NUCLEAR FUEL BEHAVIOUR ACTIVITIES

at the OECD/NEA

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Abstract

The work programme in the field of fuel behaviour carried out at the OECD/Nuclear Energy Agency during the last year is described as well as plans for future work.

The programme covers the following aspects:

- ***Nuclear Science Committee***:
  - Scientific Issues in Fuel Behaviour
  - International Fuel Performance Experiments (IFPE) database
  - Seminar on Fission Gas Behaviour in Water Reactor Fuels

- ***Data Bank***:
  - Computer Program Service

- ***Committee on the Safety of Nuclear Installations***:
  - Topical meeting on LOCA Fuel Safety Criteria
  - 2nd Meeting on Fuel Safety Margins

Nuclear Science Committee:

Scientific Issues in Fuel Behaviour & International Fuel Performance Experiments Database

The fifth meeting of the Expert Group on Scientific Issues in Fuel Behaviour was held on 25 September 2000 in Cadarache, and was chaired by W. Wiesenack. It was attended by 32 participants from 11 countries, representing 27 organisations. This meeting was co-organised by OECD/NEA and the IAEA. It was held in conjunction with the seminar on Fission Gas Behaviour in Water Reactor Fuels.

The objectives of the meeting included:

- review of progress in establishing and operating the joint OECD/NEA-IAEA International Fuel Performance Experiments (IFPE) database.
- handling of feedback from its utilisation
- identification of needs for further data to be integrated
- establishing an IFPE user forum
- cooperation with IAEA on FUMEX-II (FUel Modelling EXercise), and its relation to the IFPE database

The items discussed in detail concerned:

- *Status of the IFPE Data Base*: because the meeting was followed by the seminar on Fission Gas Release, special emphasis was given to describing the characteristics of IFPE datasets that contain relevant data for this aspect. The status, methodology, feedback procedure and QA were discussed in detail. The present status of IFPE is described in more detail in Annex 1.
- The design, functionalities and performance of the French fuel performance *database CRACO* and its use in connection with the METEOR computer code was presented, and the usefulness / feasibility of adopting this method for IFPE was debated.
- *Experiments completed* or in progress, the results of which may be integrated into IFPE were addressed. In particular, the PHWR MOX fuel experiments of CNEA, Argentina were presented in detail. This dataset was released to IFPE since then.
- Contributions to the *IFPE validation* were presented e.g. with TRANSURANUS for VVER fuels, and FRAPCON-6 for PWR fuels.
- In order to improve communication among users of the IFPE data and to establish a convenient way of providing feedback from their use for validation it was decided to establish an *IFPE Internet forum*, the address for queries and replies is ifpe@nea.fr.
- A follow-up to the FUMEX-I exercise, the *FUMEX-II exercise* was presented. CEA proposed three parts: 1) two cycle rod power transients / short holding times, 2) three/four cycle rod power transient / long hold times, 3) four cycle rod power transient / long holding times. HRP proposed to offer in-core measurements of rod pressure (FGR), fuel centre-line temperature (steady-state and scram data), and cladding elongation (PCMI). Concerning high burnup refabricated BWR fuel would be used. The burnup would range from 60-70 MWd/kg. It was recommended that the exercise be a blind test.
- *Release of new data for IFPE*. New data was offered by Studsvik: Trans - Ramp I, II, IV. Belgonucléaire agreed to offer data from the GAIN programme. INR Pitesti offered data from two experimental fuel elements i.e. RO-51 and RO-89 irradiated at element power of approximately 520-590 W/cm. JAERI offered data concerned with PCMI behaviour from experiments carried out at HRP, on different cladding thickness by means of diameter rig. NUPEC promised to look into the possibility of releasing some of their data. Other valuable candidates for inclusion would be fission gas release data from high burnup VVER-440 fuel under steady-state and transient operation conditions from NIIAR, Dimitrovgrad, ramp data on WWER-440 up to 4 cycles and WWER-1000 up to 3

cycles from VNIINM. The possibility of releasing some of the IFA-504 on He/Ar/Xe gas flow, Nb doped fuel, by HRP is being discussed.

One of the discussions concerned requirements for new data to be included into IFPE. The results are summarised in the following table:

Requirements for New Data for IFPE Database

Topic	Details / Examples
Power reactor data	Different vendors and reactor systems e.g., CANDU, WWER-1000.
Fuel temperatures	Beginning of life, dependence on design parameters, MOX, transient change in temperature with changing power, at higher linear-heat generation. Better statistics is needed
PCI	Need a large number of cases to improve statistics for failure/non-failure threshold.
MOX	All forms of performance data.
UO_2 with additives	Nb, Cr etc. and burnable poisons e.g. Gd.
Densification and solid fission product swelling	Data for different fuel variants and the effect of microstructure.
Sweep gas experiments	Data on release of radioactive fission products in intact rods.
Out-of-pile experiments	Dedicated experiments e.g. FGR in high temperature transients, laser flash determination of thermal conductivity.
Detailed PIE	Data on swelling , pore sizes inter- and intra-granular bubble distributions after specific irradiation test conditions.
High Burn-up	All forms of data where ‘rim’ structure has developed

Conclusions and Recommendations

- Progress since the last meeting was reviewed: the number of cases has doubled since the last meeting and reached at the time of the meeting 387 (416 in April 2001).
- Comprehensiveness has been improved, but some gaps have been identified and requirements for new data were discussed.
- The main scope will continue on water moderator reactor fuels, but extensions are being considered, in particular to GAGR and FR fuels, laser flash diffusion data, high temperature transient, etc.
- Format of data will be kept simple with some improvement for the user to navigate through the large number of files. Moving to more sophisticated methods, e.g. Oracle or XTML will be considered later
- It was agreed that the evaluation of the experiments should always record the history of changes in the data sets, and the uncertainty in the power history.
- An Internet IFPE User Forum should be set up for discussing and exchanging information on specific topics and/or experiments, discuss difficulties, reporting feedback on IFPE use, and improve / enforce gathering feedback from users. A questionnaire should be prepared for the gathering of feedback.
- Further data being released or offered to IFPE was identified: Studsvik TransRamp (received since), Gain (UO_2-Gd_2O_3) from Belgonucléaire (end of May 2001), IFA's on PCMI by JAERI, data owned by NUPEC, data from INR Pitesti, Romania, data released within the FUMEX-II activity, etc. More cases will improve statistics.
- More frequent meetings are recommended (once/year). One meeting should be held in Spring 2002 in conjunction with the seminar on PCMI, Cadarache.
- Co-operation with the FUMEX-II project is recommended.
- Participants asked the chairman to report the results to NSC and emphasise that a strong support is needed, that a financial backing should be available, and that the scope be extended in the future to other type of data.

The latest edition of IFPE, issued in January 2001, was requested by 72 establishments in 20 countries who are members of both IAEA and OECD/NEA: Argentina, Bulgaria, Belgium, Canada, Czech Republic, Finland, France, Germany, Hungary, Japan, Republic of Korea, Poland, Russian Federation, Rumania, Slovak Republic, Spain, Sweden, Turkey, United Kingdom and United States of America.

Seminar on Fission Gas Behaviour of Water Reactor Fuel

This seminar was co-organised by CEA, OECD/NEA and IAEA and sponsored by EDF, COGEMA and Framatome. It was held from the 26 to 28 September 2000 at Cadarache. The seminar was attended by about 100 participants from 24 countries, representing 46 organisations.

The aim of the seminar was to establish a comprehensive picture of the current understanding of fission gas behaviour and its impact on the fuel rod, how the implementation of increasingly sophisticated experimental techniques (both in and out of pile) improves understanding and anticipates how fission gases behave. It addressed high burn-up behaviour, MOX and other advanced fuel concepts. The broadest range of operating conditions (i.e. normal, off-normal and accidental) and the more recently considered long-term storage conditions, were discussed and found as relevant.

At the end of the seminar there was a panel discussion where the results of the different sessions were presented. The proceedings will be published early 2001. The full proceedings will contain, besides a summary, the 38 papers presented at the seminar. The executive summary of that seminar was published as NEA/NSC/DOC(2000)20 (21 November 2000).

A further seminar is planned on **pellet clad mechanical interaction**. It is scheduled for spring 2002, also at Cadarache.

Expert Group on Reactor-based Plutonium Disposition

One activity of this Group concerns MOX fuel behaviour. At present a benchmark study addresses the behaviour of full and hollow MOX fuel pellets. In a first phase the results of comparing different codes for given fuel characteristics and power history are compared against each other. In the final phase these results are compared against data from experiments, which will be made available explicitly for this study. This work is in progress and will be finalised during 2002.

Meetings of Interest

During 2001 two meetings of interest will take place as far as innovative fuels are concerned:

- Workshop on ***Advanced Reactors with Innovative Fuels (ARWIF-2001)*** scheduled for 22-24 October 2001 at Chester, UK, co-organised by BNFL and the OECD/NEA
- The Seventh Meeting on ***Inert Matrix Fuels (IMF-7)*** scheduled for 25-26 October 2001 at Petten, The Netherlands.

Data Bank

Computer Program Service

The NEA Data Bank acquires, tests and distributes on behalf of its member countries and through a cooperative agreement with the IAEA computer codes in the field of nuclear technology applications. It includes also computer codes for fuel behaviour modelling. Codes distributes through the Computer Program Services are provided as Annex 2.

Committee on the Safety of Nuclear Installations (CSNI)

Special Expert Group on Fuel Safety Margins

Topical Meeting on LOCA Fuel Safety Criteria

A topical meeting on LOCA Fuel Safety Criteria was held in Aix-en-Provence on 22-23 March 2001. It was organised under the auspices of the CSNI and its Special Expert Group on Fuel Safety Margins in co-operation with IPSN Cadarache. The meeting was chaired by Dr. Georges Hache from IPSN.

In total 53 participants attended. Research and industry organisations from France, Hungary, Japan, the Russian Federation and the USA presented 14 papers in all. The papers covered three main issues:

1. post-quench ductility;
2. the impact of axial constraint; and
3. fuel relocation.

Proceedings of the meeting, along with conclusions and recommendations, will be published shortly. Table 1 shows the authors, their organisation and the title of the papers presented.

A number of issues was identified, resolution of which must come from ongoing and planned national and international experimental programmes:

- ***High burnup Zircaloy***: Combined effects of H and O uptakes on post-quench ductility
  - Role of axial restraints during quench (internal versus external forces)
  - Role and consequences of fuel relocation during quench
- ***New alloys*: *(ZIRLO, M5^{TM})*** Combined effect of Nb addition along with H and O uptakes on post-quench ductility
  - Verification of PCT criterion
  - Differences in behaviour compared to Russian E110 alloy with 1%Nb.

The Second Plenary Meeting of the Special Expert Group on Fuel Safety Margins

The LOCA Topical Meeting was followed by the 2nd plenary meeting of the Special Expert Group on Fuel Safety Margins which reviewed the status of ongoing activities and discussed future activities, in particular:

- Preparation of a CSNI report of fuel safety related R&D programmes in NEA Member States (ongoing)
- Preparation of a CSNI report on fuel safety criteria used in NEA Member States (ongoing)
- Initiation of ISP based on a CABRI test (still pending)
- Organization of the next topical meeting on RIA fuel criteria (approved)
- Co-sponsoring the IAEA TCM on fuel behaviour under transient and LOCA conditions, Halden, September 10 – 14, 2001, (supported)

Table 1:

Author	Organisation	Paper Title
G. Hache	IPSN, France	The rationale of the LOCA 10CFR50.46b criteria for Zircaloy and comparison with E110 alloy
Ralph Meyer	NRC, USA	NRC program for addressing effects of high burnup and cladding alloy on LOCA safety assessment
Laszló Maróti	AEKI Budapest, Hungary	Ring-compression test results and experiments supporting LOCA PCT, oxidation and channel blockage criteria
A. Le Bourhis	Framatome-ANP, France	Justification of the M5TM behavior in LOCA
W.J. Leech	Westinghouse, USA	Ductility testing of Zircaloy-4 and ZIRLO^{TM} Westinghouse cladding after high temperature oxidation in steam
Hee Chung, R.V. Strain, T. Bray, M.C. Billone	ANL, Argonne, USA	Progress in ANL/USNRC/EPRI program on LOCA
Yu. K. Bibilashvili, N.B. Sokolov, L.N. Andreeva-Andrievskaya, et al.	VNIINM, Russian Federation	Thermomechanical properties of Oxydized Zirconium based alloy claddings in LOCA conditions
Nicolas Waeckel, Patrick Jacques	EDF, France	Analysis of fuel rod axial forces during LOCA quench
Rosa Yang, Robert Montgomery	EPRI, USA	
Hiroshi Uetsuka, F. Nagase,	JAERI, Japan	Progress in JAERI program on high burnup fuel behavior under a LOCA transient
Michel Lambert, Yann Le Hénaff	EDF/SEPTEN, France	Synthesis of an EDF and FRAMATOME ANP analysis on fuel relocation impact in large break LOCA
Jean-Luc Gandrille	Framatome-ANP, France	
C. Grandjean, G. Hache, C. Rongier	IPSN, France	High burnup UO_2 Fuel LOCA calculations to evaluate the possible impact of fuel relocation after burst
Hiroshi Hayashi	NUPEC, Japan	High Burnup Phenomena Affecting the Failure Mode of Fuel Japan Rods During LOCA
V.P. Smirnov et al.	RIAR, Russian Federation	Results of the experimental research on high burnup VVER-type fuel behaviour in LOCA conditions
A. Mailliat	IPSN, France	IPSN analysis of experimental needs requested for solving pending LOCA issues

Annex 1

Update: 12 May 2001

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

I F P E

The Aim of the IFPE Database Project

The aim of the project is to provide in the public domain, a comprehensive and well-qualified database on Zr clad UO_2 fuel for model development and code validation. The data encompasses both normal and off-normal operation and include prototypic commercial irradiations as well as experiments performed in Material Testing Reactors. This work is carried out in close co-operation and co-ordination between **OECD/NEA**, the **IAEA** and the **IFE/OECD/Halden Reactor Project**

Activities within the IFPE Database Project

- **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.
- **distribution** to interested parties and assistance where necessary in use of datasets.

Composition of the IFPE Database

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. At present two sets of data concerned with MOX fuel experiments are included. Emphasis has been placed on including well-qualified data that illustrate specific aspects of fuel performance. Of particular interest to fuel modellers 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 caesium, xenon, other fission product and actinides.

Data Currently Available

To date datasets about 416 rods/samples from various sources encompassing PWR, BWR, PHWR and WWER reactor systems have been included:

Experiments	Cases
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 Programme	12 rods
The High Burn-up Effects Programme	81 rods
WWER rods from Kola-3	32 rods
Rods from the TRIBULATION programme	19 rods
Studsvik INTER-RAMP BWR Project	20 rods
Studsvik OVER-RAMP PWR Project	39 rods
Studsvik SUPER-RAMP PWR Sub-Programme	28 rods
Studsvik SUPER-RAMP BWR Sub-Programme	16 rods
Studsvik DEMO-RAMP I - BWR	5 rods
Studsvik DEMO-RAMP II - BWR	8 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
CEA failed PWR rods irradiated in SILOE: EDITH-MOX 01	1 rod
CNEA six power ramp irradiations with (PHWR) MOX fuels	5 rods
Total	416 cases

Data in Progress of being Processed or Released

Released and/or being processed

- BR-3 High Burnup Fuel Rod Hot Cell Program (DOE/ET 34073-1, Vol. 1 & 2)
- RISØ-I experiment
- IFE/OECD/HRP FUMEX 1-6
- Studsvik/SKI data from TRANS-RAMP I, II and IV

Data release in progress or requested

- HRP He/Ar/Xe gas flow, Nb doped fuel, IFA-504
- VNIINM ramp data from WWER-440 up to 4 cycles and WWER-1000 up to 3 cycles
- IFA-508 and IFA-515 conducted by JAERI at HRP - PCMI behavior data on different cladding thickness by means of diameter rig
- NUPEC owned irradiation experiments
- CEA failed PWR rods irradiated in SILOE
  EDITH-3
  EDITH- MOX 02
- Belgonucleaire GAIN programme
  fabrication, irradiation and PIE of 4 ramped UO_2-Gd_2O_3 fuel rods with Zr4, cladding, irradiated in BR3 PWR (50 GWd/tU for peak pellet)
- INR Pitesti - RO-51 and RO-89 CANDU fuel type in-reactor measurement of internal gas pressure
- IMC (UK) swelling data from ramping CAGR UO_2 fuel in the Halden Reactor
- NIIAR (RF) fission gas release data from high burnup VVER-440 fuel under steady-state and transient operation conditions from,
- Data from laser flash measurements
- Data from high temperature transients

Contributing Organisations

AEA Technology
Atomic Energy of Canada Ltd. (AECL)
Battelle Pacific North-West Laboratory (PNL)
Belgonucleaire
British Nuclear Fuels Ltd. (BNFL)
Comision Nacional d'Energia Atomica (CNEA)
Commissariat a l'Energie Atomique (CEA)
EC Institute for Transuranium (ITU)
Electricite de France (EdF)
Framatome
Halden Reactor Project (HRP)
Imatran Voima Oy (IVO)
Institute of Nuclear Research Pitesti (INR-Pitesti)
Kurchatov Institute Moscow (INR RCC)
Research Institute of Inorganic Materials (VNIINM)
Risoe National Laboratory
Siemens Power Corporation
Studsvik AB
Swedish Nuclear Power Inspectorate (SKI)
TVEL - Joint Stock Company, Moscow
UK Health and Safety Executive
UK Industry Management Committee (IMC)
US Department of Energy(DOE)

Further information can be found in

http://www.nea.fr/html/science/fuel/ifpelst.html

Annex 2

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
COBRA	transient thermal-hydraulics fuel element clusters, sub-channel analysis
COMET	mechanical & thermal stress in fuel element clad
COMTA	ceramic fuel element stress analysis
DRUCK	thermal - mechanical stress of PWR fuel rod during LOCA blow-down
FAMREC	PWR lateral mechanical fuel rod assembly response
FASTGRASS	gaseous FP release in UO_2 fuel
FEMAXI-V	thermal & mechanical behaviour of LWR fuel rods
FRANCO	FEM fuel rod anal for solid & annular configurations
FRAPCON2	steady-state LWR oxide fuel element behaviour, FP gas release, error analysis
FRAP-T	temperature & 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 & transient thermal behaviour - stress analysis
GRASS-SST	fission gas release & fuel swelling in steady-state & transients
GTR2 GAPCON-THERMAL2	steady-state fuel rod thermal behaviour & FP gas release
HASSAN	time dependent temperature distribution & stress & strain in HTGR fuel pins
LIFE-1	stress analysis swelling, cylindrical fuel element performance in fast reactors
MARGE SLUMP	radial temperature distribution & void diameter, MOX LMFBR fuel pin
MOXY/MOD-1	thermal analysis swelling & rupture of BWR fuel element during LOCA
PIN99W	modelling of VVER and PWR fuel rod thermal & mechanical behaviour
RODBURN	power profiles & 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

For more information see: http://www.nea.fr/html/dbprog/