THE PUBLIC DOMAIN DATABASE ON NUCLEAR FUEL PERFORMANCE EXPERIMENTS (IFPE) FOR THE PURPOSE OF CODE DEVELOPMENT AND VALIDATION

P. Menut
Division of Nuclear Fuel Cycle
and WasteManagement
IAEA Wagramestrasse 5
P.O. Box 100
A-1400 Vienna
Austria
+43 1206 022763

E. Sartori
OECD Nuclear Energy Agency
Le Seine St. Germain
12, Boulevard des Iles
92130 Issy-les-Moulineaux
France
+33 (0)1 45 24 10 72

J.A. Turnbull
Independent Consultant
Cherry-Lyn
The Green
Tockington
Bristol BS32 4NJ
UK
+44 1454 620887

ABSTRACT

The paper describes the compilation and maintenance of a database on nuclear fuel performance for the purpose of code development and validation. This work is carried out in close cooperation and coordination between the OECD/NEA, the IAEA and the IFE/OECD/Halden Reactor Project. The paper outlines the background behind the formation of the database and the progress made to date in assembling data sets from various sources encompassing PWR, BWR, WWER and PHWR reactor systems. The aim of the project is to provide a comprehensive and well-qualified database on Zr clad UO_2 fuel for model development and code validation for the public domain. The data encompasses both normal and off-normal operation and include prototypic commercial irradiations as well as experiments performed in material testing reactors. The paper outlines recent acquisitions, highlighting specific topics of interest to modelers.

INTRODUCTION

With increasing emphasis on economics as well as safety, reactor operation is heavily supported by “best estimate” code calculations. The principal function of a fuel performance code is to describe the behavior of reactor fuel in the most accurate way possible under whatever conditions – both normal and off-normal – that are required by the licensing authority. In this way, all aspects of fuel performance are treated simultaneously and in a self-consistent manner. By aiming toward a “best estimate” calculation or one that is intentionally biased, the uncertainties in the conditions under which the code is applied are under the control of the user.

The need for calculations to be best estimate necessitates that the code be developed and validated against good quality data. The most obvious source of these is the power reactors for which the calculations are to be applied. However, this source is insufficient, as data are also needed for fuel experiencing transients and other off-normal operating conditions which cannot be reproduced under experimental conditions in power reactors. For this reason, code development and validation must have access to both types of information, and it is therefore of importance to include data obtained from dedicated experiments in test reactors.

This requirement was identified by the OECD/NEA Nuclear Science Committee (NSC) Task Force who, in their report recommended the compilation of a public domain database on fuel performance for the express purpose of fuel performance code development and validation.[1]

The task force ran concurrent with the IAEA FUMEX program, in which “blind” predictions were compared with a selection of in-pile data from Halden Project experiments.[2] During this exercise it became apparent that many countries did not possess an adequate code for predicting LWR fuel rod behavior, particularly during off-normal conditions. The principal reason for this was the inadequacy of the data available to these countries on which to develop and validate models. This was convincingly demonstrated by the improvements made by the end of the program when advantage had been taken of the experimental data issued to participants.

Following the recommendations of the NSC task force, compilation of the International Fuel Performance Experiments Database (IFPED) commenced. The work comprises:

- *Acquisition* of data through discussion and negotiation with originators.
- *Compilation* of the data into a standard form and content as agreed by a task force set up for supervising the work.
- *Peer review* of the data by independent experts.
- *Integration* and indexing of the data into the IFPE database, inclusion of all used reports in electronic form.

These activities were first discussed at the previous ANS meeting held in Portland, Oregon.[3] Thus, the present paper reports the progress to date and provides examples of the more interesting and important topics covered by recently acquired data sets.

CHOICE OF DATA

From the outset it was recognized that the database should apply to all commercially operated thermal reactor systems and that the data should be both prototypic, originating from power reactor irradiations with pre- and post-irradiation characterization, and test reactor experiments with in-pile instrumentation and PIE exploring normal and off-normal behavior. It is recognized that experiments have been performed where the data remain of commercial interest, for example, much of the details of modern MOX performance remains proprietary to the manufacturers, and it is not the intention to compromise such arrangements. However, zircaloy clad UO_2 pellet fuel can be largely regarded as a “standard product” and as such, release of what was previously proprietary data can only benefit the nuclear community at large.

A particular aspect of the compilation is the inclusion of data generated within internationally sponsored research programs whose confidentiality agreements have expired. Such data, although available in principle, have not been widely used. The inclusion of such data is of particular importance in situations where the originating organization has lost staff through retirement or has changed its terms of reference. For example, the Risø laboratories in Denmark no longer perform nuclear research and find difficulty in resourcing the supply of information for their three fission gas release projects. In such cases, there exists the danger of losing access to the data altogether.

EXTENT OF PARAMETERS INCLUDED

The database is restricted to thermal reactor fuel performance, principally with standard product zircaloy clad UO_2 fuel, although the addition of advanced products with fuel and clad variants is not ruled out. The data encompass normal and off-normal behavior but not accident conditions entailing melting of fuel and clad, resulting in loss of geometry. Early data sets concentrated on measurements of fuel temperatures and fission gas release. Recently, more emphasis has been placed on dimensional changes, oxide thickness measurements and hydrogen pickup for commercial fuel and the behavior of defect fuel with respect to coolant activity.

Of particular interest to fuel modelers are data on: fuel temperatures, fission gas release (FGR), fuel swelling, clad deformation (e.g. creep-down, ridging) and mechanical interactions. Data on these issues are of great value if measured in-pile by dedicated instrumentation and in this respect, the IFPE Database is fortunate in having access to several diverse experiments. In addition to direct in-pile measurement, every effort is made to include PIE information on clad diameters, oxide thickness, hydrogen content, fuel grain size, porosity, Electron Probe Micro Analysis (EPMA) and X-Ray Fluorescence (XRF) measurements on cesium, xenon, other fission product and actinides.

Emphasis has been placed on including well-qualified data that illustrate specific aspects of fuel performance. For example, cases are included which specifically address the effect of gap size and release of fission gas on fuel-to-clad heat transfer. In the context of thermal performance, the effect of burn-up on UO_2 thermal conductivity has also been addressed. This is illustrated by cases in which fuel temperatures have been measured throughout prolonged irradiation and at high burn-up where sections of fuel have been refabricated with newly inserted thermocouples.

Regarding fission gas release, data are included for normal operations and for cases of power ramping at different levels of burn-up for fuel supplied by several different fuel vendors. In the case of power ramps, the data include experiments where in-pile pressure measurements show the kinetics of release and the effect of slow axial gas transport due to closed fuel-to-clad gaps. In order to provide data on transients, out-of-pile annealing experiments are included for more detailed information on isothermal fission gas release and the effect of material and operating parameters.

The IFPE database now includes measurements on the release of radioactive fission products from two sorts of experiments. The first are data from swept fuel experiments where released gases are entrained in a sweep gas (usually helium) and measured at a gamma detector situated outside the reactor. The second relate to the escape of gaseous and volatile fission products through cladding defects in failed fuel into the coolant. In this case, the data are in the form of coolant activity levels under different operating conditions.

Fuel is at risk during power ramps to pellet clad interaction (PCI) and failure by stress corrosion cracking (SCC). There have been many ramping programs conducted to investigate propensity to failure of both PWR and BWR fuel. Discussion with Studsvik Nuclear has resulted in a representative number of cases being released and included in the database.

Although zirconium based alloys have proved an excellent trouble free cladding material for low and medium exposure, the extension of fuel cycles to high burn-up has uncovered problems with oxidation, hydrogen pickup and degraded mechanical properties. High burn-up clad properties have been addressed in several irradiation programs and appropriate data have been acquired for inclusion in the database.

BRIEF DESCRIPTION OF DATA

At present, the IFPE database holds some 381 cases comprising the following:

Halden irradiated IFA-432	5 rods
Halden irradiated IFA-429	7 rods
Halden irradiated IFA-562.1	12 rods
Halden irradiated IFA-533.2	1 rod
Halden irradiated IFA-535.5 &.6	4 rods
The Third Risø Fission Gas Release Project	16 rods
The Risø Transient Fission Gas Release Project	15 rods
The SOFIT WWER Fuel Irradiation Program	12 rods
The High Burn-up Effects Program	81 rods
WWER rods from Kola-3	32 rods
Rods from the TRIBULATION program	19 rods
Studsvik INTER-RAMP BWR Project	20 rods
Studsvik OVER-RAMP PWR Project	39 rods
Studsvik SUPER-RAMP PWR Sub-Program	28 rods
CEA/EDF/FRAMATOME Contact 1 & 2	3 rods
AEAT-IMC NFB 8 and 34	22 samples
CEA/EDF/FRAMATOME PWR and OSIRIS ramped fuel rods	4 rods
CENG defect fuel experiments	8 rods
CANDU elements irradiated in NRU	36 rods
Siemens PWR rods irradiated in GINNA	17 rodlets

Many of these data sets have been discussed in a previous paper.[3] Here we confine attention to the most recent additions, concentrating on the most interesting aspects of the data.

The 87 PWR and BWR rods included from the Studsvik ramp tests are a sub-set of the total data available on the failure propensity in power ramps by PCI and SCC. In all cases, the in-pile testing was followed by an extensive PIE program including diameter changes before and after ramping and fission gas release measurements on unfailed rods.

The OVER-RAMP program utilized KWU and Westinghouse rods encompassing design variables which were irradiated in KK Obrigheim or BR-3 in the range 12-31 MWd/kgU before ramping. Five of the 25 KWU variants failed, as did seven of the 15 Westinghouse variants.

The SUPER-RAMP PWR sub-program consisted of six groups of rods supplied by Westinghouse and KWU with variations in design covering: standard design, gadolinium doped UO_2, large grain size and annular fuel. The rods were irradiated in either KK Obrigheim or BR-3 at modest ratings to levels of burn-up in the range 33-45 MWd/kgU before ramping in Studsvik to final power levels in the range ~40 to ~50 kW/m. KWU supplied ten standard rods and four rods containing 4 wt% gadolinium of which none failed. However two rods out of five containing large grain fuel did fail. Westinghouse supplied five standard rods of which three failed, and four rods containing annular pellets which all failed.

The INTER-RAMP BWR program was designed to establish the failure threshold of 20 standard un-pressurized ASEA-ATOM fuel rods in the range 10-20 MWd/kgU. The test matrix contains three main variables: 1) clad heat treatment (a comparison of recrystallized annealed “RX” versus cold worked plus stress relief annealed “SR”), 2) pellet-clad diametral gap, and 3) fuel density. The rodlets were base irradiated in boiling capsules and fast ramped in a pressurized loop in Studsvik. Seven of the 12 variants irradiated to 10 MWd/kgU failed, as did four of the eight variants irradiated to 20 MWd/kgU.

As discussed later, it is planned to add further Studsvik ramp cases to the database, though the cases already included constitute a substantial volume of data on which to develop or validate PCI models for failure assessment.

The CONTACT series of experiments was a program of in-pile tests conducted in the SILOE reactor in Grenoble, France, funded jointly between CEA and Framatome. Short rods with Zr-4 clad UO_2 pellets of typical PWR 17×17 design were irradiated under conditions designed to simulate those of commercial PWRs. Each rod was equipped with a fuel centerline thermocouple, diameter gauge, gas lines providing a flow of gas through the rod and internal pressure gauges to measure pressure drop along the fuel stack. The gas flow entrained released fission gases which were measured by a gamma detector installed in the out-of-reactor gas handling system. The experiment is unique in that the rods operated under near constant powers for the majority of their lives. CONTACT 1 operated at a constant 40 kW/m up to a burn-up of ~22 MWd/kgU whilst CONTACT 2 and 2bis operated at 25 kW/m to burn-up levels of 5.5 and 12.4 MWd/kgU respectively. The data include temperatures as a function of burn-up, clad diameter changes as a function of power and burn-up, stable (^{85}Kr) and radioactive fission gas release as a function of center temperature and burn-up. The change in clad diameter as a function of burn-up for CONTACT 1 and 2bis is given in Figure 1. Both rods showed an initial decrease in diameter resulting from clad creep down. This continued to the end of irradiation for the low power rod 2bis, but for the high power rod 1, after a burn-up of ~5 MWd/kgU the diameter gradually increased as a result of fuel swelling. The kinetics of stable fission gas release as measured with the long half-life ^{85}Kr is shown in Figure 2 and the release of the short half-life ^{85m}Kr as a function of centerline temperature is shown in Figure 3. These last data are useful for developing and validating models for calculating “gap inventories” of radiologically significant species like ^{131}I, for example, see Ref. [4].

Figure 1. Clad diameter change measured at zero power as a function of burn-up for CONTACT 1 and 2bis

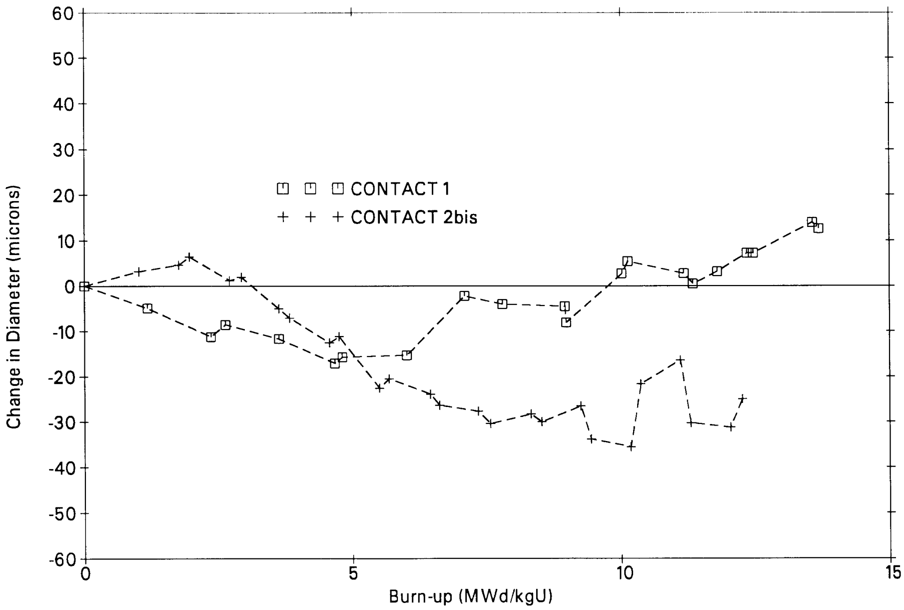


Figure 2. ^{85}K release as a function of burn-up for the three CONTACT experiments 1, 2 and 2bis

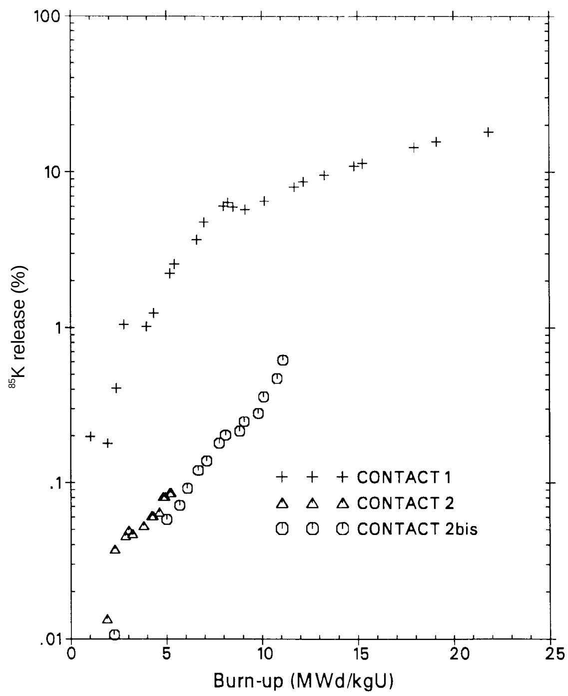
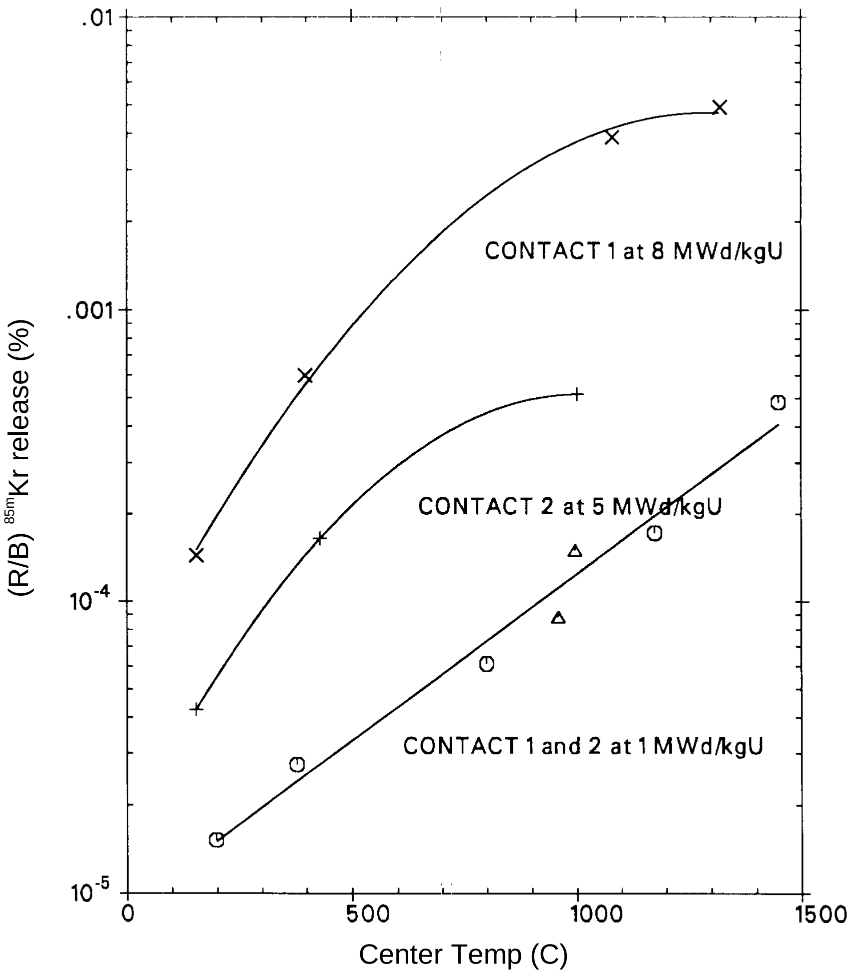


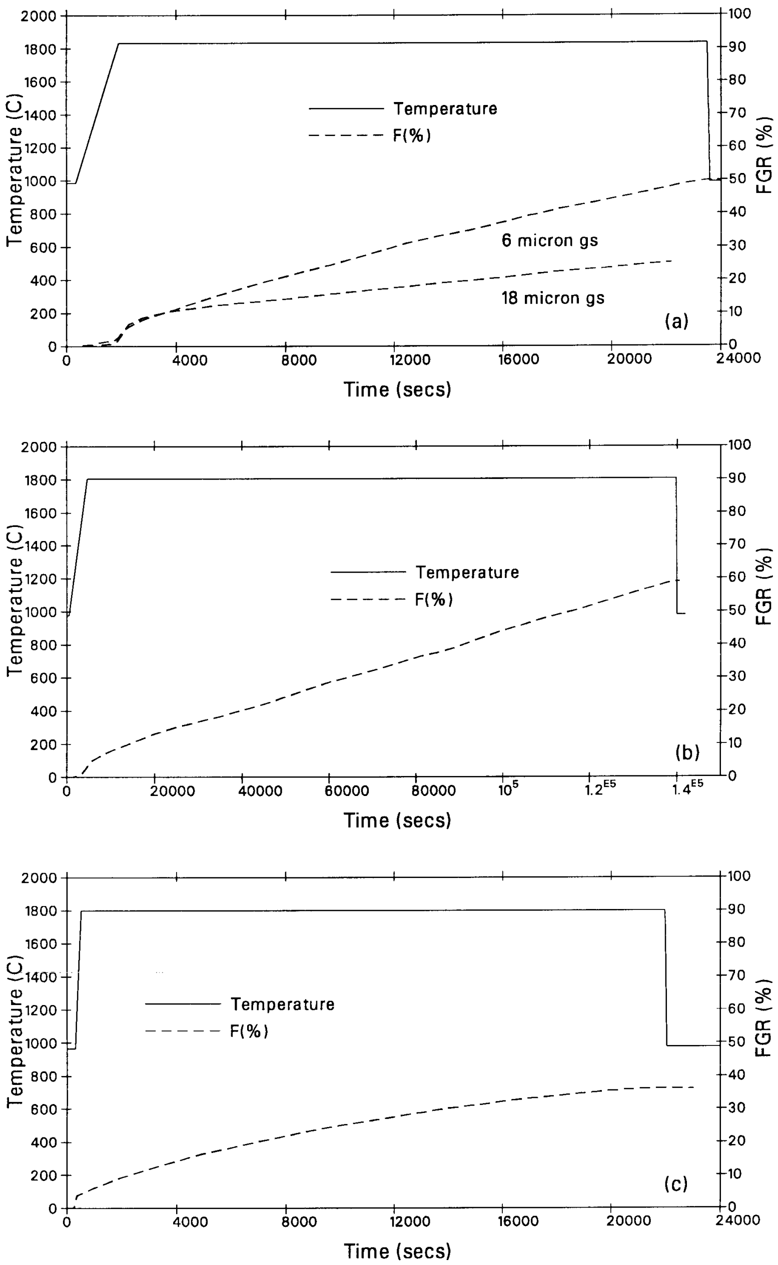
Figure 3. (R/B) for the radioactive ^{85m}Kr as a function of center temperature at different levels of burn-up for CONTACT 1 and 2



The next data set is the first departure in type of data from previous additions as it comprises a set of measurements made of fission gas release during out-of-pile annealing experiments. AEA Technology at Harwell carried out the experiments in two campaigns under funding from the Industry Management Committee (IMC): 10 tests as NFB-8 and 12 tests as NFB-34. The data were released for inclusion in the database by the Health and Safety Executive (HSE) of the UK. Small samples of UO_2 fuel were extracted from CAGR fuel pins and annealed in helium at pre-defined temperatures of 1500 to 1900°C for periods between 2 and 40 hours. The rate at which the final temperature was attained varied from 0.1 to 8.0°C/s in order to determine whether or not the behavior of intragranular bubbles, which were present during and after irradiation, was sensitive to changes in temperature ramp rate within this range. The fuel used for this study had a burn-up of ~17 MWd/kgU and a mean linear intercept (mli) grain size of 6 or 18 μm. Fission gas release at the end of the base irradiation was low in both cases, <0.1%, with no visible grain boundary porosity. Examples of the results are illustrated in Figures 4(a)-(c). Figure 4(a) shows the effect of grain size for a ramp rate of 0.5°C/s followed by a hold at 1800°C for six hours. Figures 4(b) and 4(c) both show the gas evolution from 18 μm ramped to a final temperature of 1800°C. In Figure 4(b), the temperature ramp rate was 0.2°C/s, followed by an extended anneal for 38 hours. Figure 4(c) shows the

Figure 4. The fraction of ^{85}Kr released during out-of-pile annealing experiments as a function of time

(a) Effect of grain size
(b) Kinetics for an extended hold at 1800°C
(c) Effect of a rapid increase in temperature 8°C/s prior to a hold at 1800°C



results of a more rapid increase in temperature at a rate of 8°C/s, followed by a hold for six hours. These experiments provide valuable data for developing models of fission gas release during rapid high temperature transients where the effect of the irradiation conditions is small compared to that of temperature and time. Note that the classical parabolic diffusion release kinetics evident at short times changes to a more linear behavior at long times.

Data on prototypic commercial PWR fuel performance were obtained with agreement from the EDF, CEA and Framatome. The data are for four full-length rods irradiated in EDF reactors. Rods K11 and J12 were irradiated for two cycles in Gravelines 3 and 5, respectively, to ~24 MWd/kgU, G07 was irradiated for three cycles in Gravelines 3 to ~35 MWd/kg and rod H09 was irradiated in Cruas 2 for four cycles to ~46 MWd/kgU. Prior to irradiation, these rods were well characterized, and were subjected to extensive PIE after irradiation. During PIE measurements made included: diameter change, oxide thickness, hydrogen content of cladding, length change, fission gas release, pellet density, radial distribution of fission products and actinides and metallography. Figure 5 shows the relative axial power profile for rod G07 and oxide thickness along the rod. Care was taken to accurately reproduce the axial power profile when processing the data by constructing 18 axial zone power histories reflecting the difference in power at and between the grids.

Sections of rods J12 and K11 were cut from span 5 and re-fabricated for ramp testing in the CEA OSIRIS reactor at Saclay. Rodlet J12-5 was conditioned to 21 kW/m before ramping to 39.5 kW/m without failure. Rodlet K11-5 was conditioned at 24 kW/m and ramped to 43.7 kW/m without failure. Subsequent PIE provided measurements of clad diameter changes, fission gas release and metallography of the fuel structure at different elevations. Figure 6 shows the measured changes in clad diameter for J12-5, whose magnitude can be seen to follow the axial power profile.

A series of loop experiments have been performed by the CEA in France on short, intentionally defected fuel rods with the aim of measuring and interpreting the rate of escape of fission gases and iodine under a range of experimental conditions of linear power, defect type, gap dimensions, etc. This body of data is largely unpublished, but with the agreement of the EDF and FRAMATOME, data from some of the experiments have been acquired for inclusion in the database. The rods were specially fabricated with typical 17×17 geometry for irradiation in dedicated loops to simulate PWR thermal hydraulic conditions. The EDITH series were performed specifically to investigate the steady state escape from pre-defected fuel rods.

Figure 5. Axial relative powers and measured oxide thickness as a function of axial height for EDF irradiated rod G07

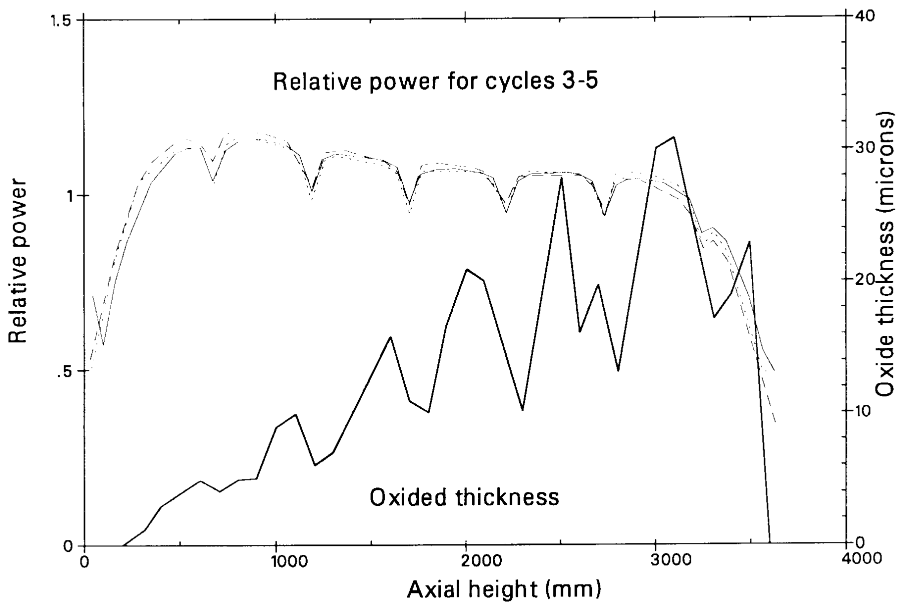
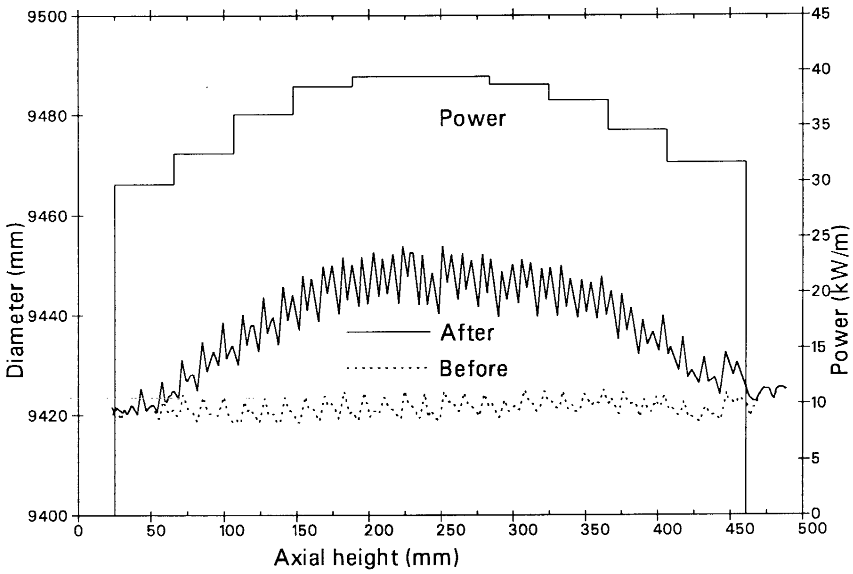


Figure 6. The power distribution and measured diameter as a function of axial height for the OSIRIS ramped rodlet J12-5



The CYFON series investigated escape during variable power/load following sequences, and the CRUSIFON series investigated the escape during steady power from rods initially intact but defected during irradiation. The experiments were all performed in the SILOE reactor in one of two water loops, one called Bouffon and the other Jet Pompe. Both loops operated at 155 bars pressure, but the flow rate in Jet Pompe was higher by a factor of three than that of Bouffon. Additionally, the Jet Pompe operating temperature of 280°C was more typical of PWR conditions than the 150°C used for Bouffon. The concentration of fission products in each loop circuit was measured by diverting a small flow through an out-of-pile circuit and past a gamma spectrometer. Because of the different flow rates, the characteristics for fission product detection were different in the two loops. Although Bouffon had a response closer to the PWR, the greater flow rate in Jet Pompe allowed a better resolution of the time dependence of fission product release.

The results from these experiments, which cover steady state and transient power operation on new and pre-irradiated fuel, provide invaluable data for modeling the escape of fission products from defective fuel rods. In a commercial reactor, such modeling is important for assessing the occurrence and characteristics of defective fuel from analysis of coolant activity. Knowledge of the presence of defective fuel can eliminate the need for premature shut down of the reactor and also minimizes outage time, thus improving the safety and economics of power generation.

The results of these experiments show that a complete description of the escape of fission products through a clad defect is substantially more complicated than release into an intact fuel rod. They show that:

- The rate of release from the fuel is enhanced as the UO_2 is oxidized by the coolant.

- Fission product behavior within the fuel-clad gap is dictated by the prevailing internal thermal hydraulic conditions, i.e. the ratio of the water to steam. The data show that the behavior of the iodine isotopes is distinct from that of the gases through their capacity to be trapped on internal surfaces of the rod. It would appear that iodine escape from a defective rod is enhanced when water is present in the gap.

- The rate of escape through the clad defect will vary depending on the local pressure difference across the hole created by the thermal hydraulic conditions and also by the physical size and location of the defect. In the case of the latter the proximity to the plenum, as a potential internal reservoir for water, is important.

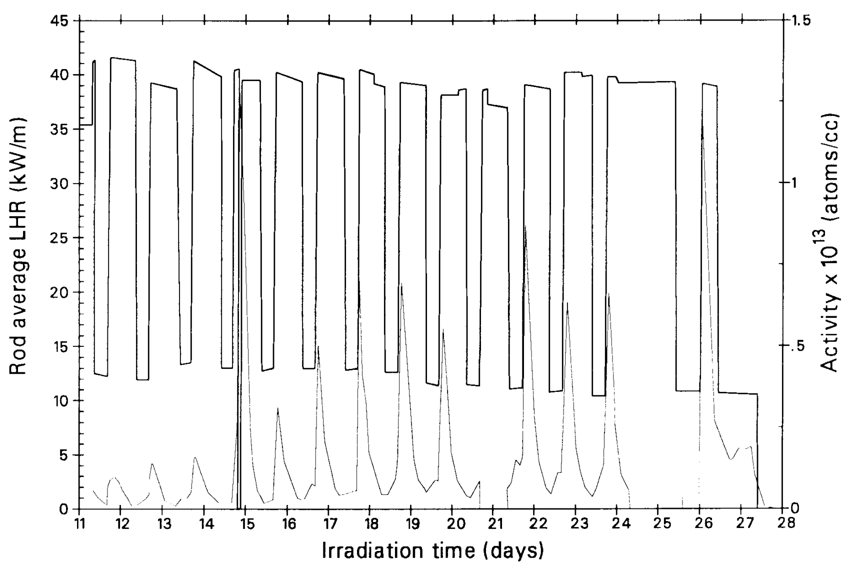
An example of the loop concentration of ^{131}I during power cycling of CYFON 2 is shown in Figure 7. It is to be noted that "spikes" of activity occurred on each instance of power reduction when coolant was first drawn into the internal volume and then expelled as the water was converted to steam. This is the origin of "iodine spikes" observed on shutdown at the end of reactor cycles.

One of the objectives pursued in compiling the database is to include data from all commercial reactor systems. AECL Chalk River has supplied information pertinent to CANDU fuel in the form of data on 18 elements (rods) from each of two bundles irradiated under appropriate conditions in the NRU test reactor.

Bundle NR was a prototype 37-element fuel bundle for the CANDU 600 reactor. For irradiation in the NRU reactor, the center fuel element was removed and replaced by a central tie rod for irradiation purposes in the vertical test section. Coolant for the test was pressurized light water under typical PHWR conditions of approximately 9 to 10.5 MPa and 300°C. The fuel elements used 1.41 wt% ^{235}U enriched UO_2 fuel pellets and were clad with zircaloy-4 material, the inner bore of which was coated with a graphite layer. The outer elements used three types of pellet-stack-to-end-cap geometries: a 350 mm^3 plenum insert (six elements), a 580 mm^3 plenum insert (six elements) and no plenum insert (six elements). Intermediate and inner element rings had no plenum insert. Outer element burn-ups reached average measured levels of 235 MWh/kgU. Outer element powers were steady during the irradiation and ranged between 58 and 62 kW/m.

Bundle JC was a prototype 37-element fuel bundle for the Bruce-A Ontario Hydro reactors irradiated in the NRU reactor in a similar fashion to Bundle NR. The fuel elements used 1.55 wt% ^{235}U enriched UO_2 fuel and were clad with zircaloy-4 material with a graphite coated inner bore. The fuel was somewhat atypical of 37-element type fuel, as the length to diameter ratio (l/d) was large (1.73) due to the pellets being ground down from an OD of 14.3 mm to 12.12 mm. The outer element burn-up averaged

Figure 7. Power history and ^{131}I loop activity during power cycling of the intentionally defected rod CYFON 2 in January 1977



approximately 640 MWh/kgU on discharge. Outer element powers varied between 57 kW/m near the beginning of life and 23 kW/m at discharge. Due to the long irradiation, Bundle JC experienced 153 short shutdowns, and 129 shutdowns of longer duration.

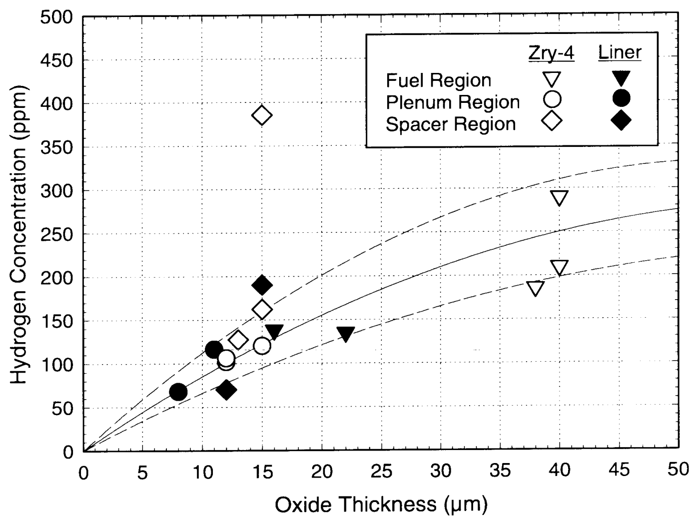
No element instrumentation was used during the irradiation. However, the bundles were subjected to extensive PIE that included dimensional changes, fission gas release, fuel burn-up analysis and metallography.

Under a cooperative agreement among Siemens Power Corporation, Empire State Electric Energy Research Corporation (ESEERCO) and Rochester Gas and Electric Corporation (RG&E), the RE Ginna reactor was loaded with four 14 × 14 demountable test assemblies (DTAs) in 1985 and irradiated for four cycles with one LTA irradiated for a further cycle.[5] Two of the DTAs contained 11 segmented rods each. The segmented rods consisted of four segments, the center two of which were pre-characterized and examined in detail after irradiation. Three combinations of pellet design (annular and solid) and cladding type (Zr-4 with sponge zirconium liner, and through wall Zr-4) were used in the fabrication of the segmented rods. The fuel-to-clad gap was another design variable with values of 160, 190 or 216 μm. The objective of the program was to develop a fuel design with increased margin to failure, increased high burn-up potential and to obtain performance data up to high burn-up for use in fuel modeling. The power history during the 4-5 cycle irradiation was quite onerous, as the following guide shows:

Cycle	BOC power kW/m	EOC power kW/m	Assembly burn-up MWd/kgU
1	21	18.5	8.9
2	32.5	28	23.9
3	25.5	24	37
4	10	10.5	42.5
5	21	22.5	52.1

The PIE included the usual measurements of dimensional changes, oxide thickness, fission gas release and metallography and data for 17 rodlets are included in the database. Of particular interest in the data set are measurements of hydrogen content, pickup fraction and mechanical properties. The mechanical properties of the irradiated material were measured on 5 mm wide ring samples tested at both room temperature and 350°C. As an example of the data, Figure 8 shows hydrogen concentration in various regions of the rodlet as a function of oxide thickness.

Figure 8. Hydrogen concentration in the cladding as a function of oxide thickness for an SPC fuel segment irradiated in the RE Ginna reactor



FORMAT OF FILES

The criterion adopted for the file format was one primarily of simplicity. It was considered that users should be able to read the files independent of commercial software. It was further recognized that the majority of codes were written in FORTRAN and therefore all files are in simple ASCII format for easy interrogation by file editors. Even text files are of this format, despite the enforced limitations imposed by this approach. If so desired, the adopted ASCII format does not preclude reformatting into a commercial database system sometime in the future. All databases have the following common elements:

- *Summary file*. This is a text file which describes the purpose of the experiment or test matrix and the scope of the data obtained.

- *Index file*. This is also a text file and lists all file titles and gives a brief summary of their contents.

- *Pre-characterization*. This includes information on the fuel pellets and cladding used, their manufacturing route, dimensions and chemical composition including impurities. For the fuel, this is augmented by details of enrichments, porosity distribution, re-sintering test data and microstructure. For the cladding, additional information includes mechanical properties, corrosion characteristics and texture as and when available. Details of the fuel rod geometry include: relevant dimensions, fuel column length, weight, fill gas composition and pressure. Details of reactor irradiation conditions are also provided.

- *Irradiation histories*. All histories are in a condensed form, with care to ensure that all important features are preserved. Where there was a significant axial power profile the history is provided in up to 18 axial zones in the case of the full length EDF PWR rods. For each time step, the data are provided as: time, time increment over which power was constant, clad waterside temperature and local heat rating for the prescribed number of axial zones. Information is provided to calculate the fast neutron flux and its spatial variation if the information is available.

- *In-pile data*. Separate files tabulate data from in-pile instrumentation as a function of variables such as time, burn-up or power, when applicable.

- *PIE data*. Where such examinations were made, data are recorded either in tabular or text form. Dimensional data include axial and diametral dimensions before and after irradiation, and post irradiation ridge heights where available. When available, data are tabulated for the axial variation of oxide thickness, hydrogen pickup and associated mechanical properties if measured. Data on fission gas release include rod averaged values obtained by puncturing and mass spectrometry, local values from whole pellet dissolution and across pellet spatial distributions as measured by gamma scanning, EPMA and XRF. Before and after porosity and grain size distributions are given, as well as the radial position for the onset of grain boundary porosity when such measurements have been made from metallographic examination.

All data files are held centrally on the OECD/NEA Data Bank computer system from which all files are dispatched. This single source for distribution is necessary for quality assurance purposes, particularly for the tracking and release of upgraded or corrected files. With the OECD's experience of data bank management, this arrangement ensures long term availability of the service.

Often PIE data are only available in graphical form and photomicrographs, the details of which are impossible to preserve in the current ASCII file format. For this reason, all reports available for compiling the data sets have been scanned and copied onto a single CD-ROM. The files are all in PDF (Acrobat Reader) format and can easily be read and printed using a wide range of software.

FIRST USE OF DATABASE

IAEA 4/012 Program

The database (at the time, 137 rods of diverse origins) was first proposed to countries for inclusion in the IAEA regional technical cooperation program for Europe RER 4/012. The aim of this program was to transfer to participants a nuclear fuel modeling computer code, and develop it for application to WWER reactor systems. Countries involved in this technical cooperation program include Armenia, Bulgaria, Czech Republic, Hungary, Poland, Romania, Slovakia and Ukraine.

The choice of the European developing countries, which, as far as nuclear energy is concerned, means mainly the east European countries, was dictated by the urgency of the situation. In the period 1990-1992, following the political changes in Eastern Europe, the situation in the area of nuclear safety was fairly precarious. Nuclear regulatory authorities were non-existent and nuclear fuel was loaded in the reactors without any licensing procedures. In addition, the concept of a quality assurance system was completely foreign.

With the IAEA's help, regulatory authorities were progressively set up in these countries. After completion of this task, the next step was to ensure that these national authorities function in a normal way. The IAEA has therefore embarked on a series of programs addressing different aspects of fuel behavior modeling, with the objective of providing as much available information as possible on WWER fuel through literature, meetings or by providing the tools necessary to perform fuel and fuel computer codes licensing. Amongst these tools the transfer of a mature computer code (TRANSURANUS developed by TUI Karlsruhe) and of the fuel database were two corner stones of the project. Training in the use of the code took place during an IAEA training course in Karlsruhe, Germany in June 1996, and a presentation and hand-over of the IFPE database occurred during a workshop at the Halden Reactor Project, Norway, in September 1996. After running for two years, the program was successfully concluded, and a successor program RER 4/19 aimed at QA and code application has been initiated.

FOR THE FUTURE

The creation of the database has met with universal approval, and consequently there has been no difficulty experienced in obtaining data for inclusion. Current work focuses on the Studsvik ramp projects, and the database will soon include: INTER RAMP (20 rods), DEMO RAMP I (5 rods), DEMO RAMP II (8 rods), OVER RAMP (39 rods) and SUPER RAMP (44 rods). Agreement has also been reached to include cases from the MR research reactor of the Kurchatov Institute, Russia, and further CANDU experimental data from Romania. Negotiations are also underway to obtain data from other sources, including China and Argentina.

ACKNOWLEDGEMENT

The authors wish to acknowledge the cooperation of the following participating organizations for the supply of their data and assistance in preparing the database: Risø National Laboratory, Denmark; Halden Project, Norway; Imatran Voima Oy, Finland; Kurchatov Institute, Russian Federation; Battelle North West, USA; Belgo-Nucleaire, Belgium; Studsvik Nuclear AB, Sweden; the CEA, EDF and Framatome, France, AECL Canada, HSE/IMC of the UK, SPC of the USA and Siemens Power Corporation. It should be noted that several organizations are preparing data for further inclusion. These are: CIAE, Beijing, China and INR, Pitesti, Romania.

REFERENCES

[1] "Scientific Issues in Fuel Behaviour", a report by an NEA Nuclear Science Committee Task Force, *OECD Documents*, January 1995, ISBN 92-64-14420.

[2] P. Chantoin, J.A. Turnbull, W. Wiesenack, "Summary of the Findings of the FUMEX Programme", Paper presented at the IAEA Technical Committee Meeting on Water Reactor Fuel Element Modelling at High Burn-up and its Experimental Support, Windermere, UK, Sept. 1994, IAEA-TECDOC-957, August 1997.

[3] P. Chantoin, E. Sartori and J.A. Turnbull, "The Public Domain Database on Nuclear Fuel Performance Experiments (IFPE) for the Purpose of Code Development and Validation", ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, 2-6 March 1997.

[4] J.A. Turnbull, "A Model for Calculating the Gap Inventory of ^{131}I in Intact Fuel Rods", paper 56, BNES Conference on Nuclear Fuel Performance held in Stratford upon Avon, UK 1985.

[5] L.F. Van Swam, A.A. Strasser, J.D. Cook and J.M. Burger, "Behaviour of Zircaloy-4 and Zirconium Liner Zircaloy-4 Cladding at High Burn-up", ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, 2-6 March 1997.