calculated or imposed
-Axial force calculation: slip or noslip option.
-External clad corrosion, fission gas retention, U and Pu isotopic concentration are calculated
-Evolution of fuel and clad properties, respectively with burn up and fast fluence are taken into account.

LANGUAGE : FORTRAN

Running on : DEC station

Number of instructions: 30000

Typical running times :
10 mn for a standard case
up to 60000 MWd/TU.
(30 axial slices, 15 radial meshes)

Link with other tools :
Data Base CRACO
Fuel rod mechanics finite elements code
CASTEM 2000 (TOUTATIS module :PCMI).

Type of numerical treatments :
-Finite differences
-Thermal behaviour: implicit calculation in steady state conditions,
Crank-Nicholson resolution in transient conditions.
-Mechanic : implicit calculation for elastic and creep strains, explicit for other type of strains.
-Explicit treatment for physicochemical phenomena
-Integrated time step control

Brief description of

MECHANICAL treatment

Fuel and clad stresses and strains are calculated in the three directions (axial, tangential, radial)

Main hypothesis :
-Plane strain
-Craked fuel is considered as an isotropic material for which elastic properties are reduced.
- The total strain is the sum of elementary contributions (elastic, creep and plastic, thermal dilatation, swelling , densification and relocation for fuel)
-Non elastic strains are supposed to be known at a given time step (explicit treatment).
-The clad creep is calculated with the hypothesis of strain hardening.

THERMAL treatment

Main hypothesis:

-The calculations are done in 1,5 D dimension with axial coupling by coolant.
- axisymmetric geometry
- constant flow rate and section for the coolant.
- axial thermal conduction, gravity and kinetic energy are neglected

Main options
Heat exchange between coolant and clad -> COLBURN formula
-> JENS-LOTTES correlation in case of boiling
Heat exchange between fuel and clad is considered as conduction in a gas slide.
The radiative exchange and contact heat transfert are also taken into account.

The fuel conductivity is calculated by Martin formula and the Lucuta recommendation for the evolution with burn-up is used.

MAIN MODELS

Radar (Palmer) modified by CEA for the Pu formation and the radial distribution of power.

Fission gas release : CEA model developped in the frame of RNR studies

External clad corrosion : COCHISE (Beslu, Billot, Giordano)

Fuel clad gap heat transfert : URGAP (Lassman)

DOMAIN OF USE:

Light water reactors
- UO2 and MOX fuel
- Up to extended burn up (60000 MWd/TU)
- Ramp tests

VALIDATION FIELD:

Validation is in progress

PWR UO2 in standard conditions
based on :
-The CEA analytical experiments CONTACT, GRIMOX
-Post irradiation examination on rods irradiated up to extended burn-ups at different power levels in french PWR up to 60000 MWj/t

KEY REFERENCES

-P.T Elton , K. Lassman
Calculation methods for diffusionnal gas release Nu. Eng and design 101 (1987) 259-265

-K. Lassman, F. Hohlefeld
The revised URGAP Model to describe the gap conductance between fuel and cladding
Nu. Eng and design 103 (1987)

-J H Harding, D. G. Martin
A recommendation for the thermal conductivity of UO2; J.N.M. 166 (1989) 223-226

-P.G. Lucuta and al
Thermal Conductivity of SIMFUEL; Strasbourg conference;E- MRS Fall Meeting , Symposium E Nov 5-8 1991

-K. Lassman, H Blank
Modelling of Fuel Rod Behaviour and Recents Advances of the TRANSURANUS code
Nu. Eng and design 106 (1988) 291-313

-I.D. Palmer and al
RADAR, a model for predicting the radial power profile in a fuel pin.

-Ph Billot and al
Development of a mechanistic Model to asses the external corrosion of the zyrcaloy cladding in PWR.
Zirconium in Nuclear Industry ; Eighth International Symposium ASTM STP 1023. 1989

-J C Melis, J P Piron, L Roche
Fuel Modelling at high burn up , Recent development of Germinal code;
ANS/ENS Internationnal Meeting Nov 15-20 1992, Chicago Illinois

CODE NAME: TRANSURANUS

Date: August 1993

Version: V1M1J93

Development team:

ITU Modelling group

Originally based on:

URANUS code [1,2]

Address:

Commission of the European Communities
Joint Research Centre
Institute for Transuranium Elements
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D-76125 Karlsruhe
Tel (x) 07247 84 297
Fax (x) 07247 84 4046

OBJECTIVES:

TRANSURANUS is a computer program for the thermal and mechanical analysis of fuel rods in nuclear reactors and was developed at the European Institute for Transuranium Elements (ITU). The TRANSURANUS code consists of a clearly defined mechanical-mathematical framework into which physical models can easily be incorporated [3]. Besides its flexibility for different fuel rod designs the TRANSURANUS code can deal with very different situations, as given for instance in an experiment, under normal, off-normal and accident conditions. The time scale of the problems to be treated may range from milliseconds to years. The code can be employed in different versions (as a deterministic and a statistical code [4]).

MAIN ASSOCIATED OPTIONS:

The code has a comprehensive material data bank for oxide, mixed oxide, carbide and nitride fuels, Zircaloy and steel claddings and different coolants. During its development great effort was spent on obtaining an extremely flexible tool which is easy to handle, exhibiting very fast running times. The total development effort is approximately 40 man-years. In recent years the interest to use this code grew and the code is in use in several organisations, both research and private industry. The code is now available to all interested parties.

The whole TRANSURANUS system consists of several pre-and postprocessors (MACROH, AXORDER, URPLOT, URSTAT, URNOIS) and specific testprograms for all models, material data etc.

LANGUAGE:

FORTAN 77 with extensions

Running on:

PCs
workstations
main frame computers

Number of instructions:

78 000 statements,
30 000 executable instructions

Typical running time:

1-2 min on a workstation (20 radial, 10 axial nodes, 1000 time steps)

The radial and axial discretization is very flexible through the usage of pseudo-variable dimensioning: All dimensions are defined by a few PARAMETER statements which can easily be adapted to specific users needs. Thus, an analysis using 100 axial slices or even more does not cause any problems.It is important to note that all techniques were chosen in such a way that the computer costs depend more or less linearly on the discretization.

Type of NUMERICAL treatments:

Deterministic version: FD and FEM methods (explicit, implicit, Crank Nicholson)
Statistic version: Monte Carlo statistics, Numerical Noise analysis
Convergence: several different techniques, sophisticated time step control is applied.

Brief description of MECHANICAL treatment:

The mechanical analysis consists of the calculation of stresses, strains and the corresponding deformations. Dynamic forces are in general not treated and the solution is therefore obtained by applying the principal conditions equilibrium and compatibility together with constitutive relations. The following assumptions are made:

a) The geometric problem is confined to one-dimensional, plane and axisymmetric idealization, i.e. the axial deformation is constant across the radius (modified plane strain condition).
b) The elastic constants are isotropic and constant within a cylindrical ring, a so-called coarse zone.

One important theoretical concept of the TRANSURANUS code is that all volume changes due to different processes such as densification and swelling, cracking etc. are expressed via strains. The assumptions made and the equation of equilibrium lead to the classical semianalytic solution of the problem. From the theoretical concepst above it follows that fuel and cladding are divided into an arbitrary number of rings (coarse zones) which are further subdivided in fine zones in order to allow for the numerical integration. Since any discretization can be chosen the concept is called a variable multizone concept.
The integral fuel rod is treated by a superposition of a 1d radial and 1d axial description. A sophisticated theory of treating axial friction forces (URFRIC model) is applied [5].

THERMAL treatment:

Thermal analysis of the whole fuel rod is obtained by a superposition of one-dimensional radial and axial energy conservation equations [6] The energy equation (heat conduction equation) for fuel, cladding and structure is applied. The solution is based on a quasi steady-state approximate function within a small cylindrical ring. This equation together with inner and outer boundary conditions leads to a nonlinear system of equations with tridiagonal structure. The main features of the thermal analysis are summarized as follows:

a) The method includes the well-known Finite Difference Method and the Finite Element Method as special cases. The standard usage is an optimum combination of both which makes the solution extremely accurate.
b) The methods includes explicit, implicit or Crank-Nicholson integration procedures. Standard usage for transient conditions is the Crank-Nicholson scheme.
c) Phase changes (melting, boiling) are considered.
d) An important aspect of the thermal analysis is the heat transfer between fuel and cladding includes a detailed model for this gap conductance (URGAP model) [7].

Special emphasis was given to the problem of obtaining convergence and to the total amount of numerical effort. It was found that a special iteration scheme together with a sophisticated time step control gave optimum performance. The applied iteration scheme for solving the nonlinear heat conduction in a fuel rod is based on a Newton-Raphson technique and is fully described in Reference [8]. In conclusion, the thermal analysis has been carefully designed to be fast and reliable. This is the prerequisite for transient analyses in which the thermal analysis has to be done up to several thousand times for a single analysis.

MAIN MODELS:

The general concept of TRANSURANUS is that the basic equations apply to all type of fuel rods and reactor conditions. However, specific models are needed for specific problems. The complete set of models and options available is fully described in Table 1 of Ref. [9]. The basic equations of material conservation are applied to plutonium redistribution, to pore migration and redistribution of oxygen (OXIRED model) [10]. Different models are optional for densification, gas release [11] and swelling, relocation, for predicting the fuel structure, cladding failure, Pu build-up, the radial power density, the formation and closure of the central void (FBR), the waterside corrosion etc.. In order to be as flexible as possible the relevant material properties such as the elastic constants, the thermal conductivity, the specific heat, the density etc. are formulated in specific subroutines which allow for the incorporation of different correlations. At present up to 30 different correlations for fuel and cladding are possible and the TRANSURANUS code includes material data for all reactor types. Possible fuel materials are oxide, mixed oxide, carbide and nitride, the cladding materials are Zircaloy, steel and niobium and the coolant may be water, sodium, sodium-potassium and helium.
Similar to the formulation of the basic material properties, the general boundary conditions were formulated as flexible as possible. Different geometries and boundary conditions can easily be treated.

DOMAIN OF USE:

All types of deterministic and statistical fuel rod analyses

VALIDATION FIELD:

The TRANSURANUS code has been verified extensively by the following steps:

1) Verification of the numerical techniques by comparison with analytic solutions or by comparison with other techniques. Through these comparisons the optimum techniques could be identified and their proper use investigated.
2) Verification of specific models with experiments.
3) Verification by code-to-code comparisons.
4) Verification of the TRANSURANUS code by comparison with irradiations which is an ongoing activity.

based on:

LWR conditions have been analysed by experiments from the following projects: Halden, Risoe, Studsvik, Tribulation and others, FBR conditions were mainly tested within the CABRI project.

KEY REFERENCES:

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CODE NAME: PROFESS

DATE: 28th June, 1993.

VERSION: 1.0

DEVELOPMENT TEAM:
C.S.Viswanadham, D.N.Sah and C.Ganguly

ORIGINALLY BASED ON: None

ADDRESS:
Radiometallurgy Division
Bhabha Atomic Research Centre
BOMBAY-400 085
INDIA

OBJECTIVES: The main objective of PROFESS is to interpret the post-irradiation examination results on irradiated water reactor fuel pins which are examined in our hot cells. The code lays emphasis on the prediction of fuel temperature, fission gas release, fuel restructuring and fuel and cladding dimensions.

MAIN ASSOCIATED OPTIONS: None.

LANGUAGE: FORTRAN

RUNNING ON: Norsk Data 570

NUMBER OF INSTRUCTIONS: 1600 (approx.)

TYPICAL RUNNING TIME: Depends on the number of axial segments and the number of time steps of irradiation considered. CPU time required for 10 axial segments and 50 time steps is about 5 minutes.

LINKS WITH OTHER TOOLS: None.

TYPE OF NUMERICAL TREATMENTS:

PROFESS is one-dimensional and axisymmetric. Integral fuel pin is treated by dividing its length into a number of axial segments. Pin power in a given segment is considered to be uniform. Power history is divided into small time-steps of constant power. The cold dimensions of the fuel and the cladding are updated for irreversible changes at the end of every time step. By repeating the calculations for all the segments and for all the time steps, the code can predict the behaviour of the integral fuel pin over the entire irradiation period. All self-consistent solutions are obtained by successive bisection method.

BRIEF DESCRIPTION OF MECHANICAL TREATMENT:

Under open gap conditions, internal gas pressure (updated for fission gases released) and coolant pressure are used to calculate the three principal components of stress on the cladding. For closed gap, in addition, the fuel-cladding contact pressure is self-consistently calculated using successive bisection method. Elastic, thermal, plastic and creep strains are then considered for cladding strain calculations.

BRIEF DESCRIPTION OF THERMAL TREATMENT:

The thermal treatment in PROFESS considers thermal neutron flux depression and fuel-cladding gap conductance/interfacial resistance. The gap conductance model in PROFESS is based on Ross and Stoute Model with modifications incorporated to account for the effect of fission gases, gas pressure and pellet eccentricity. The heat transfer through the fuel clad gap is considered to take place by conduction and radiation in the open gap condition. When the gap is closed the fuel clad contact pressure is considered to evaluate the interfacial resistance. The gap conductance is iteratively calculated by successive bisection method. Temperature distribution in the fuel is calculated by dividing the fuel cross section into annular regions. Neutron flux depression near fuel centre is considered by modified Bessel functions.

MAIN MODELS:

Fuel-cladding heat transfer;
Thermal neutron flux depression near fuel centre;
Solid and gaseous fission product swelling;
In-pile fuel densification;
Fuel relocation;
Steady state fission gas release (5 optional models);
Transient fission gas release;
In-pile creep of cladding;
Equiaxed grain growth in fuel;
Columnar grain growth in fuel.

DOMAIN OF USE:

The model can be used for modelling zircaloy-2 clad UO_2 or MOX fuel pins. Fuel pins with both free-standing and collapsible cladding can be modelled. The code has so far been used for BWR and PHWR fuel pins.

VALIDATION FIELD:

1) Fission gas release during base irradiation and power ramp in three pre-pressurised minifuel pins irradiated to a burnup of 32000 MWD/MTU.

2) Fuel temperature in an instrumented fuel pin irradiated to a burnup of 12980 MWD/MTU.

3) Fuel temperature and restructuring in PHWR fuel pins irradiated to a burnup of 3700 MWD/MTU.

BASED ON:

1) Experimental data on D-Com Blind problem circulated under IAEA-CRP.

2) EPRI - Case C.

3) Post-irradiation metallography done at BARC, Bombay which generally validated the temperature predictions.

KEY REFERENCES:

1. D.N.Sah and D.Venkatesh, "A Brief Description of PROFESS and Its Submodels", Proc. IAEA Specialists' Meeting on Water Reactor Fuel Element Performance Computer Modelling, IWGFPT/19, International Atomic Energy Agency, Vienna, 1984, p.237.

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CODE NAME: FAIR

Date: 28th June 1993

Vesion: 1.0

Development Team:
P.Swami Prasad, B.K.Dutta, K.Anantharaman, H.S.Kushwaha, A.Kakodkar.

Originally based on :Nil

Address:
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Hall No.7
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Trombay, Bombay - 400 085
India.

OBJECTIVES: The validation of code FAIR at high burnups is important for us in view of our main interest in extending burnup of Pressurised Heavy Water Reactor fuel with the use of advanced fuel cycles and also to develop high burnup fuels for Advanced Heavy Water Reactors.

MAIN ASSOCIATED OPTIONS: These objectives are desired to be achieved by generating in pile and out of pile experimental data for high burnup fuels internally . The code will be tested against this data.

LANGUAGE: Fortran

Running on: Norsk Data 570, Risk based LandMark systems.
Under commisioning on : 32 node parallel processor.

Number of Instructions: 7000 (approx)

Typical running time: Depends on discretisation of pellet and clad and burnup. For a typical case consisting of 21 elements and 103 nodes *CPU* time required is 90 minutes for modelling upto a burnup of 13000 MWD/TeU on Norsk Data 570 and 20 minutes on Landmark System.

Links with other tools: Graphics support for pre and post processing is available through *Interactive Graphics Library (IGL)* and the output is directed to either Calcomp plotter or Tektronix terminal.

Type of NUMERICAL treatments: Two dimensional axisymmetric Finite Element Technique is used for both thermal and mechanical analyses. Transient temperature analysis is carried out by either Galerkin method or Crank-Nicholson method. Nonlinear behaviour of fuel pins because of thermal viscoplastic loads is analysed using Newton-Raphson method. System of equations are solved by either Banded Gaussian solver or Frontal Technique (Ref.1 & Ref.2)

Brief description of
MECHANICAL treatment: The module for predicting the mechanical behaviour of fuel pins is based on the principles of thermal viscoplasticity and Finite element technique. Two dimensional ;plane stres, plane strain and axisymmetric behaviour of fuel pins can be analysed using this module. Incremental theory of plasticity is applied for analysing elastoplastic behaviour. Von-Mises yield criterion, Prandtl-Reuss flow rule and isotropic hardening rule are used for modelling this behaviour. Temperature dependant mechanical material properties are considered. A material propety data base for pellet and clad materials is incorporated in this module from MATPRO (Ref.4). Pseudo strains arising out of thermal, creep loads, change of yield surface with temperature and also because of swelling are treated as initial strains.

THERMAL treatment: The treatment for thermal calculations consists of Finite element technique based formulation for plane/ axisymmetric structures. All the three modes of heat transfer are considered in this module. Iterative techniques are used to consider material property dependance on temperature. Temperature dependant material properties are incorporated based on MATPRO (Ref.4). Bessel functions are used to calculate the flux depression in pellet. Heat loads because of volumetric heat generation, edge heat flux, point flux can be considered. Isothermal, time varying convective and radiative boundary conditions can be analysed.

MAIN MODELS:

1) Fission gas release model based on Ref.5 & Ref.6.
2) Model for gap conductance based on Ross and Stoute (Ref.7).
3) Densification, Swelling, and creep models for pellet and clad based on Ref.4.
4) Model for equiaxed grain growth based on Ref.5.
5) Crack propagation and ultimate rupture model for clad bassed on Ref.8.
6) Model for treating radial cracks in pellet by changing material property matrix from axisymmetric state to that of plane stres state, with zero hoop stress.

DOMAIN OF USE:
The code can be used for modelling water cooled reactor fuel pins. Fuel pins with both free standing and/or collapsible clads can be analysed. Pellets made of UO_2 or UO_2/PUO_2 and sheath made of zircaloy-2 or zircaloy-4 can be modelled. The code can take care of power ramps with any given ramp speed. Presently the code has been validated by analysing bench mark problems with a maximum burnup of 5000 MWD/Teu and linear heat ratings of 70 KW/m.

VALIDATION FIELD:
1) The code has been validated by mechanistic simulation of threshold power ramp criteria for PHWR fules in the form of P_c, ΔP_c curves and comparing them with experimentally available data (Ref.9).
2) Bench mark problems as suggested in Ref.10 have been analysed and the results were found to be in good agreement with those quoted in Ref.10.

based on:
1) The first validation is based on the inpile data of Pickering and Douglas point reactors given in Ref.9.
2) The second validation is based on Ref.10, wherein a comparison of four fuel rod modelling codes is compared with experimental data.

KEY REFERENCES:

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DESCRIPTION OF THE CODE

CODE NAME: FUDA (FUel Design Analysis)

Version : MOD 1

Originally based on : FUDA - MOD 0

Development team: M.Das, P.N.Prasad, S.A.Bhardwaj, B.V.Arunakumar

OBJECTIVES:

The objective of the code is to carry out design calculations and analyses for licensing submissions. The code is also used for fuel performance evaluation of the operating stations and feed back to design. Additionally the code is used for optimizing the fuel design and fabrication parameters for improved performance.

MAIN ASSOCIATED OPTIONS:

This code has the following options:

1. Different types of pellet geometries

2. Different fuel and sheath materials

4. Applicable for both PHWRs (primarily) and LWR

5. Analysis for load following or base load operation.

LANGUAGE: FORTRAN

Running on :(i) i860 workstation with UNIX operating system
(ii) NorksData computer with SINTRAN-III operating system
(iii)PC-386/486 with MS-DOS operating system

Typical running time: 3 min CPU time on i860 workstation(approx.)

Number of instructions: 2000

Links with other tools : Graphic package for pre and post-processing

Type of NUMERICAL treatments:

Code uses Finite Difference Method for temperature, thermal expansions, and sheath stresss calculations. Local stress and ridge analysis are carried out by finite element technique.

Brief description of MECHANICAL treatment:

Fuel expansion is calculated using a two zone model in which the stresses in UO_2 are ignored. The model assumes that above a certain temperature, the UO_2 deforms plastically and below that temperature, it cracks radially and behaves as an elastic solid. The extent of plasticity is governed by the temperature of the UO_2, the stress imposed on it by sheath strength and the coolant pressure and is a function of time. In a collapsible sheathing with low diametral clearances, the firm contact between the pellet and the sheath due to external pressure and diametral expansion, limits the relative fuel movement.

The fission gas generated is a function of burnup. The fission gas generated for each radial burnup zone is calculated as suggested by Southier.

Fission gas pressure is calculated based on mass of the fission gas release, mass of the fill gas, available void volumes and temperature of storage locations.

Global sheath stresses and strains due to fuel thermal expansion,swelling and densification are calculated. The creep and stress relaxation in the time zone at constant power operation is calculated using semi-empirical formula considering athermal and thermal creep including effect of irradiation. Fuel sheath interfacial pressure is then calculated based on gas pressure and strains. This shows whether the expanded fuel is touching the sheath or gas gap exists between them.

Using global diametral changes, local deformations of the pellet and sheath are calculated considering hourglass phenomenon of the pellet. Finite element method using axisymmetric 8 noded isoparametric elements is used for calculating deformation, stress and strain in pellets and sheath.

THERMAL treatment:-

The radial temperature distribution across the pellet and sheath are calculated for the given inputs of linear heat rating, coolant temperature and heat transfer coefficient. The components of the fuel-sheath heat transfer coefficient across the gas filled fuel-sheath gap and through the solid-solid contact points are calculated using Ross and Stoute equation.

Using the new fuel-sheath heat transfer coefficient, new temperature distribution across fuel and sheath are calculated and the cycle is repeated iteratively. The iteration is terminated when two successive calculations of internal gas pressure agree within 5%. For any power change, the above iterative procedure is repeated for a given time zone till the required convergence is obtained. For improving accuracy, the pellet is divided into 100 rings radially and all the parameters are calculated for each ring. The radial flux depression in the element is taken into account in estimating powers in different rings. For radial flux depression calculation two options are available. The first one is using Bessel function with a kappa value. The second option is using an equation fitted to the results of the neutron-physics code. This takes into account variation with burnup. Axial flux gradient in the fuel element can be handled by splitting the element into a number of equal lengths and considering each as being at an uniform power output.

MAIN MODELS:

Following are the main models used in the code:

1. Flux depression in the pellet: The local flux perturbations affecting fuel design are radial flux depression through the bundle and flux peaking at the interface between adjacent fuel bundles in a channel. The radial distribution of flux is a function of pellet diameter, UO_2 enrichment and burnup and plutonium build up. Two options are included in this model viz., (i) Bessel function based and (ii) Based on PHWR physics codes which varies with burnup.

2. Film heat transfer co-efficient: Depending on the coolant condition and environment, sheath-to-coolant heat transfer is calculated. For forced flow through rod bundles, Dittus-Bolter equation is used. For BWRs, Jens-Lottes correlation is used.

3. Fuel-sheath gap heat transfer coeff.: Pellet clad gap conductance is calculated by Ross and Stoute model (Ref. 4 & 5) taking care of the physical gap existing between the pellet and the clad. Pellet-clad gap conductance consists of three parts

$$hg = hs + hf + hr$$

a) Conduction through solid-solid contact points (hs)

b) Convection through solid-gas interface (hf) and

c) Radiation exchange between pellet outer surface and clad inner surface (hr)

The heat transfer coefficient between the sheath and the pellet is a function of:

- radial gap/contact-pressure between the pellet and the sheath;
- the composition of gases inside the fuel element; and
- the initial roughnesses of the surfaces of the sheath and of the pellet.

4. Fuel Thermal Conductivity: Thermal conductivity variation with temperature and porosity is considered. The low temperature radiation damage is considered by assuming the UO_2 conductivity below 500 °C as constant. Pellet temperature profile is

calculated by dividing pellet into a number of concentric rings, normally 100. Temperature is calculated from surface to centre using finite difference method.

5. Fission Gas Release: There are two models incorporated in FUDA for fission gas release.

i) Temperature dependent release mechanism.

ii) Physical model based on diffusion and grain growth mechanism - Both equiaxed and columnar grain growth are treated. Equiaxed grain growth is calculated based on local temperature, UO_2 enrichment and grain size. Columnar grain growth is calculated based on local temperature and its gradient, pressure in the gas bubble and grain size. Fission gases are assumed to diffuse through UO_2 grains, the amount of diffusion being dependent amongst others on local temperature and grain diameter. The bubbles accumulated on the grain boundary grow in size and coalesec before releasing to the gap through the tunnels/cracks.

DOMAIN OF USE:

FUDA is a computer programme to predict the performance of UO_2 fuel elements irradiated at high power outputs upto 55 KW/m for PHWR fuel to high burnups upto 15000 MWD/TeU.

VALIDATION FIELD:

Different parameters like fuel centre temperature, surface temperature, fission gas release, sheath strain(plastic upto a burnup of 15,000 MWD/TU against end of life parameters measured after irradiation experiments. Variation of fission gas pressure with burnup is compared with available instrumented data from irradiation experiments. The different parameters are also compared with results obtained with other similar computer codes like ELESIM, ELESTRES etc.

Validation based on:

The validation is based on
1) Power reactor data
2) Comparison with other codes
3) Literature on results of inpile experiments conducted abroad.
4) Inpile experiments conducted in India
5) Post irradiation examination

REFERENCES:

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12. M.Das, "Design Aspects of PHWR for improved performance", Material Science Forum Vol. 48 & 49, Trans Tech Publication, Switzerland, (1989).

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20. P.B.Desai, V.G.Date, M.Das, P.N.Prasad, K.S.Prasad, "Balloning and Rupture Behaviour of PHWR Fuel Cladding", Third International Conferecne on CANDU Fuel, 1992.

CODE NAME: TRUST

Date: 30-MAY-1993

(Thermo-mechanics of Rod Under Steady-states and Transients)

Version: 1.0

Development team:

13 Group, 1'st Research Department
Nippon Nuclear Fuel Development Co.,Ltd.

Originally based on:

Developed by the team from scratch

Address:

2163 Narita-cho Oarai-machi,
Higashi-ibaraki-gun, Ibaraki-ken, 311-13
JAPAN

OBJECTIVES:

To predict thermo-mechanical behavior of nuclear fuel element used in water-cooled reactors throughout its life up to 70 GWd/t under steady-state and/or (normal) transient irradiation conditions. The prediction item includes fuel temperature, FP gas release, stress and strain in pellet and cladding.

MAIN ASSOCIATED OPTIONS:

None

LANGUAGE: FORTRAN-77

Running on: VAX Station 4000 model 60

Number of instructions: 11656
(executable statements)

Typical running time: 10 minutes

Links with other tools: A program called TIG (TRUST Input Generator) is operative. Graphic tool to display results is connected to the code.

Type of NUMERICAL treatments:

Finite element method is adopted to discretize thermal, mechanical and gas-transport equations. Fuel rod is spacially discretized radially and axially to form ring elements for thermal and mechanical analyses. The elements for FP gas transport analysis is axially aligned. The backward Euler formula is adopted to discretize time domain for numerical stability.

The thermal, mechanical and gas-transport elements are assembled into a set of global equations, and the system is numerically solved. Iterative technique is adopted to get the solution because the equations are non-linear. The calculation procedure for the fuel behavior in a time increment is repeated until the fuel irradiation history ends.

Brief description of
MECHANICAL treatment:

Mechanical status of fuel is given by solving the overall stiffness equations. The finite elements for pellet and cladding as well as the gap elements to convey force between them are assembled into the system. An axisymmetric plain strain model is adopted to form the pellet and cladding elements. The ring-shaped element has 4 connection nodes: inner and outer surface for radial deformation and top and bottom surface for axial deformation. There are four varieties of gap elements handling gap status open, close, axial slip and locking. Element for the spring or sleeve device to hold down pellet stack can also be assembled.

Various phenomena are taken into account to assess the inelastic strain in the finite elements: pellet and cladding thermal expansion, creep, plastic deformation, densification, swelling, caused by the accumulation of solid FP and the growth of FP gas bubble, and cladding irradiation growth. Pellet is treated as non-tension material to simulate crack. Pellet relocation is treated as a part of crack behavior. Coolant and inner gas pressures as well as pellet weight are exerted on element nodes. Overall force equilibrium is assumed in forming the stiffness equations.

THERMAL treatment:

Fuel temperature distribution is given by solving the overall thermal equations. The thermal elements for heat conduction in pellet and cladding as well as for the heat transfer at interfaces between pellet, cladding, and coolant are assembled into the system. The elements are derived from radial axisymmetric heat flow equation with transient (or heat capacity) term. No axial heat flow is assumed. Radial heat generation distribution in pellet is expressed in a Bessel function for heavy water reactors and a polynominal function for LWRs. The Ross and Stout gap conductance model, the Jens-Lottes and Dittus-Boelter cladding-to-water heat transfer coefficients (for BWR and PWR, respectively) are adopted to form the thermal gap element. There is a special variable-width element to express the pellet rim structure. For PWR fuel behavior analysis, axial coolant temperature rise is modeled.

FP gas concentration to assess the gap conductance is calculated by a FP gas axial transport model. The model gives axial distribution of FP gas concentration caused by localized FP gas release and its slow diffusion in axial direction. Axial FP gas diffusion speed is assessed by the free volume from mechanical analysis and by its temperature from thermal analysis.

MAIN MODELS:

A simple yet mechanistic FP gas release and swelling model is incorporated into the code. The model describes FP gas atom generation and diffusion in pellet grains, formation and growth of FP gas bubbles on grain boundary, FP gas release from pellet through linkage of the bubbles and microcracking of pellet by thermal stress.

In thermal analysis, the thermal conductivity of pellet is assessed by a model depicting its decrease by buildup of FP atoms and irradiation defects, and by growth of FP gas bubbles. The gas bubble swelling strain from the FP gas model is used to assess the part of the decrease in pellet thermal conductivity.

DOMAIN OF USE:

The code is for the R&D and design of high burnup LWR fuel.

In R&D field, The code is to be used in a variety of sensitivity studies. The assessment of the advanced pellet performance in comparison with the conventional pellet is one application. The code will give insight into the effect of pellet grain enlargement or thermal conductivity enhancement. Another application is to evaluate the effect of the rim structure on high burnup fuel behavior.

In design field, a feasibility study will be performed by the code to search for a design of high burnup LWR fuel with its average burnup 70 GWd/t. Combination of the advanced pellets and other design specifications such as fuel dimensions will be optimized to satisfy design criteria in the study.

VALIDATION FIELD:

Pellet center line temperature, FP gas release, fuel and cladding elongation, fuel radius change, FP gas local retension in pellet, gas inter-mixing speed, and ridging height on cladding outer surface.

based on:

Data from instrumented assemblies and rods from HALDEN and Riso projects are the main source of code validation. PIE data on commercial fuel rods gathered in Nippon Nuclear Fuel Development Co. as well as data from other sources have been used to verify FP gas release and fuel deformation.

KEY REFERENCES:

[1]ZIENKIEWICZ,O.C.: "The Finite Element Method", 3rd Edition, McGraw-Hill, 1977.

[2]ROSS,A.M. and STOUT,R.L.: "Heat Transfer Coefficient between UO2 and Zircaloy-2", AECL-1552, 1962.

[3]JENS,W.H. and LOTTES,P.A.: "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High Pressure Water", ANL-4627, 1951.

[4]DITTUS,F.W. and BOELTER,L.M.K: Univ. Calif. Pubs. Eng. 2, 443, 1930.

[5]KOGAI,K.: "A Symple Fission Gas Release/Gaseous Swelling Model", Paper presented IAEA Technical Commitee Meeting, 27 April-1 May 1992, Pembroke, Ontario, Canada.

[6]ISHIMOTO,S.,HIRAI,M. and ITO,K.: "Effect of Soluble Impurities on Thermal Conductivity of Nuclear Fuel Pellet", A304, The Twelfth Japan Symposium on Thermophysical Properties, 1991.

[7]MATPRO-Version 11 (Revision 2): "A Handbook of Material Properties for Use in the Analysis of Light Water Reactor Fuel Behavior", NUREG/CR-0497 TREE-1280, Rev.2, 1981.

[8]KNUDSEN,P.,BAGGER,C.,CARLSEN,H.,JOHANSEN,B.S.,MISFELDT,I. and MOGENSEN,M.: "Final Report on the Riso Transient Fission Gas Release Project", Riso-TFGP-P29, Vol.1 and 2, 1986.

[9]The Third Riso Fission Gas Project: "Final Report: The Project", RISO-FGP3-FINAL,Pt.1, 1991.

[10]LANNING,D.D.,CUNNINGHAM,M.E.,BRADLEY,E.R. and BARNER,J.O.: "Qualification of Fission Gas Release Data from Task 2 Rods", HBEP-25(2P4), 1987.

[11]CUNNINGHAM,M.E.,LANNING,D.D. and BARNER,J.O.: "Qualification of Fission Gas Release Data from Task 3 Rods", HBEP-60(3P26), 1990.

CODE NAME: EIMUS	Date: 1981 - 1994
Version: 4 Originally based on: FEMAX-III	**Development Team** Mr. Motoyasu Kinoshita Mr. S. Kitajima **Address:** Nuclear Science Department Komae Research Laboratory, CRIEPI 2-11-1 Iwatokita Komaeshi Tokyo, 201 Japan

OBJECTIVES: The EIMUS code has been developed and utilized for the following objectives:

(i) Evaluate LWR fuel performance at operational conditions, including transients, especially to meet utility's needs.
(ii) Evaluate fuel reliability at load following operation and at extended burnup.
(iii) Investigate detailed thermo-mechanical behaviour at PCMI using finite element method.
(iv) Investigate effects of developed fuel/cladding micro-structure to the thermal and mechanical performance at high burnup and long term utilization.

Main associated options:

- Reactor type: BWR, PWR, HBWR
- Pellet shape: flat, dished, chamfered solid/hollow
- Internal gas composition: Helium, Nitrogen, Xenon, Krypton
- Pellet additives: Gadolinia
- Re-fabrication: Change of plenum gas composition and pressure during irradiation

LANGUAGE: FORTRAN-4 **Running on:** FACOM (M380,VP2000) SUN

Number of instructions: ~20000 FORTRAN Statements

Typical running time: 30 minutes

Links with other tools: Calcomp Plotter

Type of NUMERICAL Treatments

For the thermal analysis a fuel rod is divided into 12 axial segments maximum and pellets are divided into 20 radial sections maximum. At each time-step the results of thermal analysis are transferred to the mechanical analysis part. The iterative calculation is made to converge to center temperature at each axial segment. The results of mechanical analysis are not transferred back to the thermal part to affect this conversion process.

For the mechanical analysis axi-symmetric finite element method (FEM) is used. An axial half of a pellet and corresponding cladding is divided into FEM segments. Creep and plasticity of pellet and cladding is formulated by Cyr's method with Newton-Raphson iteration scheme.

MECHANICAL treatment:

Cladding ridge formation at pellet-pellet interface is calculated by the axi-symmetric 2-dimensional FEM with boundary conditions of contact/sliding at pellet-pellet and pellet-cladding interfaces. This detailed analysis is made for a half height of one pellet. The axial position of the concerned pellet is selected and usually it is placed where the highest PCMI is expected.

Radial relocation of pellet fragments and resultant reduction of pellet-cladding gap is assumed at the start-up of the irradiation. Fission gas swelling is assumed to be suppressed when pellets are mechanically restrained by contact with the cladding. Pellet hot-pressing is calculated using Mohr-Coulomb creep potential surface.

THERMAL treatment:

Radial heat transfer calculation is made by one-dimensional thermal analysis. Though usual calculation is neglecting time dependence, the code switches to transient mode, including the effect of heat capacity, when the increase of heat rating is fast enough. The radial shape of heat generation is calculated with effects of Pu build-up at rim area due to epithermal neutron absorption.

The thermal conductivity of fuel is calculated by a model in which the conductivity decreases by development of micro-cracks or separated grain boundaries and by solid solution of fission product in fuel matrix as the burnup increases.

The gap conductance calculation is related to the geometry and stress, such as the cladding creep-down, pellet densification and pellet-cladding contact pressure. The local gas composition and pressure in free volume, that consists of plenum, pellet-cladding gap, and pellet dish is calculated by axial gas transport model.

MAIN MODELS

- Thermal conductivity degradation

  Burnup dependence of the thermal conductivity is modelled by a mechanistic model. It assumes development of micro-cracks and/or grain boundary spacing and the resultant effect of thermal barrier is calculated. The other effect, fission product solid solution in fuel matrix, is also taken into account.

- Radial heat generation due to neutronic rim effect

  The increase of local fission rating at pellet rim region is calculated by an empirical correlation which is derived by fitting results of neutronic code calculation. The void ratio is included as a parameter for BWR results.

- Fission gas release

  One dimensional diffusion equation is calculated for a fuel grain assuming spherical geometry. The intra-grain diffusion coefficient of fission gas is based on Matzke's experiment. Diffusion delay due to trapping by intra-bubble formation is included empirically up to 20 MWd/kgU. The incubation period of gas release is modelled by saturation of grain boundary. The gas concentration is calculated by the diffusion equation.

- Axial gas mixing

  Fission gas is released from the fuel high temperature region. The released gas is transported by axial flow and mixed with Helium. The local pressure increase is due to the gas released and the reduction of the free volume. The rate of axial flow is also dependent on the flow resistance at pellet-clad gap.

- Pellet-cladding gap

  A pellet is cracked into fragments and relocates at start-up of the irradiation. Pellet densification, pellet swelling, cladding creep-down are taken into account.

DOMAIN OF USE

The code is used for:

- Evaluation of fuel performance and reliability at extended burnup up to 55 MWd/kgU assembly average. However the calculated temperature at burnup exceeding 45 MWd/kgU is made by extrapolation.

- Evaluation of fuel reliability in load-following operation and in faster power transient operation. The key analyses of the code are to evaluate thermal feedback due to gas release and its dilution by axial gas mixing, and to estimate local stress increase at pellet-pellet interface due to PCMI.

- Piloting calculation for new design for higher burnup up to 70 MWd/kgU based on mechanistic modelling.

VALIDATION FIELD

The thermal calculation was mainly verified versus RISØ-3 and Halden data which are obtained from temperature measurement at fuel centerline during irradiation. The mechanical calculation was verified versus Halden data. The Studsvik ramp experiments were also utilized to verify cladding local strain development at failure.

Based on:

- OECD Halden Reactor Project

  Fission gas release and fuel center temperature between 0 to 30 MWd/kgU were benchmarked. The fuel cracking model for thermal conductivity degradation was verified versus fuel temperature data from gas flow rigs and local cladding deformation was verified versus IFA-508 diameter measurements conducted by JAERI.

- Risø Fission Gas Release Project

  Fission gas release model was verified against data obtained from the Risø 2 programme between 15 and 50 MWd/kgU and Risø 3 provided the necessary data to benchmark FGR and fuel centre temperature models between 15 and 45 MWd/kgU.

- Battelle High Burnup Programme

  Local burnup development at fuel rim region is verified versus EPMA data of high burnup rods (up to 85 MWd/kgU) of PWR, irradiated in the BR-3 reactor at MOL (Belgium).

- Studsvik Inter-Ramp, Over-Ramp Projects

  Used to verify data on local cladding strain due to strong PCMI in ramping conditions.

KEY REFERENCES

[1] KINOSHITA, M., Development of LWR Fuel Performance Analysis Codes, J. Nucl. Sci. and Tech. Vol. 30, No.1, Pp.1-17, January 1993.

[2] KITAJIMA, S., MATSUMURA, T., KINOSHITA, M., Reduction of Effective Thermal Conductivity in High Burnup Fuels, IAEA Technical Committee Meeting on Fuel Performance at High Burnup for Water Reactors, Nykoping, Sweden, 5 - 8 June 1990.

[3] MATSUMURA, T., KAMEYAMA, T., Burnup and Plutonium Distribution near the Surface of High Burnup LWR Fuel, IAEA Technical Committee Meeting on Water Reactor Fuel Element Computer Modelling in Steady-State, Transient and Accident Conditions, Preston, U.K., 19 - 22 September 1988.

[4] KINOSHITA, M., Development of High Burnup Fuel Analysis Code EMUS, CRIEPI Report ET88002, July 1988.

[5] KINOSHITA, M., High Burnup Fission Gas Release Model FGRBEM, CRIEPI Report No. T86003, May 1987 (in Japanese).

[6] KINOSHITA, M., Evaluation of Axial Fission Gas Transport in Power Ramping Experiments, IAEA Specialists Meeting on Water Reactor Fuel Performance Computer Modelling, Windermere, U.K., April 1984. also in Res. Mechanica 19(3) 1986.

[7] KINOSHITA, M., TANAKA, H., Axial Fission Gas Transport in LWR Fuel Rods (II) - Elementary Solution for Unsteady State, CRIEPI Report No. 284020 December 1984 (in Japanese).

CODE NAME: FRAPCON-2 (modified)

Date: 1993/06/14

Version: V0

Development team:
Fuel Performance Research Laboratory
China Institute of Atomic Energy

Originally based on: FRAPCON-2
Originally developed by US Idho National Eng. Laboratory, modified by China Institute of Atomic Energy.

Address:
P. O. Box 275 (64), Beijing 102413
China

OBJECTIVES:

The code is used to analyze the behaviour of LWR fuel rods with UO_2 pellets during long-term burnup for design, operation, and safety evaluation of NPP fuel system. Besides, it is also used to provide initial conditions for RELAP4 and FRAP-T6 which are used for reactor accident analysis.

MAIN ASSOCIATED OPTIONS:

- PWR or BWR;
- Mechanical models: 1-D fuel and cladding deformation analysis,
  2-D Finite Element Method for PCMI anlysis;
- Fission Gas Release: several models for selection.

LANGUAGE: FORTRAN IV

Running on: CYBER, IBM, or PC

Number of instructions: 35,000

Typical running time: about 30 minutes on CYBER-825

Links with other tools:

If it is used to provide initial conditions for FRAP-T6 transient analysis, it links with FRAP-T6 through its produced data file.

Type of NUMERICAL treatments:

General FORTRAN numerical treatment:

- Input data in NAMELIST format,
- Produces: formated output file to be printed,
  binary plot file for plotting.

Brief description of
MECHANICAL treatment:

The mechanical analysis models consider elastic, plastic, and creep deformation, fuel-cladding mechanical interaction (PCMI). Several options are as follows:

- FRACAS-I: 1-D rigid pellet model which does not consider stress-driving deformation of pellet,
- FRACAS-II: 1-D deformable pellet model,
- PELET/RADIAL: modified 1-D Finite Element Method mechanical analysis model taken from GAPCON code.
- AXISYM: 2-D Finite Element Method PCMI analysis model. Because of some mistakes, it was not available, it has been modified.

THERMAL treatment:

Thermal analysis models include:

- 1-D fuel temperature calculation,
- Heat transfer through fuel-cladding gap, cladding to coolant,
- Coolant temperature calculation,
- Plenum temperature calculation,
- Calculation of fuel rod internal pressure,
- Fission gas production and release from fuel to gap,
- Besides, the code has a model for analyzing fuel rod failure histories as function of time-dependent fuel rod power and coolant boundary conditions.

MAIN MODELS:

- Thermal analysis models (temperature calculation of fuel, cladding, plenum, gap, coolant etc.),
- Mechanical analysis models,
- Internal pressure calculation,
- Fission gas release models,
- Fuel rod failure analysis model.

DOMAIN OF USE:

- Steady-state operation,
- UO_2 and Zr-4 or Zr-2 temperature should be less than their melt temperature,
- Deformation: small deformation (cladding strain should be less than 5%),
- Fission gas release: for general burnup, for higher burnup, US NRC modification factor should be used (it is more conservitive).

VALIDATION FIELD:

1. Lower burnup validation;
2. Higher burnup validation.

   (Fuel center temperature, cladding deformation, internal pressure, FGR)

based on:

1. Case 3 and 5, provided by Halden;
2. Case 1, 2, 4, 6, provided by Halden.

KEY REFERENCES:

1. Berna, G. A. et al., FRAPCON-2: A Computer Code for the Calculation of Steady State Thermal- Mechanical Behaviour of Oxide Fuel Rods, NUREG/CR-1845, January, 1980.

Filled by Zhang Yingchao

CODE NAME:
ROFEM

Version:
1B

Originally based on:
FEMAXI - III

Date:
1993.06.22

Development team:
D.R.Moscalu
G.Horhoianu
Thermal Reactor Fuel Performance Analysis Group

Address:
INSTITUTE FOR NUCLEAR RESEARCH
P.O.BOX 78, 0300 PITESTI
ROMANIA

OBJECTIVES:

The ROFEM computer code provides a detailed thermal and mechanical analysis of PHWR-CANDU type fuel-elements behavior as a function of the reactor power history under normal operating conditions.

MAIN ASSOCIATED OPTIONS:

LANGUAGE:
FORTRAN IV

Number of instruction:
13000

Links with other tools:

Running on:
CDC - Cyber 830

Typical running time:
1-30 minutes,depending on the type of analyse and the number of time steps.

Type of NUMERICAL treatments:

-classical numerical algorithms for the one-dimensional thermal and mechanical calculations;
-two-dimensional axisymmetric finite element method for detailed mechanical calculations;
-an implicit iterative procedure for solving non-linear equations

Brief description of

MECHANICAL treatment:

The detailed local mechanical analysis is performed separately in the mechanical part of the code by means of two-dimensional axisymmetric FEM. The mechanical part utilizes as input the temperature distribution and inner gas pressure transferred from the thermal subcode at the completion of the thermal calculations. The region of a half pellet height is analyzed in detail assuming axisymmetry and a plane symmetry at the mid-plane of a pellet. Both fuel and cladding are divided in quadratic isoparametric ring elements which are linked by continuity laws of force and displacements. Elasto-plasticity, creep, thermal expansion, fuel cracking and crack healing, relocation, densification, swelling and fuel-clad mechanical interaction are modelled.

THERMAL treatment:

The thermal part of the code have been recently modified in order to simulate the behavior particularities of the PHWR-CANDU type fuel elements. The integral fuel element thermal and mechanical behavior is analyzed in one-dimensional axisymmetric approach. The calculations are performed for one axial segment, dividing the fuel pellet in 100 concentric rings. After the temperature distribution determination, the thermal expansion, densification, restructuring and fission gas release are evaluated for each ring. The associated inner gas pressure and gas composition are then evaluated. After updating the gap conductance, the calculations are repeated until the given convergence criterion is reached.

MAIN MODELS:

The main constituent models are physically based and include such phenomena as:

-microstructure dependent fission gas release;

-temperature dependent grain growth, both equiaxed and columnar;

-temperature,porosity and burnup dependence of thermal conductivity;

-burnup dependence radial power profile in the fuel pellet;

-pellet to clad heat transfer via solid-solid,gas and radiative components;

-stress, dose and temperature dependent constitutive equations for the cladding, including creep.

DOMAIN OF USE:

The domain of use for ROFEM code is presently limited to PHWR-CANDU type fuel behavior analysis in normal operating conditions. The main objective of ROFEM code participation in FUMEX exercise is the intention to extend the applicability of the code to burnups above 25 MWd/KgU.

VALIDATION FIELD:

The code has been validated on the available data base which include a significant number of irradiation experiments on PHWR type fuel with peak linear powers in the range 500-700 W/cm and maximum discharge burnups of about 20 MWd/KgU.

based on:

INR - experimental fuel elements irradiation program data base.

KEY REFERENCES:

[1] Nakajima,T. et al.,FEMAXI-III, A Computer Code For The Analysis Of Thermal And Mechanical Behavior Of Fuel Rods,JAERI 1298 (1985)

[2] Horhoianu,G.,Moscalu,D.R. Improvement Of The CANDU Type Fuel Element Performance In Order To Increase The Ability To Operate At High Powers And To Meet High Burnup, Final Report,IAEA Research Contract No. 6197/RB (1992)

[3] Campbell,F.R. et al.,In-Reactor Measurements Of Fuel To Sheath Heat Transfer Coefficients Between UO_2 And Stainless Steel, AECL-5400 (1977)

[4] Notley,M.J.F., Hastings,I.J., A Microstructure-Dependent Model For Fission Product Gas Release And Swelling In UO_2 Fuel, AECL-5838 (1978)

[5] Lucuta,P.G. et al., Thermal Conductivity And Gas Release From SIMFUEL, IAEA Technical Committee Meeting, Pembroke Ontario Canada, 28-april 1-may 1992

[6] Gheorghiu,C., Ciocanescu,M., Irradiation Test Programme Aimed To Check Romanian Nuclear Fuel Behavior, Third Int. Conf. On CANDU Fuel, Chalk River, Canada, 4 - 8 October 1992.

CODE NAME: TRANSURANUS — Date: July 1993

Version: Version 1
Modification 1
Year 1992

Development team:
PSI Fuel Modelling Group
Process Technology Section
&
European Institute for
Transuranium Elements
(Pr. K. Lassmann et al)

Address:
Paul Scherrer Institute
CH-5232 Villigen PSI
Tel: (056) 99 21 11
Fax: (056) 98 23 27

Originally based on:
URANUS Code developed at the European Institute for Transuranium Elements /1/.

OBJECTIVES: The code has been designed for describing the thermal and the mechanical behaviour of a whole fuel rod in any type of reactor. The code can deal with very different situations, as given for instance in an experiment, under normal, off-normal and accident conditions.

MAIN ASSOCIATED OPTIONS: The code can be employed in different versions i.e as a deterministic and as a statistical code.

LANGUAGE: Fortran 77

Running on:
CONVEX C220
SUN station IPX

Number of Instructions:
about 75300 lines of code

Typical running time:
4.5 minutes on SUN for FUMEX1

Links with other tools: PV-WAVE P & C (Precision Visuals ,Inc.)

Type of NUMERICAL treatments:

TRANSURANUS code /2/ uses a quasi two-dimensional (1-1/2d) approach to simulate the thermal-mechanical behaviour of the fuel rod. The fuel rod is divided into axial slices and at a given time the rod is analysed slice by slice. After all slices have been analysed, the slices need to be coupled together which means that quantities such as the inner pin pressure or the axial friction forces between fuel and cladding are determined (axial coupling).

Brief description of

MECHANICAL treatment:

The mechanical analysis consists of the calculation of stresses, strains and the corresponding deformations. The solution is obtained by applying the principal conditions of equilibrium and compatibility together with constitutive relations. The main assumptions are: the geometric problem is confined to one-dimensional, plane and axisymmetric idealization and the elastic constants are isotropic and constant within a cylindrical ring so-called coarse zone. Since the inelastic strains cannot be given analytically, the solution has to be evaluated numerically. The semianalytic solution uses the multizone concept. One important theoretical concept of the code is that all volume changes due to different processes are expressed via strains.

THERMAL treatment:

The thermal analysis of the whole fuel rod is obtained by a superposition of one-dimensional radial and axial energy conservation equations. The main features of the thermal analysis are the following: (1) the method includes the well-known finite difference method and the finite element method as special cases. The standard usage is an optimum combination of both which make the solution extremely accurate. (2) the method includes explicit, implicit or Crank-Nicholson integration procedures. Standard usage for transient conditions is the Crank-Nicholson scheme. (3) phase changes (melting, boiling) are considered. (4) the code includes a detailed model for analysing the heat transfer between fuel and cladding. A special iteration scheme based on a Newton-Raphson technique is used for solving the nonlinear heat conduction in a fuel rod.

MAIN MODELS:

The TRANSURANUS code consists of a clearly defined mechanical-mathematical framework into which physical models can easily be incorporated. All important physical models are included, i.e. models for thermal and irradiation induced-densification of the fuel, swelling due to solid and gaseous fission products, creep, plasticity, pellet cracking and relocation, oxygen and plutonium redistribution, volume changes during phase transitions, formation and closure of central void, treatment of axial friction forces.

DOMAIN OF USE:

The code has a comprehensive material data-bank for oxide, mixed oxide, carbide and nitride fuels, Zircaloy and steel claddings and different coolants. It can deal with very different situations, as for instance in an experiment under normal, off-normal and accident conditions.

VALIDATION FIELD:

The code has been verified extensively by comparing the numerical techniques with other techniques or with analytical solutions and by verification of the models with the experiments.(Risoe, Studsvik, Halden,....)

KEY REFERENCES:

/1/ G. Kaltenthaler et al, Final Report of a comparison of the fuel rod performance code GAPCON-THERMAL-3 and the URANUS-LWR version applied to Halden ramp tests. Tech. Hochschule Darmstadt (1983), internal report.

/2/ K. Lassmann et al, Modelling of Fuel Rod Behaviour and Recent Advances of the TRANSURANUS code, Nucl. Eng. and Design,106,291-313 (1988)

CODE NAME: PIN - W

Date: 21.6.1993

Version: 1993

Development team: M.Valach
R.Svoboda

Originally based on:

PIN micro
PIN 004

Address: NRI plc
250 68 Řež near Prague
Czech Republic

OBJECTIVES: Fuel rod thermomechanical behaviour under operational conditions. Code calculates fuel temperature, gap contuctivity, fission gas release and pressure, under steady-state conditions.

MAIN ASSOCIATED OPTIONS:

Options for Zry and Zr1Nb cladding materials Modular structure of the code (variuos calculation chains)
Interactive graphics

LANGUAGE: FORTRAN 77

Running on: tested on IBM-PC, DOS 5.0
principially platform independent

Number of instructions:

3600

Typical running time:
about 30 sec per one
time step (PC-AT386/387/40 MHz)

Links with other tools:

Lahey Fortran graphic library

Type of NUMERICAL treatments:

Temperature field in fuel is calculated by 1D FEM combined with WRM.
Physical models are based on a semiempirical approach.

Brief description of
MECHANICAL treatment:

- fuel thermal expansion
- cladding thermal expansion, radiation induced creep, strain caused by internal gas and coolant pressure

THERMAL treatment:

Temperature field in cladding, gap, fuel

MAIN MODELS:

radial power depression, fuel thermal conductivity, cladding-coolant head transfer, fission gases generation and release, fuel densification, relocation, swelling, cladding thermal conductivity, grain growth, creep, thermal expansion, fuel restructuralization.

DOMAIN OF USE:

Nuclear Fuel Rod behaviour modelling under steady state operational condictions for the NPP Safety Reports.

VALIDATION FIELD:

"ISO D-COM Blind Problem

.xperiments, performed by the RNRC "Kurchatov Institute".

based on:

KEY REFERENCES:

1. F.Pazdera,P.Strijov,M.Valach et al.: User's guide for the Computer Code PIN-micro, ÚJV 9517-T, Řež, November 1991.
2. Strijov P.,Yakovlev V.,Dubrovnin K.,Pazdera F.,Valach M.,Bárta O.: An improved version of the thermal Behaviour of an Oxide Fuel Rod. ÚJV - 6124-T, 1982.
   Strijov P.,Dubrovnin K.,Yakovlev V.,Pazdera F.: Computer and experimental VVER fuel rod modelling for extended burnup. IAEA TCM on Fuel Performance at High Burnup for Water Reactors held in Studsvik, Sweden, 5-8 June 1990. IWGFPT/36.

CODE NAME:

ENIGMA

Date:

22 July 1993

Version:

5.2

Development team:

P A Jackson, I Palmer of BNFL
R J White, W J Kilgour, J A Turnbull
of NUCLEAR ELECTRIC

Originally based on:

Completely new architecture
based on experience with
MINIPAT and SLEUTH codes

Address:

Berkeley Technology Centre
Berkeley, Glos, GL13 9PB, UK

O. JECTIVES:

To calculate fuel behaviour under steady state and transient conditions for LWR conditions in support of safety submissions for Sizewell B PWR. In particular: fuel temperatures for fault initial conditions, rod internal pressure, fission product release including ^{131}I and conditioning power for PCI analysis.

MAIN ASSOCIATED OPTIONS:

Code treats hollow as well as solid pellets, stainless steel cladding and some gas cooled reactor specific options, Code can handle different options of models at user discretion. The code has the facility to handle reactor and fill gas changes to treat refabrication experiments.

LANGUAGE:
FORTRAN 77

Running on:
platform independent; is run on mainframes, PCs and work stations.

Number of instructions:

Compiled code is 0.5 MBts.

Typical running time:

variable depending on complexity of time steps.

Links with other tools:

Type of NUMERICAL treatments:

Finite difference treatment, 1.5D stress calculation. Where ever possible sub-models are mechanistic to maximize confidence in extrapolating outside database.

Brief description of
MECHANICAL treatment:

Dimensional change of fuel and cladding by thermal expansion and creep, zircaloy growth in the cladding and fuel swelling by solid and gaseous fission products. Stress calculation with cylindrical symmetry allows pellet cracking with reduced fuel modulus in tension. A plane strain axial force balance is assumed, converging on radial displacement, axial strain and force balance. Treatment allows for both solid and hollow pellet solutions and transition from one to the other. Pellet wheatsheaf growth is an 'add–on' which does not interfere with the main calculation. Account is taken for strain concentration at pellet ends and over radial fuel cracks.

THERMAL treatment:

Axi–symmetric calculation of temperature distribution across waterside zircaloy oxide, clad thickness, fuel–pellet gap and up to 20 radial fuel zones. Transient calculation set by input flag. Iteration convergence on fuel surface temperature. Calculation includes an empirical recipe for fuel thermal conductivity decreasing with burn–up based on Halden and Riso experimental data. Gap conductance allows for parallel heat transfer by radiation, contact and gas conduction. Gas content of gap is at user discretion and allows for thermal feedback from released fission gas.

MAIN MODELS:

RADAR radial power distribution, fuel and gap conduction, thermal expansion, creep, Zr growth and oxidation, fuel densification, fission product diffusion coefficient dependent on temperature, rating and burn–up, solid fission product and gaseous swelling, integrated steady state and transient model for long lived fission gas release and swelling, steady state ^{131}I release, axial gas transport and thermal feedback, pellet wheatsheafing and stress/strain concentration at pellet ends and over radial fuel cracks

DOMAIN OF USE:

To calculate fuel behaviour under steady state and transient conditions for LWR conditions in support of safety submissions for Sizewell B PWR. In particular: fuel temperatures for fault initial conditions, rod internal pressure, fission product release including ^{131}I and conditioning power for PCI analysis.

VALIDATION FIELD:

Fuel centreline temperatures in transient and steady state conditions, dimensional changes: clad diameter and creep down, long lived fission gas release under steady state and transient conditions, ^{131}I release under steady-state conditions, clad ridging.

based on:

Halden, Riso and Studsvik in-pile experiments, EPRI-NFIR I irradiation experiments, WAGR irradiation experiments, CONTACT sweep gas experiments, commercial reactor irradiation and PIE data.

KEY REFERENCES:

Capabilities and Validation of the ENIGMA Fuel Performance Code by W J Kilgour, J A Turnbull, R J White, A J Bull, P A Jackson and I D Palmer, Paper presented to the ENS International Topical Meeting on LWR Fuel Performance in Avignon, France, April 21-24 1991.

CODE NAME: ENIGMA

Date: 22 July 1993

Version: 5.9

Development team: R J White, W J Kilgour, J H Shea and J A Turnbull of NUCLEAR ELECTRIC

Originally based on: NE–BNFL Joint owned ENIGMA 5.2

Address: Berkeley Technology Centre, Berkeley, Glos, GL13 9PB, UK

,JECTIVES:

To calculate fuel behaviour under steady state and transient conditions for PWR and Advanced Gas Cooled Reactor (AGR) conditions in support of safety submissions for all NUCLEAR ELECTRIC reactors. In addition, to analyse test reactor experiments and to extend application and validation to all UO_2 fuelled reactors including BWR and VVER types.

MAIN ASSOCIATED OPTIONS:

Code treats hollow as well as solid pellets, stainless steel cladding and AGR specific models. Code can handle different options of models at user discretion. The code has the facility to handle reactor and fill gas changes to treat refabrication experiments. The code has the capability of re-starting and automatic transient calculations. Code automatically introduces sub-steps for onerous histories.

LANGUAGE: FORTRAN 77

Running on: platform independent; is run on mainframes, PCs and work stations.

ıvumber of instructions: Compiled code is 0.5 MBts.

Typical running time: variable depending on complexity of time steps.

Links with other tools: Interfaced with whole core physics code PANTHER

Type of NUMERICAL treatments:

Finite difference treatment, 1.5D stress calculation. Where ever possible sub-models are mechanistic to maximize confidence in extrapolating outside database.

Brief description of
MECHANICAL treatment:

Dimensional change of fuel and cladding by thermal expansion and creep, zircaloy growth in the cladding and fuel swelling by solid and gaseous fission products. Stress calculation with cylindrical symmetry allows pellet cracking with reduced fuel modulus in tension. A plane strain axial force balance is assumed, converging on radial displacement, axial strain and force balance. Treatment allows for both solid and hollow pellet solutions and transition from one to the other. Pellet wheatsheaf growth is an 'add-on' which does not interfere with the main calculation. Account is taken for strain concentration at pellet ends and over radial fuel cracks. The code contains options and models for treating fuel-clad bonding in AGR pins and associated strain concentrations and inner bore clad crack growth.

THERMAL treatment:

Axi-symmetric calculation of temperature distribution across waterside zircaloy oxide, clad thickness, fuel-pellet gap and up to 20 radial fuel zones. Calculation is automatically in Transient mode for short time steps. Iteration convergence on fuel surface temperature. Calculation includes a pseudo-mechanistic treatment of burn-up dependent fuel thermal conductivity allowing for fission product release and generation of gas bubble porosity as well as an empirical recipe based on Halden and Riso experimental data. Gap conductance allows for parallel heat transfer by radiation, contact and gas conduction. Gas content of gap is at user discretion and allows for thermal feedback from released fission gas.

MAIN MODELS:

RADAR radial power distribution, fuel and gap conduction, thermal expansion, creep, Zr growth and oxidation, fuel densification, fission product diffusion coefficient dependent on temperature, rating and burn-up, solid fission product and stress dependent gaseous swelling, integrated steady state and transient model for long lived fission gas release and swelling, steady state ^{131}I release, axial gas transport and thermal feedback, pellet wheatsheafing and stress/strain concentration at pellet ends and over radial fuel cracks. Stainless steel properties and AGR specific PCI models for bonded clad and clad inner bore crack growth.

DOMAIN OF USE:

To calculate fuel behaviour under steady state and transient conditions for LWR conditions in support of safety submissions for Sizewell B PWR. In particular: fuel temperatures for fault initial conditions, rod internal pressure, fission product release including ^{131}I and conditioning power for PCI analysis. Code now extended to cover AGR behaviour: pin internal pressure, ^{131}I release, clad inner bore crack growth due to power manoeuvres and PCI failure probability.

VALIDATION FIELD:

Fuel centreline temperatures in transient and steady state conditions, dimensional changes: clad diameter and creep down, long lived fission gas release under steady state and transient conditions, ^{131}I release under steady state conditions, clad ridging. AGR PCI failure probability.

based on:

Halden, Riso and Studsvik in-pile experiments, EPRI-NFIR I irradiation experiments, WAGR irradiation experiments, CONTACT sweep gas experiments, commercial reactor irradiation and PIE data. AGR PCI data and failure probability from dedicated Halden irradiation experiment.

KEY REFERENCES:

Capabilities and Validation of the ENIGMA Fuel Performance Code by W J Kilgour, J A Turnbull, R J White, A J Bull, P A Jackson and I D Palmer, Paper presented to the ENS International Topical Meeting on LWR Fuel Performance in Avignon, France, April 21-24 1991.

CODE NAME: START

Date:

Vertion: 3

Development team: S.M. Bogatir
V.B. Lagovsky
G.A. Khvostov
V.I. Kuznetsov

Originally based on: START2

Address: All Russia Scientific Research Institute of Inorganic Materials
Rogov Str.5
Moscow, 123060
RUSSIA

OBJECTIVES: Full-scale nuclear fuel elements performance code.

MAIN ASSOCIATED OPTIONS: The START3 computer code provides a full-scale thermal hydraulic, and mechanical analisis of cylindrical LWR and FBR fuel rods behavior as a function of the actual reactor operating history

LANGUAGE: FORTRAN

Running on: IBM PC AT 386/387 or higher

Number of instructions: 10000

Typical running time: 30 minutes

Links with other tools: None

Type of NUMERICAL treatments:

Spatial treatment: the finite difference scheme is used

Time treatment: The 1^{st} order Euler's method is used for integration of the set of ODEs

Brief description of MECHANICAL treatment:

The fuel rod is devided into a number of sites on length and radius. The equations of axial and radial balance are solved in axisymmetric prediction in conditions of fuel pin cracking, elasto-viscoplastic and creep deformations, and various sort of volumetric restructing (swelling, densification, thermal expantion) and ect.

THERMAL treatment:

The thermal physical computation of START3 code includes computations of temperature fields, fuel grain restructing, gas release, and fuel swelling.

The computation of temperatures is performed by the numberical solving of non-stationary heat transfer equation for fuel rod with internal heat sources.

The fuel-clad conductance is calculated with the use of Ross-Stoute method with addition of contact component.

The main ideas of fuel grain restructing calculation were reported in Preston [5].

The gas release model utilized with START3 incorporates all of the commonly known aspects which are necessery to be accounted.

It involves:

- the low-temperature release mechanisms (such as "knock-out" and "direct recoil") discription [1]
- the "diffusion-traps" model of fission gas products behaviour in the grain interior
- the development of grain boundary porosity model [2]
- the poly-granular aggregate model for gas percolation process [2]

MAIN MODELS:

1) Thermal physical model of fuel pin behavior

2) Gas release and swelling model

3) Fuel-clad gap conductance model

4) Mechanical model of fuel rod behavior

5) Clad-coolant thermal hydraulic model

DOMAIN OF USE:

The most recent version of START3 is developed for both LWR and FBR applications, involving the ability to characterize the cause and time of fuel pin failure in both steady-state and transient operation of the reactor.

VALIDATION FIELD:

START3 has been subjected to separate validation exersises in he fields of
- fuel center-line temperature
- fission gas release
- clad damage accumulation (scc)
- fuel rod diameter change
- fuel restructing

based on:

The verification of thermal physical model in START3 code was based on data of experimental programm on envestigation of VVER-1000 fuel rod behavior under irradiation in research reactor MR.

Two series of experimental data characterized by significantly different levels of measured gas releases were used to calibrate and verify gas release model integrated with full-scale computational code START3:

1) Data on tests of VVER-1000 fuel rods in research reactor MR. Measured levels of relative gas releases lay in wide range: . % up to about 60 % by the end of irradiation.

2) Data on regulary fabricated VVER-1000 fuel rods after 3-year operating in fifth unit of Novo-Voroneg NPP with low gas release levels (~1÷3%)

KEY REFERENCES:

1. D.R.Olander, "Fundamental aspects of the nuclear fuel reactor elements", US Energy Reseach and Development Administration, 1976.
2. R.J.White, M.O.Tucker, J.Nuc.Mat., 118(1983)1
3. J.Rest, J.Nuc.Mat., 120(1984)195
4. J.A.Turnbull,R.J.White and C.Wise,IAEA Tech.Com.Meeting,Preston, September,1988
5. A.V.Medvedev,V.G.Kulakov,IAEA Tech.Com.Meeting,Preston, September,1988

Appendix III

INPUT DATA FOR ALL FUMEX CASES

1 REACTOR CONDITIONS

Coolant/moderator	D_2O
Coolant pressure	33 6 bar
Coolant temperature	240°C
Heat dissipated in pellets/total energy	0 937

Cooling conditions

All experiments are cooled under D_2O boiling conditions The actual system temperature is contained as variable T1 on all data files T1 may differ slightly from the nominal value of 240°C The system pressure will then follow the equilibrium pressure (235/30 81 // 230/28 03) However, it is permissible to always use 33 6 bar since small deviations will have negligible influence on calculated results

Boiling heat transfer can be assumed all along the active fuel length A widely used correlation is due to Jens-Lottes

Hcool = 1 26*(q**0 75)*exp(P*0 01619)

with Hcool in W/(m*m K), q = cladding surface heat flux in W/(m*m), P = system pressure in bar (assume constant value of 34 bar) Note that the correlation gives an unphysical value of 0 at 0 power A lower limit could be used (the exact choice is not critical)

The data for power given in all cases represents the heat energy flowing through the cladding and therefore excludes energy loss by gammas and fast neutrons This is based on a usable energy of 192 MeV per fission compared to a total value of 205 MeV per fission

2 PELLETS AND RODS

Radial flux depression in a pellet

For use with Bessel function or RADAR, the following values may be applied

Resonance escape probability	0 92
Fast leakage factor	0 975
Inverse diffusion length	$32\ 8*(E*p)^{0\ 8} + 54*(5/R)^{0\ 8}*(E*p)^{0\ 19}$ (1/m) (R = fuel radius (MM), E = enrichment (%), p = fractional density (-))

Insulating pellets, plenum temperature

All fuel stacks have insulating pellets with negligible heat generation at both ends The pellets are of about half the length of normal pellets However, for calculating rod pressure it can be assumed that the plenum temperature is about 15°C above moderator temperature at full reactor power (based on data from a number of other experiments with plenum thermocouples) Using the moderator temperature would not introduce an appreciable error End pellets are not included in the fuel column length

Pellet dishing

FUMEX-1 has pellets with both ends dished (1 1m^3/pellet)

FUMEX-3 has pellets with one-sided dishing Assume a dishing volume of 2% of pellet volume and a shoulder width of 1 5mm All other cases have flat-ended pellets

Spring force

As a rule, the spring is loaded to at least balance the weight of the fuel

Total free volume

The total free volume is determined by measurement (evacuation/refilling) It contains thus all accessible volumes including open porosity Subtract volumes of gap, dishing, open porosity to obtain (equivalent) plenum volume Note that hollow pellet bores are more or less filled with thermocouples, they have a negligible contribution to the free volume

Densification (powder route, pore size distribution)

Since densification seems to have a critical influence on calculated results (temperatures), and since parameters required by some models are not available in many cases, the following is proposed

FUMEX-1	At a power of 20 kW/m, a temperature increase of 60°C was observed from 0 to about 3 5 MWd/kg UO_2 Select densification such that this increase is reproduced with your models
FUMEX-2	Assume a final densification of 2 2% (volumetric)
FUMEX-3	Same procedure as in FUMEX-1 Temperature increases at 20 kW/m are 80°C for rod 1 (He filled), 280°C for rod 2 (Xe, 100 µm gap) and 265°C for rod 3 (Xe, 50µm gap)
FUMEX-4	Use the porosity distribution supplied as input specification
FUMEX-5	Use the porosity distribution supplied
FUMEX-6	Information on porosity distribution or temperature is not available Emphasis in FUMEX-6 is on the behaviour after the base irradiation To avoid run-off results please tune your models such that 20% fission gas release are not exceeded at the end of base irradiation

3 CLADDING

Cladding oxidation, crud creep

Cladding oxidation crud deposition and creep should be neglected under HBWR conditions

Cladding properties

An annealing temperature of 560 - 570°C for 2 - 5 hours means fully recrystallised cladding This should also be assumed for cases where the information is not given

Except for FUMEX-5 and maybe FUMEX-4 cladding mechanical properties should have little influence on the thermal/FGR performance

FUMEX 1

Rod Design

Pellet radius, inner (mm)	0 (0.9 for thermocouple bore)
Pellet radius, outer (mm)	4.045
Pellet length (mm)	10
Pellet end geometry	Both ends dished, spherical, 11 mm^3/pellet. Land width 0.6 mm
Cladding radius, inner (mm)	4.11
Cladding radius, outer, (mm)	4.75
Fuel stack length (mm)	810 enriched
Plenum spring force (N)	14
Total free volume (cm^3)	8.2
Filling gas	He
Filling gas pressure (bar)	10
Fuel surface roughness (μm)	2.0
Cladding surface roughness (μm)	0.5
Fast flux level (n/cm^2s)	$6.3*10^{11}$*LHR (kW/m)

Material Characteristics

UO_2 powder route	AUC
Enrichment U^{235}	3.5 w/o
Fuel density (% of theor. density)	94.1
Size distribution of fuel porosity	-------
Open porosity (% of total porosity)	62
Sintering temperature (°C)	1700
Resintering results	0.10 g/cm^3 increase after 2.5 hours sintering at 1700 °C
Fuel grain size (μm)	10
Cladding type	Zr-4
Cladding metallurgical condition	Annealed 5 hours at 510 °C
Cladding yield stress $\sigma_{0.2}$ (N/mm^2)	570 at room temp., 333 at 400 °C

Variables

Local heat rates

Pos	**Name**	**Unit**
810	LHR9	kW/m
708.75	LHR8	"
607.5	LHR7	"
506.25	LHR6	"
405	LHR5	"
303.75	LHR4	"
202.5	LHR3	"
101.25	LHR2	"
0.0	LHR1	"

Other variables

Name	**Unit**	**Meaning**
HRTF	kW/m	Heat rate at pos 737
AHR	"	Average heat rate
BU	$MWd/kgUO_2$	Burnup
T1	°C	Coolant temperature

8 hollow pellets at upper end for thermocouple.
LHR9 is calculated assuming a solid pellet.
Multiply heat rates interpolated between LHR8 and LHR9 by 0.955 to get heat rates of hollow pellets.

FUMEX 2

Rod Design

Pellet radius, inner (mm)	0
Pellet radius, outer (mm)	2.96
Pellet length (mm)	7.5
Pellet end geometry	Flat ended
Cladding radius, inner (mm)	3.01
Cladding radius, outer (mm)	3.51
Fuel stack length (mm)	443 enriched
Plenum spring force (N)	5
Total free volume (cm^3)	3.1
Filling gas	He
Filling gas pressure (bar)	10
Fuel surface roughness (μm)	2.0 (assumed)
Cladding surface roughness (μm)	0.5 (assumed)
Fast flux level (n/cm^2s)	$4.4 * 10^{11} *$LHR (kW/m)

Material Characteristics

UO_2 powder route	-------
Enrichment U^{235} (%)	13 w/o
Fuel density (% of theor. density)	94.3
Size distribution of fuel porosity	-------
Open porosity (%)	10
Sintering temperature	1650 °C, 3 hours
Resintering results	-------
Fuel grain size (μm)	7-10
Cladding type	Zr-2
Cladding metallurgical condition	Annealed 4 hours at 570 °C
Cladding yield stress	-------

Variables

Local heat rates

Pos	Name	Unit
443	LHR5	kW/m
332.2	LHR4	"
221.5	LHR3	"
110.8	LHR2	"
0.0	LHR1	"

Other variables

Name	Unit	Meaning
AHR	kW/m	Average heat rate
BU	$MWd/kgUO_2$	Burnup
T1	°C	Coolant temperature

FUMEX 3

Rod design	Rod 1	Rod 2	Rod 3
Pellet radius, inner (mm)	- - - 0 (0.9 for hollow pellets) - - -		
Pellet radius, outer (mm)	5.35	5.35	5.375
Pellet length (mm)	12.7	12.7	12.7
Pellet end geometry	- - - - - - One end dished - - - - - -		
Cladding radius, inner (mm)	5.4	5.4	5.4
Cladding radius, outer (mm)	6.25	6.25	6.25
Fuel stack length, enriched (mm)	140	140	140
Plenum spring force (N)	30	30	30
Total free volume (cc)	3.8	3.8	3.8
Filling gas	He	Xe	Xe
Filling gas pressure (bar)	1	1	1
Fuel surface roughness (μm)	1	1	1
Cladding surface roughness (μm)	0.5	0.5	0.5
Fast flux level	- - - - $3*10^{11}*AHR$ (kW/m) - - - -		

Material Characteristics

UO_2 powder route			
Enrichment U^{235} (%)	10	6	10
Fuel density (% of theor. density)	95	95	95
Size distribution of fuel porosity (%)			
Open porosity	10	10	10
Sintering conditions			
Resintering results			
Fuel grain size (μm)	3.4	≈20	3.4
Cladding type	Zr-2	Zr-2	Zr-2
Cladding metallurgical condition	- - - - - Annealed 560 °C - - - - -		
Cladding yield stress			

Variables

Name	Unit	Meaning
AHR1	kW/m	Average heat rate, rod 1
AHR2	"	Average heat rate, rod 2
AHR3	"	Average heat rate, rod 3
BU1	$MWd/kgUO_2$	Burnup rod 1
BU2	"	Burnup, rod 2
BU3	"	Burnup, rod 3
T1	°C	Coolant temperature

Centre hole in end pellet and in 6 enriched pellets.

Power in 6 hollow pellets ≈0.988*AHR

Power in 5 solid pellets ≈1.014*AHR

FUMEX 4

Rod Design

Pellet radius, inner (mm)	0.0 (0.9 for thermocouple bore)	
Pellet radius, outer (mm)	5.34	
Pellet length (mm)	12.7	
Pellet end geometry	Flat ended	
Cladding radius, inner (mm)	5.45	
Cladding radius, outer, (mm)	6.39	
Fuel stack length (mm)	781 enriched	
Plenum spring force (N)	10	
	Rod A	**Rod B**
Filling gas	He	92%He, 8%Xe
Total free volume (cm^3)	8.6	8.4
Filling gas pressure (bar)	3	1
Fuel surface roughness (μm)	1	1
Cladding surface roughness (μm)	0.5	0.5
Fast flux level (n/cm^2s)	$4.8*10^{11}$*LHR (kW/m)	

Material Characteristics

UO_2 powder route	-------
Enrichment U^{235}	9.9 w/o
Fuel density (% of theor. density)	95.0
Size distribution of fuel porosity	see separate sheet
Open porosity (% of total porosity)	10
Sintering temperature (°C)	1700
Resintering results (%)	-------
Fuel grain size (μm)	12 (see also separate sheet)
Cladding type	Zr-2
Cladding metallurgical condition	Annealed, ASTM B353 - 71
Cladding yield stress $\sigma_{0.2}$ (N/mm^2)	557 at 20°C, 320 at 382°C

Variables

Local heat rates

Pos	Name	Unit
781	LHRx9	kW/m
683.375	LHRx8	"
585.75	LHRx7	"
488.125	LHRx6	"
390.5	LHRx5	"
292.875	LHRx4	"
195.25	LHRx3	"
97.625	LHRx2	"
0.0	LHRx1	"

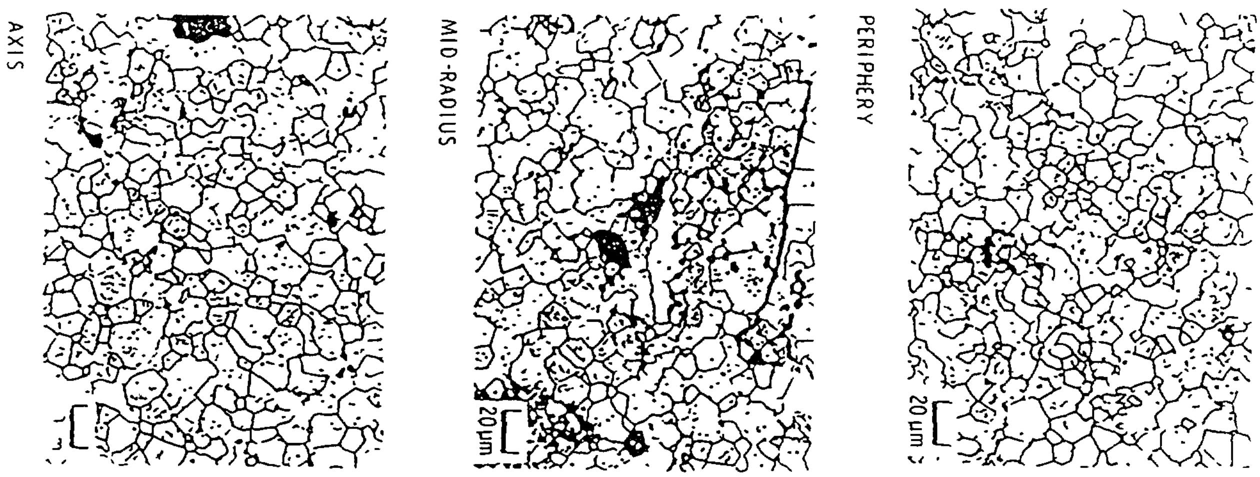
Other variables

Name	Unit	Meaning
AHRA	kW/m	Average heat rate, rod A
AHRB	"	Average heat rate, rod B
BUA	MWd/kgUO_2	Burnup, rod A
BUB	"	Burnup, rod B
T1	°C	Coolant temperature

There are nine local heat rates for each rod, named LHRA1-LHRA9 and LHRB1-LHRB9.

7 hollow pellets at lower end for thermocouple. LHRx1 is calculated assuming a solid pellet. Multiply heat rates interpolated between LHRx1 and LHRx2 by 0.985 to get heat rates of hollow pellets. Thermocouple position is 80.0.

FUMEX 4 : grain size distributions



Variation in Grain Size for 95% TD Stable UO_2 Pellets

the fuel pellets have a uniform grain size. These grains are approximately 12 µm in diameter, with a maximum grain size of less than 20 µm

TABLE Grain Size of As-Sintered Pellets

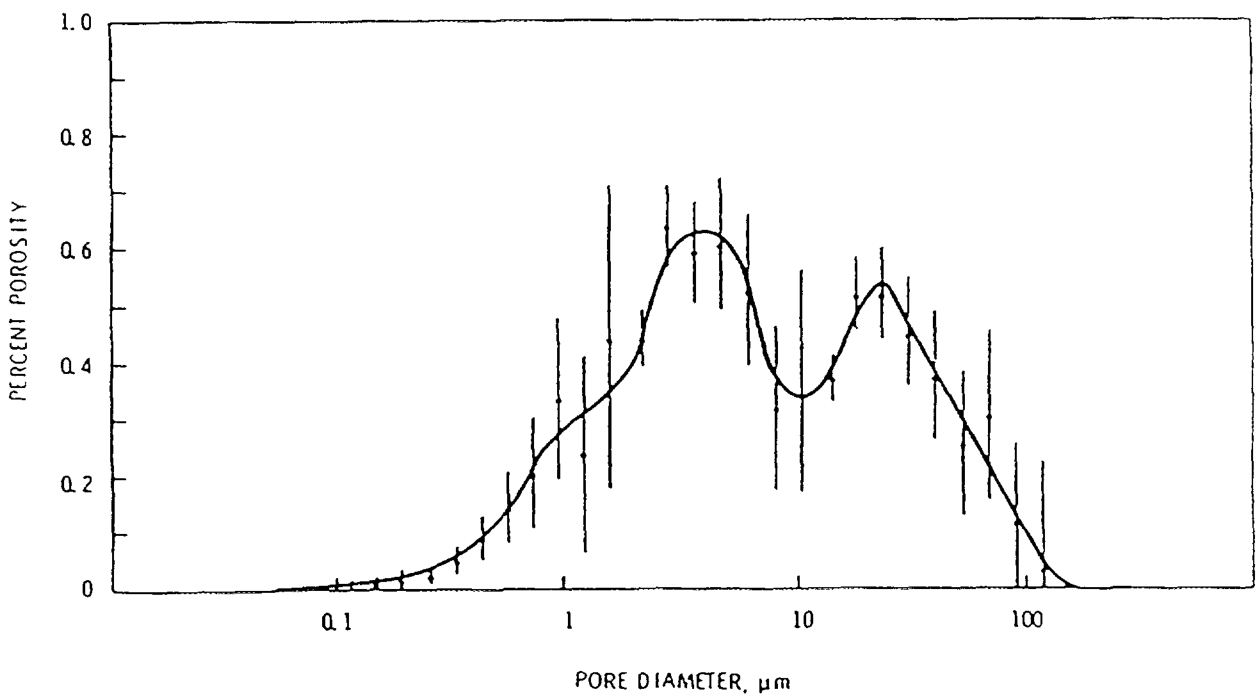
Fuel Assembly (95% Theoretical Density Stable Fuel)

Position	(Avg. dia, µm)
Peripheral	12 ± 2
Midradius	12 ± 2
Axial	12 ± 1

Averages from transverse sections of three pellets

Pore Distributions in 95% TD Stable UO_2
Fuel for Assembly

Radial Location	Periphery	Mid-Radius	Centerline
Pore Diameter, μm			
Median, All	7.3	7.3	13
Median, < 1 μm	0.7	0.7	0.9
Median, > 1 μm	7.3	7.3	13
Median, > 10 μm	37	28	28
Maximum	91	119	91



Pore Size and Volume Distribution for 95% TD Stable UO_2 Fuel (vertical lines indicate 2σ confidence limits at midpoint of each size range)

FUMEX 5

Rod Design

Pellet radius, inner (mm)	0 0
Pellet radius, outer (mm)	5.3
Pellet length (mm)	10 8
Pellet end geometry	Flat ended
Cladding radius, inner (mm)	5 405
Cladding radius, outer (mm)	6.27
Fuel stack length (mm)	457 2
Plenum spring force (N)	17
Nett plenum volume (cm^3)	2 6
Filling gas	He
Total free volume (cm^3)	4.25
Filling gas pressure (bar)	1.0
Fuel surface roughness (µm)	0.6
Cladding surface roughness (µm)	0.5
Fast flux level (n/cm^2s)	$4.8 * 10^{11} * LHR$ (kW/m)

Material Characteristics

UO_2 powder route		
Enrichment U^{235} (%)	3.93	
Fuel density (% of TD)	95.0	
Size distribution of fuel porosity	-	
Open porosity (% of total porosity)	10	
Sintering temperature (°C)	1700 (5 hours)	
Resintering results	-	
Fuel grain size (µm)	14.5	
Cladding type	Zr-2 (ASTM B353 - 71)	
Cladding metallurgical condition	570 °C annealing temp., 2 hours	
Cladding mechanical properties	**20 °C**	**340 °C**
YS (N/mm^2)	350	130
UTS (N/mm^2)	520	260

Variables

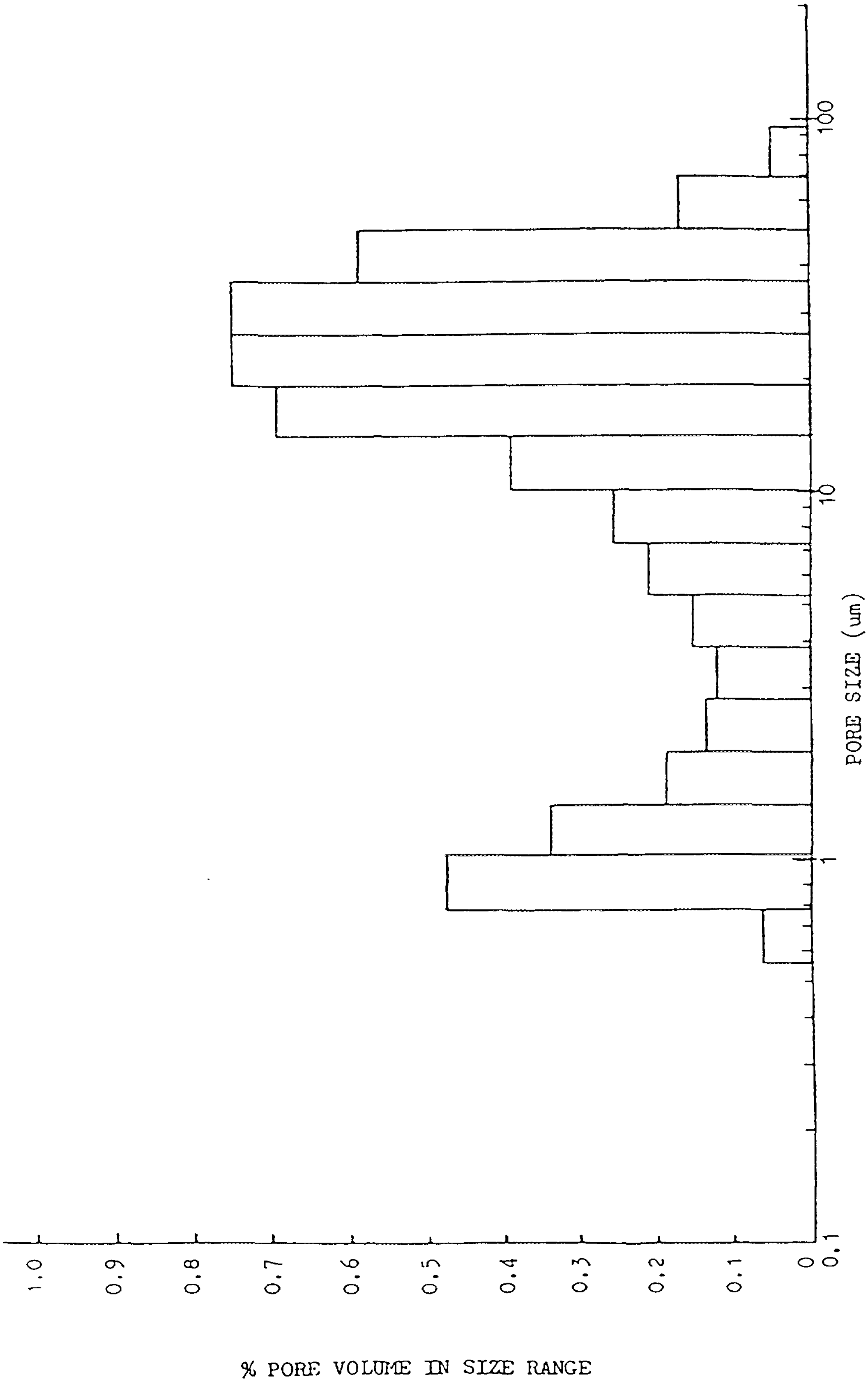
Local heat rates

Pos	Name	Unit
457.2	LHR5	kW/m
342 9	LHR4	"
228 6	LHR3	"
114 3	LHR2	"
0 0	LHR1	"

Other variables

Name	Unit	Meaning
AHR	kW/m	Average heat rate
BU	$MWd/kgUO_2$	Burnup
T1	°C	Coolant temperature

FUMEX 5 PORE SIZE DISTRIBUTION



FUMEX 6

Rod Design

Pellet radius, inner (mm)	0.0
Pellet radius, outer (mm)	5.27
Pellet length (mm)	10.4
Pellet end geometry	Flat ends, chamfered
Cladding radius, inner (mm)	5.4
Cladding radius, outer, (mm)	6.26
Fuel stack length (mm)	466 enriched
Filling gas	He
Total free volume (cm^3)	9.8 (base irradiation) *)
Filling gas pressure (bar)	1 (base irradiation) *)
Fuel surface roughness (µm)	1.0
Cladding surface roughness (µm)	0.5
Fast flux level (n/cm^2s)	$4.8*10^{11}*LHR$ (kW/m)

Material Characteristics

UO_2 powder route	-------
Enrichment U^{235}	9.88 w/o
Fuel density (% of theor. density)	94.7
Size distribution of fuel porosity	-------
Open porosity (% of total porosity)	10
Sintering temperature (°C)	1850
Resintering results (%)	0.87 $\Delta V/V$ change after 20 hours sintering at 1750 °C
Fuel grain size (µm)	16
Cladding type	Zr-2
Cladding metallurgical condition	Annealed 4 hours at 577 °C
Cladding yield stress $\sigma_{0.2}$ (N/mm^2)	333.2

Variables

Local heat rates

Pos	Name	Unit
466	LHR5	kW/m
349.5	LHR4	"
233	LHR3	"
116.5	LHR2	"
0.0	LHR1	"

Other variables

Name	Unit	Meaning
AHR	kW/m	Average heat rate
BU	$MWd/kgUO_2$	Burnup
TI	°C	Coolant temperature

*) After base irradiation, end plugs with a volume of 8.85 cm^3, filled with 50 bar helium were attached to the rods. Communication with the original rod free volume was established and pressure equilibrium between the volumes obtained. Complete mixing of the gases should be assumed.

Appendix IV

GUIDANCE FOR CALCULATIONS OF SIMPLIFIED CASES

General Specifications

Pellet inside diameter	0.00	mm
Pellet outside diameter	10.67	mm
(gap size)	(230)	(μm)
Cladding inside diameter	10.90	mm
Cladding outside diameter	12.78	mm
Dishing	none	
Plenum volume	2.5	cm^3
Fuel segment length	20	cm
Fuel density	95	% th. d.
Grain size	15	μm
Enrichment U-235	10	%
Fuel surface roughness	3	μm
Clad surface roughness	1	μm
Fill gas	He	
Fill gas pressure (20°C)	5	bar

Case 1

<u>Take the general specifications</u>
History: 20 kW/m from 0 to 50 MWd/kgUO_2.
Calculate and plot: Fuel centre temperature, rod pressure, fission gas release.

Case 2

<u>Take the general specifications</u>
History: 20 kW/m from 0 to 30 MWd/kgUO_2, 40kW/m from 30 to 50 MWd/kgUO_2.
The ramp at 30 MWd/kg, from 20 kW/m to 40 kW/m will last 0.5 hours.
Calculate and plot: Fuel centre temperature, rod pressure, fission gas release. Make also enlarged plot from 30 to 35 MWd/kgUO_2 (after ramp).

Case 3

<u>Take modified general specifications</u> as follows: pellet diameter 10.850 (gap 50 μm)
History: start up from 0 to 40 kW/m
Calculate and plot: Fuel centre temperature according to linear rate.

Case 4

<u>Take modified general specifications</u> as follows: pellet diameter 10.800 (gap 100 μm)
History: start up from 0 to 40 kW/m
Calculate and plot: Fuel centre temperature according to linear rate.

Case 5

Take modified general specifications as follows: pellet diameter 10.750 (gap 150 μm)
History: start up from 0 to 40 kW/m
Calculate and plot: Fuel centre temperature according to linear rate.

Case 6

Take modified general specifications as follows: pellet diameter 10.700 (gap 200 μm)
History: start up from 0 to 40 kW/m
Calculate and plot: Fuel centre temperature according to linear rate.

Case 7

Take modified general specifications as follows: pellet diameter 10.670 (gap 230 μm)
History: start up from 0 to 40 kW/m
Calculate and plot: Fuel centre temperature according to linear rate.

Additional plot

Please plot also
1) cases 3 to 7 on the same figure
2) temperature at 20 and 30 kW/m as function of gap size (case 3 to 7)

Case 8

Take modified general specifications as follows: pellet diameter 10.800 (gap 100 μm), gas Xe
History: start up from 0 to 40 kW/m
Calculate and plot: Fuel centre temperature according to linear rate.

Additional plot

Please plot also cases 4 and 8 on the same figure.

Statistical Analysis

As discussed at Halden during our last meeting, it is proposed to carry out these calculations including the following variations:

Power	±	5%
Initial gap	±	5μm
Thermal conductivity	±	5%

DATA ON FUEL CENTRELINE TEMPERATURES ON START-UP WITH DIFFERENT FILL GAS COMPOSITIONS

The following data are from the start-up of a single assembly containing 4 short rods which could be flushed with different gases. Each rod contained thermocouples at the top and bottom of the fuel column. At the same axial level as the thermocouple tip were rings of 3 neutron detectors. In this way the local powers were recorded with high accuracy. Measurements of centreline temperature were recorded during several slow power ramps when the rods were filled with: helium, argon or xenon at 2 atmospheres pressure. The data here are in the form of envelopes encompassing the readings of all 8 thermocouples. In the main part, the spread represents the stochastic nature of pellet cracking and fragment relocation.

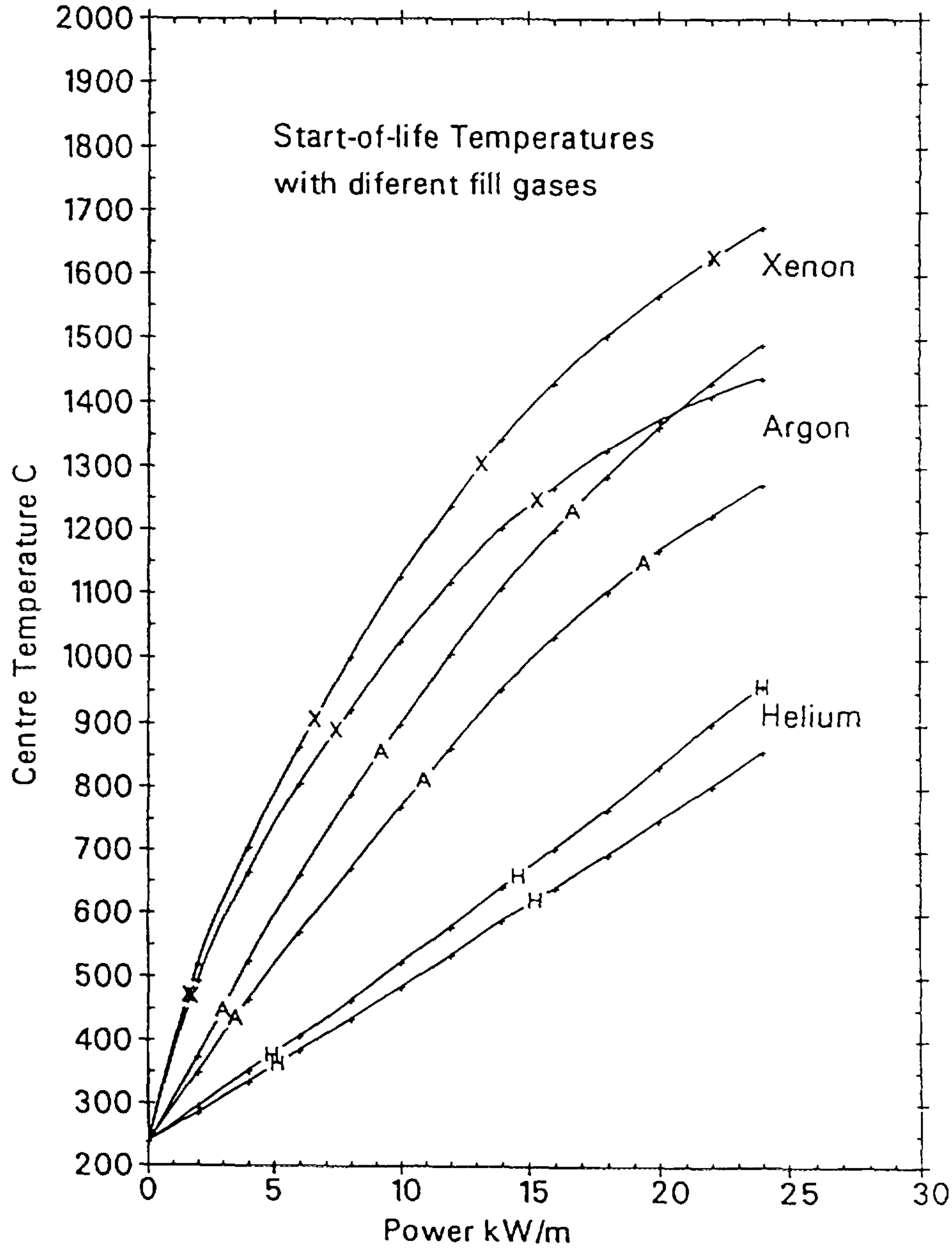
FUEL ROD PARAMETERS

Clad material	Zr-2
Outside diameter - mm	12.45
Inside diameter - mm	10.79
Thickness - mm	0.83
Fuel material	sintered UO_2 pellets
Outside diameter - mm	10.59
Inner bore diameter for T/C insertion - mm	1.8
Pellet length - mm	12.7 ± 1
Dishing - mm^3	24 ± 6
Fuel to clad diametral gap - μm	200
Fuel density	95% TD, 10.35 ± 1 g/cc
Enrichment	10 wt% U-235
Fuel length - mm	700
Coolant Temperature - °C	240
Coolant Pressure - atmos.	34

The clad is cooled under boiling conditions.

ENVELOPE OF START-UP TEMPERATURES IN DIFFERENT FILL GAS COMPOSITIONS: HELIUM, ARGON AND XENON

LOCAL LINEAR POWER KW/M	HELIUM TEMPERATURES (C)		ARGON TEMPERATURES (C)		XENON TEMPERATURES (C)	
	MIN.	MAX.	MIN.	MAX.	MIN.	MAX.
0	240	240	240	240	240	240
2	285	295	350	375	495	520
4	335	353	465	525	665	705
6	385	407	570	660	807	865
8	436	465	672	790	923	1003
10	485	523	770	900	1025	1127
12	535	580	865	1008	1120	1240
14	590	643	955	1110	1205	1345
16	642	705	1034	1203	1270	1432
18	695	768	1105	1287	1328	1505
20	750	835	1170	1364	1375	1568
22	805	903	1225	1432	1412	1625
24	860	960	1275	1492	1440	1675



Start-of-life Temperatures
with diferent fill gases
Xenon
Argon
Helium
Centre Temperature C
Power kW/m
2000
1900
1800
1700
1600
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
0
5
10
15
20
25
30

Appendix V

SUMMARIES OF PARTICIPANTS COMMENTS AS PROVIDED AT THE SECOND RESEARCH CO-ORDINATION MEETING

The comments are presented in the order below as given in Table II

Country	Organization	Code
Argentina	CNEA	BACO
Bulgaria	INRNE	PIN micro
Canada	AECL	ELESIM.MOD11
Finland	VTT	ENIGMA 5.8f
France	EdF	TRANSURANUS-EdF 1.01
France	CEA/DRN	METEOR-TRANSURANUS
CEC	ITU	TRANSURANUS
India	BARC	PROFESS
India	BARC	FAIR
India	NPC	FUDA
Japan	NNFD	TRUST 1b
Japan	CRIEPI	EIMUS
China	CIAE	FRAPCON-2
Romania	INR	ROFEM-1B
Swiss	PSI	TRANSURANUS-PSI
Czech Republic	NRI Rez	PIN/W
United Kingdom	BNFL	ENIGMA 5.2
United Kingdom	NE	ENIGMA 5.8 D
Russia	IIM	START 3

BACO Code - A.C. Marino, CNEA, Argentina

1. Code strong points

- Good thermomechanical performance in the range of low and intermediate burnup (Cases 1,2,4,5 & 6)
- Modular structure (very helpful with cases 3 & 6)
- Compatible with our Atucha and CANDU fuels (for instance self-standing and high pressure of filling gases firstly, and secondly with a collapsible cladding and normal pressure of filling gases).

Code weak points

- Poor agreement at high burnup (cases 4.1, 3.1 & 6)
- Mechanical axial treatments (case 5)
- Gap conductance when the filled gases include a high percent of Xe and Kr.

2. Improvements in the Code due to the CRP FUMEX

- New pre-processor of input data in the power history aspect (compatible with the models included in the BACO code).
- New post-processor of output data. Friendly and easy to use, based on Windows 3.1
- New mesh points distribution.
- General revision of the convergence criteria, fission gas release, creep and "cracking" models.
- We introduce the option of a greater number of axial zones
- Some secondary FORTRAN corrections.
- Real compatibility with several FORTRAN compilers.

3. Follow-up to FUMEX

A new version of the FUMEX cases only for code comparison using very limited and hyper-reduced power history. This type of calculation will leave just the pure behaviour of the models without the masked effects of the "real" irradiation (shutdowns, low variation in power, waiting times, etc).

4. FUMEX strong points

- All cases are stringent tests of code performance

Weak points

- Power history treatments, to many time steps making data handling difficult
- Densification parameters
- Clad elongation data parameters

PIN-micro Code, Svetlana Stefanova, Bulgaria

Our participation in FUMEX up to now has been very difficult, but very useful. We have got first hand knowledge and experience in fuel rod behaviour modelling and prediction.

1. **We obtained very satisfactory pre-test results, because of the following:**

   i) We improved the most important ("key") models in PIN-micro and processed them to be adequate to the FUMEX fuel rods.

   ii) We used special engineering estimations with the aim of overcoming the imperfections and the inherent restrictions of PIN-micro.

   iii) Our Russian co-authors are very experienced.

2. **The weak points are:**

   i) PIN-micro is a very simple code, validated against WWER fuel. The included fuel models are so simplified, that the difference between different fuel fabrication cannot be modelled.

   ii) The modification of the "key" models must be completed with clearly defined data.

3. **Recommendations**

   i) Further participation in FUMEX is very useful and important for us.

   ii) Recalculations of the first FUMEX cases with simplified power histories and using the finally modified models.

   iii) Would like new FUMEX cases with gas content and gap size variation.

   iv) Carry out a sensitivity study: power $\pm 5\%$, K fuel $\pm 5\%$, Gap FGR conductance $\pm$ 10%.

4. **Conclusions and lessons learned**

   i) The satisfactory pre-test results are due to:

   - the PIN-micro model improvements and modifications
   - special engineering estimations
   - otherwise, the authors are not experienced and it was very hard to participate in this CRP, but very, very fruitful.

   ii) The original PIN-micro version had to be improved to overcome some of its imperfections such as:

   - FGR and gap conductance models not adequate for extended burnup
   - fuel thermal conductivity not dependent on burnup
   - radial neutron flux depression model not accounting for rim effect and extended burnup
   - inconsistent burnup determination
   - lack of fuel creep model
   - improper cladding elongation model
   - cladding creep inadequate to the Halden fuel rods
   - others.

   iii) The simplified fuel material behaviour models built in PIN-micro are based on Russian fuel and are different from models necessary to run the FUMEX cases.

ELESIM Code - W. R. Richmond, AECL Research, Canada

The ELESIM CANDU fuel modelling code was modified to enable it to model the FUMEX irradiation cases. Our initial results for FUMEX cases 1 to 3, inclusive, showed large temperature under predictions compared with the measured temperatures. Upon re-examination of our computer code, we realized that we had not permitted the fuel-to-sheath heat transfer model to account for the fuel-to-sheath gap. Correcting this permitted ELESIM to considerably reduce the amount of temperature under prediction. The comments below, therefore, reflect this correction.

The strengths of the code were:

i) Its ability to handle a PWR type irradiation, without major changes. This is despite its development and validation specifically as a CANDU fuel code.

ii) The models' sensitivity to microstructural changes. This was shown by the big improvement in the calculation of temperatures in FUMEX 3.2 when the initial grain size was corrected to 20μm from 3.2μm.

iii) Reasonable fission gas release predictions for FUMEX cases 1 and 2. This is tempered, however, by the low results for fission gas release in FUMEX cases 3.1 and 3.2.

The shortcomings of the ELESIM code in this FUMEX exercise were:

i) Calculation of the fuel-to-sheath gap. The code provided reasonable agreement of the fuel centre temperature when the predicted fuel-to-sheath gap was closed. When it was open, the differences between measured and predicted temperatures were large.

ii) The steady-state nature of many of the models in ELESIM. This would likely have the greatest impact of the fuel pellet model and consequently affect the fuel-to-sheath gap.

As a result of the FUMEX exercise, as well as other research recently conducted by AECL, the fission gas release model has been changed to a more analytic form which has eliminated its strong dependence on the number and duration of the time steps. Development of a new pellet model is presently well underway. The FUMEX cases will provide valuable information to compare against a revised version of ELESIM, containing this new model.

Our recommendations for future work in the FUMEX programme are:

i) Conduct a sensitivity analysis of the codes to small changes in power and UO_2 thermal conductivity.

ii) To make available, as PC compatible data, the plots of centreline and internal gas pressure as a function of burnup. The other information given in the charts would also be very useful, and would permit easier comparison to predicted data.

ENIGMA Code - S. Kelppe, VTT, Finland

1. Strong and weak points

The ENIGMA code is very flexible as far as simulation of different reactor and irradiation conditions are concerned. Among others, the capability of describing details such as partially annular fuel stack, re-instrumented rods with gas change etc. has proven particularly handy for this exercise. Thus, practically no formal difficulties were encountered in the execution of the FUMEX cases.

In the results, the strong points are definitely the accurate temperature estimation for start-up conditions and, for helium and small gap xenon rods, also for temperature histories. For the xenon-filled rods with a larger gap there is under prediction of temperatures.

The burnup trend of thermal behaviour is generally right in all cases. The effect of densification seems to be over-emphasized early-in-life in xenon rods, which however is of minor importance in practical power reactor applications.

The under prediction of FGR in FUMEX 1 is not to be rectified with a modification in diffusion release. The mechanism of this low temperature release needs more studies. Similar behaviour has been encountered by VTT in some BWR experience.

Description of the mechanical behaviour is far from complete. In reality the PCMI must be largely governed by local phenomena and perhaps also pellet stack effects, not to be explained by overall fuel swelling and clad creep alone. There may also be a strong dependency on fuel type. Well characterized data, closely representative of the fuel used would be required.

2. Improvements considered

Addition of description for the rim, special features for Gd absorber rods and a model for BWR thermal hydraulic boundary conditions are being considered.

There may be a need to review the gap conductance model for conditions of a closing gap (solid contact) with simultaneous gas release, typical of strong power increase later-in-life.

Mechanisms of athermal release and their modelling need more work.

3. Recommendations

At the end of the FUMEX exercise, in the summary of the results, the behaviour of the codes could be tested to reproduce the burnup trend of temperatures at fixed power levels or temperature versus power dependency at several burnup stages. That could provide useful information on the status of the code capabilities. In the parameter studies the participants might consider the possible 'tuning' should report on the goal of improving these models.

The participants could also report, when available, their experience outside of FUMEX, with special attention to possible differences that might be addressed to fuel types and irradiation conditions other than those of Halden.

TRANSURANUS-EDF 1.01 Code - EDF, France

1. Introduction

We present here a brief analysis based on the comparison between FUMEX experimental results and the calculations performed with TRANSURANUS-EDF 1.01. The aim of this analysis is to identify some points to be discussed during the next TCM. The following topics are presented:

- problems met during the constitution of input data
- comparison between calculated and experimental results.

2. Problems met during the constitution of input data

The main problem we encountered during the constitution of input data, and which concerned more or less all FUMEX cases, was the lack of data on fuel densification. This is an important parameter for the prediction of thermal performance and, hence, fission gas release and rod internal pressure. Although the information provided allowed us to perform the calculations, we are not sure that densification is correctly modelled. This phenomenon is strongly correlated with pellet fabrication route and out-of-pile representative re-sintering tests results. But these data were generally not precise. Moreover, the model calibration based on a temperature jump is insufficient to correctly evaluate the amplitude and kinetics of densification.

The second problem, which only concerned the FUMEX-4 cases, was the evaluation of fuel porosity. The porosity spectrum given leads to a total porosity of 7.8%, which seems to be a rather high value. This has an impact on fuel thermal conductivity and centerline temperatures.

3. Comparison between calculated and experimental results

3.1 Thermal Performance

In general, the centerline temperatures are quite correctly evaluated by the code, during rod base irradiations. However, some specific aspects must be pointed out:

- In FUMEX 1, the temperatures are, more or less close to the experimental results. The slight under prediction is probably due to the fact that the amplitude and kinetics of fuel densification are not correctly evaluated. The densification may be higher than the value used in the data (1.84%).

- In FUMEX 3, the centerline temperatures are correctly predicted, even for the rods filled with xenon. However, the temperature during the final overpower period are, in the first two cases, over predicted (+200-250°C). The reason for this over prediction may be an over prediction of pellet-clad gap, due to an under prediction of fuel gaseous swelling during the power transient.

- In FUMEX 3.2 and FUMEX 3.3, the temperature decrease between BOL and EOL is predicted by the code. The differences between experiments and calculations are probably due to uncertainties on the modelling of phenomena which govern the pellet-clad gap (densification, fuel relocation). Gap conductance is very sensitive to the gap width in xenon-filled rods.

- In FUMEX 4A and FUMEX 4B, the temperatures are systematically over predicted, especially during and after the ramps. Here, this may be correlated with the uncertainties on fuel porosity and densification (cf § 2 above). On the other hand, experimental centerline temperature seems to be very low, compared with other results from Halden experiments (eg. IFA-432).

In conclusion, the centerline temperatures are generally evaluated with a good accuracy in "normal" (steady-state) operating conditions, and over predicted during power transients. The discrepancies between measured and calculated values may be explained, in steady-state conditions, by uncertainties on some input data. The systematic over prediction during transients may be due to a poorly modelled gaseous swelling. This point could be confirmed by comparisons between measured and calculated pellet-clad gaps at EOL.

3.2 Fission gas release and rod internal pressure

In normal conditions, fission gas release is accurately evaluated by the code. The rod internal pressure at BOL is also correct, indicating that internal free volumes are correctly calculated. In FUMEX 2, the over prediction of rod internal pressure at EOL is consistent with the over prediction of fission gas release. In FUMEX 4A, rod internal pressure is correctly calculated before the transient.

However, a general trend of the calculations is the systematic under prediction of fission gas release during power transients, and as a consequence of this, too low internal pressure jumps. Once again, this may be related to the lack of a convenient gaseous swelling and fission gas release model during transients.

3.3 Clad diametral deformations (FUMEX 5)

The code doesn't calculate any clad deformations during the final overpower periods. Again, the problem of transient gaseous swelling may be the cause of this poor prediction.

3.4 Axial rod elongations

Axial rod elongations are systematically over predicted. The axial growth model of TRANURANUS-EDF is empirical and rather complex: fluence, temperatures of clad material final annealing and irradiation are taken into account. But the irradiation conditions of the Halden reactor (low fluence and temperature, compared to PWRs) are very far from the domain of validity of this model.

4. Conclusion

In general, the thermal performance (evolution of fuel conductivity vd. burnup, gap conductance) and fission gas release (especially at low temperature), in normal steady-state operating conditions, are accurately evaluated by TRANSURANUS-EDF. Moreover, rod internal free volumes and pressures are correct (the results are consistent with fission gas release). These aspects are the strong points of the models included in the code.

The weakest point seems to be the modelling of gaseous swelling and fission gas release during off-normal situations. It should be noticed here that these two phenomena are strongly (and in a complex way) coupled during power transients, and this is not correctly taken into account in the code. This point is still under development. Fuel densification modelling is also rather controversial, and the calculated results are strongly dependent on the models and data used in the code.

5. Proposals for the next TCM and the future of the programme

According to the results of our analysis, we wish that the following points be discussed during the next TCM:

- input data concerning fuel densification and porosity;
- modelling of gaseous swelling and fission gas release during transients.

In order to give answers to our questions concerning the first point, it would be very useful to have a discussion about the data and results of other codes. We also wish to know whether other PIE results are available (missing PIE on FUMEX 2, pellet-clad gap, fuel column axial elongation and density measurements, etc.).

For the future, and before examining other cases, a detailed analysis could be performed and presented by the participants. The analyses could be completed by sensitivity studies (especially on fuel densification and porosity), and comparisons with other experimental results, as mentioned above.

METEOR/TRANSURANUS Code - CEA, FRANCE

The cases chosen for the FUMEX exercise are very different from those chosen to validate the METEOR/TRANSURANUS code (powder type, fuel design, power level, cladding materials, cladding mechanical conditions). The lack of data about fuel densification (kinetics and maximum value) induces uncertainties on fuel temperature.

After careful comparison between the experimental results METEOR/TRANSURANUS calculations, we can point out:

Strong points

- The thermal behaviour of fuel is calculated if the experimental conditions are close to PWR ones.
- The thermal threshold for fission gas release is well estimated (FUMEX 1, FUMEX 5).

Weak points

Fuel conductivity

For very low power levels and high burnup, fuel conductivity appears to be overestimated (FUMEX 1).

Fuel cladding gap conductance

Up to now, we tended to consider fuel and cladding surface roughness values as parameters of the gap conductance model. FUMEX 3 seems to show that we have to consider them as measurable physical data.

Fuel relocation

We have not provided FUMEX 4 calculations for the comparison exercise. However these cases show that for this type of geometry (large pellet) the code considerably overestimated the fuel temperature. These two cases demonstrate the need to introduce the pellet radius in modelling of the relocation displacement.

Fission gas release

Thermal fission gas release seems overestimated in the case of FUMEX 2. Release by recoil and knock out in case of large open porosities is under estimated (FUMEX 1).

In power transient cases, the code does not correctly account for the gaseous swelling of the fuel and the transient release, which is all the more regrettable as these phenomena have consequences on fuel temperature.

Code modifications

- From now on use the exact surface roughness values.
- Introduce the pellet radius in relocation modelling.

These two changes are directly deduced from FUMEX exercise and if they have a strong influence on the thermal behaviour, they remain minor for the code.

For fission gas release, FUMEX confirms the need of a special model for power transient and a more physical model for steady state conditions. But this work is part of a wider project already in progress.

TRANSURANUS Code - EC, K. Lassmann

Difficulties encountered

We ran all FUMEX cases with identical models and identical parameters corresponding to the input data. Detailed power histories were used for all cases. Some Monte Carlo statistic runs were performed. No difficulties in using the data provided or running the cases were encountered.

The predicted temperatures for FUMEX 1, 3.1, 3.2, 3.3, 4a and 4b are in good agreement with experimental evidence over the full range of linear rating and burnup. A slight general over prediction of temperatures (≈10%) is found at high burnup and high linear rating which is believed is due to an under prediction of the gap conductance.

Fission gas release can be compared for FUMEX 1, 2, 5 and 6 with experimental evidence. With the exception of FUMEX 1, the predicted fission gas release is in agreement with experimental evidence. A modification of the athermal part of the diffusion coefficient gave better agreement for FUMEX 1 which indicates that the athermal gas release is under predicted by present TRANSURANUS standard models.

New developments of TRANSURANUS models will concentrate on LWR high burnup models. It is intended to perform detailed re-calculation of all FUMEX cases including probabilistic analysis using the two statistical techniques employed int the TRANSURANUS code: Monte Carlo method and Numerical Noise Analysis.

The follow-up to FUMEX should include the following aspects:

- development of a broad data base of well defined fuel experiment irradiations
- a detailed discussion of theoretical concepts and models
- definition of specific experiments which allow clarification of possible open questions.

<u>FAIR Code - India</u>

Experience with FUMEX cases

All the FUMEX cases could be analyzed successfully. The modifications needed to handle certain cases (such as for refabricated rods after base irradiation) could be done easily due to modular structure of the code FAIR. The fuel pins with different filler gas compositions and various pressures could be handled satisfactorily. The predictions of FAIR are generally in agreement with the majority of the codes. From the results the swelling model is dependable. The physically based fission gas release model performed well for general operating temperature range and non-severe power ramps. The pressure calculations generally followed the pattern of fgr and experimental findings qualitatively.

It has been found necessary to review the low temperature release model, especially the low temperature grain boundary saturation limit. Similarly the fission gas release mechanism during power ramps involving high temperature requires review. A densification model should be developed, which should be applicable to all the FUMEX cases. The inherent assumption in the calculation of instantaneous mixing of fission gases should be modified to consider realistic mixing time. The computer time required for gap conductance convergence should be brought down by using a better convergence procedure.

Improvements to be considered in the codes:

- single densification model for all the FUMEX cases
- fission gas release during power ramps
- improved gap conductance model during PCMI
- diffusion coefficient calculation at low temperature and high burnup
- realistic procedure for axial gas mixing

Follow-up of FUMEX cases

- sensitivity analysis to identify region of uncertainties
- tuning of different models for better predictions
- availability of more transient experimental data, such as fgr, rod pressure and PCMI.
- incorporating the relevant physical phenomena in modelling the above concepts.

<u>FUDA MOD 1 Code - M. Das, India</u>

The computer code FUDA (Fuel Design Analysis) participated in all the FUMEX cases. In general, the code predictions compared well with the experiments at Halden under various parametric and operating conditions. In particular, the fuel temperature predictions at relatively higher ratings, lower burnup and under steady state conditions, matched very well with the experimental results (eg. case 3.1 at 5 MWd/kg burnup). This is understandable as originally the code was developed for fuel behaviour analysis of PHWR (CANDU) fuel which normally operate at ratings up to 50 kW/m and burnup up to 15,000 MWd/TeU. The weak point of the code version MOD 1 was the transient fission gas releases.

Since the original code FUDA MOD 0 was specifically written and validated for PHWR fuel, it was recognized at the time of participating in FUMEX cases that certain modifications to suit LWR fuels and high burnup needed to be carried out. Hence certain modifications, particularly in the gap heat transfer model, (eg. URGAP developed by K. Lassmann) was incorporated in FUDA MOD 1. Modifications in the other models such as fission gas release, could not be completed well in time for prediction of FUMEX cases.

FUDA MOD 2 version now incorporates the URGAS model developed by K. Lassmann and certain other tuning in the major models and submodels has also been carried out.

With regard to the follow up of FUMEX in the future, the following are suggested:

i) All the FUMEX cases need to be re-run after modifications and tuning.

ii) Physical interpretation of the behaviour under the various cases should be emphasized.

iii) Behaviour of annular pellets will be of interest.

iv) Sensitivity analysis may be performed.

v) New cases (simplified) are welcome.

vi) Some cases on PHWR fuel (CANDU) may be considered.

TRUST VIb Code - M. Ishida, NFD, Japan

The FUMEX benchmark test results have shown strong and weak points of the TRUST VIb code. In comparing the various aspects of TRUST code predictions with the predictions of other codes, **the strong points of the TRUST VIb code seems to be:**

i) in predicting temperature of fuel in relatively low burnup
ii) in predicting fuel deformation by PCMI

One exception of the successful prediction of startup temperature was for xenon-filled narrow gap rod (FUMEX 3.3). **Other weak points of the code seem to be:**

i) in predicting fuel temperature at ramp in high burnup
ii) in predicting FGR with sufficient accuracy.

Although the two items above are mutually dependent, the FGR model seems the main culprit of the inaccuracy.

Though we cannot develop a fuel performance code with the information supplied by the FUMEX programme alone, it prompted us to improve the FGR model in the code. Other improvements are being made in the data handling facilities for preparing input and processing output to prevent errors.

For the future of FUMEX, the following programmes would be helpful for improving the predictive capability of the codes.

i) Benchmark the codes with destructive PIE data such as xenon retention or porosity in pellet radial direction in addition to the overall FGR from rod.

2) Benchmark the codes from other data sources than Halden, if possible.

The first item is for looking into mechanisms of FGR, and the second is to add variety in the benchmarking cases.

EIMUS Code - S. Kitajima, CRIEPI, Japan

1. The code was able to calculate fuel temperatures and fission gas releases for the rods which were irradiated at moderate temperatures and burn-up (FUMEX 1, FUMEX 6).

   In the code,the pellet cladding gap model may be insufficient at low and moderate burnup (FUMEX 3).

   The code was verified with normal diameter rods, therefore the code did not provide good predictions of fission gas release for the thin rod FUMEX 4.

   Calculated fuel temperatures were higher than measured ones at very high L.H.R, because the code did not include the effect of thermal conductivity recovery process (FUMEX 3, FUMEX 4).

2. The fuel temperatures is the most important parameter to evaluate fuel behaviour. Therefore we must improve the accuracy, the applicability for different design and burnup range of the thermal conductivity model.

3. For future study, code improvements, detailed PIE, and data analysis by code are necessary and international co-operative efforts should be taken as systematically as possible.

FRAPCON-2 Code - Zhang Yingchao, Zhang Shishun, China Institute of Atomic Energy, China.

1. FRAPCON-2 is US NRC code used for analyzing fuel performance in Steady State conditions. Through the FUMEX exercises it can be found that:

Strong points of the code:

- the code can predict very well the fuel temperature at low burnup, especially when FRACAS-I mechanical model is selected for the exercises.
- in this code several parallel models can be selected to model the same physical phenomenon.

Weak points of the code:

- FGR prediction for some FUMEX cases, consequently the internal gas pressure is also not well predicted.
- fuel temperature at extended burnup.

Therefore, some models of the code will be improved or replaced by new models, such as:

- fuel thermal conductivity, considering the degradation of fuel thermal conductivity with burnup.

- fission gas release model, considering burnup effect.
- relocation model
- 2-D finite element model for PCMI analysis.

2. **Modifications have been done through FUMEX exercises: FRAPCON-2 is a very old code, the version was released in 1981. Through the FUMEX exercises, some shortcomings have been corrected:**

    - transplantation of the code on PC computer
    - improvement of uncertainty analysis model
    - recovery of 2-D finite element PCMI analysis model since the model could not be run due to some shortcomings in the code.

3. **Recommendations**

    - FUMEX programme is very useful to verify the FRAPCON-2 Code, so, the programme should be continued to improve the code by rerunning the FUMEX cases after blind test.

    - the IAEA should recommend or organize a new task on the development of some elementary models related to extended burnup through the FUMEX programme.

<u>ROFEM-1b Code - G. Horoianu, INR- Romania</u>

Summary

In accordance with INR objectives for these exercises we have selected for blind predictions with ROFEM-1b code five fuel rods: Nos. 2, 3.1, 3.2, 3.3 and 5. The principal reasons for this selection were the short fuel stack lengths and the relative flat axial power profiles; these are requirements of the particular version of the code used.

The comparison between blind code predictions and measured values show a reasonable capability of ROFEM-1b to analyze the thermal and mechanical behaviour of fuel rods irradiated in normal conditions up to 50MWd/KgUO_2.

Strong points of ROFEM-1b Code

Modelling of cladding deformation gives good results.

Weak points of ROFEM-1b Code

Generally, the code has a tendency to overestimate the fuel temperatures, especially above linear powers of 30kW/m. Among possible causes, we mention here the input limitations of the actual version of the code and the fuel to cladding heat transfer coefficient model.

Future improvements in ROFEM-1b Code

A new version of the code will be developed in order to eliminate the weak points mentioned above.

Difficulties in the FUMEX exercise

INR Romania has no experience in LWR type fuel rod behaviour analysis and has no access to the Halden project's experimental data. The successive revisions of the ROFEM code have been permanently verified using the results of the instrumented experiments on CANDU type fuel performed in the TRIGA research reactor of INR Pitesti.

Recommendations for the FUMEX Programme

Supplementary PIE examination seems to be necessary for improvement of the comparison possibilities of code predictions and to identify possible modelling problems. In this respect, the determination of fission gas release at EOL for fuel rods 3.1, 3.2 and 3.3 is of prime necessity.

TRANSURANUS/PSI Code - C. Ott, Paul Scherrer Institute, Switzerland

FUMEX Calculations

With the help of the TRANSURANUS/PSI code, we could perform each FUMEX text case without any difficulty. I take this opportunity to thank the Halden Project for having delivered well documented input and results data.

Strongest point:

The fuel centre-line temperature predictions are in quite good agreement with the experimental results, except in two cases (FUMEX 1 at EOL and FUMEX 3.1 EOL). This is a very positive result which should allow us to refine and improve some of the sub-models of the code because most of the physical phenomenon occurring during irradiation are strongly temperature dependent.

Weakest point:

The most important shortcoming is in the area of pellet-cladding mechanical interaction. Due to some convergence problems encountered for a few test cases using the standard friction force model of the code, a nonslip model has been applied for each calculation. Such an option suffers, of course, from severe limitations.

Follow-up of the Programme:

The first step will consist of a detailed comparison between our code predictions and the experimental results, in order to define more precisely the model improvements which have to be carried out. From a preliminary comparison, it is already possible to draw out areas in which detailed investigations and model calibrations will be performed:

- contact component of our gap conductance model
- fission gas release and fuel swelling models during ramp
- effective diffusion coefficient
- calculation of fuel-clad friction forces

It is planned to re-calculate all FUMEX test cases including also some sensitivity analyses.

PIN-W Code - Radek Svoboda, NRI, Czech Republic

1. Difficulties with exercises

- code primarily designed for PWR energetic reactors
- unable to model ramps, rim effect, fuel cladding axial interaction
- under predicted fuel centerline temperature
- under predicted fission gas release
- over predicted cladding elongation
- difficulties with input and output data handling

2. Code improvements to introduce

Already realized: fuel conductivity degradation
Zry material properties
relocation model tuning

To be done: fuel densification and swelling
gap conductivity
fission gas release at low power, high burnup, under power cycling

3. Recommendations

- More detailed experimental data (fission gas release vs burnup, fuel surface temperature, gas composition, fuel micro-structure).
- Smoother power history.

ENIGMA (NE) Code - P. Tempest, NE/UK

Both the ENIGMA codes were able to complete all the power histories in the FUMEX exercise. The main inconvenience was to convert ENIGMA output into the plotting format required by the IAEA.

1. Comments on code performance

Strong points

- Good fission gas release calculations in general - especially FUMEX 6
- Reasonable clad creep down - FUMEX 5
- Good agreement - xenon small gap FUMEX 3.3

Weak points

- Low fgr for FUMEX 1. Fuel temperature satisfactory.
- Poor agreement in early irradiation in FUMEX 3.2. Xenon rod - Large grain size - would like PIE evidence of 20μm grain size.
- Over prediction of fuel temperature at high rating and high burnup.

2. Improvements being considered following FUMEX exercises

Re-analysis of degradation of thermal conductivity variation with burnup. Also a review of fission gas porosity coupling with fuel conductivity and assess the effect of annealing of irradiation damage in fuel at high temperature.

3. Follow-up Suggestions

Repeat some of the blind exercises with a modified version of the code to improve performance.

START 3 Code - Russia

The predictions of the cases considered in the exercise are generally satisfactory.

The strong points:

- prediction of fuel central temperatures under moderate burnup
- prediction of the increase of fission gas release during ramp

The weak points:

- predictions of fuel central temperature at high burn-up.

The following models need improvement as a high priority:

- thermal conductivity degradation; dependence on burnup
- fission gas release under extended burnup particularly in the field of higher values.

The proposals for future activities on FUMEX:

- to obtain more detailed PIE data, including fuel structure
- calculation of the FUMEX cases with an improved code version
- a discussion among participants of the developments made to improve code predictions
- calculations of feedback effect under extended burnup and comparison with experimental data where possible.

Appendix VI

SUMMARY OF THE FINAL RESEARCH CO-ORDINATION MEETING

SESSION I

FUEL THERMAL PERFORMANCE

It was clear that there were similarities between the methods adopted by the codes. Several codes relied upon fuel fragment relocation to reduce the fuel-to-clad gap, thus increasing the gap thermal conductance. There was some discussion regarding the concept of 'hard' and 'soft' contact between fuel and cladding and the necessity to provide a mechanism for reducing relocation once the gap began to close. Numerical values for the relocation distance were around 30% of the initial gap size.

An alternative approach, but sometimes used in conjunction with pellet fragment relocation was the parallel heat path through regions where fuel and cladding were in contact. This component increased as the gap size decreased and became the main heat transfer route for small and closed gaps. Out-of-pile experiments have been performed to measure the magnitude and the contact conductance of flat plates, and these data demonstrated a $(\text{pressure})^{1/2}$ dependence on its value. Most codes adopted this dependence on interfacial pressure, but in one case a linear dependence was assumed and in another code, there was no dependence on pressure.

There was some discussion on the effect of surface roughness particularly for the situation of a closed fuel-to-clad gap. Whereas the laboratory experiments demonstrated the importance of the surface finish of the two materials in contact, in-pile experiments under high burn-up and closed gap conditions show temperatures to be substantially independent of the fill gas composition and hence little or no dependence on surface roughness. It was postulated, by Mr. Sah of BARC, and supported by PIE evidence that the clad roughness was less after irradiation than before. The possibility exists that the surface roughness changes with exposure but the process operating and its kinetics remain unresolved.

The process of fuel densification and swelling are addressed by all codes. Densification and low temperature swelling are usually treated using simple non-physical correlations. Some codes differentiate between small ($< 1\mu m$) and large fabrication porosity, allowing only the small pores to disappear. In some codes there is a correlation between in-pile densification and the standard 24 hours at 1700°C re-sintering test. In this way, a distinction is made between fuel pellet manufacturing routes. Intergranular gaseous swelling is more difficult to treat. Only a few codes attempt a detailed treatment of porosity evolution while many codes adopt empirical correlations as a function of local temperature.

The degradation of fuel conductivity is now well accepted and data are available with which to develop empirical modifications to the conductivity equation. Evidence was suggested for the effect of low temperature degradation due to irradiation damage. This saturates after only a brief exposure, typically ~20 days irradiation. In some codes this was treated by assuming that the UO_2 conductivity remained constant at the 500°C value for all temperatures <500°C.

Dr. Lucuta of AECL Canada has proposed an alternative formulation for UO_2 conductivity taking account of: porosity, temperature, impurity level, irradiation damage and O/U ratio. When compared to diverse experimental observations, prediction of the model compared very favourably with the data.

The principle topics where improvements could be made:

- The dependence of surface roughness on burnup
- The formulation of a reversible relocation
- The dependence of contact conductance on surface roughness and interfacial pressure
- Improvement in the treatment of densification/swelling
- The influence of the high burnup 'rim' structure
- Most appropriate formulation of UO_2 fuel thermal conductivity in terms of Temperature, Burnup, Irradiation damage, Additives, Density (including gaseous swelling), O/M etc.

SESSION II

FISSION GAS RELEASE

Experimental evidence suggests that there are three distinct regimes of fission gas release (FGR).

- Low power, low temperature, therefore low FGR. This can be treated as remedy dependent only on burnup and the majority of codes employ an empirical correlation for these conditions.

- High fractional release at high temperature with kinetics suggesting a diffusion rate controlled phenomena. Hence most codes adopt a similar approach where the release is controlled by an 'effective' single gas atom diffusion process. Under certain circumstances, this is augmented by a grain sweeping mechanism. Only one family of codes treats intergranular bubble formation to obtain the 'effective' diffusion co-efficient.

- The two regimes are separated by a threshold which has been established experimentally. Fuel operating above a critical temperature at a given burnup will trigger the high release mode, and the transition can be quite abrupt. Most codes that employ a diffusion model treat this threshold in terms of grain boundary saturation above which gas atoms are released along interconnected grain boundary porosity.

Despite the similarity in approach, the codes produce a diversity of predictions. This is because the values and functional dependence of critical parameters are not known with accuracy. Also, it is possible that other mechanisms, at present ill defined, play important roles under particular circumstances. In addition to grain boundary sweeping, other potential mechanisms include thermal resolution, bubble mobility and dislocation sweeping.

There is a clear need for further information on gas release and also the need for a comprehensive database for code and model validation. Specific issues include:

- Evidence for or against grain boundary sweeping
- Data on intergranular bubbles, their size and concentration as a function of temperature and burnup
- Availability of a comprehensive database for code validation
- Data on additional phenomena - thermal resolution/dislocation sweeping etc.

At high burnup there is the additional complication of 'rim' formation. The structure, formation and properties of this structure need further investigation.

SESSION III

MECHANICAL BEHAVIOUR

Compared to the thermal analysis, the treatment of mechanical behaviour has to deal with a more complex system including the cladding and the axial dimension. This may have contributed to the fact that not all participants treated the mechanical behaviour aspects of the FUMEX cases. These include radial deformation (F5) as well as PCMI manifested by cladding elongation (F1 & F4). There were clear deficiencies of code predictions compared to measurements.

A common feature of all codes participating in the FUMEX project is axial symmetry, but otherwise a variety of solutions can be found, including FEM, FD and semi-analytical treatments in one or two dimensions. Only few codes include axial sliding for non-zero contact forces. Ridging is treated by all 2-D codes and is approximated by some 1-D codes. Other features pertaining to mechanical behaviour modelling are compiled in the attached Table.

Several examples of the influence of mechanical behaviour on the results of thermal performance calculations were given, emphasizing the interlinked nature of fuel behaviour. However, a unified treatment poses problems and approximate solutions are therefore still applied. An example is the different thermal and mechanical gaps which are assumed in some codes to obtain satisfactory results. Some codes are used to supply input to other codes which treat special aspects of fuel behaviour, e.g. ridging and stress concentration. This kind of co-operation is quite useful to reduce complexity and computation time.

Good solutions for treating mechanical interaction could be found in several codes. It was suggested that less well developed codes should take advantage of this but avoid over-simplification. The result would be better predictions of mechanical and thermal behaviour and fission gas release.

Recommendations:

- A good understanding of mechanical behaviour is essential for overall successful predictions

- Both 1-D and 2-D models have their merit, but in either case the treatment of structurally weak cracked pellets (beyond the assumption of free thermal expansion) seems important

- Unified thermal and mechanical modelling, avoid different assumptions for these two areas

- Include axial sliding as a possible condition for the case of non-zero contact forces

Data: Sufficient material is included in the FUMEX cases

CODE	based on	Mechanics									Contact		Failure			FUMEX mechanical	
					FUEL			CLAD									
		axial symmetry	FEM	FD	plain strain	plain stress	cracked pellet	plain strain	shell	ridging	stick	sliding (nonzero contact force)	SCC	damage accum.	failure model	satisfied	to be improved
BACO	-	X		X	X	X	-p	X		X			X			?	?
EIMUS	Femaxi III	X	r-z				X			X	X	X					
ROFEM	Femaxi III	X	r-z							X	X	X	X		?	Y/r	
ELESIM	-	X															
ENIGMA/BNFL	ENIGMA	X		X	X		E	X		X	X			?	X		
ENIGMA/NE	ENIGMA	X		X	X		E	X		X	X			?	X		
ENIGMA/VTT	ENIGMA	X		X	X		E	X		X	X			?	X	?	?
FAIR	-	X	r-z		X	X	X			X	X				X		
FRAPCON-2/China	FRAP-CON	X	(x)												X		
FRAPCON-2/Turkey	FRAP-CON	X													X		
FUDA	Ele sim	X	(x)	X						X							
PIN-MICRO	GT-2	X							X								
PIN-W	GT-2	?															?
PROFESS	-	X						?									
START 3	-	X															
TRANSURANUS/CEC	TRANSURANUS	X	semi-analytical		X		X	X			X	X	X	X	X		Y/z
TRANSURANUS/EDF	TRANSURANUS	X	semi-analytical		X		X	X			X	X	?		X	?	
TRANSURANUS/CEA	TRANSURANUS	X	semi-analytical		X	?	-p	X			X		?				
TRANSURANUS/PSI	TRANSURANUS	X	semi-analytical		X		X	X			X	X				N/z	Y/z
TRUST 1b	-	X	r-z		X			X			X	X				yes!	

r = radial
z = axial
-p = pressure at crack surface
E = modified Young's modulus
Y = yes
N = no

Features marked according to code descriptions submitted for FUMEX. Codes may have features even if not marked in table.

SESSION IV

MATHEMATICAL METHODS

Fuel modelling computer codes normally use a one-dimensional, axis-symmetric model to simulate the thermo-mechanical behaviour of the fuel. The fuel rod is divided into a number of axial segments (so called zones, slices or nodes). The pellet and clad are divided with concentric rings (annuli). Thermo-mechanical equations are solved using a finite-difference iterative procedure through successive clad and fuel annuli. After all the segments are analysed, the segments are coupled together (axial coupling) with quantities such as rod internal pressure, fission gas content or the axial friction forces between fuel and cladding. due to this axial-radial coupling, this type of representation is described as quasi two dimensional or one and a half dimension. Out of 19 codes that participated in the FUMEX exercise, only 1 or 2 codes used two dimensional finite-element method for thermo-mechanical fuel behaviour analysis. A few codes use a two-dimensional finite element method for local analysis of stress and strain and thus perform a ridging analysis and pellet hourglassing. Some of the codes have automatic time step control where certain key physical parameters are checked to see if they have converged or nearly converged. Standard numerical methods such as Bisection, Regula falsi, Newton-Raphson methods are used. Explicit, implicit or semi-implicit integration procedures are adopted. Some codes use a Crank-Nicholson scheme for transient conditions. One code uses the secant method.

There was some discussion on the specific advantages of one numerical method over another, if any. Advantage or otherwise of 2-D codes was briefly discussed. It appeared that mathematicians are generally not involved for advice on solution techniques of specific problems.

The following areas need further consideration:

- Code structure
- Time step features
- Numerical stability (robustness)
- Convergence
- Run-time
- Modifications made after the FUMEX exercise
- Code objective vs. recommended modelling technique (1D/1.5D/2D/3D).

SESSION V

SPECIFIC ISSUES

Several features of the codes can be considered under the five headings below:

Specific Problems on Clad Behaviour

Some codes attempt to model clad failure by pellet-cladding interaction. Two approaches are used:

a) An empirical treatment using calculated stress as a criterion. A threshold stress for failure is derived from calculations applied to a series of experimental power ramps. This approach may be used even if the maximum stress is systematically under predicted, as is the case for 1D mechanical computation.

b) A more mechanistic approach is based on results of crack propagation rate in out-of-pile tests. To apply such models, the real maximal stress (at ridge level) is needed. This stress can be obtained either by 3D computation or by a transfer function from stresses computed by 1D, to stress computed by 3D calculations. Because of the difficulty of measuring clad stress, none of these models can be considered as validated today.

One code addresses interaction at pellet end-caps for design purposes.

A model calculating the growth of external corrosion in PWR has been presented, based on out-of-pile loop test results and validated against in-pile results. The main difficulty of such a calculation is the very high sensitivity to temperature.

Specific Problem on Fuel

Most of the codes are calculating radial depression flux and Pu formation. Many of them use the RADAR model with more or less improvements.

In some codes, a special effort has been made in order to give a good description of radial Pu formation and burnup in the rim. The corresponding swelling is modelled and the consequences on thermal conductivity may be assessed.

A semi-empirical model of iodine released has been presented, taking into account non-interlinked and interlinked fuel.

No comment was made on the fuel chemistry, suggesting that it is sufficient to assume that in UO_2 fuel the O/M remains equal to 2.0.

Special Fuels

Only a few codes model the behaviour of MOX fuel, introducing its properties and taking into account the influence of this fuel on the neutronics and the different models.

One code models the heterogeneities of such a fuel, dealing with the evolution of agglomerates.

An effort is still to be made on the modelling of this type of fuel in order to have a good descriptions of gas behaviour and fuel chemistry.

Modelling fuel with gadolinium is in progress in a few codes but is not at an advanced stage or validated.

Special Calculations

Some codes have the possibility of performing calculations for refabricated pins, which is necessary if the validation database includes such experiments.

Statistical Analysis

This is considered in more detail in Section 6.3.

Most of the participants have made an uncertainty analysis which overall are in good agreement. Few codes have made probabilistic assessments which, when compared, give some scatter in the standard deviation.

Depending on user requirements, the main areas requiring further attention are:

- Evaluation of maximal clad stress during power ramp in order to improve PCI failure prediction. 2D or preferably 3-D mechanical computation is probably necessary.

- A more mechanistic description of gas behaviour in MOX fuel. The specific microstructures of this fuel is probably to be taken into account

- Modelling of gadolinia bearing fuel is still at an initial stage. Much effort is still required.

- Statistical analysis is needed at the validation stage and for use of codes in safety analysis.

SESSION VI

QUALITY ASSURANCE

Quality Assurance is very important during the development and use of fuel performance codes.

The topics requiring attention include configuration management software, code maintenance, testing, validation, verification, code release, user support, training and portability.

The level of quality assurance used by different code developers vary widely with only three codes using configuration management software. However, it is clear that validation of a fuel performance code against experimental data is a fundamental and universal feature of all the quality assurance systems.

The use of various tools to aid in code development and quality assurance include compiler options and debugging tools.

With the advent of FORTRAN 90, it was also recognized that in the long term the code developers should move away from FORTRAN 77 and start to use FORTRAN 90 as a new standard for scientific software.

Overall it was clear that the role of quality assurance is of fundamental importance in developing and maintaining any fuel performance code, particularly if it is used for nuclear safety calculations.

The principle topics required for development and use of codes include:

1. A document describing the implementation of software quality assurance
2. Increased awareness of the software tools available to aid quality assurance
3. Set up a well defined database of experimental results to aid in code validation
4. Consider using FORTRAN 90 as the code language.