THE RELEVANCE OF THE IFPE DATABASE TO THE MODELLING OF VVER-TYPE FUEL BEHAVIOUR

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1. Introduction

The aim of the International Fuel Performance Experimental Database (IFPE Database) is to provide, in the public domain, a comprehensive and well-qualified database on zircaloy-clad UO_2 fuel for model development and code validation. The data encompass both normal and off-normal operation and include prototypic commercial irradiations as well as experiments performed in Material Testing Reactors. To date, the Database contains over 800 individual cases, providing data on fuel centreline temperatures, dimensional changes and FGR either from in-pile pressure measurements or PIE techniques, including puncturing, Electron Probe Micro Analysis (EPMA) and X-ray Fluorescence (XRF) measurements. This work in assembling and disseminating the Database is carried out in close co-operation and co-ordination between OECD/NEA and the IAEA.

The majority of data sets are dedicated to fuel behaviour under LWR irradiation, and every effort has been made to obtain data representative of BWR, PWR and VVER conditions. In each case, the data set contains information on the pre-characterisation of the fuel, cladding and fuel rod geometry, the irradiation history presented in as much detail as the source documents allow, and finally any in-pile or PIE measurements that were made.

The purpose of this paper is to highlight data that is relevant specifically to VVER application. To this end, the NEA and IAEA have been successful in obtaining appropriate data for both VVER-440 and VVER-1000-type reactors. These are:

- Twelve (12) rods from the Finnish-Russian co-operative SOFIT programme;
- Kola-3 VVER-440 irradiation;
- MIR ramp tests on Kola-3 rods;
- Zaporozskaya VVER-1000 irradiation;
- Novovoronezh VVER-1000 irradiation.

Before reviewing these data sets and their usefulness to modellers, the paper touches briefly on recent, more novel additions to the Database and on progress made in the use of the Database for the current IAEA FUMEX II Project. Finally, the paper describes the Computer Program Service of the OECD/NEA Data Bank relative to fuel behaviour modelling.

2. Recent additions

In addition to data on performance of fuel in standard rod geometry irradiated in normal and off-normal conditions, the database now contains data for out-of-pile annealing tests, defect fuel behaviour, single-effect dedicated PIE studies and fuel experiencing accident scenarios (outlined below). Apart from the NRU LOCA tests, which are dedicated to the validation of transient fuel performance codes, the other data sets are “single effects” and are most useful in developing specific models.

AEAT-IMC NFB 8 and 34, released March 1999

These experiments were performed by AEA Technology at Harwell and released by the Health and Safety Executive. Their aim was to investigate the relative importance of the effective gas atom diffusion coefficient, the capacity of grain boundary bubbles and the fuel grain size under conditions more nearly representative of a fault transient rather than of normal operation. Small samples of UO_2 fuel were obtained from CAGR fuel pins and annealed in a pre-defined high temperature (1 500° to 1 900°C) for periods of 2-40 hours. The release rate of 85 Kr was monitored continuously throughout each test. The rate at which the final temperature was attained varied from 0.1-8°C/s in order to determine whether or not the behaviour of intergranular bubbles was sensitive to changes in temperature ramp rate within this range. The fuel used for these tests had a burn-up of ~17 MWd/kgU and had two mean linear intercept grain sizes, 6 and 18 microns.

CENG defect fuel experiments, released February 2000

CEA failed PWR rod irradiation in SILOE: EDIT-MOX 01, released June 2000

These are data from a series of loop experiments performed on short fuel rods by CEA in France with the aim of measuring and interpreting the release rate of fission gases and iodine under a range of experimental conditions (linear power, defect type, gap dimensions, etc.). The results from these experiments, which cover steady state and transient power operation on new and pre-irradiated fuel, provide invaluable data for modelling the escape of fission products from defective fuel rods. In a commercial reactor, such modelling 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 shutdown of the reactor and also minimises outage time, thus improving the safety and economics of power generation.

IMC (UK) swelling data from ramped CAGR UO_2 fuel in Halden reactor, released December 2003

This data set comprises measurements of intra- and intergranular porosity, and associated swelling from an extensive study of UO_2 fuel power ramped in the Halden reactor. The ramp tests were performed to study the mechanisms of PCI in Advanced Gas-cooled Reactor (AGR) fuel, with the clad deformation measurements supplemented by the use of transmission electron microscopy (TEM) and scanning electron microscopy (SEM). Although the cladding in this case was stainless steel, the data on fuel swelling are generic and equally applicable to LWR fuel modelling. The SEM study comprises nearly three thousand micrographs. The study was augmented by use of the ENIGMA fuel modelling code to obtain estimates of local temperatures and conditions from which the SEM/TEM samples were obtained. In addition, unramped samples of the same fuel were annealed to temperatures of 1 600-1 900°C in a combination of different temperature ramp rates, maximum temperature attained and hold times.

NRU MT-4 and MT6A simulation tests, released December 2003

The MT tests were conducted in a specially designed test train in the NRU reactor to supply the specified coolant conditions of flowing steam, stagnant steam, and then the re-flood characteristic of a LOCA. Provided here is information on two materials tests, MT-6A and MT-4, considered to be well-characterised for the purposes of setting up computer cases. Typical instrumentation for the MT tests included: fuel centreline thermocouples, cladding inner surface thermocouples, cladding outer surface thermocouples, rod internal gas pressure transducers or pressure switches, coolant channel steam probes and self-powered neutron detectors. This instrumentation allowed for determining rupture times and cladding temperature. After the experiments, the test train was dismantled, cladding rupture sites were determined and fuel rod profilometry was performed in the spent fuel pool. Only limited destructive post-irradiation examination was performed on these two tests.

Information on the complete set of data compiled and reviewed for the IFPE Database can be found at http://www.nea.fr/html/science/fuel/ifpelst.html.

3. IAEA FUMEX II

The International Atomic Energy Agency is sponsoring a Coordinated Research Project on Fuel Modelling at Extended Burnup (FUMEX-II). Nineteen fuel-modelling groups are participating with the intention of improving their capabilities to understand and predict the behaviour of water reactor fuel at high burnups. The exercise is carried in coordination with the OECD/NEA.

The participants are using a mixture of data derived from actual irradiation histories of high burnup experimental fuel and commercial irradiations where post-irradiation examination measurements are available, combined with idealised power histories intended to represent possible future extended dwell commercial irradiations and test code capabilities at high burnup. All participants have been asked to model nine priority cases out of some 27 cases made available to them for the exercise from the IAEA/OECD Irradiated Fuel Performance Experimental Database.

The focus of the high priority cases is on the topics:

- Thermal performance
- Fission gas release
- Pellet to clad interaction (PCI)

at extended burn-up above 50 MWd/kg

The priority cases were designed to be generic in character, and several idealised cases were used. This approach allowed codes designed for very different regimes (PWR, VVER and CANDU reactor systems) to demonstrate their capabilities against simplified histories and well defined experimental conditions. There were further cases within the FUMEX-II data sets that were specific to commercial irradiations, including VVER conditions, and the participants have been using these cases to validate and improve VVER fuel modelling. In particular, the EEC, Karlsruhe fuel modelling code TRANSURANUS has been supplied to several participants from VVER operating countries. The code has been modified to allow modelling of VVER conditions, and has been shown to perform extremely well with the FUMEX-II cases. The data sets described in section 4, with the exception of the data from the SOFIT programme, are all being used in the FUMEX-II programme.

The FUMEX-II Coordinated Research Project is still in progress. Results of code predictions against a set of priority cases show that fuel temperature modelling is much improved since the previous FUMEX-1 CRP. Fuel centre temperature predictions are now generally good, and match the data well up to burnups of around 60MWd/kgU.

Fission gas release measurements are also generally satisfactory within normal operating burnups, up to around 50MWd/kgU, but accurate modelling the release at higher burnups is problematic, particularly in the absence of data that could help to discriminate between the various modelling options currently being used by the different teams. These different options lead to different release behaviour when the available burnup range is extrapolated.

4. Data of specific interest for VVER applications

4.1. Twelve (12) rods from the Finnish-Russian co-operative SOFIT programme

The SOFIT programme was a series of experiments on VVER 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 endurance of instrumentation	Parametric studies based on the irradiation of re-instrumented high burn-up rods	Irradiation tests under transient conditions

SOFIT 1 was comprised of four assemblies (1.1-1.4), each with 18 rods arranged in hexagonal geometry around a central non-fuelled tube used for instrumentation, including neutron detectors and coolant temperature thermocouples [1,2]. The fuel-to-clad gaps ranged between 140 and 290 microns, and the fill gas (both helium and xenon were used) varied in pressure between 0.1-2 MPa. The UO_2 density varied between 10.4-10.75 g/cc and the fabricated grain size was ~5 microns. The fuel active length was 1 000 mm within a total rod length of 1 200 mm.

In most cases, six of the 12 outer rods were instrumented with thermocouple hot junctions located between 300 and 500 mm above the lower end of the fuel stack and near the maximum power position. For SOFIT 1.1, the instrumented rods were as indicated in the following table.

Rod no.	Pellet OD (mm)	Gas gap (microns)	He fill pressure (MPa)	Relative power	Instrumentation
1	7.54	210	0.1	0.95	TC
2	7.60	150	0.1	0.85	TC
3	7.54	210	0.1	0.8	2 × TC
4	7.54	210	1.5	0.75	TC
5	7.48	270	0.5	0.75	TC
6	7.48	270	0.1	0.9	TC

Rods 1 and 3 represent the standard design for VVER-440 fuel with nominal parameter values. The equivalent table for SOFIT 1.3 is shown below.

Rod no.	Pellet OD (mm)	Gas gap (microns)	He fill pressure (MPa)	Fuel density (g/cc)	Instrumentation
1	7.60	150	0.1	10.60	TC+ EF+EC
2	7.55	200	0.1	10.55	TC
3	7.55	200	0.1	10.55	TC+EF+EC
4	7.48	270	0.1	10.45	TC+PF
5	7.60	150	0.1	10.50	TC
6	7.55	200	0.1	10.45	TC+PF

In these tables, TC, EF, EC and PF refer to thermocouple, fuel elongation detector, clad elongation detector and internal pressure monitor.

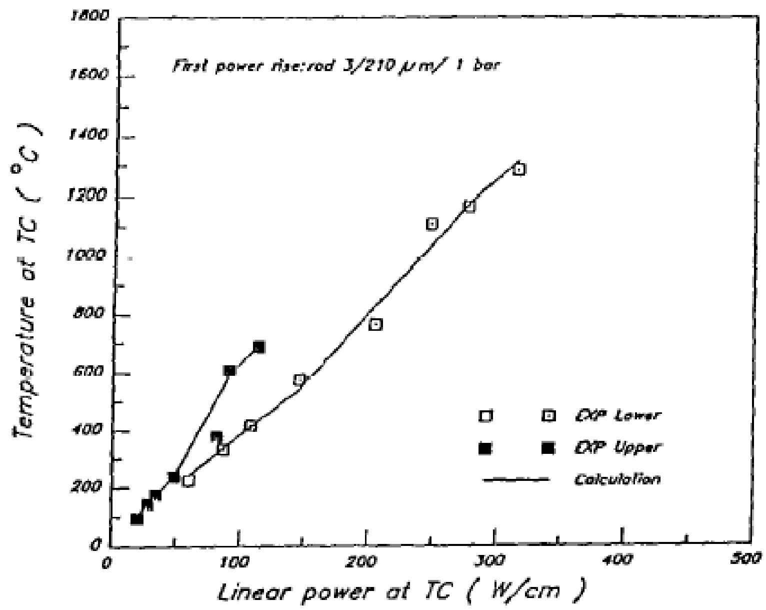
The data set available to date contains:

- Pre-characterisation and irradiation histories for SOFIT 1.1 rods 1-6, 7 and 12;
- In-pile temperatures for rods 1-6 and PIE FGR from non-instrumented rods 7 and 12;
- Pre-characterisation and irradiation histories for SOFIT 1.3 rods 1, 3, 4 and 5;
- In-pile fuel and clad extension measurements for rods 1 and 3;
- In-pile temperature data for rods 4 and 5.

The temperature data are very much as expected, with the effect of differing gap size clearly seen in the first ramp to power (**Figure 1**). From the rods fitted with extensometers, the early life behaviour shows negligible change in the clad length apart from thermal expansion, while there is evidence of significant fuel column shrinkage due to densification. Through life data at constant power, 30-32 kW/m are available for rods 3, 4 and 5 of SOFIT 1.1 up to 12, 3.5 and 6 MWd/kgU, respectively. Although rods 3 and 4 have the same gap dimension (210 microns), it can be seen that the higher fill pressure of rod 4 gives slightly lower temperatures than for rod 3, but this difference disappears with irradiation.

Regarding Fission Gas Release (FGR), two rods in the SOFIT 1.1 assembly show extremes in fission gas release behaviour. Rod 12 with a peak power of ~39 ± 4 kW/m to 16 MWd/kgU gave a measured release of 7-9%, while rod 7 with a peak power of ~28 ± 4 kW/m to 12 MWd/kgU, gave a measured release of 0-1%. It is clear that the higher fuel temperatures induced in rod 12 due to the higher power, some 10 kW/m higher than in rod 7, were responsible for the difference in fission gas released. It should be noted that destructive PIE of rods from SOFIT 1.1 showed negligible grain growth even for high power rods, ~40 kW/m; maximum observed grain size was 7 microns.

Figure 1. Measured temperatures for upper and lower thermocouples as a function of power during the startup of SOFIT 1.1 rod 3



Densification was measured by fuel stack shortening. Densifications of 0.8-2% were measured for initial densities of 10.65-10.7 and 10.55 g/cc. The major part of the densification took place early in life and was substantially complete by 1.5 MWd/kgU. The largest densification was seen in a xenon-filled rod that experienced high fuel temperatures.

Clad elongations are available for large and small gap rods, SOFIT 1.3 rod 1 with 270 micron gap, and rod 3 with 150 micron gap, respectively. In the first case, the clad elongation was very small throughout the 80 day irradiation. For the small gap rod, elongations were small and of the same order up to 48 days (2 MWd/kgU) when significant PCI occurred and clad elongation increased significantly.

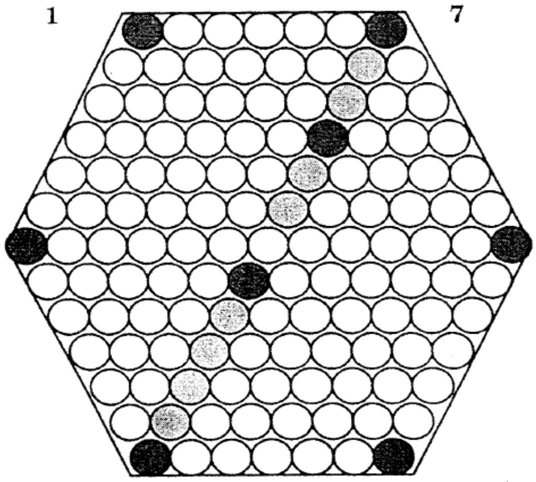
4.2. Kola-3 VVER-440 irradiation

The Database includes data for two pre-characterised standard VVER-440 fuel assemblies – FA-198 and FA-222, which were manufactured by the Russian fuel vendor, Elektrostal, and irradiated in the Kola-3 reactor. These assemblies were the centre of a programme called Blind Calculations for the VVER-440 High Burn-up Fuel Cycle Validation, and were initiated in the spring of 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 ~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 21-26 kW/m were experienced at the beginning of the second cycle, followed by steady state operation at LHGR of 10-22 kW/m, before gradually decreasing to ~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 VVER-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 2**). In addition to 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.

Figure 2. The arrangement and numbering of rods in assemblies FA-198 and FA-222 irradiated in the Kola-3 VVER-440



4.3. MIR ramp tests on Kola-3 rods

In 1996-1997, a number of over power transient tests were carried out in the MIR reactor (SSC RIAR) using high burn-up VVER-440 fuel. Fuel rods employed were from FA-198 and FA-222 and had operated under normal operating conditions at Kola NPP Unit 3 during four and five fuel cycles up to a maximum burn-up of ~50 and 60 MWd/kgU (see preceding Section 4.2). The tests were carried out under single ramp conditions (RAMP experiment) and step-by-step power increase (FGR-1 and FGR-2 experiments). The objectives of the experiments were to determine the influence of power ramps on fuel rod behaviour, to evaluate the threshold value of linear power for promoting FGR, and to study the influence of the structure and properties of the fuel [3,4].

The rods for which data are available in the data set are identified in the following table.

MIR test	Base rod FA/rod no.	Stack length (mm)	Instrumentation	PIE
FGR1 32	222/6	950	None	No
FGR1 41	198/99	750	PF	Yes
FGR1 48	222/2	750	PF	Yes
FGR2 50	222/25	400	TC	No
FGR2 51	198/20	400	TC	No
FGR2 52	222/46	400	None	No
RAMP 33	198/76	950	None	Yes
RAMP 37	222/5	950	None	Yes
RAMP 38	222/3	950	None	No

The RAMP test can be divided into three stages:

Stage 1 Irradiation at initial power level (duration ~346 hours);

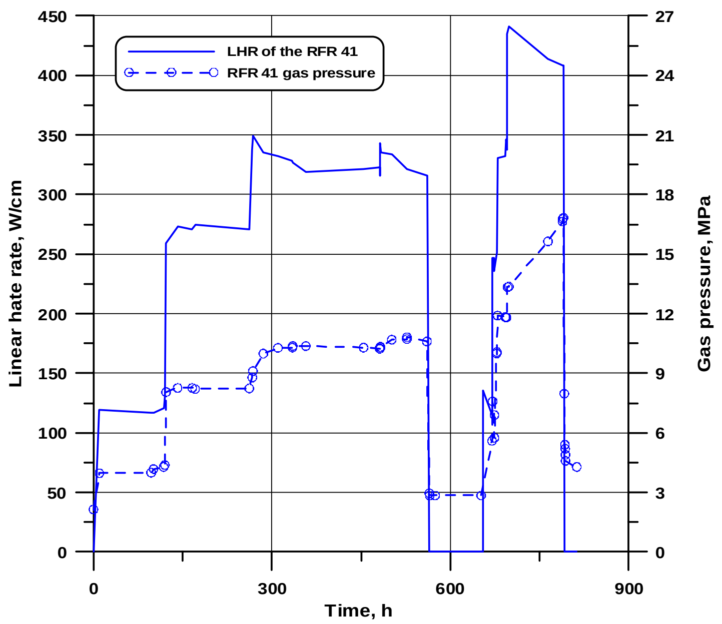
Stage 2 Power ramp from initial to maximal level (duration ~23 minutes);

Stage 3 Hold stage after ramp (duration ~107 hours).

Power levels of the rods at Stages 1 and 3 of the RAMP test are presented in the table below:

Rod no.	**33**	**37**	**38**
Stage 1 LHR, kW/m	17.0	14.2	12.7
Stage 3 LHR, kW/m	36.1	30.3	27.1
Power ramp magnitude, kW/m	19.1	16.1	14.4

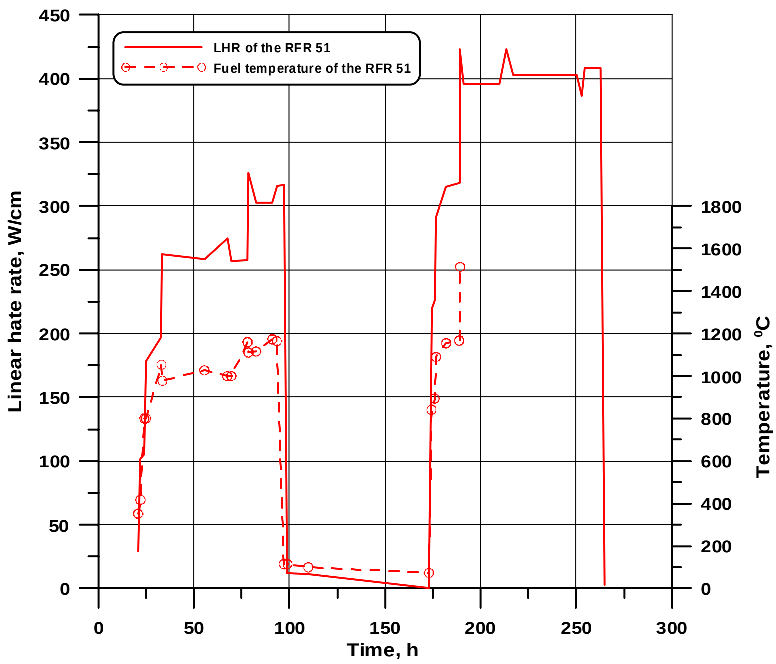
Figure 3. The evolution of power and pressure sensor response for rod 41 during FGR-1



The FGR-1 tests were carried out in two stages, with each of these divided into several sub-stages. Part way through the tests there was a short shutdown (~0.5 hours), with the maximum power of the rods reached in the final stage. The evolution of power and pressure sensor response for rod 41 is shown in **Figure 3**.

The FGR-2 experiment was the second of the test series directed to investigate the thermal-physical behaviour of VVER fuel at the different power levels. The test rig included three rods: 50, 51 and 52, with rods 50 and 51 re-instrumented with thermocouples. As in the case of FGR-1, the FGR-2 tests were carried out in two stages, each divided into four sub-stages. The power evolution at the level of the thermocouple and the thermocouple response of rod 51 is shown in **Figure 4**.

Figure 4. The evolution of power at the level of the thermocouple and the thermocouple response of rod 51 in FGR-2



FGR values for all tests are given in the following table. On discharge, the thermocouples of rods 50 and 51 were found to be damaged; although an FGR value for rod 50 was possible, this was not the case for rod 51. Where available, further details of the destructive examination are given in the data set.

MIR test	Base rod FA/rod no.	Instrumentation	FGR (%)
FGR1 32	222/6	None	46.6
FGR1 41	198/99	PF	47.5
FGR1 48	222/2	PF	50.0
FGR2 50	222/25	TC	48.37
FGR2 51	198/20	TC	?
FGR2 52	222/46	None	48.37
RAMP 33	198/76	None	31.3
RAMP 37	222/5	None	16.9
RAMP 38	222/3	None	19.6

4.4. Zaporozskaya VVER-1000 irradiation

Fuel assembly FA-0325 was operated in the first unit of VVER-1000 Zaporozskaya NPP during the fuel cycles 4-8. It was irradiated for 1142.1 effective days, to an assembly averaged burn-up of 48.9 MWd/kgU with a peak burn-up of 51.3 MWd/kgU in the hottest fuel rod. The maximum local linear power in any fuel rod was 29 kW/m, while the average linear power for the whole irradiation varied within the limits of 7.1-26.6 kW/m. The operational conditions of the assembly corresponded to a four-year fuel cycle, both in dwell time and discharge burn-up of the fuel. After the base irradiation, the assembly was subjected to PIE in the hot cells of SSC NIIAR Dimitrovgrad (Russian Federation).

During PIE, the following measurements were made:

- Burn-up by non-destructive gamma scanning of ^{137}Cs;
- Axial evolution of clad diameter;
- Rod length;
- Fuel-to-clad residual gap by radial deformation; the gap size was determined from the force-displacement relationship;
- FGR.

The data set contains pre-characterisation, 10-zone irradiation histories and PIE measured data for 312 rods from this assembly. Rod length changes varied between 0.35-0.46%, diameter decrease was 0.55-0.87% and FGR values ranged between 0.2-2.5%.

4.5. Novovoronezh VVER-1000 irradiation

FA-4108 was operated in the fifth unit of Novovoronezh NPP during fuel cycles 7-9 to an average fuel burn-up of ~47 MWd/kgU. A plot of the reactor power during that time is given in **Figure 5**.

The irradiation histories of 317 fuel rods are presented in separate files where variables are given for 30 axial zones in the following forms:

- Bu – local burn-up, MWd/kgU;
- Ql – local heat rate, kW/m;
- T – temperature of cladding inner surface, °C;
- Ft – neutron fluence (× 10-22), n/s/cm2 (E > 0.1 MEv).

The method of numbering rods for this assembly is given in **Figure 6**.

After irradiation, the assembly was shipped to the hot cells of NIIAR Dimitrovgrad and an account of the PIE was published by A. Smirnov, *et al*. [5].

Figure 5. Reactor power for cycles 7-9 of the Novovoronezh NPP unit 5

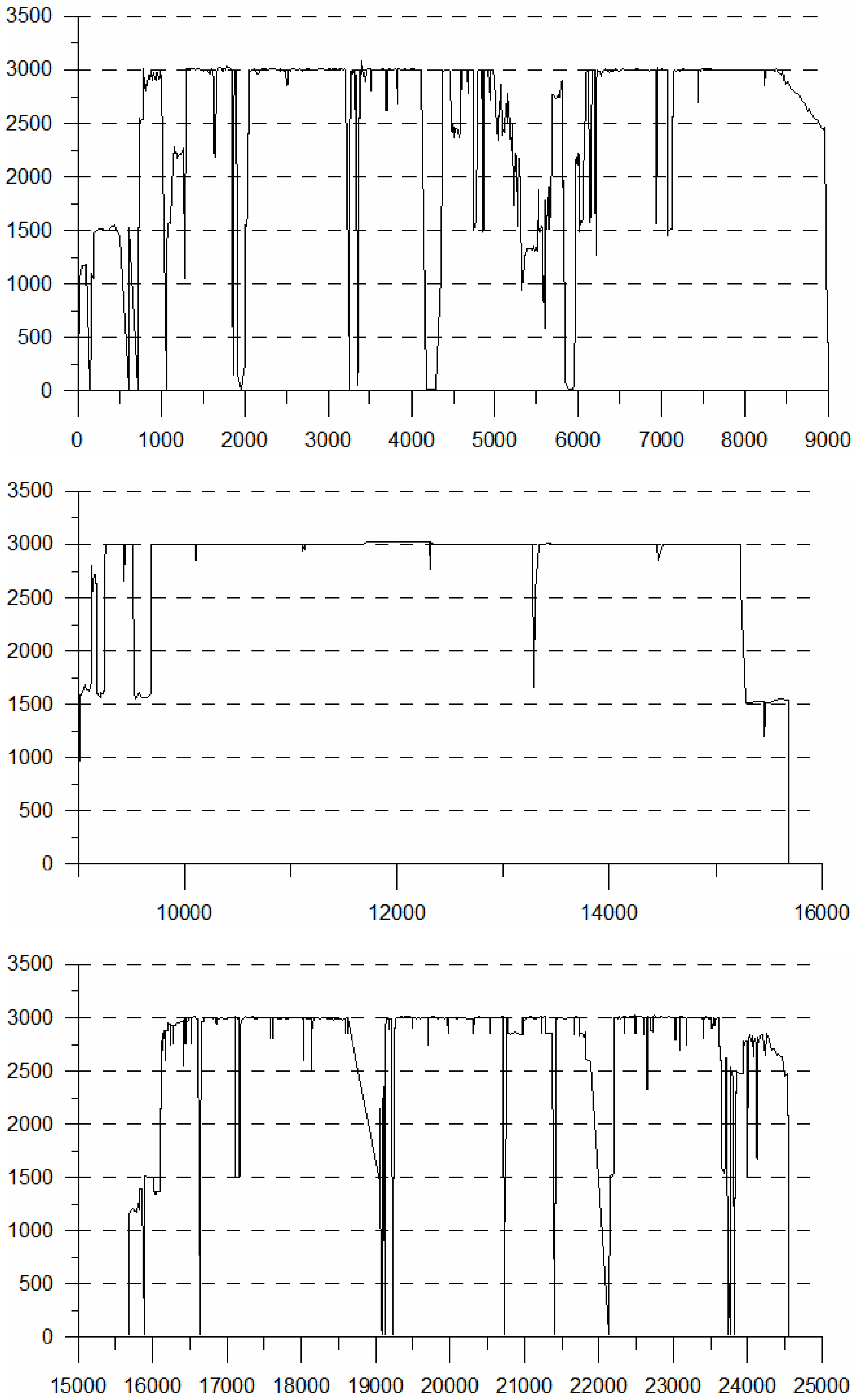
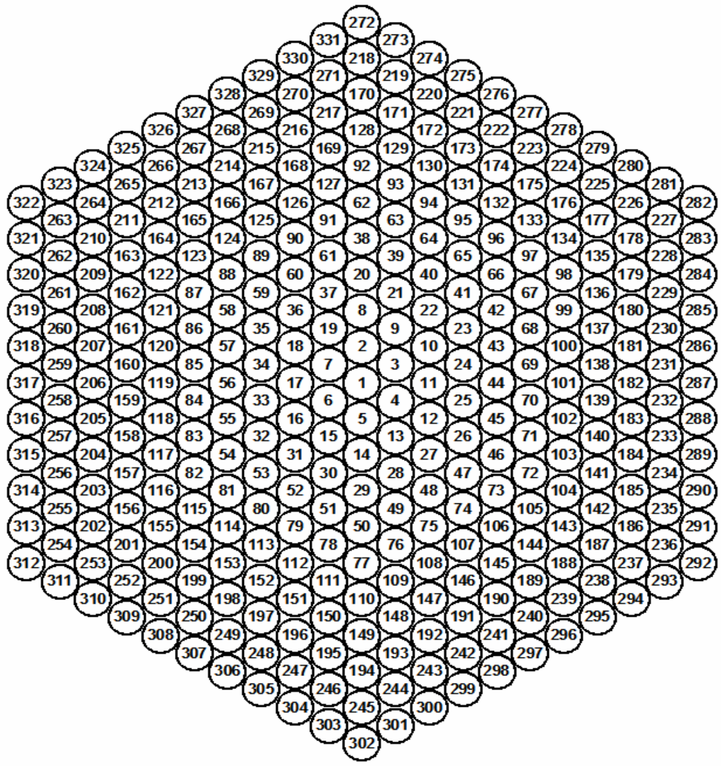


Figure 6. Identification numbers for rods in assembly FA-4108 irradiated during cycles 7-9 in the Novovoronezh VVER-1000 NPP unit 5



4.6. HRP IFA-597.4 rod 1 and 2 (MOX) – Solid and hollow MOX pellet behaviour

Additional fuel behaviour studies of relevance for VVER-1000 reactors are carried out under the aegis of the NEA Nuclear Science Committee in the framework of the Expert Group on Reactor-based Plutonium Disposition (TFRPD). Work is concentrated on modelling the behaviour of full or hollow MOX pellets using experimental data from the Halden Reactor Project. The objective is to validate the models and codes used for MOX fuel behaviour and to create a favourable licensing environment in the Russian Federation and USA for loading MOX fuels in PWRs and VVERs. Future work will concentrate on transients and RIA conditions. A final report of this study and eventually the data will be integrated into the IFPE database.

5. OECD/NEA Data Bank

The OECD/NEA Data Bank acquires tests and distributes computer codes in the field of nuclear technology applications on behalf of its member countries and through a co-operative agreement with the IAEA. The OECD/NEA Data Bank distributes in particular non-commercial computer codes for fuel behaviour modelling. Codes distributed through Computer Program Services are provided in the Annex.

6. Conclusion

The effort carried out at an international level to ensure that valuable data from fuel behaviour tests are preserved and shared internationally has proven beneficial to modellers in research and industry. In fact, ten editions of IFPE have been issued since 1996. The present version contains data on 850 rods or samples. IFPE data were requested by ~100 different establishments in 32 countries. The creation of the database has met with universal approval and, consequently, there has been no difficulty in obtaining data. However, there is a need to make the database more comprehensive and therefore new data are always welcome. Exercises in the FUMEX series have contributed to model development and verification against good quality data. These exercises have helped to share understanding in the basic phenomena underlying fuel behaviour and to improve the predictive power of the models.

Access to this joint database is through the NEA, which provides the data and documentation on CDs. While every care has been taken in preparing complete error-free data sets, there is sometimes a need to issue revisions. It is therefore important that all users provide feedback that would help remove possible inconsistencies/errors and thus contribute to the improvement of the database.

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Annex

FUEL BEHAVIOUR COMPUTER CODES
DISTRIBUTED BY THE OECD/NEA DATA BANK

Program name	Description
ANSCLAD-1	Creep strain in fuel pin zircaloy clad during temperature transient
BUST	Elastic stress in HTGR pressurised fuel element
COMET	Mechanical and thermal stress in fuel element clad
COMTA	Ceramic fuel element stress analysis
DRUCK	Thermal-mechanical stress of PWR fuel rod during LOCA blowdown
FAMREC	PWR lateral mechanical fuel rod assembly response
FASTGRASS	Gaseous FP release in UO_2 fuel
FEMAXI-V	Thermal and mechanical behaviour of LWR fuel rods
FRANCO	FEM fuel rod anal for solid and annular configurations
FRAPCON2	Steady state LWR oxide fuel element behaviour, FP gas release, error analysis
FRAP-T	Temperature and pressure in oxide fuel during LWR LOCA
FRETA-B	LWR fuel rod bundle behaviour during LOCA
FREVAP-6	Metal FP release from HTGR fuel elements
GAPCON-THERMAL3	Fuel rod steady state and transient thermal behaviour – stress analysis
GRASS-SST	Fission gas release and fuel swelling in steady state and transients
GTR2 GAPCON-THERMAL2	Steady state fuel rod thermal behaviour and FP gas release
HASSAN	Time dependent temperature distribution, stress and strain in HTGR fuel pins
LIFE-1	Stress analysis swelling, cylindrical fuel element performance in fast reactors
MARGE SLUMP	Radial temperature distribution and void diameter, MOX LMFBR fuel pin
MOXY/MOD-1	Thermal analysis swelling and rupture of BWR fuel element during LOCA
PIN99W	Modelling of VVER and PWR fuel rod thermal and mechanical behaviour
RODBURN	Power profiles and isotopics in PWR-BWR fuel rods (input for FEMAXI-V)
SPAGAF	PWR fuel, cladding behaviour with FP gas
STOFFEL-1	Steady state in-pile behaviour of cylindrical water-cooled oxide fuel rod
TAPIR	Thermal analysis of HTGR with graphite sleeve fuel element
TEMPUL	Temperature distribution in fuel element after pulse
THETA-1B	Fuel rod temperature distribution by 2-D diffusion, heat transfer to coolant
WELWING	Material buckling for HWR with annular fuel element
WREM TWODEE-2/MOD3	2-D time-dependent fuel element thermal analysis after LOCA
ZZ FUELS-DATA	Data library for LWR fuel behaviour for FRAP program

FP= Fission Products; LOCA= Loss of Coolant Accident

For more information, see the following website: http://www.nea.fr/html/dbprog/.