Status of Nuclear Science Committee Activities in the Field of Fuel Behaviour

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Content

1 INTRODUCTION

The OECD/NEA Nuclear Science Committee (NSC) has long recognised the need for improved knowledge and international co-ordination of important scientific issues related to fuel behaviour. Consequently, a special Task Force was set up by the NSC in late 1993.

The objectives of the Task Force, as endorsed by the NSC, were to identify areas of high priority to Member countries, which would benefit from international co-ordination and co-operation on studies of the basic underlying phenomena of fuel behaviour under normal operating conditions.

The Task Force was asked also to advise the NSC on developments needed regarding data, models and experiments to meet the requirements for better understanding of fuel behaviour and for improved predictive models.

The main findings and conclusions of the Task Force were summarised in a report [1], reviewed in Section 2 below, which set the scene for a series of actions taken by the NEA over the succeeding years. The present report reviews these initiatives and activities, highlighting the achievements already made as well as indicating for the future, the direction where there is benefit for continuing and extending the current activities.

2 IMPORTANT ISSUES FOR FUEL PERFORMANCE MODELLING

The first action of the Task Force was to identify topics and potential authors to review the then current state of understanding, and to identify priority areas where further work was required. This was accomplished and a report addressing the following topics was issued [1]:

- *Thermal performance;* the calculation of fuel temperatures, the effect of design parameters and the effect of irradiation
- *Fission Gas Release;* the release under different operating regimes and the effect of high burn-up
- *Fission Product Swelling;* the distinction between inexorable solid fission product swelling which is predominantly a function of burn-up and fission product accumulation, and gaseous swelling which is also dependent on high power operation
- *Stress Corrosion Cracking;* the conditions of stress, time and fission product release on the propensity for clad failure from this mechanism

- *Water Chemistry*; current practice and monitoring, its effect along with Zr composition and final treatment on clad corrosion
- *Hydrogen in Cladding*; its distribution and measurement
- *Failed Fuel*; detection and modelling degradation processes
- *Spent fuel*; long term storage under ‘wet’ and ‘dry’ conditions
- *Quality Assurance*; as applied to materials, experimental tests and data production.

The report concluded that although fuel in nuclear reactors had proven to be highly reliable in its performance and safety, reactor operation was often supported by rather generous bounding operating conditions. In the future the requirements would be more onerous. The economic requirement to increase efficiency meant that fuel performance must be calculated accurately on a ‘best estimate’ basis. To do this requires good fuel performance computer codes validated by high quality data.

As a result of this report the Task Force arrived at the following recommendations for future NEA activities:

i. Countries and organisations should be encouraged to make special research efforts to reduce uncertainties in modelling specific aspects of fuel behaviour, namely
   - thermal performance and calculating fuel temperatures
   - fission gas release
   - fission product swelling and creep of UO_2
   - thermo-mechanical behaviour
   - high burn-up fuel behaviour in transient conditions

ii. As a first step, a review should be made of existing data
iii. A public domain database should be assembled of well-qualified experiments and data which can be used for model development and code validation. This database should be organised and maintained by the NEA Data Bank
iv. International topical meetings covering high priority issues should be organised by the NEA in co-ordination with the IAEA. Of particular interest were workshops on: thermal performance, fission gas release (FGR) and pellet-clad mechanical interaction (PCMI)

3 IDENTIFICATION OF AVAILABLE EXPERIMENTAL DATA

The second report commissioned by the Task Force and issued by the NEA was the review of existing data that could be made available to fulfil the aim of improving code performance. The report [2] concentrated primarily on data produced in the Joint Programme carried out by the Halden Reactor Project, Norway.

Conditions in the Halden reactor are particularly well suited to studies of fuel performance. The boiling conditions ensure a constant coolant temperature, and hence a well-defined boundary condition from which to assess thermal performance from measurements of centreline fuel temperatures. Also, the low system pressure of 32 bar and low fast flux ensures negligible clad creep down, thus removing one parameter from through life assessments of fuel dimensions and temperature. However, when more prototypic conditions are required, dedicated in-pile loops are available to simulate the thermal hydraulic conditions of temperature and pressure as well as neutron flux spectrum for PWR, BWR and most recently, Advanced Gas Cooled Reactors (CAGR) as operated in the UK.

The report outlined some of the experiments that have been performed in the Halden reactor and the most important data that they have provided which are available for use in fuel performance studies. As to be expected in a Project that has many years of experience and data production, several reviews of individual topics have already been written. This present report made extensive use of these with the aim of demonstrating the extent of the information available collated in a form that is most useful for the development of fuel performance codes and their validation.

The data can be divided into a number of categories depending on the use to be made of them. In this respect, the data are not of universal interest. The simplest division is into the three broad categories:

Data useful for model development and validation, e.g.: radial flux depression, fuel creep, fuel densification and swelling and clad creep down. These are the 'unsung needs' of the code developer, not often apparent in code predictions but essential to obtain a good description if reliable predictions are to be obtained. Very often these data require special measuring techniques and non-prototypic rig designs.

- *Data of direct relevance to licensing requirements*, e.g.: fuel temperatures, stored heat, fission product release and rod internal pressure, waterside corrosion. Such data are of particular use for validation purposes.

- *Data for fuel development and optimisation*, e.g.: performance of fuel variants and new cladding materials, effect of changes in fuel design. These data are of most interest to fuel vendors in developing and supporting new products.

Within these categories, the report discussed data under the following phenomenological headings:

- Radial Flux Depression
- Thermal Performance
- Fuel Densification and Swelling
- UO_2 Grain Growth
- Fission Product Release
- Clad Properties
- PCMI - Pellet Clad Mechanical Interaction
- Integral Behaviour
- High Burn-up Effects.

The report reviewed experiments that addressed ‘single effects’, i.e., where single parameters were isolated for study, (fuel centre temperature as a function of fuel pellet to clad gap), and experiments which addressed ‘integral behaviour’, where several effects were studied simultaneously, (fuel temperature, fuel and clad axial extension and rod internal pressure as a function of power and burn-up).

In addition to Halden Project data, the report briefly reviewed data from certain Studsvik ramp tests and from the three Risø fission gas release projects. These had already been shown to be compatible with Halden experiments, particularly in the areas of PCI/SCC failure and PIE of ramp tested fuel.

Despite the comprehensive nature of the data available as outlined in the report, it was clear that there were a number of areas where further experiments were necessary. With the move to higher discharge burn-up, it is necessary to extend the data to cover the extremes of burn-up and power expected in commercial reactors. This implied a progressive extension of well qualified data in excess of 70 $MWd/kgUO_2$. New effects were being discovered at these levels of burn-up and further data were required on such topics as: 'the rim effect', further definition of the effect of burn-up on fuel thermal conductivity, clad corrosion and hydriding and clad mechanical properties. Although there were data on temperature changes during reactor scrams, for transient analysis, there was a need for data on the evolution of temperatures during rapid power changes. There were (and still are) few data on this last topic and there is clearly a benefit to be claimed in reactivity faults for example, to exploit the lag between power increase and consequent increase in temperature.

The report supported the Task Force initiative to assemble an International Fuel Performance Experiment (IFPE) database, and recommended the inclusions of experimental data identified in the report from all three international programmes.

4 IFPE DATABASE

From the outset it was recognised that the database should apply to all commercially operated thermal reactor systems and that the data should be both prototypic, originating from power reactor irradiations with pre- and post-irradiation characterisation, and test reactor experiments with in-pile instrumentation and PIE exploring normal and off-normal behaviour. It was recognised that experiments have been performed where the data remain of commercial interest, for example, much of the details of modern MOX performance remains proprietary to the manufacturers, and it was not the intention to compromise such arrangements. However, Zircaloy clad UO_2 pellet fuel can be largely regarded as a 'standard product' and as such, release of what was previously proprietary data can only benefit the nuclear community at large.

A particular aspect of the compilation was the inclusion of data generated within internationally sponsored research programmes whose confidentiality agreements had expired. Such data, although available in principle, have not been widely used. The inclusion of such data is of particular importance where the originating organisation has changed its terms of reference. For example, the Risø laboratories in Denmark no longer perform nuclear research and find difficulty in resourcing the supply of information for their three fission gas release projects. In such cases, there exists the danger of losing access to the data altogether.

4.1 Extent of parameters included

The database is restricted to thermal reactor fuel performance, principally with standard product Zircaloy clad UO_2 fuel, although the addition of advanced products with fuel and clad variants is not ruled out. Recent additions to the Database include MOX fuel, $(U,Gd)O_2$ and defect fuel. Data encompass normal and off-normal behaviour but not accident condition entailing melting of fuel and clad, resulting in loss of geometry.

Of particular interest to fuel modellers are data on: fuel temperatures, fission gas release (FGR), clad deformation (e.g. creep-down, ridging) and mechanical interactions. In addition to direct measurement of these properties, every effort is made to include PIE information on the distribution of: grain size, porosity, Electron Probe Micro Analysis (EPMA) and X-ray Fluorescence (XRF) measurements on caesium, xenon, other fission product and actinides.

Emphasis has been placed on including well-qualified data that illustrate specific aspects of fuel performance. For example, cases are included which specifically address the effect of gap size and release of fission gas on fuel-to-clad heat transfer. Also in the context of thermal performance, the effect of burn-up on UO_2 thermal conductivity has been addressed. This is illustrated by cases where fuel temperatures have been measured throughout prolonged irradiation and at high burn-up where sections of fuel have been refabricated with newly inserted thermocouples. Regarding fission gas release, data are included for normal operations and for cases of power ramping at different levels of burn-up for fuel supplied by several different fuel vendors. In the case of power ramps, the data include cases where in-pile pressure measurements show the kinetics of release and the effect of slow axial gas transport due to closed fuel-to-clad gaps. Supplementing these in-pile studies, there are datasets of Out-of-Pile annealing studies measuring FGR under well defined conditions of temperature and time.

To date datasets about 418 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
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
BN GAIN (U,Gd)O_2 fuel	4 rods
INR Pitesti - RO-89 and RO-51 CANDU fuel type	2 rods
Total	422 cases

4.2 Format of files

The criterion adopted for the file format was one primarily of simplicity. It was considered that users should be able to read the files independent of commercial software. It was recognised that the majority of codes were written in FORTRAN and therefore all files are in simple ASCII format for easy interrogation by file editors. Even text files are of this format despite the enforced limitations imposed by this approach. The adopted ASCII format does not preclude reformatting into a commercial database system sometime in the future if so desired. All databases have the following common elements:

- ***Summary file***. This is a text file which describes the purpose of the experiment or test matrix and the scope of the data obtained.

- ***Index file***. This is also a text file and lists all file titles and gives a brief summary of their contents.

- ***Pre-characterisation***. This includes information on the fuel pellets and cladding used, their manufacturing route, dimensions and chemical composition including impurities. For the fuel, this is augmented by details of enrichments, porosity distribution, re-sintering test data and microstructure. For the cladding, additional information includes mechanical properties, corrosion characteristics and texture as and when available. Details of the fuel rod geometry include: relevant dimensions, fuel column length, weight, fill gas composition and pressure. Details of reactor irradiation conditions.

- ***Irradiation histories***. All histories are in condensed form with care to ensure that all important features are preserved. Where there was a significant axial power profile the history is provided in up to 12 axial zones. For each time step, the data are provided as: time,
  time increment over which power was constant, clad waterside temperature and local heat rating for the prescribed number of axial zones. Information is provided to calculate the fast neutron flux and its spatial variation if the information is available.

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

- ***PIE data***. Where such examinations were made, data are recorded either in tabular or text form. Dimensional data include axial and diametral dimensions before and after irradiation, and post irradiation ridge heights where available. Data on fission gas release include rod averaged values obtained by puncturing and mass spectrometry, local values from whole pellet dissolution and across pellet spatial distributions as measured by gamma scanning, EPMA and XRF. Before and after porosity and grain size distributions are given as is the radial position for the onset of grain boundary porosity when such measurements have been made from metallographic examination.

All data files are held centrally on the OECD/NEA Data Bank computer system from which all files are dispatched. This single source for distribution is necessary for Quality Assurance purposes, particularly for the tracking and release of upgraded or corrected files. With the OECD's experience of data bank management, this arrangement ensures long term availability of the service.

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

5 INTERNATIONAL WORKSHOPS

5.1 Thermal Performance

The initiative to hold international workshops commenced with Thermal Performance of High Burn-up LWR Fuel held 3-6 March 1998 at CEA Cadarache. The workshop was co-organised by the OECD NEA in co-operation with the IAEA and hosted by the Département d'études des Combustibles (DEC). The meeting was attended by 66 experts from 19 countries, and four international organisations.

The first two sessions were devoted to *Fuel Thermal Conductivity Data* and *Thermal Conductivity Modelling.* This was followed by a first panel discussion, chaired by Hubert Bairiot and Louis-Christian Bernard addressing *Solving and Emerging Thermal Conductivity Questions.* The third session covered *Fuel Clad-Gap Evolution Modelling* followed by a second panel discussing *Gap Evolution and Heat Transfer Questions,* chaired by Gary Gates and Marc Lippens. The two final sessions covered *Experimental Databases* and *Advances in Code Development on Thermal Aspects.*

A final panel, chaired by J.A. Tumbull and Daniel Baron, reviewed the conclusions of the two panels on thermal conductivity and gap conductance and summarised - after discussion among all participants - the key conclusions of the seminar. These conclusions are appended below:

Fuel thermal conductivity

Fuel thermal conductivity correlations
The initial thermal conductivity correlations are well established for almost all fuel oxides loaded in commercial reactors. The influence of parameters such as temperature, stochiometry, plutonium content and gadolinium content are relatively well known. Some models are able to account for all these parameters simultaneously.

MOX fuels
The non-homogeneity of MOX fuel can be accounted for, using mathematical homogenisation techniques. However, even for high Pu content the fuel thermal conductivity is close to the uranium oxide fuel conductivity providing O'M close to 2.000. Participants seemed to agree on a degradation of 4 to 5% per 10% Pu. However, it was shown that a deviation from stochiometry has a stronger effect.

The burn-up effect
The burn-up effect on fuel thermal conductivity has been assessed up to 80 MWd/KgU, thanks to in-pile centreline temperature measurements and out-of-pile thermal diffusivity measurements. There is some agreement between these two methods.

Cp variation with burn-up
Analysing shut-down temperature records in the Halden reactor, a slight increase of the Cp with burn-up is likely to happen. Nevertheless the discussion outlined that the Cp variation with burn-up could be neglected. The experimental data expected in the near future will certainly allow checking this assumption.

Heat transfer in the rim
The rim thermal conductivity evolution is not yet clearly known. The thermal degradation due to the onset of numerous micrometer porosities could be balanced by the cleaning of the matrix subsequent to this porosity build-up. In order to investigate the net effect of the rim structure, samples irradiated to high burn-up and temperatures Tower than 800 K are

required. Other questions are still open about the rim fission gas release and the rim volume variation during the porosity build -up.

Further needs
A lack of data is identified on the burn-up degradation at temperatures higher than 1800 K. A decrease of the conductivity improvement due to the electronic heat transport is likely to happen with burn-up. This effect was already observed when increasing the gadolinium content.

<u>Gap conductance</u>

Pellet fragmentation and relocation
The stochastic nature of this phenomenon prohibits analytical modelling of gap conductance. Reliance must be placed on benchmarking models against in-pile data. There is no general consensus as to a definitive formulation for gap conductance. Fortunately a large database exists and providing this is employed, there are limited difficulties in formulating an adequate gap conductance model which correctly reflects the effects of gap dimension, fill gas composition and pressure.

Surface roughness
In principle this should be an important contributor to gap conductance by its effect of inhibiting heat flow. In practice, at both the beginning of line and at high burn-up, there appears to be a little effect on heat transfer between the fuel pellet and the cladding.

Formation of inner clad oxide layer
At high burn-up, when the fuel and clad have been in intimate contact for some time, an inner oxide layer is formed, 6-10 microns thick. It has been shown that this has a complex structure, which can include ZrO~, uranium and caesium, depending on the heat rating and burn-up. It is supposed that its contribution to gap heat transfer is small and if anything, beneficial, as it would tend to eliminate effects due to surface roughness and misaligned pellet fragments.

Closed gap conductance
There is evidence to suggest that under these conditions, heat transfer is good and substantially independent of the fill gas composition and pressure, since it corresponds to a solid bound between fuel and cladding. There is some debate as to whether or not the conductance depends on interfacial pressure.

5.2 Fission Gas Release

The second workshop in the series was the Fission Gas Release Workshop, held at CEA Cadarache, 26-29 September 2000, organised and hosted under the same arrangement as before. The seminar was attended by about 100 participants from 24 countries, representing 46 organisations. The objective of this Seminar was to emphasise the more recent developments, from both the experimental and modelling points of view. The areas covered by the seminar included: diffusion coefficient of rare gases including helium, properties of bubbles, the effect of irradiation on re-solution and diffusion, FGR as a function of operating conditions, modelling and code validation.

The meeting was divided into four sessions which occupied three days of presentations. The session titles were: *Feedback from Experience, Basic Mechanisms, Analytical Experiments, Industrial Modelling* and *Software Packages*. Summaries of each session were prepared by session chairpersons and were appended to the final report. On the fourth and final morning, there was a session entitled: *Panel Discussion: Conclusions and Perspectives* where each chairperson gave an oral presentation of their sessions which was discussed among the participants. In view of the 'maturity' of the subject, each chairperson was requested to address the three questions:

- What do we know, what is new?
- What don't we know, what are the uncertainties and what remains unclear?
- What initiatives should be implemented in the future?

From the first session, *Feedback from Experience*, it was concluded that there was an enhancement of fission gas release at high burn-up, >50 MWd/kg, both in steady state operation and in transient overpower, but there was no accepted phenomenon to explain this. One result of this was that the well established Halden 1% FGR criterion relating fuel centreline temperature to burn-up overestimated the onset temperature at these levels of burn-up. That is, the temperature for the onset of release at high burn-up was *lower* than predicted by this criterion. It has been observed that high burn-up fuel contains a region of restructured fuel microstructure close to the pellet periphery, the so-called 'rim' structure. It was tempting to use this as a reason for the enhanced release, but a direct correlation could not be made. Indeed a measure of the ratio of released krypton and xenon isotopes implied that their origin was the hot central regions of the fuel and not the cold plutonium rich restructured region in the pellet rim. The implication of this observation was that the restructured rim served mainly to increase the thermal resistivity of the fuel, thus increasing fuel temperatures.

A discussion of this dilemma continued into the second session on *Basic Mechanisms*. Investigations showed that this rim structure consisted of a refinement of the original ~ 10 micron diameter grains into ~0.1 micron diameter grains and the introduction of a micron size population of bubbles which could account for a swelling up to ~10 volume %. In addition, the fission gas concentration in the matrix fell to a low level. The impression of most workers was that the shortfall in fission gas resided in the bubbles and had not been released from the fuel, however the exact distribution of gas remained uncertain. The fission gas release from the rim was not more than 15 to 20 %, showing a high retention capacity of such a restructured material. The mechanism of rim formation was not clear; one suggestion was that the build up of irradiation damage caused the grain refinement and that this was followed by the collection of gas into the porosity. Alternatively, it has been postulated that the porosity was the first to form and that the sub-micron grains were nucleated from the pore surfaces. It remained a matter of debate whether this rim restructuring was beneficial or detrimental.

Several presentations addressed FGR from the rim both in steady state and transients such as RIA, concentrating on mechanisms and experimental observations. Measurement of krypton and xenon isotopic ratios in released gas was clearly a valuable technique to determine the relative contribution of the rim region to FGR, and application to RIA tests had proved informative. It was concluded that although we knew details of this new high burn-up structure at the micron level, it was clear that a goal for the future was to understand it at the nanometer or atomic level, and that there was scope for simulation studies using high energy accelerators to reproduce the restructuring without the constraints of time and radioactivity.

A major contribution to the release process is known to be single gas atom diffusion and quite satisfactory models could be constructed using this along with irradiation resolution from grain boundaries. Many models employed the diffusion coefficient formulated by Turnbull, White and Wise as described in the IAEA Preston Meeting in 1988. Here the authors gave a two term diffusion coefficient with a third low temperature term to be applied close to surfaces and used for short lived radioactive species. There was much discussion on this topic in the meeting and it transpires that White has an alternative description using only the two high temperature terms with a 'fractal' treatment of the surface to account for the difference in kinetics between long and short lived species. Subsequent to the meeting, this has now been published in the external literature.

Extensive study of retained caesium and xenon using Electron Probe Micro Analysis (EPMA) carried out at ITU showed that systematically, caesium diffusion was some 3 times slower than for the rare gases. This was an important result since in many accident calculations it is assumed that their diffusivities are comparable; the assumption therefore is reassuringly pessimistic.

In addition we know from electron microscopy that intragranular bubbles are formed within grains and intergranular bubbles and tunnels along grain boundaries. Theoretical studies of krypton atoms in the UO_2 matrix concluded that both krypton and xenon were insoluble thus casting doubt on the existence of 'thermal resolution' as a means of moderating absorption of single gas atoms into intragranular bubbles. In the absence of intragranular bubble mobility, such a mechanism was used in several models in order to predict substantial release in transients. It was agreed that observations support only slow or limited random migration of small bubbles, but a 'directed' movement towards grain boundaries due to a vacancy concentration gradient was an interesting proposal, (Evans, UK). If this mechanism had a significant contribution, the resulting microstructure would be quite distinctive and amenable to future testing. Topics which required further attention was the re-solution of gas atoms from grain boundary bubbles, particularly for large grain fuel where the grain size was greater than the fission fragment range and the response of grain boundary porosity in fast transients.

It was clear that there was much more information available on these features than currently accessible through the open literature. There was general support for a more open distribution of information and all participants were urged to publish as much information as possible in the interest of a general improvement in models and their application to safe reactor operation. It was noted that although the level of fundamental research in many establishments had been reduced, there were still original and comprehensive works going on in ITU, for example, improving the investigation techniques (microhardness tests, lattice parameter measurements). Also, there was a refreshing intake of young engineers and scientists within CEA Cadarache. The participants looked forward to publication of new and original work from these quarters.

As a digression from fission gas, it was clear that helium generation in long term storage of high burn-up fuel can pose a significant problem. In particular for high burn-up MOX. Further data were required on low temperature helium diffusion coefficients.

The session entitled *Analytical Experiments* started with two presentations describing a novel method of determining the disposition of gas between the matrix and the grain boundaries. Such studies complement those performed with X-Ray Fluorescence (XRF) and EPMA and provide vital information for modellers. The techniques still need refining, particularly regarding the effect of small intragranular bubbles on the results, but promise an evaluation of the grain boundary capacity for gas prior to and during interlinkage. As mentioned previously, Halden have obtained data from high burn-up fuel which suggests that the temperature for the onset of FGR is lower than previously expected. It was clear that further data were required to substantiate this observation. In addition, there was a need to obtain further data on MOX fuel to compare it with UO_2 fuel performance under identical conditions.

Enhanced release at high burn-up can cause the rod internal pressure to exceed the coolant pressure. The effect of this requires attention. So far, experimental data suggests that the gap does not reopen by clad creep-out and positive feedback does not occur. However, the data are sparse and further experiments are necessary.

Both mechanistic and empirical approaches to FGR modelling were presented in the session: *Industrial Modelling* and *Software Packages*. It was clear that empirical modelling can be very valuable, but is limited in applicability and is essentially only valid within the confines of parameters and irradiation conditions covered in the database on which it is developed. Modellers were also cautioned about employing multiple mechanistic models and expecting to obtain good predictions by using appropriate fitting parameters. There is a case for independent assessment of mechanistic models and their supporting data before consideration for inclusion within fuel performance codes. The requirement for further data comparing UO_2 and MOX behaviour was re-affirmed in a paper by Struzik of CEA which showed that there was a significant difference in behaviour in ramp test above 30 MWd/kg. Below 30 MWd/kg the differences in behaviour could be explained in terms of the radial power profile but at higher burn-up FGR and swelling in MOX was greater than expected.

In conclusion there was unanimous agreement that the Workshop had been a success with an excellent range of presentations which initiated extensive and in-depth discussion. It was clear that the topic of fission gas release modelling was in a mature state with agreement about most of the important mechanisms contributing to the phenomena. High burn-up and MOX behaviour are the current challenges where there is an incomplete understanding, but the meeting produced some good suggestions where future work could be focussed. Participants were urged to publish their work in the open literature so that all could benefit towards the goal of safe operation of commercial reactors.

5.3 Pellet-Clad Interaction

The third and final workshop is to be on Pellet Clad Mechanical Interaction as this was deemed the least advanced area of modelling. The date for this meeting is scheduled for Autumn 2003 to be held at CEA Cadarache.

References [3,4] concern the proceedings of these two seminars.

6 SURVEY OF SAFETY RELATED ACTIVITIES IN MEMBER COUNTRIES

While the NSC Expert Group covers mainly scientific issues underlying fuel behaviour in normal and transient operating conditions (see Annex), fuel safety issues are addressed within the work programme of the Committee on the Safety of Nuclear Installations (CSNI). NSC and CSNI co-operate and co-ordinate activities in fuel behaviour area.

Under the auspices of CSNI a report has been issued reviewing the status of safety related research activities in member countries [5]. This report is in response placed on members of a a Special Experts' Group on Fuel Safety Margins (SEGFSM) to compile ongoing and planned fuel safety research in NEA member states with the aim of providing CSNI an overview on related R & D international programmes and projects, along with the identification of current and future needs and priorities.

The report is based on a questionnaire distributed to SEGFSM members in October 2000, requesting them to identify fuel safety research programmes and to provide information on achievements and future plans. The questionnaire required respondents to provide information on the ongoing R&D programmes under the following headings:

A. Title
B. Research Laboratory/Sponsor(s)
C. Objectives/Goals
D. Status of Work

E. Brief description/presentation of the main results achieved
F. Future plans
G. References

Replies were received from organisations in the following countries:

Canada, Czech Republic, France, Germany, Hungary, Japan, Korea, Norway (Halden Reactor Project), Switzerland, United Kingdom, and USA.

The report is based only on the information provided in the replies received, as a consequence it cannot be viewed as comprehensive; programmes may well be in progress in addition to those detailed here. It is also possible that the detailed results of some programmes may remain proprietary and therefore not available in the short term.

The report is organised in topic sections relating to: fuel and clad studies, integral fuel rod tests and PIE, LOCA and RIA studies including whole rods and bundles as well as single effects studies of fuel and cladding, code development for both steady state and transient fuel behaviour, thermal hydraulics, reactor physics codes and finally severe accident studies.

The main issues for the current generation of reactors are those of high burn-up performance in normal operations, LOCA and RIA conditions and the main goal for the industry is to consolidate the safety issues to bring all countries up to a licensed discharge burn-up of ~60 MWd/kg and possibly 65 MWd/kg. The principal issues requiring attention can be broken down as follows:

Normal operation:

- fission gas release and rod over-pressure,
- properties of the High Burn-up Structure (HBS) at the pellet rim, its effect on thermal performance and fission gas release,
- cladding oxidation, hydriding and embrittlement.

Loss of Coolant Accidents (LOCA)

- the possibility of fuel slumping into the ballooned region; the effect of fuel-clad bonding at high burn-up,
- increase of pressure in the ballooned region due to fission gas release from slumped fuel,
- response of irradiation hardened and hydride embrittled cladding.
- review of the 17% Equivalent Clad Reacted (ECR) criterion,

Reactivity Initiated Accidents (RIA)

- PCMI (Pellet-Clad Mechanical Interaction) loading mechanism(s) on the cladding, the effect of the HBS at the pellet rim,
- effect of HBS on fuel dispersal,
- response of embrittled cladding to transient PCMI.

In addition to these, for those countries that load both MOX and UO_2 fuelled assemblies, there is a requirement to bring the MOX database to the same level as that for UO_2 fuel with the aim of treating MOX indistinguishably from that of UO_2 as far as safety is concerned.

The survey of international research programmes outlined in the report demonstrates the large element of activity to address these issues. When put together, the individual programmes add up to a tremendous effort in both time and money and will ultimately lead to a much better understanding of materials and component behaviour in a wide range of postulated scenarios. It is to be noted that all countries have extensive modelling and code development programmes to best utilise the data generated from the experimental programmes.

It is very important therefore, that these activities are well supported and that their results should be made available to the widest possible audience. Only in this way can there be a global common culture of safe and economic production of electricity from nuclear power generation.

7 FUTURE ACTIVITIES

7.1 The IFPE Database

Now that the IFPE Database is firmly established, there is a continuing need for its support and improvement. Indeed, on the assumption that the Database will continue to be maintained by the NEA, several organisations are using the database as the sole source for their code development and validation. Further data will be added to the Database in the future including the following for which agreement for release has already been obtained:

- 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
- Zaporoshye VVER1000 fuel behaviour data (4-8 cycles, Bu » 50 MWd/kgUO_2
- 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 behaviour data on different cladding thickness by means of diameter rig.
- CEA failed PWR rods irradiated in SILOE; EDITH-3 and EDITH- MOX 02
- IMC (UK) swelling data from ramping CAGR UO_2 fuel

Based on the success of the FUMEX-1 Code Comparison Exercise, the IAEA have agreed to launch a initiate project, FUMEX-2. The global outcome to the FUMEX 1 programme was very encouraging, with a high degree of participation from Member countries. All agree that it was a worthwhile exercise, and that the cases chosen were stringent tests of models and codes performances. The exercise was useful in demonstrating the strong points of the codes as well as highlighting deficiencies where improvements were necessary. As a consequence, most of the codes underwent some development during this programme. It was also apparent that many of the codes have been developed on a limited database and that the FUMEX 1 cases provided a valuable addition. However developments pointed out that limited knowledge have been gained in the extended burn-up range (above 50 MWd/kg). Burn-up extension corresponds to a general trend in the field of fuel management, and reliable prediction of fuel behaviour at high burn-up constitutes a basic demand for safe and economic operation of nuclear fuel.

This was the basis for launching FUMEX-2 which will run from December 2002 to its completion in 2006. Central to the project are selected cases in the IFPE Database. This will help promote the use of the Database and at the same time, will generate feedback and peer review from users, thus assisting the Quality Assurance of the datasets within the Database. The cases chosen for FUMEX-2 are tabulated below, with cases already in the Database shown in bold type.

It is important therefore that the NEA provide additional support to maintaining and improving the Database over this period in addition to the introduction of further data as they become available.

No.	Case identification	Measurements made for comparison
1.	Halden IFA 534.14, rod 18	EOL FGR and pressure, grain size 22 μm, Bu ≈ 52 MWd/kgUO_2
2.	Halden IFA 534.14, rod 19	EOL FGR and pressure, grain size 8.5 μm, Bu ≈ 52 MWd/kgUO_2
3.	Halden IFA 597.3, rod 7	Cladding elongation, at Bu ≈ 60 MWd/kgUO_2
4.	Halden IFA 597.3, rod 8	FCT, FGR at Bu ≈ 60 MWd/kgUO_2
5.	Halden IFA 507, TF3	Transient temperature during power increase
6.	Halden IFA 507, TF5	Transient temperature during power increase
7.	GONCOR	FGR and cladding diameter during and after transient at Bu ≈ 48 MWd/kgUO_2
8.	**Kola-3, rod 7 from FA222**	**FGR, pressure and creepdown at Bu ≈ 55 MWd/kgUO_2**
9.	**Kola-3, rod 52 from FA222**	**FGR, pressure and creepdown at Bu ≈ 46 MWd/kgUO_2**
10.	**Kola-3, rod 86 from FA222**	**FGR, pressure and creepdown at Bu ≈ 44 MWd/kgUO_2**
11.	**Kola-3, rod 120 from FA222**	**FGR, pressure and creepdown at Bu ≈ 50 MWd/kgUO_2**
12.	**Risø -3 AN2**	**Radial distribution of fission products and FGR-EOL, Bu ≈ 37 MWd/kgUO_2**
13.	**Risø -3 AN3**	**FGR and pressure-EOL, FCT, Bu ≈ 37 MWd/kgUO_2**
14.	**Risø -3 AN4**	**FGR and pressure-EOL, FCT, Bu ≈ 37 MWd/kgUO_2**
15.	**HBEP, rod BK363**	**FGR-EOL, Bu ≈ 67 MWd/kgUO_2**
16.	**HBEP, rod BK365**	**Fission products and PU distribution, FGR-EOL, Bu ≈ 69 MWd/kgUO_2**
17.	**HBEP, rod BK370**	**Fission products and Pu distribution, FGR-EOL, Bu ≈ 51 MWd/kgUO_2**
18.	**TRIBULATION, rod BN1/3**	**Pressure, FGR, cladding creepdown, Bu ≈ 52 MWd/kgUO_2**
19.	**TRIBULATION, rod BN1/4**	**Pressure, FGR, cladding creepdown, Bu ≈ 51 MWd/kgUO_2**
20.	**TRIBULATION, rod BN3/15**	**Pressure, FGR, cladding creepdown, Bu ≈ 51 MWd/kgUO_2**
21.	EDF/CEA/FRA, rod H09	Fission products and Pu distribution, FGR-EOL, Bu ≈ 46 MWd/kgUO_2
22.	Kola-3 + MIR test	Temperature during ramp, FGR-EOL, Bu ≈ 55 MWd/kgUO_2
23	Kola-3 + MIR test	Pressure-EOL, Bu ≈ 55 MWd/kgUO_2
24.	RIA	to be specified (real data or simplified case)
25.	LOCA	to be specified (real data or simplified case)
26.	Simplified case	Temperature vs. Bu for onset of FGR

7.2 Other Activities

Based on the survey of member countries and their safety related R & D activities, the NEA formed a good idea of priorities for current and future work. They are active therefore in supporting these activities particularly related to high burn-up studies, the introduction of MOX into commercial reactors and the issues arising in the use of weapons grade fuel, W-MOX in order to reduce the inventories held by both the US and Russia.

Two workshops were held on "Advanced Reactors With Innovative Fuels (ARWIF) in 1999 and 2001. Fuel behaviour issues were also discussed. For details see Ref. [6,7].

The commercial nuclear industry is currently at a very critical stage. The concern over Global Warming has focussed attention on fossil fuel usage and its emissions. As a result, many countries are extending the life of their current installed plant as well as showing a renewed interest in building new nuclear generators. It is clear however that this is unlikely to be in the form of current designs. There are several new designs, some are extensions of the current Generation II LWR, but some are more innovative and as such are classified under the generic Generation IV designs. Common features of these new designs include passive safety, high thermodynamic efficiency and proliferation resistant fuel. Before entering full scale commercial operation, these will require extensive qualification both in their design and in the materials used in their construction and operation. It is a goal for the NEA to assist in the quest for improved nuclear design and installation by supporting R & D into these new concept and new fuel types and cladding alloys.

8 REFERENCES

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Annex

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.