THE COMPILATION OF A PUBLIC DOMAIN DATABASE ON NUCLEAR FUEL PERFORMANCE FOR THE PURPOSE OF CODE DEVELOPMENT AND VALIDATION

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Abstract

This paper describes the compilation of a database on nuclear fuel performance for the purpose of code development and validation. The paper begins with the background behind its formation and the progress made to date in assembling 291 datasets from various sources encompassing PWR, BWR and WWER reactor systems. Agreement has been reached for the inclusion of further cases which will be processed into a common format for storage and retrieval in the OECD/NEA Data Bank.

Introduction

With increasing emphasis on economic as well as safe reactor operation, the approach to fuel performance assessment is now very much more detailed and closer to 'best estimate' evaluation than in the past. Not only are the safety requirements more extensive, but the economics of nuclear power production are constantly under scrutiny to ensure minimum unit cost. This is particularly the case in countries where nuclear stations must compete with conventional ones. For these reasons it is no longer possible to support operation with anything other than well qualified code calculations where all aspects of fuel performance are treated simultaneously and in a self-consistent manner. Also, there is a move in some countries to 'shadow' reactor operation with an on-line evaluation of fuel performance and a continuous assessment of core conditions against fixed limits for a large fraction of fuel assemblies in the reactor.

The principal function of a fuel performance code is to describe the behaviour of reactor fuel in the most accurate way possible under whatever conditions – both normal and off-normal – that are required by the licensing authority. By aiming to be 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 therefore it is 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 [1] recommended the compilation of a public domain database on fuel performance for the express purpose of fuel performance code development and validation.

The Task Force ran concurrent with the IAEA FUMEX programme where '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 behaviour, 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 programme 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, and this paper provides a brief description of its structure and the progress made. The list of experiments incorporated in the database to date is given in Table 1. The database is operated by the OECD/NEA in co-operation with the IAEA. Although compiled for general use without restrictions, its success relies on participation. Therefore, incumbent on all users is the requirement to contribute whatever well-qualified data they too can make available. Only in this way can the objective of a standard comprehensive database be realised on which all codes can be developed and against which they can be judged.

Choice of data

From the outset it was recognised 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 characterisation, and test reactor experiments with in-pile instrumentation and PIE exploring normal and off-normal behaviour. It is recognised 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 programmes 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 where the originating organisation 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 behaviour but not accident condition entailing melting of fuel and clad, resulting in loss of geometry. Recently the behaviour of defect fuel has been considered and it is planned to include such cases in the near future.

Of particular interest to fuel modellers are data on: fuel temperatures, fission gas release (FGR), clad deformation (e.g. creep-down, ridging) and mechanical interactions. In addition to direct measurement of these properties, every effort is made to include PIE information on the distribution of: grain size, porosity, Electron Probe Micro Analysis (EPMA) and X-ray Fluorescence (XRF) measurements on caesium, 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. Also in the context of thermal performance, the effect of burn-up on UO_2 thermal conductivity has been addressed. This is illustrated by cases where 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 cases 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.

Brief description of data

At present, the database holds some 291 cases comprising the following:

Halden irradiated IFA-432	5 rods
The Third Risø Fission Gas Release Project	16 rods
The Risø Transient Fission Gas Release Project	15 rods
The SOFIT WWER fuel Irradiation Programme	12 rods
The High Burn-up Effects Programme	81 rods
Rods from Kola-3 FA-198 and 16 rods from FA-222	32 rods
Rods from the TRIBULATION programme	19 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
Studsvik INTER-RAMP BWR Project	20 rods
Studsvik OVER-RAMP PWR Project	39 rods
Studsvik SUPER-RAMP PWR Project	28 rods

These are described briefly below highlighting some of the more important aspects of the data.

The first experiment to be addressed was the Halden irradiated IFA-432 commissioned by the USNRC for irradiation between December 1975 and June 1984. The main objectives of IFA-432 were measurement of fuel temperature response, fission gas release and mechanical interaction on BWR-type fuel rods up to high burn-ups. The assembly featured several variations in rod design parameters, including fuel type, fuel/cladding gap size, fill gas composition (He and Xe) and fuel stability. It comprised six BWR-type fuel rods with fuel centre thermocouples at two horizontal planes. Rods were also equipped with pressure transducers and cladding extensometers.

Data from five rods have been included in the database providing in-pile and limited PIE data up to 46 MWd/kg UO_2. An illustration of the type of data available on thermal performance is given in Figure 1. Here, the measured fuel centreline temperature normalised to 20 kW/m is plotted as a function of burn-up for rod 1 (230 micron gap) and the small gap rod 3 (80 micron). At the start of the irradiation, the difference in temperature is due to the 150 micron difference in gap size. This difference persists to such an extent that fission gas release commences in rod 1 at around 5 MWd/kg UO_2 resulting in a 'poisoning' of the gap conductance and increased fuel temperatures, reaching a maximum at around 11 MWd/kg UO_2. Meanwhile, the low temperatures sustained in rod 3 prevented significant gas release and the slow monotonic increase in temperatures is predominantly due to degradation of the UO_2 thermal conductivity with irradiation damage and accumulation of fission products. Because of the difference in gap size, greater axial interaction was observed in the small gap rod 3 compared to the other larger gap rods.

The Risø Transient Fission Gas Release Project (Risø II) was executed in the period 1982-86. Short lengths of irradiated fuel were fitted with in-pile pressure transducers and ramped in the Risø DR3 reactor. The fuel used came from either IFA-161 irradiated in the Halden reactor or from segments irradiated in the Millstone BWR. Using this refabrication technique, it was possible to back fill the test segment with a choice of gas and gas pressure and to measure the time dependence of fission gas release by continuous monitoring of the plenum pressure. The short length of the test segment was an advantage because, depending on where along the original rod the section was taken, burn-up could be a chosen variable, and during the test the fuel experienced a single power. Some segments were tested without refabrication. Here the fuel stack was longer than in the case of the refabricated tests and hence the segments experienced a range of powers during the ramp depending on axial position in the test reactor. These 'un-opened' segments were used to confirm that refabrication did not affect the outcome of the tests. Extensive hot cell examination compared the fuel dimensions and microstructure before and after the tests.

Some 17 tests were performed and all but one (which failed) have been included in the database and provide valuable information on fission gas release during power transients at high burn-up and also clad diametral deformation and fuel swelling as a function of ramp power and hold time. Figure 2 shows the evolution of fission gas release as a function of time during the power ramp for one of the tests using fuel from IFA-161. At each step, the fractional gas release shows a square root dependence on time which is characteristic of release by a diffusion type process. The sudden increase in release at the end of the test on decreasing power shows that towards the end of the hold time, there was contact between the fuel and the cladding, thus causing a restriction to the axial communication to the plenum where the pressure transducer was situated. This was confirmed by a comparison of diameter traces before and after the test which showed no significant diameter increase but a significant increase in permanent ridge height. The database also includes diametral profiles of retained fission products measured by EPMA and XRF. The radial position for the onset of release is clearly evident in Figure 3. The difference between the two types of measurements on retained xenon can be taken as the gas residing on grain boundaries; i.e. the difference between the total gas content (XRF) and the gas residing only in the matrix (EPMA).

The third and final Risø Project which took place between 1986 and 1990, bump tested fuel re-instrumented with both pressure transducers and fuel centreline thermocouples. The innovative technique employed for re-fabrication involved freezing the fuel rod to hold the fuel fragments in position before cutting and drilling away the centre part of the solid pellets to accommodate the new thermocouple. The fuel used in the project was from: IFA-161 irradiated in the Halden BWR between 13 and 46 MWd/kg UO_2, GE BWR fuel irradiated in Quad Cities 1 and Millstone 1 between 20 and 40 MWd/kg UO_2 and ANF PWR fuel irradiated in Biblis A to 38 MWd/kg UO_2. The data from the project are particularly valuable because of the in-pile fuel temperature and pressures measurements as well as extensive PIE. The database includes seven cases with ANF PWR fuel, six cases with GE BWR fuel and two cases using fuel from IFA-161. Within the test matrix it was demonstrated that the refabrication did not interfere with the outcome of the tests and that there was a correspondence with the results of the previous project.

Figure 4 shows a comparison of two tests using ANF fuel of near identical pre-test characteristics and power history during the power ramps. Both were refabricated with thermocouples and pressure transducers but one section was filled with helium whilst the other was filled with xenon. Diameter measurements before and after the ramp showed that the fuel-to-clad gap was closed. There is only a small difference in temperatures between the two cases, much smaller than predicted by many models. This shows that contrary to laboratory tests which demonstrate an effect of surface roughness on heat transfer between two surfaces in contact, at high burn-up, the surface roughness of the fuel and cladding for closed gap situations has little effect on fuel temperatures.

The HBEP was an international, group-sponsored programme managed by Battelle North West Laboratories whose principal objective was to obtain well-characterised data on fission gas release for typical LWR fuel irradiated to high burn-up levels. The programme was organised into three tasks, the first of which was a review of existing data. Tasks 2 and 3 comprised the experimental work carried out by the HBEP. Under Task 2, 45 existing fuel rods, either at moderate burn-up levels or undergoing irradiation to higher burn-up levels, were identified, acquired and subjected to PIE. Some rods were also subjected to power-bumping irradiations. Under Task 3, a series of fuel rods were built for irradiation in BR3 to high burn-up levels. Four design variations and three variations in operational history were used to study the effect of design and operation parameters on high burn-up FGR. This programme provides a substantial amount of data on fission gas release and the variations observed with different manufacturing routes, different reactor systems and different characteristics of the fuel. In the latter case it is possible to quantify the difference release between: high and low internal pressure, hollow versus solid pellets, the effect of different grain sizes and the effect of adding gadolinia to the UO_2 pellets. A total of 45 rods were considered under Task 2 and 37 rods for Task 3 and all have been included in the database.

The SOFIT programme was a series of experiments on WWER fuel carried out in the MR pool type research reactor at the Russian Research Centre Kurchatov Institute (IRTM) in co-operation with the Finnish Utility, Imatran Voima Oy (IVO). The programme was divided into three distinct phases, each addressing specific objectives:

SOFIT 1	**SOFIT 2**	**SOFIT 3**
Parametric fuel rod irradiations with basic steady state power histories up to moderate levels of burn-up as dictated by instrumentation endurance.	Parametric studies based on irradiation of re-instrumented high burn-up rods.	Irradiation testing under transient conditions.

At the moment, the database contains detailed information on six rods instrumented with centreline thermocouples and fission gas release data for two uninstrumented rods from SOFIT 1.1, centreline temperature data from two rods and fuel and clad elongation data from another two rods of SOFIT 1.3. It is hoped that further data from this programme will be released in the near future.

WWER rods are clad in ZR-1% Nb and contain hollow pellets. The temperature data are very much as expected, with the effect of differing gap size clearly seen in the first ramp to power. From the rods fitted with extensometers, the early life behaviour shows negligible change in the clad length apart from thermal expansion, whilst there is evidence of significant fuel column shrinkage due to densification, Figure 5.

The Database also includes data for two pre-characterised standard WWER-440 fuel assemblies: FA-198 and FA-222 manufactured by the Russian fuel vendor Electosal and irradiated in the Kola-3 reactor. These assemblies were the centre of a programme called Blind Calculations for the WWER-440 High Burn-up Fuel Cycles Validation, initiated in Spring 1994 with the objective of testing the predictive capabilities of several Russian codes.

The maximum linear heat generation rate (LHGR) of FA-198 was <31 kW/m at the beginning of life and decreased to about 14 kW/m by the beginning of the fourth cycle and 11 kW/m at the end of life. In FA-222, peak LHGR values of 21 to 26 kW/m were experienced at the beginning of the second cycle, followed by steady state operation at LHGR of 10 to 22 kW/m before gradually decreasing to around 8 kW/m at the end of life. During the whole of the irradiation, both assemblies were located remote from any control rods. Consequently, the irradiation conditions are considered representative of base load operation for WWER-440 reactors.

The database contains details of 16 rods from each assembly; these are the two corner rods 7 and 120 and rods along the diagonal, Figure 6. As well as comprehensive pre-characterisation, the data include detailed 10 zone irradiation histories and PIE observations of dimensional changes and fission gas release. The measured gas release varied from ~0.5% for the diagonal rods to ~1.2% for the FA-198 corner rods, and 1-1.6% for the diagonal rods to 2.3-3.7% for the corner rods of the higher burn-up FA-222.

The objectives of the TRIBULATION programme were twofold. It was primarily a demonstration programme aimed at assessing the fuel rod behaviour at high burn-up, when an earlier transient had occurred in the power plant. The second objective was to investigate the behaviour of different fuel rod designs and manufacturers when subjected to a steady state irradiation history to high burn-up.

The first objective was met by irradiating fuel rods under steady state conditions in the BR3 reactor and under transient conditions in BR2. The effect of the transient was determined by comparing data from four identical rods tested as follows:

i) BR3 irradiation followed by PIE;

ii) BR3 irradiation followed by BR2 transient then PIE;

iii) BR3 irradiation followed by BR2 transient and re-irradiated in BR3 before PIE;

iv) BR3 irradiation and continued BR3 irradiation to maximum burn-up before PIE.

The Database contains data from 19 cases using rods fabricated by Belgo-Nucleaire (BN) and Brown Boveri Reactor GmbH (BBR). The matrix provides good data on clad creepdown and ovality as a function of exposure as well as the effect of different irradiation histories on fission gas release.

IFA-429 consisted of PWR type fuel rods assembled in three axially separated clusters of six rods and irradiated in the Halden Boiling Water Reactor. The eighteen-rod assembly had been designed to investigate gas absorption, fission gas release and thermal behaviour of UO_2 fuel during both steady state and a period of repeated rapid power transients. In order to achieve the objectives, the assembly was instrumented with nine vanadium neutron detectors, one cobalt detector. Two rods were instrumented with fuel centre line thermocouples and nine rods were instrumented with null-balance gas pressure transducers monitoring rod internal gas pressure.

This database contains power histories for seven rods as well as the measured temperature history for one of the middle cluster rods up to the burn-up level of 53 MWd/kg UO_2 and the measured internal pressure data for three upper and three lower cluster steady-state irradiated fuel rods subjected to two series of rapid power transients. Manually initiated gas pressure measurements were performed during the testing sequence allowing comparison of FGR data at three different fuel densities (91% TD, 93% TD and 95% TD), at two different grain sizes (6 and 17 microns) and at two different fuel-cladding gap sizes (200 and 360 microns) are possible. The power histories for the two series of power ramps are given schematically in Figure 7 whilst the measured rod internal pressure is shown as a function of cumulative time at high power for the first ramp series in Figure 8. The parabolic form of these curves is indicative of release by a diffusion controlled process.

IFA-562.1 contained twelve fuel rods assembled in two axially separated clusters irradiated to investigate the effect of pellet surface roughness on both fuel thermal and rod mechanical performances. As a second experimental objective the fuel grain growth caused by a power ramp at the end of the irradiation period was intended for investigation by PIE. To achieve the objectives fuel centre temperatures and clad elongation were measured in the fuel rods containing either rough or smooth pellets and filled with xenon or helium gas. To provide ideal conditions for the comparison, the rods were fabricated with small fuel-clad gap size and the fuel temperatures were kept below 1200°C during the base irradiation. The small initial gap size allowed gap closure, and the low fuel centre temperature prevented both fission gas release and grain growth. Nevertheless, no systematic differences were observed in either thermal or mechanical behaviour of the smooth and rough pellet rods. Consequently the effect of pellet surface roughness is insignificant in comparison with other influences such as the pellet-cladding gap size change, fuel densification and relocation.

After twelve years irradiation in the Halden Boiling Water Reactor two fuel rods (Rod 807 and Rod 808) were re-instrumented with fuel centre thermocouples and reloaded as IFA-533.2 into the reactor in order to investigate fuel thermal behaviour at high burn-up. The fuel rods were pre-irradiated with four other rods in the upper cluster of IFA-409. After base irradiation the four neighbouring rods were re-instrumented with pressure transducers and ramp tested in IFA-535.5 (slow) and IFA-535.6 (fast) providing useful data about FGR at two different ramp rates, Figure 9. As the irradiation history of IFA-533.2 in the first months was very similar to the history of the ramp tests, the fuel temperature and FGR data measured in the different IFAs complement each other, although the fuel-cladding gap sizes were slightly different and due to re-instrumentation the internal gas conditions were also dissimilar.

The database currently contains data from three Studsvik Ramping Projects, these are: INTER-RAMP, OVER-RAMP and SUPER-RAMP. Data are also available for DEMO-RAMP I & II and these will be included shortly. In all cases, PIE data are available after base irradiation and after the final ramp test, as well as whether or not clad failure occurred. The tests are therefore valuable for developing or validating code predictions of dimensional changes and FGR during power ramps as well as predictions of failure by PCI and SCC.

The objective of the INTER-RAMP project was to establish the PCI failure threshold of 20 standard un-pressurised BWR fuel rods on power ramping at burn-ups of 10 and 20 MWd/kgU. The fuel rods were supplied by ASEA-ATOM of Sweden. Nine rods were of identical reference design: 150 micron fuel-to-clad gap, 95% TD fuel pellets and re-crystallised cladding. The other 11 rods differed from these by way of gap size (80 and 250 microns), fuel density (one rod at 93% TD) and cladding heat treatment (8 rods stress relieved). The rods were base irradiated in boiling capsules in the Studsvik R2 reactor at powers ranging between 40 and 22 kW/m. The final base irradiation power was between 22 and 30 kW/m in all but two cases, where the final powers were ~38-39 kW/m. Power ramping took place in a pressurised loop simulating BWR conditions. Each test comprised a conditioning period of 24 hours at a power nominally identical to the final base irradiation power. This was followed by a fast ramp at a rate 4 kW/m/min to a terminal power in the range 41-65 kW/m which was held for 24 hours or until failure was detected. In all, 11 of the 20 rods failed.

In the OVER-RAMP project, 39 rods from two sources were ramped under PWR conditions. Twenty-four rods were supplied by KWU/CE which had been irradiated at powers in the range 14-25 kW/m to burn-ups between 12 and 31 MWd/kgU in the Obrigheim reactor, Germany. The rods were in six groups of 4 rods covering different ranges of burn-up, diametral gap (137 to 171 microns) and different fuel grain sizes (4.5, 6 and 22 microns). A further 13 rods in three groups were supplied by Westinghouse after pre-irradiation in BR-3. The groups covered the burn-up ranges 18-24, 16-23 and 20-21 MWd/kgU with two different levels of fill pressure 13.8 and 24.8 bar. The power ramp was preceded by a conditioning phase lasting 72 hours where all rods were irradiated at 30 kW/m apart from

3 Westinghouse rods where 23 kW/m was chosen. The ramp rate was typically 10 kW/m/min but a small number of rods experienced slower rates. The ramp ended at a terminal power of typically ~42 kW/m but ranged between 37 and 53 kW/m. The maximum power was held for 24 hours or until failure was detected. Seven KWU/CE and 6 Westinghouse rods failed.

To date, only the PWR sub-programme of the SUPER-RAMP project has been included in the database; the BWR cases will follow soon. In the PWR sub-programme, 28 rods were divided into six groups of different designs:

- 2 groups of standard KWU/CE design PWR rods;
- KWU/CE standard design containing gadolinia doped fuel pellets;
- KWU/CE supplied large grain fuel;
- KWU/CE supplied annular pellet fuel;
- Westinghouse supplied standard design;
- Westinghouse supplied annular pellets.

The KWU/CE rods were pre-irradiated in the Obrigheim reactor and the Westinghouse rods were irradiated in BR-3. The time averaged power for all rods was 14-26 kW/m with final burn-ups in the range 33-45 MWd/kgU. The conditioning power was chosen to be 25 kW/m which was held for 24 hours. The ramp rate was 10 kW/m/min up to a ramp terminal powers of 39-50 kW/m which was held for 12 hours or until failure was detected. Two KWU/CE rods were tested at coolant temperatures lower than normal by 25 and 50°C and two other KWU/CE rods had conditioning times in excess of 24 hours. The results of the ramp tests were 2 out of 5 KWU/CE large grain fuel rods failed, 3 out of 5 Westinghouse standard rods and all KWU/CE annular rods failed.

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 recognised 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. The adopted ASCII format does not preclude reformatting into a commercial database system sometime in the future if so desired. 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-characterisation***. 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.

- ***Irradiation histories***. All histories are in 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 12 axial zones. 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***. When applicable, separate files tabulate data from in-pile instrumentation as a function of variables such as: time, burn-up or power. For example, IFA-432 fuel rods were equipped with centreline thermocouples and cladding elongation detectors. Files were created tabulating temperatures at constant powers of 20 and 30 kW/m as a function of local burn-up throughout life. At several times during the irradiation, at approximately 5 MWd/kg UO_2 intervals, temperature is tabulated against local power during slow ramps. In this case, the data are reproduced directly from the original files where signals were logged every 15 minutes. A similar procedure was adopted for clad elongation measurements; during short periods of variable power, 200-5000 hours duration, elongation is tabulated against rod average power showing cases where there was pellet-clad mechanical interaction.

- ***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. 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 is 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 which are difficult to preserve in the current ASCII file format. For this reason, the possibility of scanning figures and photographs for storage and retrieval using the medium of the CD ROM was investigated. As a result, all reports available for compiling the datasets to date have been scanned and copied onto a single CD-ROM. The files are all in Tif-Group IV format and can easily be read and printed using a wide range of software.

First use of database

IAEA 4/012 programme

The Database (at that time, 137 rods of diverse origins) was first proposed to countries for inclusion in the IAEA regional technical co-operation programme for Europe RER 4/012. The aim of this programme was to transfer to participants a nuclear fuel modelling computer code, and developing it for application to WWER reactor systems. Countries involved in this technical co-operation programme are Armenia, Bulgaria, Czech Rep., 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 quality assurance system was unknown.

With the Agency's help Regulatory Authorities were progressively set up in these countries. After completion of this task, the next step is to ensure that these national authorities function in a normal way and the Agency has therefore embarked on a series of programmes addressing different aspects of fuel behaviour modelling, with the objective of giving as much available information as possible on WWER fuel through literature or 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.

The transfer of the fuel database was made during an Agency training course, held at the Halden reactor project in September 1996, in which the data proposed to the modellers were analysed according to the type of utilisation to be made. On one hand, experiments were reviewed one by one. On the other hand they were reviewed according to their utilisation in models addressing the following parameters:

- thermal performance;
- fission gas release;
- pellet clad interaction.

In co-operation with TUI Karlsruhe and the help of the database, the exercise of developing the TRANSURANUS code for WWER applications is now underway in several countries.

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 completion of the Studsvik ramping tests of which there are 16 rods in the SUPER-RAMP BWR sub-programme, 5 rods in the DEMO-RAMP I and 8 rods from DEMO-RAMP II BWR programmes. Agreement has been reached with Framatome, EDF and the CEA for the release of 3 Fessenheim rods of 3, 4 and 5 cycles irradiation, 2 experimental gas flow rods: CONTACT 1 & 2 and 10 experimental defect fuel rods irradiated in the Siloe research reactor. This will bring the number of datasets within the database to 335. In addition, discussions are in hand for the inclusion of WWER fuel ramped in the MR reactor at the Kurchatov Institute, Moscow, and CANDU data from both Romania and AECL Canada. Interest in the database has been shown from China, Argentina and India with promises of yet further data to add to the list of cases.

Finally, it may be concluded that the initiative shown for the creation of the publicly available database has been given world-wide acceptance from countries whose nuclear industries and fuel performance assessment capabilities are in various of stages of maturity. It is hoped therefore that wide dissemination of the database will contribute to a wholesale improvement in code capabilities and that these will work their way through to a safer and more economic use of nuclear fuel in all reactor systems.

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Table 1. International Fuel Performance Experiments Database (IFPE)

Operated by the OECD/Nuclear Energy Agency in co-operation with the International Atomic Energy Agency (IAEA)

List of compiled data as of May 1997

Origin of data	Data-set-name	Package-Id
IFPE/IFA, Halden Reactor Project Fuel Performance Experiments Data	IFPE/IFA-432 IFPE/IFA-429 IFPE/IFA-562.1 IFPE/IFA-533.2 IFPE/IFA-535.5 & 6	NEA 1488/02 NEA 1488/03 NEA 1488/04 NEA 1488/05 NEA 1488/06
IFPE/RISØ II, Fuel Performance Data from Transient Fission Gas Release	IFPE/RISØII-A IFPE/RISØII-B IFPE/RISØII-E IFPE/RISØII-GE-A IFPE/RISØII-GE-B IFPE/RISØII-GE-G IFPE/RISØII-GE-H IFPE/RISØII-GE-I IFPE/RISØII-GE-K IFPE/RISØII-GE-L IFPE/RISØII-GE-M IFPE/RISØII-GE-N IFPE/RISØII-GE-O IFPE/RISØII-H IFPE/RISØII-I IFPE/RISØII-IFA161B IFPE/RISØII-K IFPE/RISØII-L	NEA 1502/01 NEA 1502/02 NEA 1502/03 NEA 1502/08 NEA 1502/09 NEA 1502/10 NEA 1502/11 NEA 1502/12 NEA 1502/13 NEA 1502/14 NEA 1502/15 NEA 1502/16 NEA 1502/17 NEA 1502/04 NEA 1502/05 NEA 1502/18 NEA 1502/06 NEA 1502/07
IFPE/RISØ-III, Fuel Performance Data from 3rd Risø Fission Gas Release	IFPE/RISØIII-AN1 IFPE/RISØIII-AN10 IFPE/RISØIII-AN11 IFPE/RISØIII-AN2 IFPE/RISØIII-AN3 IFPE/RISØIII-AN4 IFPE/RISØIII-AN8 IFPE/RISØIII-GE2 IFPE/RISØIII-GE4 IFPE/RISØIII-GE6 IFPE/RISØIII-GE7 IFPE/RISØIII-IFA161 IFPE/RISØIII-II1 IFPE/RISØIII-II2 IFPE/RISØIII-II3 IFPE/RISØIII-II5	NEA 1493/01 NEA 1493/06 NEA 1493/07 NEA 1493/02 NEA 1493/03 NEA 1493/04 NEA 1493/05 NEA 1493/08 NEA 1493/09 NEA 1493/10 NEA 1493/11 NEA 1493/16 NEA 1493/12 NEA 1493/13 NEA 1493/14 NEA 1493/15
IFPE/HBEP Batelle's High Burn-up Effects Programme for Fuel Performance	IFPE/HBEP-2 IFPE/HBEP-3	NEA 1510/01 NEA 1510/02
IFPE/SOFIT, WWER-400 & 1000 Fuel Performance Experiments Database	IFPE/SOFIT-1.1 IFPE/SOFIT-1.3	NEA 1310/01 NEA 1310/02
IFPE/KOLA-3, WWER-440 Fuel Performance Data from Kola-3 NPP	IFPE/KOLA-3	NEA 1532/01
IFPE/TRIBULATION Data for 11 BN and 8 ABB Fuel Rods Tested in the TRIBULATION Programme	IFPE/TRIBULATION	NEA 1536/01
IFPE/INTER-RAMP (Studsvik) Data from 20 ASEA-ATOM BWR Fuel Rods	IFPE/INTER-RAMP	NEA 1555/01
IFPE/OVER-RAMP (Studsvik) Data from 24 KWU/CE PWR Rods	IFPE/OVER-RAMP	NEA 1556/01
IFPE/SUPER-RAMP (Studsvik) Data from 24 Rods of Different KWU/CE and Westinghouse PWR Design	IFPE/SUPER-RAMP	NEA 1557/01

Figure 1. Fuel centreline temperatures at constant power as a function of burn-up for rods 1 (larg gap) and 3 (small gap) of the Halden irradiated IFA-432 experiment

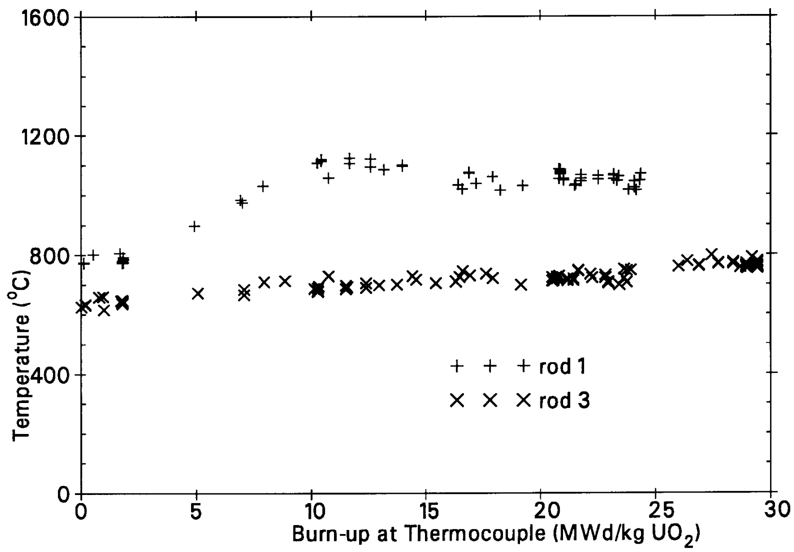

Figure 2. Fission gas release as determined from rod internal pressure for test Risø-b (Risø Transient Fission Gas Release Project) during the power ramp in the DR3 reactor

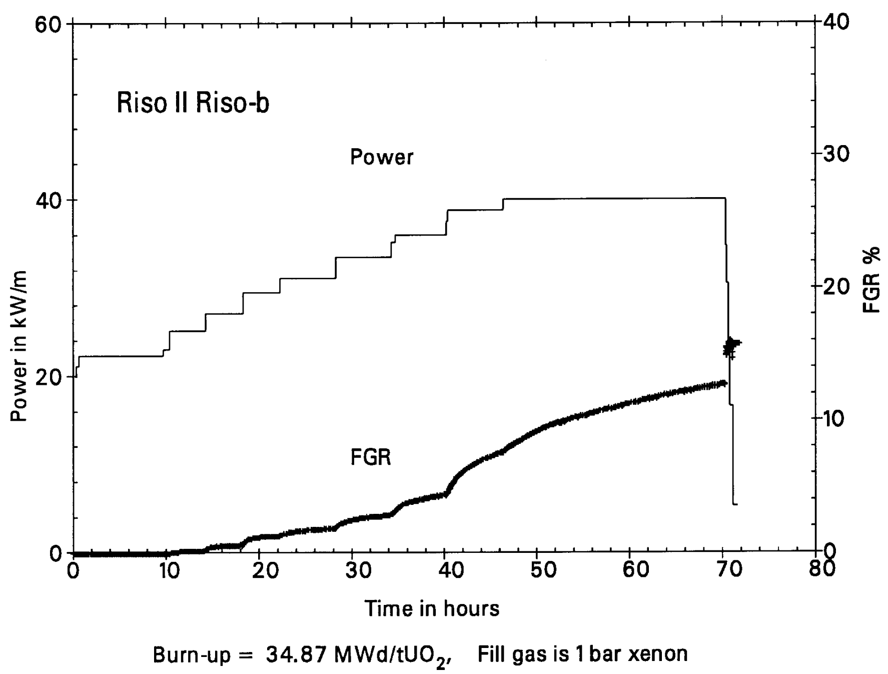

Figure 3. Comparison of EPMA and XRF retained xenon profiles for test GEa of the Risø Transient Fission Gas Release Project

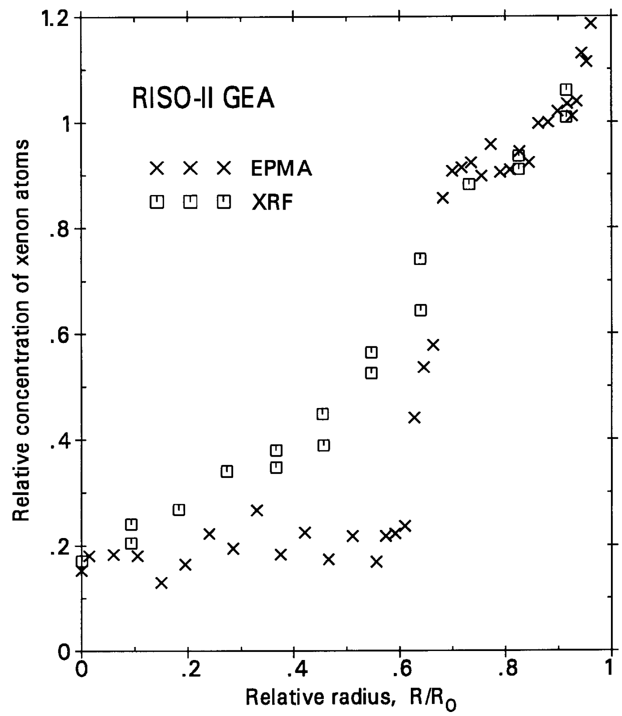

Figure 4. comparison of centreline temperatures of two nominally identical rods AN3 and AN4 filled with 15 bar He and 1 bar Ne respectively during the power ramp in DR3 for the Third Risø Fission Gas Project

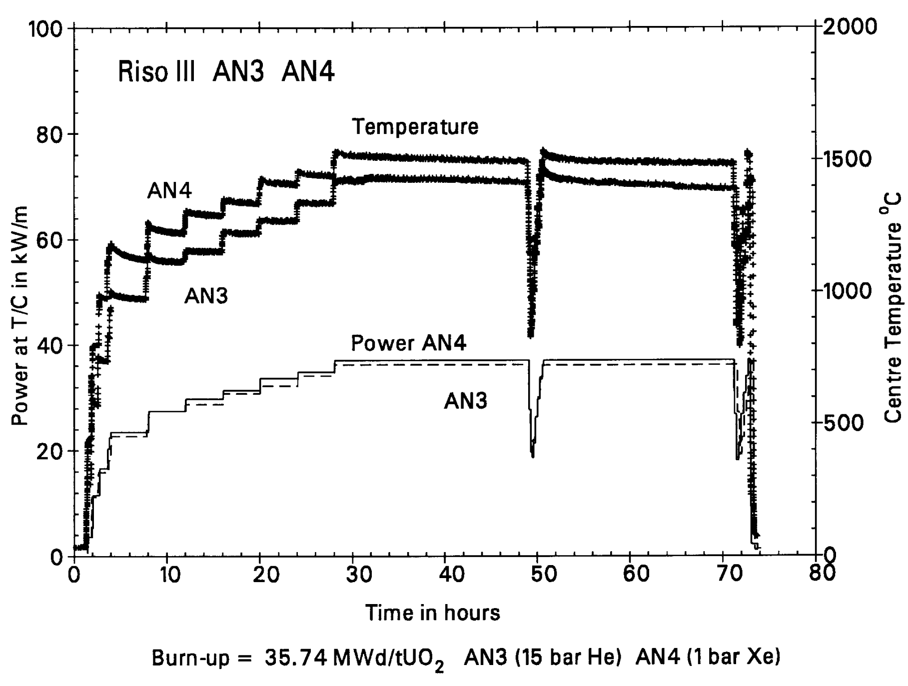

Figure 5. Axial extension of fuel cladding as a function of time for rod 3 of SOFIT 1.3

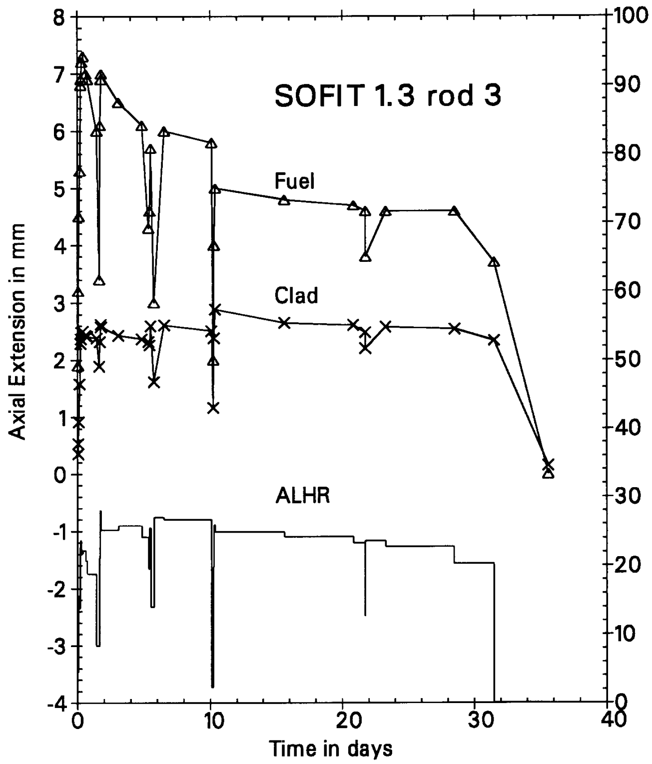

Figure 6. Arrangement and numbering of rods in the WWER-440 assemblies FA-198 and FA-222 irradiated in the Kola-3 reactor

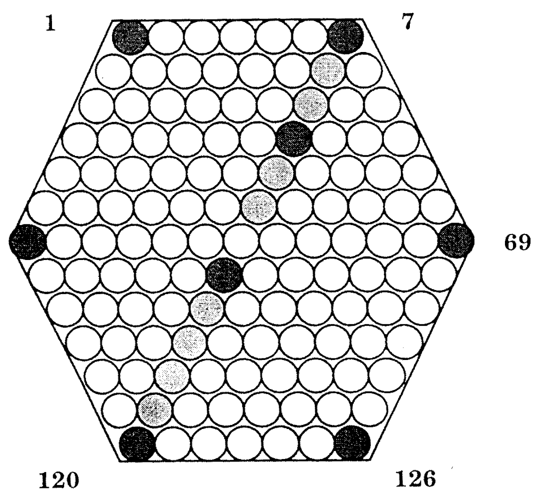

Figure 7. Schematic histories for the two series of power transients experienced by rods in IFA-429

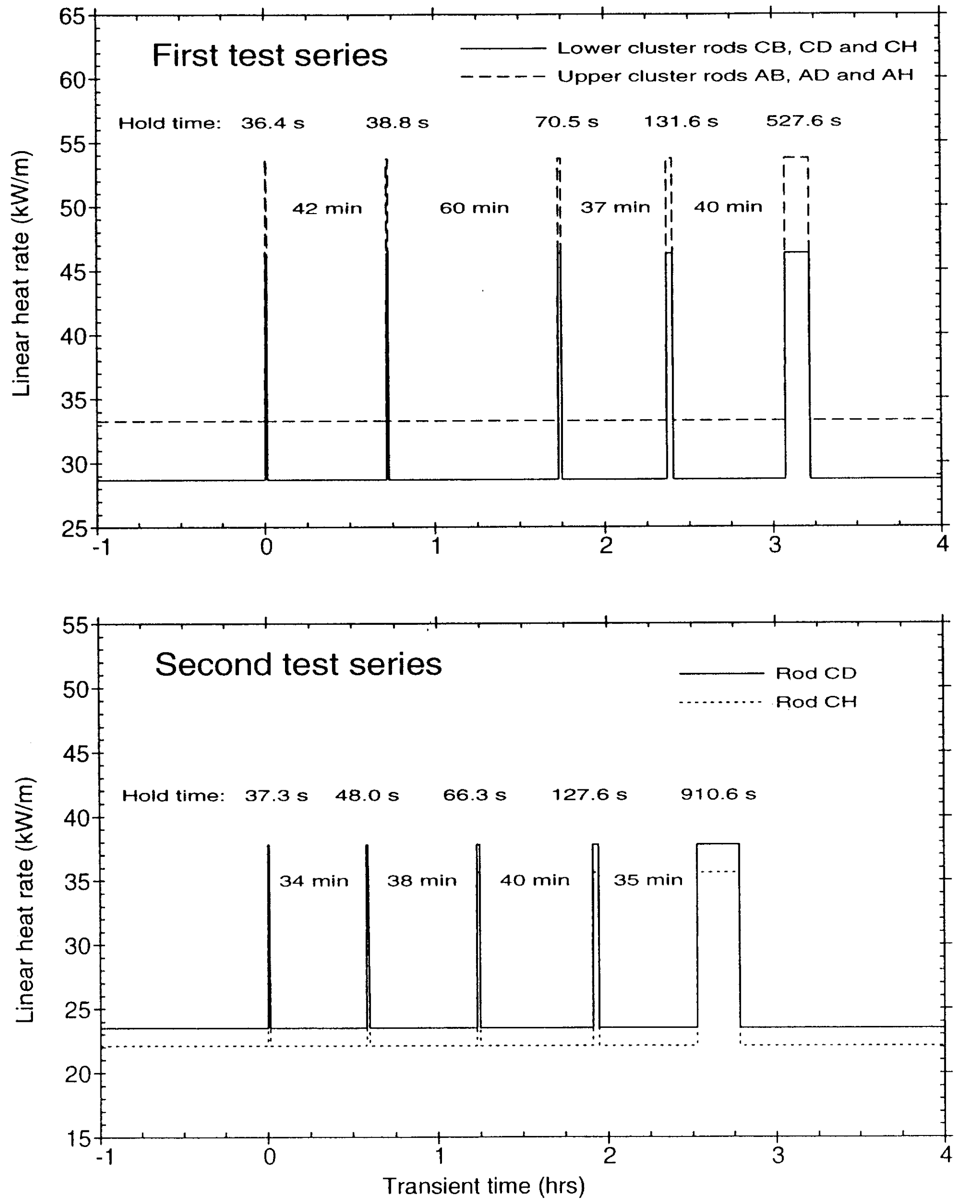

Figure 8. Measured rod internal pressure as a function of cumulative time at peak power for rods during the first series of power transients in IFA-429

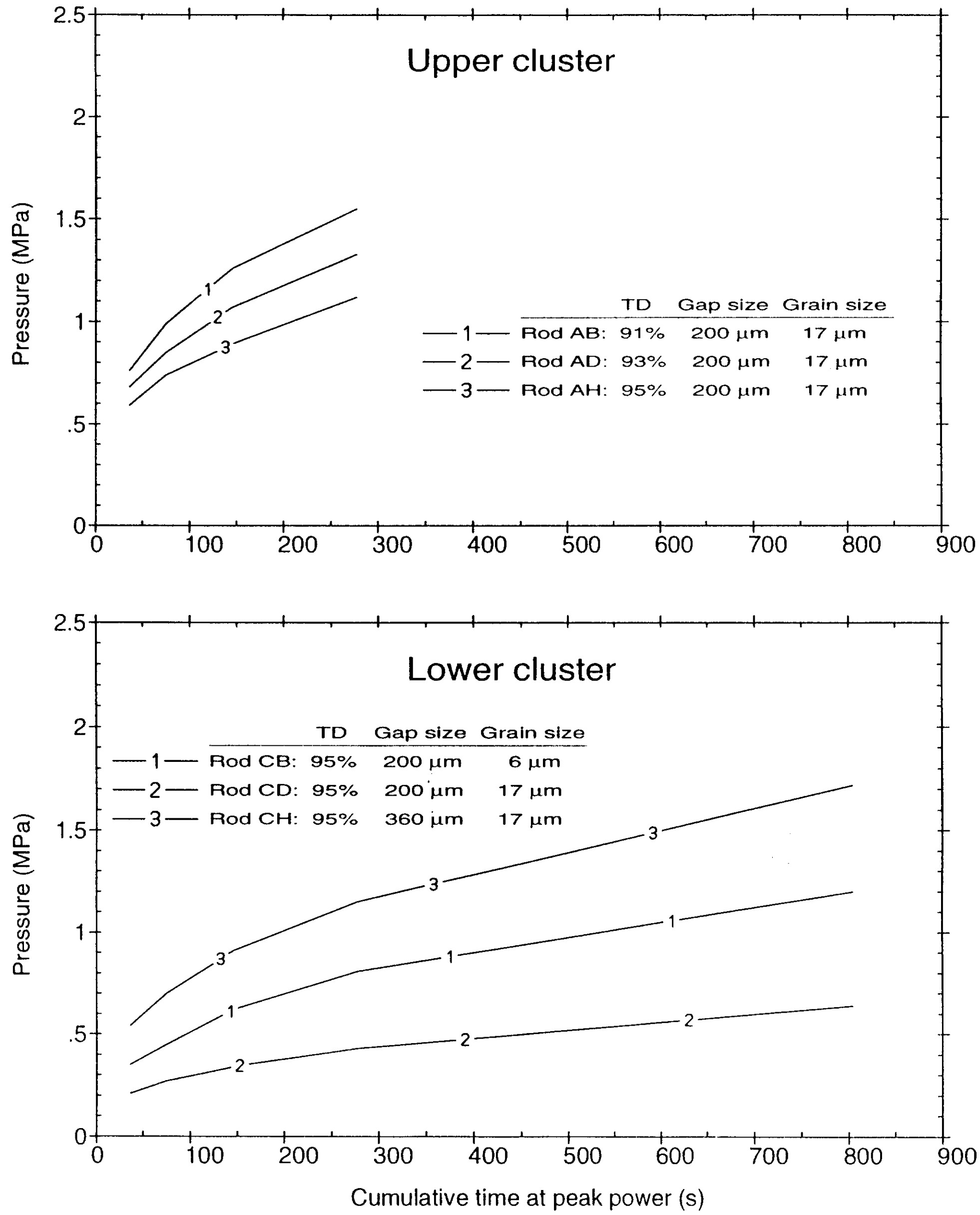

Figure 9. Fission gas released for the fast ramped IFA-535.5 and the slow ramped IFA-535.6 rod as a function of time