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NUCLEAR ENERGY AGENCY
NUCLEAR SCIENCE COMMITTEE

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SUMMARY OF THE FIFTH MEETING OF THE EXPERT GROUP ON SCIENTIFIC ISSUES OF FUEL BEHAVIOUR (EGSFB)

International Fuel Performance Experiments Data Base (IFPE)
(Status and Programme)

25th September 2000
Cadarache, France

English - Or. English

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OECD/NEA Nuclear Science Committee (NSC)
Expert Group on Scientific Issues of Fuel Behaviour
(EGSFB)

International Fuel Performance Experiments Data Base (IFPE)
(Status and Programme)

Cadarache, 25 September 2000

Summary of the Fifth Meeting

1. Opening/Introduction of Participants

On behalf of the local organisers, M. Noé of CEA welcomed all participants to the meeting.

The fifth meeting of the Expert Group on Scientific Issues in Fuel Behaviour was opened by the newly designated chairman, W. Wiesenack. He welcomed the 32 participants from 12 countries, representing 27 organisations (see Annex 1). This meeting was co-organised by the OECD/NEA and the IAEA, and was held in conjunction with the seminar on Fission Gas Behaviour in Water Reactor Fuels.

2. Objectives of the Meeting/Agenda

The proposed agenda was reviewed and endorsed (see Annex 2), as well as the objectives of the meeting as follows:

- 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.

For convenience, the Scope and Objectives of the Expert Group is attached (Annex 3).

3. Status of the IFPE Data Base

J A Turnbull (Consultant), P Menut (IAEA) and E Sartori (NEA): A Review of Fission Gas Release Data (FGR) within the NEA/IAEA IFPE Database

Turnbull reviewed the status of the data present at the time of the meeting in the IFPE database. The purpose of the IFPE database is to provide sufficient information to allow modellers to develop individual models describing the several processes contributing to release and also allow them to validate code predictions as a whole. At the time of the meeting, the database comprises some 380 individual cases, a substancial increase compared to previous editions, and covers most of the parameters of interest for FGR modelling. Concerning this phenomenon, data are required to cover **rod design**, in particular the effect of ***fuel clad gap*** size influenced by fuel temperatures and thermal feedback, the ***fill gas pressure***, where thermal feedback, rod length, and axial gas transport play an important role, and ***hollow versus solid pellets***, with differing effect of stress on FGR, i.e. a reduction in pellet clad mechanical interaction.

The data need to cover also the different **vendor/manufacturing process** such as the pellet fabrication route (IDR, ADU, wet route, dry route, etc.), the UO_2 grain size, a major factor affecting FGR and PCMI, the additions, e.g. Gd_2O_3 as a burnable poison, Nb_2O_5 etc. as a grain growth promoter, MOX, the pore size distribution, having an effect on sintering behaviour and temperatures, and the cladding properties (creepdown and effect on temperatures).

Finally, the **irradiation history** is required with the specification of the reactor type, (PWR, BWR, WWER, CAGR, CANDU), as each has different modes of operation, the effect of position in assembly, the description of speed, magnitude, duration, and repeats of power transients, the power cycling, load following (apparently not effective in promoting FGR), the loading schemes (in-out-in etc.), and the discharge burn-up (high burn-up properties, conductivity degradation, rim structure).

Highlights of the IFPE data sets adressing the FGR issues were presented, namely the:

- High Burnup Effect Programme
- Risoe Fission Gas Release Projects
- WWER Data
- Halden Project Instrumented Fuel Assemblies
- Belgonucléaire Tribulation Project
- Studsvik Ramp Test Projects
- CEA/EDF/Framatome Data
- Siemens PWR Rods Irradiated in the R.E. Ginna Reactor
- IMC Out-of-Pile Annealing Data.

Further release of experimental data on fuel behaviour has been recognised both as a useful and necessary effort contributing to model improvement and evaluation at international level.

The following issues (or problems) encountered in compiling the experimental data into the standard format were identified:

- Calculation of local clad temperature (often this information must be inferred from other data in the reports)
- Powers: "thermal" or "total" (thermal + fast neutrons + gammas)
- Geometry: which way is "up" and which way is "down"
- Reproduction of PIE data:
  - Should data be digitized, e.g. diameters, oxide thickness, etc
  - Best way of presenting metallography (optical)
- Pre-characterisation:
  - Pellet: impurity content?
  - Cladding: composition, impurities, mechanical properties

It was also noted that there is no standard way of characterising mechanical properties of materials.

Participants insisted that it is important that the information stored is complete, that the irradiation history and all other available information is present in the compilation, e.g. data on densification, porosity size distribution.

The ensuing discussion 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 are 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

A paper describing the status of IFPE was presented at the *ANS 2000 International Topical Meeting on Light Water Reactor Fuel Performance*, Park City, Utah USA, 10-13 April 2000,entitled "The Public Domain Database on Nuclear Fuel Performance Experiments (IFPE) for the Purpose of Code Development and Validation", by P. Menut, E. Sartori, J.A. Turnbull. A copy was distributed to participants. Additional information on the present status can be found by accessing on Internet: http://www.nea.fr/html/science/fuel/ as well as copies of reports produced by the Group.

4. Presentation and experience with the CRACO database (P. Obry, CEA)

CEA has developed a relational database (CRACO) that ensures a centralised access to PWR fuel rod behaviour data. Data are checked for quality, and coupling with modelling codes (e.g. METEOR) for their assessment and validation is implemented. This is done in co-operation with EDF and Framatome with the aim of simplifying the transfer of information and shared code benchmarking. The system includes a pre-processor (XMETEOR) for automatically collecting manufacturing and irradiation history data for selected rods, to propose default values when experimental data is not available and to prepare input files for METEOR. A module XSQL extracts information from the database for presentation in tables and graphs.

This project was initiated ten years ago. At present one upgrade per year is made, for sharing with the other partners. The choice of a relational database (RDB) was dictated by the need for structured, reliable, coherent data, and shareable amongst users. The modelling in addition dictates data relations and constraints which are easily implemented in an RDB. The system chosen is ORACLE for its efficiency, reliability, flexibility and portability. The broad structure of the data is organised in manufacturing data, history of irradiation, and examinations, each with its substructure such as pellet, cladding, rods and components for manufacturing or rod, assembly or reactor irradiation history, etc. A user friendly tool (C3) for capturing, modifying and browsing/reporting information was developed that includes a Q.A. loading and verification process. At the time of the presentation CRACO contained data for 200 PWR fuel rods comprising monitoring programmes (UO_2, MOX mimas and COCA, and UO_2 GdO_3) and experimental programmes (power ramps at Siloe, Osiris, R2) and specific instrumented irradiations (Siloe, Halden). Future work foresees extension of the database to results from mechanical testing of fuel pellets, cladding corrosion and inclusion of images. Also the coupling of CRACO to fuel manufacturing and mechanical cladding test databases is planned. In conclusion the experience of using a relational database running under ORACLE has been very positive and rewarding, only the loading process is a rather time-consuming one.

Participants commented that CRACO focusses on PWR rods and its design is adapted mainly to the needs of METEOR. IFPE covers a wider range of rods and is used in connection with many different codes (more than 50 copies distributed to 20 countries). It may thus be difficult to place the data into a fine structured database. The flexibility of CRACO could, however, be explored in a first step to better identify difficulties and efforts required.

5. Experiments Planned/in Progress/Further Needs - Discussion

PHWR MOX Fuel Experiments and the IFPE (A. C. Marino)

The irradiation of the first Argentine prototypes of PHWR MOX fuels began in 1986. These experiences were made in the HFR-Petten reactor, the Netherlands. The rods were prepared and controlled in the CNEA’s α Facility. The post-irradiation examinations were performed in the Kernforschungs-zentrum, Karlsruhe, Germany, and in the JRC, Petten. The first rod has been used for destructive pre-irradiation analysis, and the second one as a pathfinder to adjust systems in the HFR. Two additional rods including iodine doped pellets were intended to simulate 15 MWd/kg(M) burnup. The remaining two rods were irradiated up to 15 MWd/kg(M) (BU15 experience). One of them underwent a final ramp with the aim of verifying fabrication processes and studying the behaviour under power transients. The experiments were later included in the IFPE (International Fuel Performance Experiments Data Base). An overview of the CNEA’s PHWR MOX irradiation was provided.

There are enough reasons to complete the IFPE database with LMFBR fuel experiments (M.C. Paraschiv)

The requirements in experimental data for modelling new generation LWR fuels go beyond what is presently available in the IFPE database. The burnup extension has an important economic effect and advanced light water reactors using MOX fuels would be developed at ultra-high burnup possibly reaching 100 MWd/kg. The advanced cladding material required for ultra-high burnup fuels and/or fast neutron spectrum reactor could be selected from representative heat resistant materials with thermal neutron absorption cross-section lower than conventional cladding materials. The micro-structural change by heavy neutron irradiation, the resistance against corrosion, ductility loss and creep would then be considered to be more important than neutron economy. Major technological problems expected in advanced fuel and cladding materials would need to be addressed. In the light of this, the completion of IFPE data base with new information offered by experiencing the LMFBR fuels becomes an important objective. Some compiled information about the LMFBR fuel behavior has already been put together in connection with the standard model on fission gas release ANS5.4. In order to improve predictions of oxide fuel behavior, special attention has been given recently to the mechanisms of volatile fission products release (VFPR). This is strongly correlated to the chemical and micro-structural changes of the oxide fuel. The generation rates of FP differ from conventional calculation schemes. This means that a new strategy in modelling of VFPR is necessary.

Compiling fast reactor fuel behaviour data would benefit IFPE. Good sources for such data would be Dounreay and Phenix. It represents a vast amount of data and effort.

6. *User Experience - Code Validation Matrix*

Validation of TRANSURANUS for VVER fuels (K. Lassmann)

The IFPE database has provided a most important contribution to modelling in recent years. It has played an important role in the work at ITU for the development of the TRANSURANUS VVER version and its verification within the PECO project. The Sofit and Kola data from IFPE were used. Feedback has helped to obtain an improved version of the Kola data. Open issues concern cladding creep and swelling, transient experiments. Future work on IFPE should concentrate on release (clarification of procedures, better indication of what is new, what has changed or improved), compiled data should be checked by the originators, more information on the thermal-hydraulics should be available. It is important that the user can clarify problems encountered, how error reporting and correction procedures work. It is suggested that an Internet forum is established for this purpose. Questions to be addressed concern the format: can it be made more user friendly? The data should be extended to include transients and possibly FBR data. One feedback was provided on the Studsvik Superramp experiment. For D 171, an error was detected in reporting the grain size. It was also found that the important information of the plenum volume is missing.

C. Beyer from Battelle NW Lab reported on the use made of high burnup data from IFPE (RISOE-III, and power ramp data) for improvements in the FRAPCON-6 code. The results are very satisfactory. New data will be used now, in particular those from Siemens PWR rod irradiations in Ginna. The IFPE data have been very useful for US NRC code development.

The following discussion clarified that all changes are recorded in the QA report which ensures traceability; it may be useful though that a separate indication of changes is made up front in the compilation for easier reference and checking. The QA reports also document all manipulation of data by including the actual coding used, the calculated or inferred temperatures, etc. It was agreed that version control of the data should be enforced.

It was agreed to establish an Internet forum: the address for queries and replies would be ifpe@nea.fr.

7. *The FUMEX II exercise*

The new proposed IAEA CRP FUMEX II (P. Menut)

P. Menut recalled first the results achieved in the FUMEX-I (Fuel Modelling at Extended Burnup) exercise from 1993-1996. Nineteen codes from 15 countries had participated using experimental data with 6 cases provided by the OECD Halden Reactor Project focussing mainly on consequences of extended burnup on fuel operation. The final results were published in IAEA-TECDOC-998 in January 1998: “Report of the Co-ordinated Research Programme on Fuel Modelling at Extended Burnup - FUMEX”. This exercise involved 3 areas for the simulation: thermal conductivity, fission gas release, and mechanical interaction. The highlights on ***thermal conductivity*** concerned results on gap thermal conductance, fuel densification, fuel swelling. Improvements were achieved in the formulation of the UO_2 fuel thermal conductivity, the influence of the RIM structure, formulation of a reversible relocation, dependence of surface roughness on burnup. However, further improvements are required. Concerning ***fission gas release*** results were compared among codes carrying out the modelling on different operating regimes such as: low power, low temperature, high fractional release at high temperature, diffusion rate controlled phenomena. Further improvements would concern the functional dependence of critical parameters. As concerns the ***mechanical behaviour*** it was noted that the differences in the codes are still important especially for complex systems. Also the interdependence with thermal performance was confirmed. Improvements should concern the treatment of structurally weak cracked pellets. A unified thermal and mechanical modelling would also be necessary.

In view of the success of FUMEX I, it was proposed to start a new IAEA Co-ordinated Research Project FUMEX II in 2002, using a common fuel experiment database. It would aim at improving models and codes used both for Western and Eastern fuel design, especially to predict the fuel behaviour for high burn up. It is planned to hold a Co-ordinated Research Programme in 2001. A questionnaire was prepared and sent out to specialists to identify the key issues to be addressed. Experiments will be identified to form the basis for this exercise and which would later be integrated into the IFPE database. In order to review relevant topics, a Consultancy Meeting will be organised in 2001 in Vienna.

M. Noe: New CRP on Fuel Modelling at High Burnup (Status)

This exercise would continue along the lines of past efforts initiated by the IAEA such as D-COM (1982-84) and FUMEX-I (1993-96). A first step would imply the search for available and suitable experimental data for a new exercise FUMEX II. The topics needed for an international effort for model improvements identified as identified through a questionnaire are:

- fission gas release at high burnup
- progressing in the area of mechanical interaction
- thermal performance
- mathematical models

The proposal made by CEA includes three parts:

1. Two cycle rod power transient/short holding times.
   This would principally address the inelastic contribution to fuel and clad swelling - a pure thermal effect in PMCI. The samples would consist of 3 segmented rods (24MWd/kg) from the French 900 MW PWRs. The rods have undergone four short power ramps in OSIRIS (0, 16 minutes, 12 hours) and the ramp terminal level ranged between 400 and 450 W/cm. They have undergone both a non-destructive and a destructive examination before and after the transient. Other examinations concern diameter variation profiles, primary/ secondary ridges and dish filling.

2. Three/Four cycle rod power transient/long hold times.
   Instantaneous and cumulated fission gas release rates and fission gas related swelling are addessed. The first experiment involves a three cycle rod at 33MWd/kg burnup and the second a four cycle rods at 46 MWd/kg. The refabricated rods are ramped in the SILOE reactor. Each rod is subject to 10 power ramps in the range 200-340 W/cm. On-line measurements for various radioactive gas isotopes, global Xe and Kr release, cladding diameter variations and ceramography are carried out.

3. Four cycle rod power transient/long holding times.
   The possibility is being debated with other partners of the release of results from an experimental programme featuring four cycle rod ramping, including extensive in-pile measurements, and illustrating explicitly the effects of fission gas related fuel swelling on cladding strains.

These exercises include a precise pre- and post-test pellet + cladding characterisation. The data would be released to the IFPE database, in relation with the use of the CRACO database.

HRP Data for FUMEX-II (W. Wiesenack)

HRP proposes that FUMEX-II focus on Fission Gas Release. In FUMEX-I the predictions of FGR were of varying quality as concerns the onset of significant FGR, the amount of FGR, and short and long-term kinetics. The new exercise should address release at high burnup and look at different microstructures and fuel type.

HRP could offer in-core measurements of rod pressure (FGR), fuel centerline 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. Initial enrichment would be 3.35%.

Possible other experiments that could be made available concern the variation of fuel microstructure (grain size), and possibly sweep gas experiments.

It is recommended that the exercise be a blind test and that only the power history is provided. A measured temperature history should be specified to eliminate uncertainty of temperature calculation and to concentrate on quality of fission gas release models. As in FUMEX-I artificial cases should be included.

The possibility of including a CANDU case was mentioned (M. Tayal)

Participants in the meeting were very supportive of the FUMEX-II proposal. The advantage of bringing the IFPE interpretations into FUMEX-II was recognised especially if the exercise is not carried out blind. If the exercise is not blind, new data released for FUMEX-II should be prepared in the IFPE standard format. The FUMEX-II would provide an excellent mechanism to provide feedback for the IFPE. Thus, following the offer of data from experiments by CEA and HRP, it was agreed that further progress should be achieved by the next IAEA TWGFPT meeting, scheduled for April 2001 in Vienna. The work programme should be established commencing in 2002 and would be carried out over a period of four years. OECD/NEA would co-operate with IAEA in FUMEX-II.

8. *Release of New Data*

K. Malén: New data from Studsvik for IFPE

The IFPE already incorporates the following **Studsvik** ramp tests: Inter-Ramp, Over-Ramp, Demo-Ramp I, II and Super-Ramp. New data are suggested for inclusion from the ramp tests: **Trans - Ramp I, II, IV**. This has been approved both by Studsvik and by SKI (Swedish Nuclear Power Inspectorate). The project results have previously been presented at conferences. The data will be released under the same conditions as the previous ones. The summary characteristics are as follows:

- Trans Ramp I (1982-1984) concerns BWR rods, having been subjected to fast power ramps of about 10000 W/cm,min and to powers exceeding the PCI threshold (around 50 kW/m)
- Trans Ramp II (1984-1986) concerns PWR rods, that have been subjected to fast power ramps of about 1000 W/cm,min and to powers exceeding the PCI threshold (up to 60 kW/m)
- Trans Ramp IV (1989-1993) concerns PWR rods showing incipient crack formation by ramp, subject to irradiation (BOCA) and re-ramp to the PCI failure threshold?

Belgonucléaire agreed to offer data from the **GAIN** programme. The data cover the fabrication, irradiation and PIE of 4 ramped gadolinium doped (UO_2 - Gd_2O_3) fuel rods having the following characteristics:

- *fuel rod type*: UO2 ex-AUC
- *U-235*: 3.5%
- *Gd_2O_3*: 3 or 7 % (2 rods of each type)
- *cladding*: Zr4
- *irradiation*: BR3 PWR up to 50MWd/kgU (peak pellet)
- *PIE*: NDT + DT (puncture, burnup, ceramography, EPMA/depending on rod)
- *ramp*: fast ramp at mid life followed by re-irradiation in BR3. NDT before and after ramp
- *tentative date of release*: end of May 2001

C. Gheorghiu, M.C. Paraschiv: In-Reactor Measurement of Internal Gas Pressure at INR Pitesti

The purpose of these experiments was to provide quantitative information on the effect of UO_2 density on internal gas pressure at a heat rating of technological importance. Two experimental fuel elements i.e. **RO-51 and RO-89** were irradiated at element power of approximately 520-590 W/cm. The range of densities selected covered those of present technological interest, 10.54 (for experimental fuel element RO-89) to 10.70 g/cm^3 (for experimental fuel element RO-51). Average burnups of 159 MWh/kgU for RO-51 and 125 MWh/kgU for RO-89 were achieved. The elements were irradiated in a capsule placed in the Triga MTR-Pitesti. The fuel element RO-51 was irradiated at a linear power in the range of 400 – 540 W/cm and the fuel element RO-89 was irradiated at a linear power in the range of 515 – 590 W/cm. The internal gas pressure was measured, during irradiation by a pressure transducer. In addition, the capsule was equipped with four self-powered detectors to monitor neutron flux. Post irradiation examinations were performed for both experimental fuel elements. The results will be submitted with details to IFPE including the analysis and the interpretation of fission gas pressure evolution during irradiation.

Other Data

Other valuable data for inclusion or possible release were identified as follows:

IFA-508 and **IFA-515** conducted by JAERI at HRP, and concerning PCMI behaviour on different cladding thickness by means of diameter rig. These tests were performed between 1977 and 1981, as IFA508.1, 508.2, 508.3, 508.4 and IFA-515.1, 515.2, 515.3. The main results were published in the following papers:

- M.Uchida and M.Ichikawa: "In-pile diameter measurement of Light Water Reactor test fuel rods for assessment of pellet-cladding mechanical interaction", J. Nucl. Technol.51(1980) 33-44.
- K.Yanagisawa: "An evaluation of the influence of fuel design parameters and burnup on pellet/cladding interaction for boiling water reactor fuel rod through in-core diameter measurement", J. Nucl. Technol. 73(1986) 361-377.

These papers contain the results of 8 rods (11,12,13,14,18,29,32,21). The main results cover diameter measurements during ramp tests. The original diameter measurement data, power histories, and other instrumentation data are stored at HRP.

K. Kamimura of **NUPEC** informed the OECD/NEA of current discussions going on concerning release of irradiation experiments data owned by NUPEC.

The possibility of obtaining swelling data obtained when ramping CAGR UO_2 fuel in Halden RP are investigated at **IMC, UK**.(M. El-Shanawany, R. White)

V. Smirnov from **NIIAR**, Dimitrovgrad was contacted to investigate the possible release of fission gas release data from high burnup VVER-440 fuel under steady-state and transient operation conditions.

Participants agreed that more data from VVER fuels would be needed to fill existing gaps: in particular of interest are the **VNIINM** ramp data from WWER-440 up to 4 cycles and WWER-1000 up to 3 cycles

IFPE

Other areas that would fill existing gaps are data from laser flash measurements, some of which seems already to have been condensed, high temperature transients (also here the risk of data loss is high and should be addressed).

The Chairman recalled that the database at HRP is large: in fact there are about 1000 instrumented rods. Experiments concern:

- Mainly temperatures. BOL data should not be difficult to obtain
- Some PCMI data: gap size, type of fuels
- Data on UO_2 additives and MOX

However, information cannot be released before 5 years. In addition the extraction of the data for integration in IFPE requires the allocation of manpower.

9. Reporting to the Seminar on 'Fission Gas Behaviour in Water Reactor Fuels'

Participants found it important that the results of the meeting be presented briefly at the seminar on Fission Gas Behaviour to be held from 26 - 29 September at the same premises, the objective being to make known the work on the IFPE and to incite the release of further data to the project.

This seminar was co-organised by CEA, OECD/NEA and IAEA and sponsored by EDF, COGEMA and Framatome. 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 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 and 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** in spring 2002, also at Cadarache.

10. *Conclusions and Recommendations, report to the NEA Nuclear Science Committee*

- Progress since the last meeting was reviewed: the number of cases has doubled since then and reached 387 at the time of the meeting
- Comprehensiveness has been improved, but some gaps have been identified and requirements for new data discussed (see table under item 3)
- The main scope will continue on water moderator reactor fuels, but extensions are being considered, in particular to CAGR and FR fuels, laser flash diffusion data, high temperature transient, etc.
- Format of data will be kept simple with some improvement to enable 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 (more than 50 recipients) for discussing and exchanging information on specific topics and/or experiments and difficulties, reporting feedback on IFPE use, and improving/enforcing gathering feedback from users, for which a questionnaire should be prepared.
- Further data being released or offered to IFPE were 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 were recommended (once/year). A 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 the NSC and emphasise that strong support is needed, that financial backing should be made available, and that the scope should be extended in the future to include other types of data.

11. *Date and Place of Next Meeting*

Participants agreed that there is a need for more frequent meetings (one per year) to ensure that feedback from IFPE users is obtained, an essential aspect that contributes to the improvement of IFPE. Where possible this meeting should be held in conjunction with other meetings dealing with fuel behaviour issues.

Annex 1

List of participants

Fuel Performance (IFPE) Meeting, Cadarache, 25 September 2000

ARGENTINA

MARINO, Armando Carlos
CNEA
Centro Atomico Bariloche
8400 San Carlo de BARILOCHE
Rio Negro
Tel: +54 2944 445 256
Fax: +54 2944 445 178
Eml: marino@cab.cnea.gov.ar

BELGIUM

* LIPPENS, Marc
Belgonucleaire
Avenue Ariane, 4
B-1200, Brussels
Tel: +32 2 774 0625
Fax: +32 2 774 0614
Eml: m.lippens@belgonucleaire.be

VAN UFFELEN, Paul
SCK/CEN
Fuel Research Unit
Reactor Materials Research Department
200 Boeretang
B-2400 MOL
Tel: +32 (14) 33 22 07
Fax: +32 (14) 32 15 29
Eml: pvuffele@sckcen.be

BULGARIA

DJOURELOV, Nikolay
Institute for Nuclear Research
72 Tzarigradsko Shosse
1784 SOFIA
Tel:
Fax:
Eml: nikdjour@inrne.bas.bg

CANADA

LEWIS, Brent J.
Department of Chemistry and
Chemical Engineering
Royal Military College
KINGSTON Ontario K7K 5L0
Tel:
Fax:
Eml: lewis-b@rmc.ca

TAYAL, Mukesh
Atomic Energy of Canada Limited, SP2F4
Sheridan Park Research Community
2251 Speakman Drive
Mississauga Ontario L5K 1B2
Tel: +1 905 823 9040, X 4652
Fax: +1 905 822 0567
Eml: tayalm@aecl.ca

CZECH REPUBLIC

* NOVOTNY, Nichal
Nuclear Research Institute
Dept. of Reactor Technology
Nuclear Safety & Power Division
25068 - REZ U PRAHY
Tel:
Fax:
Eml:

* VALACH, Mojmir
Nuclear Research Institute Rez, plc.
Dept. of Reactor Technology
25068 - REZ U PRAHY
Tel: +420 (0)2 20 941 114
Fax: +420 (0)2 20 940 960
Eml: val@nri.cz

FINLAND

KELPPE, Seppo
Techn. Res. Centre of Finland
VTT Energy
P.O. Box 1604
FIN-02044 VTT
Tel: +358 9 456 5026/9 40 5899932 M
Fax: +358 9 456 5000
Eml: seppo.kelppe@vtt.fi

FRANCE

AGARD, Mireille
DRN/DEC/SESC/LSC
CEN Cadarache
Bat. 315
F-13108 St. Paul-lez Durance Cedex
Tel:
Fax: +33 4 42 25 48 78
Eml: agard@drncad.cea.fr

BLANPAIN, Patrick
FRAMATOME Nuclear Fuel
Fuel Technologies
Design and Sales Division
10, rue Juliette Recamier
F-69456 LYON Cedex 06
Tel: +33 4 7274 8927
Fax: +33 4 7274 8145
Eml: pblanpain@framatome.fr

BOUSQUET, Regine
DRN/DEC/SDC
Bat. 315
CEA Cadarache
F-13108 St. Paul-lez Durance Cedex
Tel: +33 (0)4 4225 3862
Fax: +33 (0)4 4225 2949
Eml: bousquet@michigan.cad.cea.fr

NOE, Maxy
CEA DRN/DEC/SEC
CE Cadarache
Bat. 315
13108 St-Paul-lez-Durance
CEDEX
Tel: +33 4 4225 3924
Fax: +33 4 4225 2949
Eml: NOE@DRNCAD.CEA.FR

OBRY, Patrick
DRN/DEC/SESC/LSC
CEN Cadarache
Bat. 315
F-13108 St. Paul-lez Durance Cedex
Tel: +33 4 42 25 73 34
Fax: +33 4 42 25 48 78
Eml: obry@drncad.cea.fr

* PAGES, Jean-Paul
DRN/P
CEN-Cadarache
Bat. 707
F-13108 St. Paul-lez Durance Cedex
Tel: +33 4 42 25 72 28
Fax: +33 4 42 25 48 47
Eml: jean-paul.pages@cea.fr

GERMANY

LANDSKRON, Horst
Siemens AG
KWU NBTM
P.O. Box 3220
D-91050 Erlangen
Tel: +49 9131 18 84532
Fax: +49 9131 18 4799
Eml: Horst.Landskron@erl19.siemens.de

INDIA

ANANTHARAMAN, Sundaresan
Bhabha Atomic Research Centre
Radiometallurgy Division
Radiological laboratories
400 085 MUMBAI
Tel: ++91 (22) 550 5050 X 2447
Fax: ++91 (22) 550 5151
Eml: saraman@apsara.barc.ernet.in

KOREA (REPUBLIC OF)

SONG, Kee-Chan
Korea Atomic Energy Research Institute
Backend Fuel Cycle Research Group
P.O. Box 105, Yusung
Taejon 305-600
Tel: +82 42 868 2352
Fax: +82 42 868 8370
Eml: kcsong@nanum.kaeri.re.kr

MEXICO

HERNANDEZ-LOPEZ, Hector
Instituto Nacional de
Investigaciones Nucleares
Carretera Mexico-Toluca
km. 36.5
Tel: +52 (5) 518 2360 EXT. 431
Fax: +52 (5) 521 9408
Eml: hhl@nuclear.inin.mx

Salazar, MEXICO 52045

P.R. OF CHINA

SHEN, Lixin Deputy Director China national Nuclear Corporation 1, Nansanxiang, Aanlihe, Xicheng POB 2102-12 100822 BEIJING	Tel: ++86 10 6853 1904 Fax: ++86 10 6852 9603 Eml: shenlx@cnnc.com.cn
present address: C/o DRN CEN Cadarache Bat. 707 F-13108 St. Paul-lez Durance Cedex	 Fax: +33 4 4225 3781 Eml: shen@drnmail.cad.cea.fr
ZHANG, Zhongyue China Institute of Atomic Energy P.O. Box 275(64) 102413 BEIJING	Tel: ++86 (10) 935 7725 Fax: ++86 (10) 935 7008 Eml: wuzu10@mipsa.ciae.ac.cn

RUMANIA

* GHEORGHIU, Constantin Institute of Nuclear Research POB 78 0300 PITESTI	Tel: ++40 482 62599 Fax: ++40 482 62449 Eml: dirsticn@cyber.ro
PARASCHIV, Marius Institute for Nuclear Research P.O. Box 78 03000 Pitesti	Tel: Fax: Eml: mariusp@cc.nuclear.ro

RUSSIAN FEDERATION

SMIRNOV, Aleksei V. Head of Laboratory, Deputy Head of Department, Research Institute of Atomic Reactors NIIAR 433510, Dimitrovgrad, Uljanovsk region	Tel: +7 84235 32021 Fax: +7 84235 35648 Eml: gns@niiar.simbirsk.su

SPAIN

GARCIA-CUESTA, Juan Carlos CIEMAT Avda Complutense 22 28040 MADRID	Tel: +34 91 346 6233 Fax: +34 91 346 6005 Eml: jc.garcia@ciemat.es

SWEDEN

MALEN, Karl Studsvik Nuclear AB SE-611 82 NYKOEPING	Tel: +46 (155) 221 670 Fax: +46 (155) 263 156 Eml: karl.malen@studsvik.se

SWITZERLAND

NORDSTRÖM, Lars Ake Paul Scherrer Institute CH-5232 Villigen PSI	Tel: +41 56 310 2737 Fax: +41 56 310 2205 Eml: ake.nordstroem@psi.ch

UNITED KINGDOM

PALMER, Ian Manager, Fuel Modelling LWR Business Unit BNFL Int. Group B709 SPRINGFIELDS, Preston Lancashire PR4 0XJ	Tel: +44 1772 76 4113 Fax: +44 1772 76 2470 Eml: ian.d.palmer@bnfl.com

TURNBULL, James Anthony
Nuclear Fuel and Materials Specialist
Cherry-Lyn
The Green
Tockington, Bristol BS32 4NJ
Tel: +44 (1454) 620 887
Fax: +44 (1454) 620 887
Eml: tonyt@cherry-lyn.demon.co.uk

UNITED STATES OF AMERICA

BEYER, Carl E.
Pacific Northwest National Laboratory
Battelle Boulevard
P.O. Box 999
Richland, WA 99352
Tel: +1 (509) 372 4605
Fax: +1 (509) 372 4439
Eml: carl.beyer@pnl.gov

BILLAUX, Michel R.
Siemens Power Corporation
Nuclear Division
2101 Horn Rapids Road
P.O.Box 130
RICHLAND, WA 99352-0130
Tel: +1 (509) 375 8130
Fax: +1 (509) 375 8036
Eml: Michel_Billaux@nfuel.com

International Organisations

CEC/ITU

LASSMANN, Klaus
Institute for Transuranium Elements
CEC Joint Research Centre
Postfach 2340
D-76125 KARLSRUHE
Tel: +49 (7247) 951297
Fax: +49 (7247) 951379/593
Eml: k.lassmann@itu.fzk.de

SCHUBERT, Arndt
European Commission
Institut for Transuranium Elements
Postfach 2340
D-76125 KARLSRUHE
Tel: +49 (7247) 951 289
Fax: +49 (7247) 951 595
Eml: schubert@itu.fzk.de

IAEA

MENUT, Patrick
Nuclear Fuel Cycle and Waste Management, IAEA
Wagramerstrasse 5
P. O. Box 100
A-1400 VIENNE
Tel: +43 (0)1 2600 22763
Fax: +43 (0)1 2600 7 22763
Eml: P.Menut@IAEA.ORG

OECD/IFE/HRP

WIESENACK, Wolfgang   Chairman
Project Manager
OECD Halden Reactor Project
Institutt for Energiteknikk
Os Alle 13 - P.O. Box 173
N-1751 HALDEN
Tel: +47 (69) 21 2347
Fax: +47 (69) 21 2201
Eml: wolfgang.wiesenack@hrp.no

OECD/NEA

SARTORI, Enrico   Secretary
OECD/NEA Data Bank
Le Seine-Saint Germain
12 boulevard des Iles
F-92130 ISSY-LES-MOULINEAUX
Tel: +33 1 45 24 10 72 / 78
Fax: +33 1 45 24 11 10 / 28
Eml: sartori@nea.fr

* regret not to have been able to attend.

Annex 2

Expert Group Meeting on

Scientific Issues in Fuel Behaviour

- International Fuel Performance Experiments Data Base (IFPE) -

(Status and Programme)

Cadarache, 25 September 2000

Agenda

1. Opening (W. Wiesenack - Chairman)
   - Introduction of Participants

2. Objectives of the Meeting and of Expert Group (W. Wiesenack, E. Sartori)
   - Approval of Agenda

3. Status of the IFPE Data Base (J.A. Turnbull, E. Sartori)
   J A Turnbull (Consultant), P Menut (IAEA) and E Sartori (NEA): A Review of Fission Gas Release Data within the NEA/IAEA IFPE Database
   E. Sartori: Nuclear Fuel Behaviour Activities at OECD/NEA
   - Compiled Data
   - Format evaluation
   - Peer Review and QA reports
   - Data Being Released
   - Comprehensiveness

4. *Presentation and experience with CRACO database (P. Obry, CEA)*

5. Experiments Planned/in Progress/Further Needs - Discussion
   - *C. Marino PHWR MOX Fuel Experiments and the IFPE*
   - *M.C. Paraschiv: There are enough reasons to complete the IFPE database with LMFBR fuel experiments*

6. User Experience - Code Validation Matrix
   - *Contribution to the IFPE and Validation of TRANSURANUS for VVER fuels (K. Lassmann)*
   - Validation and use with other codes

7. The FUMEX-II exercise
   - *Patrick Menut: Information about the new proposed IAEA CRP FUMEX II*
   - *M. Noé: New CRP on Fuel Modelling at High Burnup (Status)* Proposal from CEA
   - *W. Wiesenack: HRP Data for FUMEX-II*
   - other proposals

8. Release of New Data
   - *Karl Malén: New Data from Studsvik for IFPE*
   - Belgonucléaire
   - NUPEC
   - HRP
   - *C. Gheorghiu, M.C. Paraschiv: In-Reactor Measurement of Internal Gas Pressure at INR Pitesti*

9. Reporting to Seminar on ‘Fission Gas Behaviour in Water Reactor Fuels’

10. Conclusions and Recommendations, report to NEA Nuclear Science Committee

11. Date and Place of Next Meeting - Closing

Annex 3

OECD/NEA Nuclear Science Committee (NSC)
Expert Group on Scientific Issues of Fuel Behaviour
(TFSFB)

SCOPE and OBJECTIVES

- **Scope**

  The Expert Group deals with the status and trends of scientific issues of fuel behaviour.

- **Objectives**

  - Compile high quality experiments for the International Fuel Performance Experiments (IFPE) database. Emphasis is given to high burn-up in water reactors under normal operating conditions. Priority goes to completed international programmes, the data of which would otherwise be lost, but also to data released from national programmes.

  - Co-ordinate the qualification of these data through review and by organising user group meetings. Take initiatives so that it may be adopted as an international standard.

  - Co-ordinate computer code validation and benchmark studies.

- **Work Plan**

  - Organise co-ordination meetings combined with IFPE database user group meetings every 12-18 months.

  - The IFPE database will be consolidated during the next two years through updating and addition of new experiments.

  - Seminars and Experts' meetings will be held as needed to address high priority issues for validating the modelling of phenomena of particular concern (thermal performance, fission gas behaviour, pellet-clad mechanical interaction, etc.)

- **Co-operation**

  This activity will be carried out in co-ordination with the IAEA, the NEA/CSNI Special Expert Group on Fuel Safety Margins and the OECD/IFE/ Halden Reactor Project.

- **Duration:** two years

  Revision approved by NEA NSC: 4 June 1998
  Mandate extended for two years on 9 June 2000