NUCLEAR FUEL BEHAVIOUR ACTIVITIES

at the OECD/NEA

(May 2000)

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Abstract

The work programme in the field of fuel behaviour carried out at the OECD/Nuclear Energy Agency is described in addition to plans for future work.

It covers the following aspects:

- Scientific Issues in Fuel Behaviour
  - International Fuel Performance Experiments (IFPE) database
- Computer Program Service
- Fuel Safety Criteria and Margins
  - fuel related safety parameters and / or phenomena in transient and accident conditions

Scientific Issues in Fuel Behaviour - International Fuel Performance Experiments Database

A series of three Seminars concerned with the state of the art of modelling some basic phenomena in fuel behaviour was defined already in 1997. The proceedings of the Seminar on **Thermal Performance of High Burnup LWR Fuel** held at Cadarache in 1998 organised by CEA, OECD/NEA and IAEA and sponsored by EDF, COGEMA and Framatome was published and is available. A second Seminar will be devoted to **Fission Gas Behaviour of Water Reactor Fuel** with the same sponsorship. It will be held from 26-29 September 2000 also at Cadarache. The technical programme committee has met on 26 April in Paris to review the abstracts submitted and to shape a preliminary programme. About 30 presentations have been proposed. A third seminar is planned on **pellet clad mechanical interaction**. It is scheduled for 2002, also at Cadarache.

The meeting of the **NSC Task Force on Scientific issues in Fuel Behaviour** will be held in conjunction with the Cadarache Seminar on 25 September 2000. It will be a joint OECD/NEA - IAEA meeting. The main objective is to review the status of the International Fuel Performance Experiments (IFPE) database, to report feedback from its utilisation and to propose the release and compilation of further data to improve comprehensiveness. The scope and objectives of this task force are enclosed as Annex 1. This text will be reviewed by the NEA Nuclear Science Committee during its June 2000 meeting. Annex 2 describes the IFPE project and the status of the data it includes or data in process of being included. Requirements for including data in the IFPE Database are described in Annex 3.

The role of the NEA Data Bank in this context consists of carrying out or arranging of data compilations from experiments, of operating the database, including management, updating, integrating of feedback, distributing, and electronic scanning including **optical character recognition** of all archive reports to be stored in the database.

The contribution of the IAEA has been essentially that of arranging the release of data from the non-OECD area and to provide assistance to experts in these countries. In this context and within technical co-operation programmes, several data sets have been used for code validation and for providing feedback for the database with the objective of removing some inconsistencies or deficiencies. Several compilations have also been arranged by the IAEA, in particular within the framework of training scientists in interpreting experiments and in QA procedures.

The database has been sent on request to more than 50 establishments in 21 countries who are members of both IAEA and OECD/NEA: Argentina, Bulgaria, Belgium, Canada, Czech Republic, Finland, France, Germany, Hungary, Japan, Korea, Mexico, Norway, Poland, Russian Federation, Rumania, Slovak Republic, Sweden, Turkey, United Kingdom and United States of America.

Computer Codes

The NEA Data Bank acquires, tests and distributes on behalf of its member countries and through a cooperative agreement with the IAEA computer codes encompassing nuclear technology applications. It includes also computer codes for fuel behaviour modelling. Codes today distributes through the Computer Program Services are provided as Annex 4.

Fuel Safety Criteria and Margins

The concern whether with the advent of advanced fuel and core designs and operating modes, safety criteria margins remain adequate was expressed. The CSNI Task Force on Fuel Safety Criteria and Margins was then set up with the mandate to identify and address fuel safety issues connected with “new” design elements and operating modes.

The ***working approach*** adopted is:

- Select fuel safety related criteria
- Identify “new” design elements and operating modes having potential impact on fuel safety criteria
- Assess adequacy of each safety related fuel criterion vs. new design elements and operational modes
- Identify effort required to ensure adequate basis for criteria, or the need to revise criteria

The ***technical review*** includes:

- CPR/DNBR
- Reactivity coefficient
- Shutdown margin
- Enrichment
- Crud deposition
- Strain level
- Oxidation and Hydriding
- Internal gas pressure
- Thermal mechanical loads, PCMI
- PCI
- Fuel fragmentation (RIA)
- Fuel failure (RIA)
- Cladding embrittlement / PCT (non-LOCA run away oxidation
- Cladding embrittlement / oxidation (LOCA)
- Blowdown / seismic loads
- Assembly holddown force
- Coolant activity
- Gap activity
- Source term

The ***"New” Design Elements and Operating Modes*** identified are:

- New fuel designs
- New core designs
- New cladding materials
- New manufacturing procedures
- Long fuel cycle
- Uprated power
- High burnup
- MOX
- Mixed core
- Water chemistry changes
- Current / new operating practices

The **main conclusions of the CSNI Report on *Fuel Safety Criteria Technical Review*** are:

- The basis for current fuel and core operation, with the use of extrapolated safety criteria in the transient/accident area, is still justifiable for fuel/core designs currently licensed.
- It is considered that the current framework of fuel safety criteria remains generally applicable, being largely unaffected by the 'new' or modern design elements; the levels (numbers) of the individual safety criteria may, however, change in accordance with the particular fuel and core design features.
- Some of these levels have already been - or are continuously being – adjusted; level adjustments of several other criteria (RIA, LOCA) also appears to be needed, on the basis of experimental data and the analysis thereof. Only in a few cases (e.g. oxidation, hydriding, crud deposition) the Task Force is not convinced that having a safety criterion is highly effective; a criterion such as fracture toughness, which really pertains to the phenomenon that causes the fuel cladding to fail, would be much more appropriate.
- In view of the further improvements in design, notably the pursuit of higher burnups, fuel safety criteria should continue to be (re)assessed with the support of experimental research. Also the analysis methods and modeling needs further improvement, with suitable benchmarking against experimental data. Therefore a well selected number of experiments should be carried out; with good and well benchmarked models the amount of tests may eventually be reduced.

Special Expert Group on Fuel Safety Margins (SEG - FSM)

In accordance with the *CSNI Strategic Plan* a SEG FSM will be established The current Task Force on Fuel Safety Margins will form the nucleus of this Group. It will address cross cutting issues related to fuel behaviour, including the work on relevant aspects of thermal-hydraulics, oxidation, chemistry, mechanical behaviour and reactor physics.

The Mandate is as follows

- Assess the technical basis for current safety criteria and their applicability to high burn up, and to new fuel designs and materials.
- Determine needs and priorities for future safety research programmes in the area of fuel safety behaviour.
- Review from a safety point of view the methodologies used for reactor care assessments related to complex configurations, different fuel designs and types.
- Provide a forum to address emerging safety relevant fuel issues.

At the present time 35 experts from 15 member states and 2 international organisations have been nominated. Industry represented by:TRACTEBEL, EDF/SEPTEN, SIEMENS, PREUSSEN Elektra, EPZ Netherland, CIEMAT, IBERDOLA, EPRI

The first meeting scheduled on June 21-23 June 2000

Annex 1

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

SCOPE and OBJECTIVES

- **Scope**

The task force deals with the status and trends of scientific issues of fuel behaviour.

- **Objectives**

To compile high quality experiments for the International Fuel Performance Experiments (IFPE) database, with emphasis placed on high burn-up in water reactors under normal operating conditions. Completed international programmes - the data of which would have otherwise been lost - data released from national programmes have been given priority.

To co-ordinate the qualification of these data through review and the organisation of user group meetings. Initiatives should be taken to ensure that these data may be adopted as an international standard.

To co-ordinate computer code validation and benchmark studies.

- **Work Plan**

Co-ordination meetings, in association with IFPE database user group meetings, will be organised every 12-18 months.

The IFPE database will be consolidated during the next three 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 release, pellet-clad interaction, etc.)

- **Co-operation**

This activity will be carried out in co-ordination with the IAEA and the NEA Task Force on Safety Criteria in Fuel Behaviour.

- **Duration:** three years

Revision drafted on 2 March 1998
Revision approved by NEA NSC on 4 June 1998

Annex 2

22 May 2000

I F P E

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

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. 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 381 rods/samples from various sources encompassing PWR, BWR, PHWR and WWER reactor systems have been included:

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
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
Total	381

Data in Progress of being Processed or Released

Released and/or being processed

- Studsvik Ramp Experiments
  - DEMO-RAMP I (5 BWR Rods)
  - DEMO-RAMP II (8 BWR Rods)
  - SUPER-RAMP (16 BWR Rods)
- CEA failed PWR rods irradiated in SILOE
  - EDITH- MOX 01
- 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

Data release in progress or requested

- CNEA six power ramp irradiations with (PHWR) MOX fuels
- 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.
- CEA failed PWR rods irradiated in SILOE
  - EDITH-3
  - EDITH- MOX 02

Contact for obtaining information about IFPE:

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More information can be found at:

http://www.nea.fr/html/science/fuel/
http://www.nea.fr/html/science/fuel/ifpe.pdf
http://www.nea.fr/html/science/fuel/IFPE-ans-2k2.pdf
http://www.nea.fr/html/dbprog/

Annex 3

Requirements for including data in the IFPE Database

Introduction

The purpose of the database is to preserve information from fuel performance experiments such that the data are useful and in a form that can be used for code development and validation. Thus, each dataset should include sufficient information to assemble an input file to run a code and information against which to compare predictions. The exception to this rule is for data appropriate to dedicated laboratory experiments which can be used to develop specific models. For example, the IMC data describing out-of-pile annealing experiments where fission gas release from small samples was measured as a function of time at different isothermal annealing temperatures.

Each dataset comprises individual files written in ASCII format. The following is a good method of presenting the data:

1. Summary of data
2. Pre-characterization data
3. Irradiation histories, one for each case
4. Poolside examination and/or non destructive testing (if applicable)
5. Post Irradiation Examination (PIE)
6. READ.ME
7. QA file (acceptable also in WORD, WordPerfect or PDF)
8. Relevant reports and documents (on paper and/or WORD, WordPerfect or PDF)

1. Summary

This is a text file describing the experiment, its objectives and main results. This file contains information on rod identification, material/rod variants etc. This file also contains details of the irradiation and/or test facilities and the conditions employed. For reactor irradiations, relevant information includes: coolant temperature, pressure, mass flow, coolant composition, position of rods in respect to axial height, details of cycle length etc.

2. Pre-characterization

This contains details of the materials used e.g., fuel and cladding, their dimensions, chemical composition, microstructure, impurity levels and any other relevant information. For instance, in the case of cladding, mechanical properties if known and relevant, for the fuel, this could include the results of re-sintering tests, pore size distribution, O/M. The following is a typical check list:

Rod I/D
FUEL PELLET
Composition
Fabrication lot no.
Method of manufacture
Diameter: Max, average, min. mm

Length: Max, average, min. mm
Geometry
Dimple diameter: Max, average, min. mm
Dimple depth: Max, average, min. mm
Uranium isotopic composition wt%: U234, U235, U236, U238
Pore size distribution, % porosity in size range, e.g, 0 – 2, 0 – 5, 0 – 10, >10 microns
Open porosity
Immersion density g/cc
Density %TD
Geometric density %TD
Stoichiometry, O/M
Chemical analysis
Resintering test, e.g., 24 hours at 1700 C in wet hydrogen *(specify)*
Change in %TD:Max, average, min.
Grain size in microns and distribution if available:
Centre pellet, max, av, min
Mid pellet, max, av, min
Pellet periphery, max, av, min

CLADDING
Composition
Heat treatment
Fabrication lot no.
External Diameter: max, average, min. mm
Internal Diameter: max, average, min. mm
Clad wall thickness: max, average, min. mm
Chemical analysis

ROD PARAMETERS
Total length mm
Length of fuel stack mm
Length of plenum mm
Volume of plenum mm^3
Fill gas, composition and pressure at STP:
Number of spacers
Assembly geometry

IRRADIATION
Reactor:
BOC, EOC, EFPD (effective full power days)
Cycle no. EFPD
End of cycle burn-up, MWd/kgU:
Cycle no. Burn-up
Measure of fast flux/fluence
Ratio of thermal heat to total heat for rods
Inlet temperature C
Outlet temperature C
System pressure MPa
Mass flow $Kg/m^2/s$

A re-statement of cross referencing rod identification and design parameters is advisable to aid data retrieval. Where known, uncertainties should be included.

3. Irradiation histories

One file for each case. Histories are constructed with sufficient axial zones as to adequately represent axial power variation. Where segments are cut from larger rods for further irradiation, e.g., in a test reactor, it is advisable to provide a history for the whole rod plus a further history for the segment in question. This should include both the base irradiation and test reactor data in the same format.

The history is provided as a histogram of time and power; a common format which has proved successful is as follows:

TIME STEP NO. TIME(DAYS) DT(HRS) BU(GWd/tU) FLUENCE (*E21 n/cm^2)

TCLAD (DEG C) POWER (kW/m), NODES 1-6

TCLAD (DEG C) POWER (kW/m), NODES 7-12

This is for a full length rod where the history is provided for 12 axial zones Each time step has entries on 3 lines as given in the header. The objective is to provide a format which can be transformed to the required input format of a fuel performance code with the minimum of pre-processing. In this example, the first line starts with a time step number followed by:

TIME	this gives the time from the beginning of the irradiation until the end of the current time step.
DT	this is the duration of the current time step; convention is to report this in hours
BU	burn-up at the end of the current time step
FLUENCE	fast neutron fluence at the end of the current time step

The second line contains entries for local clad temperature and local power for each of six axial zones 1-6. This is continued on the third line for the remaining zones 7-12. For each zone, the temperatures and powers are average values for the given time period. The temperatures are waterside values calculated from the prevailing thermal hydraulic conditions. The powers are thermal powers, i.e., describe the heat flowing out through the cladding and contributing to the temperature distribution across the fuel rod.

4. Poolside examination and/or non destructive examination

For each case, details of visual inspection dimensional measurements, gamma scanning and neutron radiography where applicable. A correspondence between zone and axial height must be given in order for predictions using the history can be compared with measurements. Useful measurements include rod growth, fuel column growth, axial variation in oxide thickness, diameter and ridge height. Gamma scan data are useful for describing axial variation in power and burn-up. Neutron radiography is useful for fuel stack length measurement, hydriding of cladding, pellet dish filling and pellet fragmentation (cracking). Details of failure sites, crack length, appearance and position are important observations if applicable using visual inspection.

5. Post irradiation destructive examination

The type of information that is required:

Results of rod puncturing,	gas pressure, internal volume, gas volume, gas isotopic composition.
Ceramography,	state of pellet, radial and axial distribution of grain size, porosity, pores size distribution, onset radius for grain boundary and/or inter granular porosity. Evidence of dish filling. Evidence of restructuring, e.g., ‘rim’ structure formation.
Metallography,	State of clad, any hydriding, blisters, fuel clad interaction compounds.
EPMA,SEM, TEM etc	any further information that describes the fuel and cladding which is relevant to fuel modelling.

6. READ.ME

This file acts as a contents list and contain a list of files in the dataset, their title and brief description. It contains the date of compilation and the person responsible for the compilation. It contains a comment on the state of verification of the data, any known shortcomings, an identifier and date in the case of re-issue.

7. QA File

The purpose of this file is to assist a third party to verify the methods used in creating the dataset. Sufficient information should be included to link the original data with that in the final dataset.

The file describes the form in which the data was received and the methods used to compile the data into the final form of the dataset. Where possible, reference should be made to reports, tables and/or diagrams used to extract data so that correct retrieval can be checked by a third party. A listing should be included of any program used to process the data, e.g., for compiling the history files, calculating time periods, local powers and temperatures. Specimen input and output files or parts thereof should be included as an appendix.

Where figures have been scanned and digitized, it is worth creating a new diagram from the data for inclusion in this QA file for visual comparison with the original.

8. Relevant reports and documents

Reports used in compiling the items 1. - 7. should also be provided in hard copy or in computer readable form. In fact these reports will be included in the database in order to facilitate access to the original documentation and to facilitate peer review or later improvements.

Annex 4

Fuel Behaviour Computer Codes Distributed by the OECD/NEA Data Bank

Recent acquisitions are:

FEMAXI-V, thermal & mechanical behaviour of LWR fuel rods
RODBURN, power profiles & isotopics in PWR, BWR fuel rods (input for FEMAXI-V)
PIN99W, modelling of VVER and PWR fuel rod thermal & mechanical behaviour

Codes acquired previously:

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, subchannel analysis method
COMET, mechanical & 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
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, FP 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 & performance of cylindrical fuel_element in fast_reactors
MOXY/MOD-1, thermal analysis swelling & rupture of BWR fuel element during LOCA
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