WORKING PARTY ON SCIENTIFIC ISSUES OF REACTOR SYSTEMS (WPRS)

Summary of Fifth Meeting and Status Report of Projects

WPRS05 Meeting held at Fukui, Japan
from 18 to 19 February 2008

hosted by JAEA

I. Introduction

The mandate of the WPRS was extended at the last NSC meeting in June 2007 for three years representing recent restructuring of the activities and adjustments.

WPRS held its fifth meeting (WPRS05) from 18 to 19 February 2008 at Fukui, Japan, hosted by the Japan Atomic Energy Agency (JAEA). This meeting was held in conjunction with the Workshop on Advanced Reactors with Innovative Fuels (ARWIF-2008).

At the fifth meeting, K. Hesketh was confirmed Chairman of the group, assisted by P. D'hondt as Vice-chairman. The chairman thanked Takamasa Mori (JAEA) for hosting the meeting. T. Mori welcomed participants and stated that his organization was glad to host WPRS05.

The agenda was structured to include technical Working Party issues which it follows directly including, for the first time, in a special session, the activities related to reactor-based plutonium disposition. One part was devoted to reports from expert groups that now form part of the Working Party, divided in the following 3 areas: Experiments**,** Reactor and Fuel Analysis, and Radiation Transport and Dosimetry. The agenda was approved and is included as Annex I.

The meeting was attended by 20 participants from 11 countries. A list is provided as Annex II.

II. WPRS Technical Session

Reactor and Fuel Analysis

1. Recently published reports and articles from WPRS and related activities

- Physics of Plutonium Recycling - Volume VIII : Results of a Benchmark Considering a High-temperature Reactor (HTR) Fuelled with Reactor-grade Plutonium, 21-JUN-07, 102 pages, ISBN: 978-92-64-99007
- Physics of Plutonium Recycling - Volume IX, Benchmark on Kinetic Parameters in the CROCUS Reactor 21-JUN-07, 94 pages, ISBN: 978-92-64-99020-3
- VVER-1000 Coolant Transient Benchmark - Phase 1 (Vol. 3) (V1000CT-1), Summary Results of Exercise 2 on Coupled 3-D Kinetics/Core Thermal-hydraulics 16-NOV-07, 92 pages, ISBN: 978-92-64-99035-7
- Mixed-oxide (MOX) Fuel Performance – Summary of the Results for the Halden Reactor Project MOX Rods, ISBN: 978-92-64-99019-7.
- The paper "Mixed-Oxide (MOX) Fuel Performance Benchmarks ", was presented at ARWIF-2008 and will be published in the proceedings (authors: L. J. Ott, Terje Tverberg, Enrico Sartori)

2. HTR Benchmarks

The "Shut-down Rod-Worths in LEU-HTR Configurations" Benchmark from PROTEUS was evaluated via the IRPhE Project and is to be included in the fourth edition of the IRPhE Handbook (March 2009). The evaluation was carried out at INL and the reviewed process is being carried out during 2008. In this benchmark the following physical parameters are measured: criticality, control rod and safety rod worths, kinetics, reaction rates and water ingress effects.

Two other experimental benchmarks have been added to the 2008 IRPhE Handbook:

1) Graphite Annular Core Assemblies with Spherical Fuel Elements containing Coated UO_2 Fuel Particles, ASTRA-GCR-EXP-001
2) Initial Critical Configuration of the HTR-10 Pebble-Bed Reactor, HTR10-GCR-RESR-001

The members approved the fact that experimental benchmarks are now thoroughly evaluated and preserved in the handbook.

No new proposals were made for future HTR benchmarks. One topic that has attracted interest concerns the HTR fuel cycle. It was commented that there are few isotopic inventory data for HTR or fuel cycle data. Searches in archives of investigations carried out in the past show that there are some data in the DRAGON project archive, available from the Data Bank and on the AVR reactor. Recent data measured at ECN Petten will be released within the EC Raphael project. Some other data will be produced within the Raphael sub-project PUMA. Contacts should be made with J. Kuijpers to investigate means of accessing such data.

3. Depletion Calculation Benchmark devoted to Fuel Cycle Issues

The depletion calculation benchmark devoted to fuel cycle issues consists of 2 phases: one relative to UOX fuel; the other concerned with MOX fuel. This work is co-ordinated by staff from CEA Cadarache. B. Roque has changed her job assignment and is now replaced by L. San Felice as far as this benchmark is concerned.

The UOX depletion benchmark based on data from measurements is complete. The text was circulated to participants for review and was approved. It will be published after final editing in 2008. For the MOX depletion benchmark only two solutions have been submitted and participants were requested once more to submit further results. The additional results will be integrated into a final report and discussed at the WPRS06 meeting in February 2009.

The MOX benchmark uses a configuration that was actually measured, but the experimental data has not been released. It was suggested that for a next phase the data from the ARIANE project (initiated by Belgonucléaire to obtain extensive data for the isotopic composition of high-burnup LWR fuels) could be used, which have been released to the SFCOMPO - Spent Fuel Isotopic Composition Database, maintained at the NEA Data Bank.

4. Accelerator Driven System Physics Benchmarks

The ADS Beam Trip Benchmark had been divided into three phases: phase I concerned a comparison of codes, phase II the effect of perturbations on the material properties, and phase III the effect of burn-up. The first 2 phases were carried out in the frame of the former WPPT and because the scope of the WP that replaced it (WPFC), had changed, this subject was now under the remit of the WPRS. A specification for the phase III benchmark on burn-up was in preparation, but due to the retirement of one of

the co-ordinators, this work had been delayed. It was recommended that G. Van den Eynde of SCK-CEN be contacted to verify the progress made.

5. Benchmark on CANDU

The possibility of including a sequence of HWR benchmark problems (lattice cell, core, transient, etc.) in the WPRS programme, similar to the studies completed for LWRs, PBMR, FR etc., was suggested at WPRS04. The CANDU Owners' Group (COG) had been contacted through AECL to provide proposals on such benchmarks. The WPRS members would agree to include such HWR benchmarks provided that a detailed proposal could be presented at WPRS05. It was pointed out that in order to attract participation it would be appropriate to propose a study for which some experimental data would be made available. As the representative of Canada was unable to attend, it was agreed to contact him/her to verify whether or not benchmark specifications will be provided by WPRS06.

6. Status of other proposals from past meetings on innovative reactor systems

Minor Actinide Burning in Thermal Reactors

This task was approved by the NSC at its meeting in June 2007. At the time of the WPRS meeting this expert group had not yet started its work. The WPRS decided to start the activity during 2008 in the following way:

1) the NEA Secretariat, with the help of the members, gathers all relevant reports from studies carried out in this field and identifies members of the task force and a chairman (proposals made: ORNL-GNEP Deep Burner Program (Jess Gehin), CEA(Jean-Paul Grouiller): French R&D on the partitioning and transmutation of long-lived radionuclides , JAEA (H. Takano), Nexia (A. Worrall))
2) call for a first meeting via teleconferencing with the members of the expert group, to agree on a schedule, structure of the state-of-the-art report, tasks to be accomplished and deadlines.
3) embed a specific subgroup meeting in the sixth WPRS meeting scheduled for 2-3 February 2009 with the aim of discussing rough drafts of the chapters
4) finalize the draft by the end of 2009
5) publish the report in 2010.

The possibility of including in the report also the results of the PUMA project (Plutonium and Minor Actinides Management in Thermal High Temperature Reactors) is being investigated, as it would provide a more complete picture of the studies carried out so far.

There was no further development of the proposal made during WPRS04 in 2007 on minor actinides recycling in the whole fuel cycle (single & double strata).

A proposal for a benchmark on Pb-Bi cooled fast reactor had been presented by the Russian delegation at WPRS04. That proposal had attracted interest as it helped give a better overall balance for WPRS activities, by bringing in a Gen IV system. No follow-up was made to this proposal as the delegation from Russia did not attend WPRS05.

III. Reactor-based Plutonium Disposition

7. Brief Status Report on the Reactor-Based Plutonium Disposition

This activity was previously held in a separate meeting, though physically and in time, contiguous with the one of the WPRS for reasons of synergies and common work areas. In previous meetings a strong delegation from the Russian Federation has attended. One of the objectives was to create a favourable licensing atmosphere in the USA and Russian Federation through sharing know-how and common exercises with countries having experience in loading MOX fuels in LWRs.

The Russian delegation did not participate this time as a new, different agreement had been signed in November 2007 between the US and Russian Governments.

Under the new agreement, the disposition program now focuses on the use of Russian fast reactors, rather than LWRs, for irradiating MOX fuel fabricated from the 34 metric tons of Russian plutonium, the amount specified in a 2000 US-Russian agreement, known as the Plutonium Management and Disposition Agreement, or PMDA - for disposition in each country.

Russia is to use the BN-600 fast reactor, which has been operating since 1980, and the BN-800, which is under construction. The two reactors also are known as Beloyarsk-3 and Beloyarsk-4, after the site at which they are located.

It was emphasized that the two fast reactors would be operated as "burners" rather than breeders as long as they were part of the disposition mission and that disposition would be accomplished "without creating new stocks of separated weapon-grade plutonium".

Disposition in the BN-600 would begin "in the 2012 timeframe," with BN-800 disposition starting "soon thereafter. Russia's projection is that the BN-800 disposition would start between 2013 and 2015. The GT-MHR may create additional possibilities for speeding up plutonium disposition in the PMDA context in the time frame of 2015-2020.

As concerns the US programme, all 34 metric tons of U. plutonium are to be converted to MOX and irradiated. The construction of the MOX Fuel Fabrication Facility (MFFF), and Pit Disassembly and Conversion Facility (PDCF) is under way. The lead test assemblies (LTAs) were fabricated in Europe with US PuO_2 and NRC approved the use of 4 MOX LTAs in the Catawba-1 Nuclear Power Plant in March and was inserted in May 2005. The 1st cycle ended in November 2006 and no leaks were observed, the poolside examinations were as expected, and the MOX fuel behaved normally, as expected. The 2nd cycle ends in spring 2008 and five second cycle rods will undergo PIE in the autumn of 2008 at ORNL. The 3rd cycle will end in the autumn of 2009 and third cycle rods will undergo PIE in the spring of 2010 at ORNL

8. MOX Fuel Behaviour

Concerning publications, the status is as follows:

- the final report on Mixed-oxide (MOX) Fuel Performance – Summary of the Results for the Halden Reactor Project MOX Rods was published as ISBN 978-92-64-99019-7.
- the paper "Mixed-Oxide (MOX) Fuel Performance Benchmarks ", was presented at ARWIF-2008 and will be published in the proceedings (authors: L. J. Ott, T. Tverberg, E. Sartori)

PRIMO (Rod BD8) MOX Fuel Performance Benchmark

L.J. Ott presented the status of the PRIMO (Rod BD8) MOX fuel performance benchmark submittals, This irradiation was organized by SCK-CEN and BN with 10 additional co-sponsors. The rod was manufactured at BN using the MIMAS process; it was base irradiated in the BR3 reactor of SCK-CEN and then ramp tested in the OSIRIS reactor at Saclay. Six organisations have contributed results by using, for the simulation, nine different codes or versions. Contributors were from Belgium, Korea, Russia, Switzerland, and the USA. The results can be summarized as follows:

The calculated rod average burnups indicate that all participants are modelling the rod geometry and heating history correctly. For the calculated rod mid-plane centreline temperature, prior to ramp, one code predictions are consistently/conservatively higher than other codes; all other code predictions are within a ~125°C band. During the ramp, code predictions with one exception are within a 250°C band. For the calculated rod internal pressure, two distinct groupings are observed (with ~1 MPa difference). For the calculated rod fission gas release, prior to ramp, all code predictions with a few exceptions are close to the experimental value of 0.5%. For the ramp, the majority of results are within an acceptable band of the experimental results, but three of the codes show rather strong deviations.

The PRIMO MOX experimental data has now been compiled and integrated into the IFPE database.

The draft report of PRIMO MOX benchmark is being prepared.

DOE WG-MOX Fuel Irradiation Experiment Benchmark

These experimental data have been compiled and integrated into the IFPE database. They can now be used for a benchmark exercise. The proposal will be made at WPRS06.

Benchmarks for Transient / Accident Conditions

The experiments from the CABRI Rep-Na series with MOX fuel have been identified as candidates for a benchmark studies. A table with the main characteristics of the measured parameters was provided by C. Vitanza. The fuel used was standard MOX as loaded in French power stations. Details on fuel itself, dimensions and grains size, it was suggested, should be retrieved from the published literature. This information should be sufficient for setting up a benchmark

In recent years eight MOX fuel experiments have been released to the IFPE database. The most recent one was contributed by JAEA: LWR MOX Fuel Irradiation Tests - HBWR Irradiation with the Instrument Rig, IFA-514/565. Two more are in process of being acquired. The TFRPD activity has been beneficial in sharing MOX experimental data for model and code validation.

9. Physics of MOX loaded Reactors

The following publications were issued in 2007 relative to reactor physics benchmarks:

- Physics of Plutonium Recycling - Volume VIII : Results of a Benchmark Considering a High-temperature Reactor (HTR) Fuelled with Reactor-grade Plutonium, 21-Jun-07, 102 pages, ISBN: 978-92-64-99007
- Physics of Plutonium Recycling - Volume IX, Benchmark on Kinetic Parameters in the CROCUS Reactor 21-Jun-07, 94 pages, ISBN: 978-92-64-99020-3

Further solutions were delivered for the VENUS-7 benchmark concerning a cold moderated LWR fuel pin lattice with UO_2 and MOX fuel in the core. The progress had been reported at the IRPhE Review group meeting. The comparison of results from calculation against the experimental value will be presented also at PHYSOR-2008. The following codes were used: CASMO-4, DORT, TORT, VARIANT-8, MOCA, APOLLO-2, MCNP-4C, MCNP-5, THREEDANT, TRITON/NEWT. VENUS-7 will be integrated into IRPhE, once the uncertainty analysis is complete.

C. Grove, a MSc student, has carried the evaluation of the VENUS-9 Pu Recycling experiments contributed by SCK-CEN. The VENUS 9 configuration was designed to study boundary effects between zones with different plutonium content and the influence of perturbations at the boundary. The following nuclear parameters were measured: critical mass, spectral index $\sigma_f^{239}/\sigma_f^{235}$, and power distributions within fuel rods and by regions. The evaluation work has now been submitted for review to the IRPhE team and is scheduled to be published in the March 2009 edition of the handbook. The VENUS-17 configuration from the same series was designed for a comparative study of different types of MOX fuel with varying plutonium content and varying plutonium composition, i.e. ^{240}Pu and ^{241}Pu. All together 10 configurations were realised, in which different nuclear parameters were measured, such as critical mass, axial buckling, spectral ratio $\sigma_f^{239}/\sigma_f^{235}$, reactivity worth of different B4C rod configurations, and power distributions. This evaluation will be addressed in the near future. This work is co-ordinated by K. Hesketh, chairman of WPRS.

The benchmark on the KRITZ-2 LEU and MOX critical experiments was already published some time ago with sensitivity and uncertainty analysis relative to cross section data used. In order to include it into the IRPhE Handbook the sensitivities and uncertainties relative to technological uncertainties have recently been added. The current version of the evaluation has been submitted for review to IRPhE. It will be included, after approval, into the March 2009 edition of the IRPhE handbook

Experiments

IFPE: Fuel Performance Experiments Database

Work on expanding the comprehensiveness of the International Fuel Performance Experiments (IFPE) database has continued in cooperation with the IAEA. Some sets of data were updated with complementary or missing data or corrections obtained from user feedback and several new ones were added. At present the database contains data about 1 436 rods or samples. Fourteen new sets of experiments were added in 2007 and early 2008 and one more is being reviewed. So far 13 editions have been issued and requested by 117 establishments in 33 countries. The full set now occupies 4 full DVDs. The NEA participated in the IAEA technical meeting on status and trends in Fuel Performance and Technology (TWGFPT), held on 28 and 29 April 2008. One of the approved new activities concerns "Improvement of Computer Codes Used for Fuel Behaviour Simulation - FUMEX III". This is of particular interest to NEA as it is one activity that allows to obtain feedback on the IFPE database NEA maintains and to augment its scope and number of experiments. The Halden Reactor Project and other organisations have expressed the intention to release more data. One of the topics of investigation will be fuel clad behaviour. Several sets of data relevant to this issue have been added recently to IRPhE, on Zr-2 and Zr-4 from HRP and the AEKI Experimental Database of E110 Claddings under Accident Conditions. The FUMEX-III meeting will be held from 10-12 December 2008 at the IAEA, Vienna. More details on the IFPE database are provided in NEA/SEN/NSC/EG(2008)2 and on http://www.nea.fr/html/science/fuel/ifpelst.html

IRPhE: International Reactor Physics Experiments Evaluation and Database

The second edition, containing data from 21 different experimental series that were performed at 13 different reactor facilities, issued in March 2007, was distributed to 195 authorised requesters from 16

countries. The third edition, issued in March 2008, was distributed to 203 requesters from 23 countries in April 2008. The third edition of the handbook includes evaluated data from the following reactor types:

- Seven liquid metal fast reactors (JOYO, BFS-1, BFS-2, ZPPR, ZEBRA, ZPR, and SNEAK)
- Two gas-cooled reactors (HTR-10 and ASTRA)
- One heavy-water reactor (DCA)
- Three light-water reactors (DIMPLE, CROCUS, and IPEN/MB-01)
- Two pressurised water reactors (VENUS and CREOLE)
- Two VVER reactors (ZR6 and PFACILITY)
- Four fundamental physics evaluations of non-fast-reactor measurements performed on BFS-1 and BFS-2 are also included

Fourteen countries have so far contributed their technical expertise and/or experimental data to the IRPhE Project.

The 2009 (fourth) edition of the IRPhEP Handbook is expected to contain completed benchmarks of VENUS 3. ZPPR 10A, and SNEAK Cores 7A and 7B are currently all in draft form. Additional data are expected from the IPEN/MB01 Reactor and ZPR-6 Assembly 7 (high 240Pu). Evaluation of several other reactor/experimental assemblies are planned or are in progress, some of which may be completed for the 2009 Edition of the IRPhEP Handbook. Included on the list of "in-progress" or "planned" evaluations are the PROTEUS reactor (GCR); FFTF (LMFR); ZPPR Assemblies 9, 13, and 17 (LMFR); VENUS Configurations 1, 7, 9, and 17 (PWR); VHTCR (GCR); HTTR (GCR); KRITZ (LWR); and TCA (LWR).

More details on the IRPhE database and handbook are provided in NEA/SEN/NSC/EG(2008)2 and on http://www.nea.fr/html/dbprog/IRPhE-latest.htm.

SINBAD: Radiation Shielding and Dosimetry Experiments

The SINBAD database currently contains compilations and evaluations of experiments for 42 reactor shielding problems, 27 for Fusion Neutronics Shielding and 15 for Accelerator Shielding cases. Several new experiments were compiled and need final review. Three experiments were related to Reactor Shielding and concerned pressure vessel dosimetry for VVER-440 and 1000. Two concerned fusion neutronics and seven accelerator shielding measurements.

Since the experimental data presently available in SINBAD are of varying quality, a revision and classification of the benchmark experiments according to the completeness and reliability of information will be done to allow the users an easier choice and a better use of the experimental information. In particular all the Time-of-Flight benchmark experiments will be revisited according to the outcomes of the EC CONRAD intercomparison exercise.

The full list of currently distributed SINBAD experiments sorted by application is sprovided in NEA/SEN/NSC/EG(2008)2 and in http://www.nea.fr/html/science/shielding/sinbad/sinbadis.htm.

Radiation Transport and Dosimetry

Pressure Vessel Surveillance and Dosimetry

In the area of Pressure Vessel Surveillance and Dosimetry (PVS) benchmarks, the VENUS-3 benchmark is undergoing final evaluation for inclusion in IRPhE. The evaluation for VENUS-2 containing MOX is undergoing the same process. They are scheduled for inclusion in the fourth edition of the IRPhE Handbook. Although not issued as benchmarks, two experiments carried out at the LR-0 facility for inclusion in SINBAD have been released recently by Nuclear Research Institute Řež and the Institute of

Nuclear Reactors RRC "Kurchatov Institute" relevant for PVS in VVER-440 and VVER-1000. Discussions have taken place to allow the results of the EC FP7 PERFECT (prediction of irradiation damage effects on reactor components) project to be shared internationally through the Data Bank. These datasets and computer codes will be valuable complement to the PVS benchmarks carried out in the NSC framework.

Skyshine benchmark

Concerning the skyshine benchmark (neutron-photon skyshine experiment), because of its importance, the SINBAD compilation has now been thoroughly evaluated and will be part of the ICSBEP benchmarks for criticality alarm systems and will be issued with the September 2008 edition. Once the evaluation is published, the exercise of validating different skyshine codes can start, with the aim to validate the simplified models and codes currently used. New radiation transport codes are being developed and this benchmark will be a valuable test for their performance (e.g. the MAVRIC sequence - MC with automated variance reduction using importance calculations - combining the results of an adjoint calculation from the 3-D deterministic code TORT with MC including the use of first collision source).

Computational radiation transport benchmarks

The Expert Group on computational radiation transport benchmarks has started to submit solutions to the problem on "Benchmarking the Accuracy of Solution of 3-Dimensional Transport Codes and Methods over a Range in Parameter Space". An exact solution based on long Monte Carlo calculations for 729 cases will be compared against the deterministic solutions. This benchmark is designed to elucidate important issues necessary to judge the quality of numerical solutions obtained with particle transport software. Several solutions have been submitted so far and a status report was presented at the Joint Benchmark meeting at the ANS meeting in Anaheim this June. Several papers were submitted for presentation at the PHYSOR-2008 conference. The final report and the reference solution will be published in the first months of 2009. In the framework of training courses related to hands on use of computer codes the Analytical Nuclear Engineering Benchmark Library was developed by Prof. B. Ganapol. The motivation for this is to promote analytical thinking which is in danger of being lost to Monte Carlo. It provides an intellectual challenge for the younger generation, it demonstrates the usefulness of analytical investigation, and is an essential element for continuous improvement of transport methods development. This effort is a contribution to the promotion of preservation of know-how in the nuclear community and for code verification and validation. This Handbook will be published under the aegis of the Data Bank with the title "Analytical Benchmarks for Nuclear Engineering Applications - Case Studies in Neutron Transport Theory" in the summer of 2008.

SATIF- Shielding of Accelerators, Targets and Irradiation Facilities

The Ninth Workshop on Shielding Aspects of Accelerators, Targets and Irradiation Facilities (SATIF-9) took place on 21-23 April 2008 at Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA in connection with the joint 11th International Conference of Radiation Shielding and Radiation Protection (ICRS11 & RPS-2008). It was hosted by the Spallation Neutron Source Facility in Oak Ridge, TN, USA. It was attended by 36 participants from 10 countries. Twenty papers were presented and discussed covering the following topics:

- Source Terms and Related Topics
- Accelerator Activation Problems
- Shielding in Medical and Industrial Accelerator Applications
- Benchmarking - Calculations and Results
- Dose and Related issues
- Status of Computer Codes, Cross Sections and Shielding Data Libraries.

The final discussion concerned progress in shielding measurement campaigns and shielding benchmarks, effective shielding materials and dosimetry issues, in particular action for international co-operation, including the finalisation of the accelerator shielding handbook. A visit to SNS provided insight into the capacity of this facility and the many experimental programmes it hosts.

As to the status of the handbook on "Accelerator Shielding", designated editors N. Mokhov and P. Vaz, the general structure was refined, several chapters were delivered and for others progress is being achieved. Some delay has been introduced in the publication schedule and the publication date has been set now for May 2010 when the next SATIF10 meeting will be hosted by the CERN in Geneva.

The radiation shielding community has benefited through this series of workshops from the co-operation which takes place under the aegis of SATIF (e.g. new and improved modelling codes and user interfaces, enlarged experimental database, etc.). The group has established itself as the international forum for addressing priority issues in this area. SATIF contributes shared research results to emerging priority areas.

Topics identified for the coming years are as follows

- Activation and dose rate estimation
- Benchmarking
- Model development / improvement
- Accelerator Shielding handbook(s)
- "Heavy Ion Handbook" published
- A more general handbook on "shielding and dosimetry" is required
- Experiments and their interpretation
- Proposal for use of existing facilities

The NSC is asked to endorse the continuation of the SATIF activity and the holding of the workshops.

Expert Groups on Reactor and Fuel Analysis (*Coupled Neutronics/Thermal-hydraulics and Core-Plant benchmarks*)

PBMR Coupled Neutronics/Thermal Hydraulics Transient Benchmark

The fourth workshop OECD/NEA-NSC PBMR Coupled Neutronics/Thermal Hydraulics Transient Benchmark - the PBMR-400 Core Design – was held at NEA Headquarters, Issy-les-Moulineaux, France, from 21 to 25 January 2008 with the following scope:

PART I: Workshop on the DIREKT code system (21-23 January 2008): the DIREKT code, is being developed at the Research Centre Jülich (FZJ) in Germany, and models steady-state and transient pebble bed reactor thermal hydraulics.

PART II (24-25 January 2008): discussion of the OECD PBMR-400 Benchmark transient benchmark. The focus areas were as follows:

1. Steady State results finalization
   The final results sent for the steady-state cases were represented and discussed to facilitate the final comparisons at the meeting. The aim is to publish all the steady-state results in a PHYSOR-2008 summary paper after this meeting.
2. Transient Results
   The focus of the workshop was the comparisons of all the transient results. Exercises 1 to 6 results were presented and discussed.

A detailed summary record was distributed as NEA/NSC/DOC(2008)7 containing a list of tasks and outstanding issues. All updated results are due for end of June 2008. A last meeting is scheduled for Sunday, 14 September 2008 at Interlaken in conjunction with the Physor-2008 conference. At this meeting the latest results will be reviewed and conclusions and benchmark recommendations agreed on. A special session will be devoted to this benchmark during the Physor-2008 conference and all papers have been written and submitted for the proceedings. It was decided to publish the results of the different parts of the benchmark in a single volume, rather than several. The plan is to issue it in March-April 2009. The full data of the benchmark, with the specification and all data characterising it, and all results will be available on a DVD to participants and to researchers.

VVER-1000 Coolant Transient Benchmarks Phase 2 (V1000CT-2)

The overall objective is the assessment of coupled N/TH codes for safety analysis of VVER (RIA) and the specific objective is the testing and validation of vessel mixing models in CFD and system codes (mixing identified as unresolved issue in previous benchmarks). It is based on actual plant data (geometry and experiments) from Kozloduy-6 such as the main coolant pump start-up experiment, the steam generator isolation experiments, the measured steady states of the MSLB (main steam-line break reference core). A multi-scale approach/validation is used for 3D core neutronics, 3D core and vessel thermal-hydraulics, including CFD models, and system thermal-hydraulics.

Eleven organizations from seven countries have participated.

The lessons learned are as follows:

The post-test calculations of VVER plant mixing experiments demonstrated the capability of CFD and system codes to model single-phase vessel mixing. The CFD codes were able to accurately reproduce the fuel assembly inlet and reactor outlet temperatures as well as the azimuthal rotation of the main loop flows for main steam-line break (MSLB) mixing patterns (thermal mixing, main coolant pump trip), when using actual (as fabricated) RPV(reactor pressure vessel) geometry and not the conceptual design geometry and the use of the appropriate treatment of turbulence. The mixing test calculations show some differences between CFD results and plant data for the core inlet pressure and flow distributions (observation based on other experiments at Kozloduy-5). This observation requires further analysis. Coarse-mesh models in system codes can reasonably simulate the fuel assembly inlet and reactor outlet temperatures, with max error at the core inlet ~ 3-4 K, depending on the nodalization. The calculation is sensitive to the number of angular sectors. The recommended minimum is 18, or $\geq$ 24 for enhanced modeling of the swirl in the lower plenum. Certain knowledge of the vessel thermal hydraulics could improve the quality. The MSLB simulations with the validated mixing models clearly display the significance of core-vessel thermal-hydraulic refinement for the prediction of sector power, hot node location and peaking factor.

The status of the publications is as follows: three volumes of the phase I (V1000CT-1) have been published and the fourth on Best-Estimate Coupled 3-D Core/Thermal-Hydraulic Plant Transient Modeling is being finalized; for phase II (V1000CT-2), two volumes were submitted for printing, but have been blocked at the NEA publication section for the last six months. Volume III on Comparison of Computational Fluid Dynamics and Coarse Mesh Calculations with Measured Data is being finalized and for Volume IV and V on MSLB Coupled 3D Neutronics / Vessel TH Simulation and MSLB Best Estimate Coupled Simulation rough drafts are available. The merging of Volume III, IV and V into 1 volume is being investigated.

The NUPEC BWR Full-size Fine-mesh Bundle Tests (BFBT) Benchmarks

The fifth Workshop (BFBT-5) OECD/NRC Benchmark based on NUPEC BWR Full-size Fine-mesh Bundle Tests (BFBT) was held at Garching, Germany on 31 March and 1 April 2008, hosted by the Gesellschaft für Anlagen und Reaktorsicherheit (GRS) Germany. It was held in conjunction with other meetings, in order to facilitate co-ordination and sharing of work. The meetings concerned are the second workshop for the OECD Uncertainty Analysis in Modelling (UAM) Light Water Reactor (LWR) benchmark, which took place from 2 to 4 April 2008; in parallel with the BFBT-5 meeting also the annual meeting of Working Group D involved in VVER reactor dynamics and safety research was held at the same premises.

The BFBT benchmark is divided into two parts (phases), each part consisting of different exercises:

- Phase I – Void Distribution Benchmark

  Exercise 1 (I-1) – Steady-state sub-channel grade benchmark
  Exercise 2 (I-2) – Steady-state microscopic grade benchmark
  Exercise 3 (I-3) – Transient macroscopic grade benchmark
  Exercise 4 (I-4) – Uncertainty analysis of the steady state sub-channel benchmark

- Phase II – Critical Power Benchmark

  Exercise 0 (II-0) – Pressure drop benchmark
  Exercise 1 (II-1) – Steady-state benchmark
  Exercise 2 (II-2) – Transient benchmark
  Exercise 3 (II-3) – Uncertainty analysis of the steady state benchmark

Specific items addressed at the fifth workshop were:

- Presentation and discussion of summary of comparisons of final submitted results for Exercise 1 of Phase I (I-1); for Exercise 0 of Phase II (II-0); and for Exercise 1 of Phase II (II-1)
- Presentation and discussion of comparison of final submitted results for exercise 2, 3 and 4 of phase I
- Presentation and discussion of comparison of final submitted results for Exercise 2 of Phase II and of preliminary uncertainty results for Exercise 3 of Phase II,
- Preparing a special issue in a journal with participants' BFBT papers

In summary, at the BFBT-5 workshop 25 presentations were made by the benchmark team and by participants. A special journal issue of Nuclear Engineering and Design is planned for the OECD/NRC BFBT benchmark. Papers have to be submitted by December 2008, and the reviewing is expected by March, 2009.

The action items and schedule of benchmark activities were discussed, which include the deadline for submitting final results for phases I & II by end of December 2009 and for the corresponding sensitivity / uncertainty analysis by end of March 2009.

The sixth workshop (BFBT-6) will be held on 27 and 28 April 2009, and will be hosted by the Pennsylvania State University (PSU), USA. Also the BFBT-6 workshop will be held in conjunction with other meetings, in order to facilitate co-ordination and sharing of work. The meetings concerned are the third workshop on the OECD Uncertainty Analysis in Modelling (UAM) Light Water Reactor (LWR) benchmark (UAM-3), which will take place from 29 April to May 1, 2009, in parallel with the BFBT-6

meeting. Also the first workshop on the Kalinin-3 VVER coupled code benchmark will be held at the same premises. The objectives of the next workshop (BFBT-6) will be the following: the Kalinin-3 data will be of particular value in support of uncertainty analysis.

The BFBT database contains a wealth of detailed experimental data and their release and very large participation in the benchmarks starts to show clear benefits to detailed 2-phase-flow modeling. The donors of the data have expressed the wish that this work on 2-phase flow not be stopped, as the momentum gained is used to really advance work in this field. The NSC is asked to discuss this and approve continuation of this series of benchmarks.

A detailed summary record was distributed as NEA/NSC/DOC(2008)5

Uncertainty Analysis in Best-Estimate Modelling (UAM)

The second workshop on the OECD Benchmark for Uncertainty Analysis in Best-Estimate Modelling (UAM) for Design, Operation and Safety Analysis of LWRs (OECD LWR UAM Benchmark) was held from 2 to 4 April 2008, hosted by the Gesellschaft für Anlagen und Reaktorsicherheit (GRS), Garching, Germany

The justification for this activity is as follows: in addition to LWR best-estimate calculations for design and safety analysis, the different aspects of uncertainty analysis in modeling (UAM) are to be further developed and validated on scientific grounds in support of its performance. There is a need for efficient and powerful analysis methods suitable for such complex coupled multi-physics and multi-scale simulations. The proposed benchmark sequence addresses this need by integrating the expertise in reactor physics, thermal-hydraulics and reactor system modelling as well as uncertainty and sensitivity analysis, and will contribute to the development and assessment of advanced/optimised uncertainty methods for use in best-estimate reactor simulations. Such an effort can be undertaken within the framework of a program of international co-operation that would benefit from the coordination of the NEA/NSC and all participants by interfacing with the CSNI activities.

The general frame of the OECD LWR UAM benchmark consists of three phases with three exercises for each phase:

Phase I (Neutronics Phase)

- Exercise 1 (I-1): "Cell Physics" focused on the derivation of the multi-group microscopic cross-section libraries
- Exercise 2 (I-2): "Lattice Physics" focused on the derivation of the few-group macroscopic cross-section libraries
- Exercise 3 (I-3): "Core Physics" focused on the core steady state stand-alone neutronics calculations

Phase II (Core Phase)

- Exercise II-1: Fuel thermal properties relevant for transient performance
- Exercise II-2: Neutron kinetics stand-alone performance (kinetics data, space-time dependence treatment, etc.)
- Exercise II-3: Thermal-hydraulic fuel bundle performance

Phase III (System Phase)

- Exercise III-1: Coupled neutronics/thermal-hydraulics core performance (coupled steady-state, coupled depletion, and coupled core transient with boundary conditions)
- Exercise III-2: Thermal-hydraulics system performance

- Exercise III-3: Coupled neutronics kinetics thermal-hydraulic core/thermal-hydraulic system performance

The expected impact and benefits of the OECD LWR UAM benchmark activity for LWR safety and licensing are summarised in "Technology Relevance of the Uncertainty Analysis in Modelling Project for Nuclear Reactor Safety", NEA/NSC/DOC(2007)15. An update of this report was presented at UAM-2. It is the intention to follow in this series of benchmarks the calculation scheme for coupled calculations for LWR design and safety analysis established in the nuclear power generation industry and regulation. The specification document that covers Phase I (which includes the first 3 Exercises) was distributed to the participants - "Benchmark for Uncertainty Analysis in Modelling (UAM) for Design, Operation and Safety Analysis of LWRs. Volume 1 – Specification and Supporting Data for the Neutronics Cases (Phase I) Version 1.0", NEA/NSC/DOC(2007)23.

The workshop held an in-depth discussion on the specification and support data for Phase I of the UAM LWR benchmark, preliminary results of Phase I, output parameters and format for Phase II, priorities for Phases II and III, and the proposed work plan and time schedule for the UAM LWR benchmark activities. The participants presented their experience and expertise in uncertainty and sensitivity analysis of LWRs.

The workshop was attended by 53 participants from 30 organisations in 18 countries. The group of participants in this benchmark includes experts from different fields namely in thermal-hydraulics, neutronics and uncertainty analysis. Several expert groups had been formed previously each addressing specific benchmark problems and, now, they are combining in this long term effort aimed at establishing best-estimate simulation methods with systematic uncertainty analysis across different phenomena (multi-physics) and different scales (multi-scales).

In summary, at the UAM-2 workshop a total of 33 presentations were made by the benchmark team and by participants. The actions agreed to include the preparation of a special session on this work during the Mathematics & Computation 2009 Conference at Saratoga Springs, NY, next May, the revision of the phase I benchmark specification in the light of first experience and discussion, preparation of templates and datasets for the benchmark exercises, submission of results of phase I exercises and organisation of the third workshop (UAM-3), hosted at the end of April 2009 by the Pennsylvania State University, USA.

The objectives of the next workshop (UAM-3) will be discussion of submitted results of Phase I, discussion of draft Specification for Phase II and discussion of priorities for Phase III.

A detailed summary record was distributed as NEA/NSC/DOC(2008)6.

Comments by WPRS:

It was concluded that the expert groups on Coupled Neutronics/Thermal-hydraulics and Core-Plant benchmarks and Uncertainty Analysis in Modelling have carried out so far a large amount of relevant work and it is recommended that they continue along these lines and report annually at WPRS meetings.

Reports from other WPs

This meeting was not held in conjunction with the Working Party on Scientific Issues of the Fuel Cycle (WPFC), but the recent structure was presented:

The WPFC deals with scientific issues in various existing and advanced nuclear fuel cycles, including fuel cycle physics, associated chemistry and flowsheets, the development and performance of fuels and materials, and accelerators and spallation targets.

The structure in expert groups is as follows:

- Expert Group on Heavy Liquid Metal (HLM) Technology
- Benchmarking of Thermal-Hydraulic Loop Models for Lead-Alloy Cooled Advanced Nuclear Energy Systems (LACANES)
- Expert Group on Chemical Partitioning
- Flowsheet studies
- Separations criteria
- Expert Group on Fuel Cycle Transition Scenarios Studies

The next meeting is scheduled for 30 April 2008. The WPRS members have expressed the wish that in future the WPFC and WPRS are held in conjunction with each other.

12. Sponsorship of workshops and conferences

Meetings / conferences scheduled in co-operation with the NEA NSC and of interest to WPRS

25-30 May 2008
13th International Symposium on Reactor Dosimetry
Akersloot, The Netherlands

14-18 September 2008
PHYSOR-2008- Nuclear Power: A Sustainable Energy Resource
Interlaken, Switzerland

27-29 October 2008
High Temperature Reactor Workshop (HTGR Issues, Physics, Technology, Applications – Fuel, Safety – Materials – Gas Turbine), Oarai Japan

3-7 May 2009
International Conference on Advances in Mathematics, Computational Methods, and Reactor Physics M&C-2009, Saratoga Springs, NY. USA

10 – 14 May 2009
2009 International Congress on Advances in Nuclear Power Plants (ICAPP '09), Tokyo, Japan

6 - 10 September 2009
GLOBAL 2009 — 8th Global Congress on Nuclear Fuel Cycle: Sustainable Options and Industrial Perspectives; Paris, France,

October 2010
Supercomputing in Nuclear Applications SNA-2010 and Monte Carlo 2010
Tokyo, Japan

The *ARWIF-Workshop*

The Fourth Workshop on Advanced Reactors with Innovative Fuels (ARWIF-2008) was held from 20 – 22 February 2008 at the Monju site in Tsuruga and In Fukui, Japan, hosted by JAEA. The opening session was held at the Monju site followed by a visit of the Monju site and the Fast Reactor Training Facility.

The scope of the workshop comprised reactor physics, fuel performance and fuel material technology, core behaviour and fuel cycle of advanced reactors with different types of fuels or fuel lattices. Emphasis was put on innovative concepts and issues related to the reactor and fuel and in particular on fast

reactor issues. The objectives were to exchange information on R&D activities and to identify areas and research tasks where international co-operation can be strengthened.

The workshop was opened by Kazumoto Ito, Executive Director of JAEA. The General Chairman was Takamasa Mori, delegate from Japan in the NSC and WPRS. It was attended by 72 experts from 11 countries. In order to familiarize high school students with international co-operation, a selected group of 35 from the advanced science high school were invited to attend.

During the opening session it was announced that the restart of the Monju reactor was officially approved for October 2008. Forty presentations were made and a panel chaired by K. Hesketh (Chair of NSC-WPRS) concluded on direction and needs for experiments and computer simulation in advanced reactors: 1) H. Sekimoto (TIT): Sodium-cooled Fast Reactors and their Fuels, 2) J. Gehin (ORNL): Very High Temperature Reactors and their Fuels, 3) M. Saito (TIT): Proliferation Resistance in Advanced Reactors and their Fuels, 4) K. Hesketh (Nexia): Advanced Light Water Reactors and their Fuels, 5) E. Sartori (OECD/NEA): International Co-operation in Advanced Reactors. For details relative to the presentations and discussion at the ARWIF Panel see Annex III.

The proceedings will be prepared during 2008.

During the technical sessions, an exhibition and demonstration was held on the "Research and Test Facilities Database (RTFDB)" set up by the NEA Expert Group on Needs of R&D Facilities in Nuclear Science, giving the status of research and test facilities in the field of nuclear science and technology. Many participants have tried the database and have expressed great interest in this NEA product. The corresponding brochure was widely distributed.

ICRS-11 and RPS-15 conference

The Eleventh International Conference on Radiation Shielding, held from 13 to 18 April 2008 at Callaway Gardens, Pine Mountain, Georgia, celebrated the 50th anniversary of the first international symposium on radiation shielding. The conference started with a panel devoted to "Shielding: a look back" and concluded with a panel "Shielding: looking to the future". Norman M. Schaeffer attended the first panel, together with Betty F. Maskewitz, former Director of RSIC and Donald Dudziak, Fellow of LANL. Conference participants, mostly from the younger generations, learned about the early days of radiation shielding, when nuclear data was poor or nonexistent and many radiation shielding methods and codes where in their infancy, and the effort made then to develop this field of science and engineering. The panel with a look at the future, discussed the new applications requiring shielding investigations using high power machines such as high intensity lasers, plasma and accelerators, the importance of benchmark experiments, shield design based on effective dose rather than other dose estimators such as ambient dose equivalent. The effect of future computer chip architecture on the performance of current generation computer codes was discussed, in particular the need to write new radiation transport software for the new architectures to be effective. The increased number of papers on Monte Carlo shows that this method has gained application power, though competes only in certain fields with deterministic methods. There is a need to better understand the limitations of approximations in the codes and to ensure training and that competence in the use of codes is developed.

The conference was attended by 290 participants from 22 countries, half of which were from North America, 30% from Europe and 20% from Asia. In all, some 300 papers were presented in 50 sessions and seven tutorials were held on computer codes of importance for radiation shielding and dosimetry.

13. Report to NSC

The Chairman will report the progress and future programme of work at the next session of the Nuclear Science Committee, scheduled for 25-27 June 2008.

14. Date and Place of Sixth Meeting

The next meeting is scheduled for **26-27 February 2009**[1] and will be held at OECD/NEA Headquarters at Issy-les-Moulineaux. One specific session will be devoted to Minor Actinide Burning in Thermal Reactors. The members have expressed the wish that the next WPFC meeting be held again in conjunction with the WPRS to strengthen co-ordination and co-operation.

[1] This date was modified from 12-13 February 2009 to 26-27 February 2009 in order to have WPRS and WPFC meetings held in conjunction

Annex I

Agenda

I. Introductory Remarks

a) WPRS matters
   i. Report from the June 2007 NSC meeting and December NSC Bureau meeting
   ii. Organisation of work by sub-groups and reporting to WPRS
b) Objectives of the Meeting
c) Introduction of Participants
d) Review and Approval of the Agenda

II. WPRS Technical Session

1. Recently published reports and articles from WPRS and related activities

2. HTR Benchmarks

- Status of HTR benchmarks
  - "Shut-down Rod-Worths in LEU-HTR Configurations" Benchmark]
- Proposals for future HTR benchmarks
  - HTR fuel cycle

3. Depletion Calculation Benchmark devoted to Fuel Cycle Issues

- Final Report on UOx Phase I Results
- Specification and schedule for the Phase II (MOX) benchmark

4. Accelerator Driven System Physics Benchmarks

- Status of benchmark activities

5. Benchmark on CANDU

6. Status of other proposals from past meetings on Innovative reactor systems

- Status of Minor Actinide burning in LWRs
- Minor actinides recycling in the whole fuel cycle (single & double strata)

III. Reactor-based Plutonium Disposition Session

7. Brief Status Report on the Reactor-Based Plutonium Disposition

8. MOX Fuel Behaviour

- Publications from previous work
- Comparison against experiments
  - PRIMO (Rod BD8) MOX Fuel Performance Benchmark
  - DOE WG-MOX Fuel Irradiation Experiment Benchmark
  - Benchmarks for Transient / Accident Conditions

9. Physics of MOX loaded Reactors

- TFRPD MOX benchmarks in IRPhE
- Other MOX benchmarks

IV. Reports from the WPRS subgroups and activities

10. Experiments / Databases

- Report from the IRPhE Project(Reactor Physics Experiments)
- Status of IFPE Database (Fuel Performance Experiments)

- Status of SINBAD Database (Shielding & Dosimetry Experiments)

11. Reports from Sub-groups and other Groups

- ***Radiation Shielding, Dosimetry and Transport***
  - Status of radiation skyshine benchmark
  - Planned SATIF9 Workshop and Radiation Protection and Shielding Topical 2008
  - Calculational Radiation Transport Benchmarks
    - Status of Suite of Benchmark Problems for 3-D Transport Codes
    - Status of the Analytical Nuclear Engineering Benchmark Library / training course
- ***Coupled Neutronics/Thermal-hydraulics and Core-Plant benchmarks***
  - PBMR-400 Coupled Neutronics/Thermal-fluid-dynamics benchmarks]
  - 3D Coupled Neutronics/Thermal-hydraulics benchmarks]
  - Uncertainty Analysis in Modelling
    - Status of benchmark
    - Phase I (Neutronics) Specification
- ***Status of Report on "Needs of Research and Test Facilities in Nuclear Science"*** *and* ***RTFDB***

12. Sponsorship of workshops and conferences

- ARWIF-2008 – Japan
- Physor-2008 (Interlaken, Switzerland)
- Other Conferences

13. Report to NSC

14. Date and Place of Sixth Meeting

Annex II

List of Participants

BELGIUM
D'HONDT, Pierre Joseph — Tel: +32 14 33 25 95
Deputy General Manager — Fax: +32 14 31 89 36
SCK-CEN — Eml: pierre.dhondt@sckcen.be
Boeretang 200
B-2400 MOL

FINLAND
LEPPANEN, Jaakko Ilman — Tel: +358 20 722 5049
VTT (Technical Research Centre of Finland — Fax: +358 20 722 5000
P.O.Box 1000 — Eml: jaakko.leppanen@vtt.fi
FI-02044 VTT

FRANCE
RUGGIERI, Jean-Michel — Tel: +33 4 42 25 7052
CEA/CE-CADARACHE — Fax: +33 4 42 25 7009
DEN/DER/SPRC/LEP — Eml: jean-michel.ruggieri@cea.fr
Bat. 230
F-13108 ST. PAUL LEZ DURANCE CEDEX

HUNGARY
KERESZTURI, Andras — Tel: +36 1 392 2297
Reactor Analysis Department — Fax: +36 1 395 9293
KFKI Atomic Energy Research Institute — Eml: kere@sunserv.kfki.hu
Konkoly Th. 29-33
P.O. Box 49
H-1525 BUDAPEST 114

JAPAN
KITABATA, Takuya — Tel: +81 770 21 5060
Group leader, International Co-operation — Fax: +81 770 21 5090
Policy Planning and Co-ordination Depart. — Eml: kitabata.takuya@jaea.go.jp
Tsuruga Head Office
Japan Atomic Energy Agency (JAEA)
Kizaki, Tsuruga-shi, Fukui-ken, 914-8585

MORI, Takamasa — Tel: +81 29 282 6434
Senior Principal Researcher, — Fax: +81 29 282 6122
Nuclear Science & Technology Directorate, — Eml: mori.takamasa@jaea.go.jp
Japan Atomic Energy Agency
Tokai-mura,
Naka-gun, Ibaraki-ken, 319-1195

YAMAJI, Akifumi — Tel: +81 29 267 1919 ex. 6487
Innovative Water Reactor Design Group — Fax:
Advanced Nuclear System Research and Deve — Eml: yamaji.akifumi@jaea.go.jp
Japan Atomic Energy Agency (JAEA)
4002 Narita-cho O-arai-machi
Ibaraki 311-1393

ZUKERAN, Atsushi — Tel: +81 29 243 3990
Ph.D., Senior Researcher — Fax: +81 29 243 3990
310-0851 Senba-Cho, — Eml: azukeran@k8.dion.ne.jp
Mito-Shi
Ibaraki-ken 310-0851

KOREA (REPUBLIC OF)

*	SONG, Jae Hoon Master's student Korea Advanced Institute of Science and Technology (KAIST) 373-1 Kusong-dong, Yusong-gu Daejeon, Korea, 305-701	Tel: +82 42 869 3859 Fax: +82 42 869 5859 Eml: jh_song@kaist.ac.kr

NETHERLANDS

*	KUIJPER, Jim C. Nuclear Reactor Physics NRG Westerduinweg 3 Postbus 25 NL-1755 ZG PETTEN	Tel: +31 224 56 4506 Fax: +31 224 56 8490 Eml: kuijper@nrg.eu

SWITZERLAND

	FERROUKHI, Hakim Nuclear Energy and Safety Research Depart Laboratory for Reactor Physics and System Behaviour Paul Scherrer Institut CH-5232 Villigen PSI	Tel: +41 0 56 310 4062 Fax: +41 0 56 310 2327 Eml: Hakim.Ferroukhi@psi.ch

UNITED KINGDOM

	HESKETH, Kevin Nexia Solutions B709 Springfields Salwick Preston Lancashire PR4 0XJ	Tel: +44 1772 76 23 47 Fax: +44 1772 76 24 70 Eml: Kevin.W.Hesketh@nexiasolutions.com

UNITED STATES OF AMERICA

	DE OLIVEIRA, Cassiano University of New Mexico Department of Chemical &Nuclear Engineeri MSC01 1120; 209 Farris Engineering Center Albuquerque, NM 87131	Tel: +1 505 277 5661 Fax: +1 505 277 5433 Eml: cassiano@unm.edu
	GEHIN, Jess C. Oak Ridge National Laboratory Bethel Valley Road P.O.Box 2008 OAK-RIDGE, Tennessee 37831-6172	Tel: +1 865 576 5093 Fax: +1 865 574 9619 Eml: gehinjc@ornl.gov
	OTT, Larry J. Nuclear Science and Technology Division Oak Ridge National Laboratory MS-6167, P.O. Box 2008 Oak Ridge, TN 37831-6167	Tel: +1 865 574 0324 Fax: +1 865 574 2032 Eml: ottlj@ornl.gov
	PETROVIC, Bojan Georgia Institute of Technology 771 Ferst Dr. NW Atlanta, GA 30332-0405	Tel: +1 404 894 8173 Fax: Eml: Bojan.Petrovic@gatech.edu
*	SPELLMAN, Donald J. OAK RIDGE NATIONAL LABORATORY Nuclear Science & Technology Division PO BOX 2008, Bldg. 5700, MS-6165 OAK RIDGE TN 37831-6165	Tel: +1 865 574 7891 Fax: +1 865 574 8216 Eml: spellmandj@ornl.gov

* TAIWO, Temitope A.
Nuclear Engineering Division
Argonne National Laboratory
9700 South Cass Avenue
Building 208
ARGONNE, IL 60439-4842
Tel: +1 630 252 1387
Fax: +1 630 252 4500
Eml: Taiwo@anl.gov

ULSES, Tony P.
Office of Research
USNRC
MS T 10 K 3
Washington, DC 20555
Tel: +1 301 415 6002
Fax: +1 301 415 5160
Eml: apu@nrc.gov

International Organisations
SARTORI, Enrico
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

* regrets, not to have been able to attend

Annex III

Summary of ARWIF 2008 Discussion Panel 22 February 2008, Fukui
(prepared by Chair: K. Hesketh)

The final session of ARWIF 2008 was a panel discussion held in a plenary session.

The title was "Direction and Needs in Advanced Reactors and Fuels (experiments, computer simulations…)". Five presentations were made to set the scene for an open discussion:

- Sodium cooled fast reactors and their fuels by Prof. Sekimoto from Tokyo Institute of Technology
- Protected plutonium production by transmutation of minor actinides for sustainable growth by Prof. Saito from Tokyo Institute of Technology
- Very high temperature reactors and their fuels – directions and needs by Dr Taiwo of Argonne National Laboratory, but presented on his behalf by Dr Gehin of Oak Ridge National Laboratory
- Advanced light water reactors and their fuels by Mr Hesketh of Nexia Solutions
- International co-operation in advanced reactors by Dr Sartori of OECD/NEA

Prof Sekimoto made a general presentation on sodium cooled fast reactors and fuels in the context of the novel CANDLE reactor, which can possibly be designed for very long life cores. The most important feature of the CANDLE reactor is not long life but the very high natural uranium utilization without reprocessing and its potential to resolve all the problems of sustainability, safety, waste, and nuclear proliferation at the same time. The CANDLE fuel configuration extends over a larger axial height than the active core and implies adding a few cm/year and the active fissioning region is deliberately confined initially to the top part of the fuel. As the fuel depletes, the active fissioning region migrates towards the bottom of the fuel stack, with a lower burning front delineating the transition from unburned fuel from the active core and a depletion front at the top above which is the spent fuel region.

Prof Sekimoto identified various technology options that have been examined for CANDLE, including metallic, nitride and oxide fuels, which are preferred in that order. Sodium cooling is one of the options considered. Particular technology issues that need to be considered are the need for very high neutron economy, very high integrity fuel materials (which are required to retain their integrity to very high burnups and remain in the core after the active core region has migrated to lower levels) and the need for a very large gas plenum volume to retain the fission gases. These are generic issues that are also shared with conventional fast neutron systems, that will require international collaborative research and development efforts. These problems may be solved by changing cladding materials at certain levels of fuel burning. Up to 40% of charged natural or depleted uranium can be fissioned, without any need for conventional reprocessing.

Prof Saito's presentation highlighted the increased importance of proliferation resistance and the need to design into future fuels cycles inherent mechanisms to make nuclear materials less attractive for weapons purposes. The presentation described an approach to denaturing plutonium using minor actinides, such as Np-237 to ensure that the plutonium produced in the fuel cycle contains a high concentration of heat-producing Pu-238.

Prof Saito emphasised the need to develop a common methodology that can be used to assess the proliferation resistance of different technologies, based on commonly agreed metrics. Prof Saito presented a metric that determines the potential weapons usability of plutonium in a fuel cycle. Multi-national collaboration will be needed to develop a methodology and metric system that will be internationally recognised.

Dr Taiwo's presentation explained the background to High Temperature Reactor (HTR) research and the direction of the US Next Generation Nuclear Plant (NGNP) research and development. The NGNP R&D programme is organised under the headings fuels; graphite; design methods and high temperature materials:

Fuels

- Fuel irradiation and qualification
- Pilot scale coating
- Upgrades for post-irradiation examination

Graphite

- Complete fabrication, operation and assembly mock-up for graphite experiments
- Sample characterisation

Design methods

- Enhancement to core physics tools
- Modelling of air ingress
- Development of benchmarks
- Test plans for RCCS experiments

High temperature materials

- Creep, creep/fatigue and environmental testing of intermediate heat exchanger candidate materials
- High temperature material characterisation and testing

Dr Taiwo's presentation then turned to the Deep Burn Modular Helium Reactor (DBMHR) concept, its goal to consume 50% of transuranic (TRU) fuel in one irradiation pass. The experimental requirements identified were:

- Production of high quality TRU fuel for irradiation testing
- Flowsheet identification and demonstration of reprocessing methods for spent fuel
- Assessment of waste management and options for HTR spent fuel
- Development of first principles modelling of TRU TRISO coated particle fuels and multi-physics evaluation methods.

Dr Taiwo's presentation ended by identifying the requirements for international collaboration on HTR under the headings of facilities; experiments and analytical tools:

Facilities

- Irradiation experiments
- Sharing of post-irradiation experiments data
- Development of advanced post-irradiation examination techniques
- Helium test facility

Experiments

- Fuel power ramp tests to establish fuel design criteria
- Decay heat removal tests
- Water and air ingress tests
- Fission product transport tests

Analytical tools

- Exchange of advanced tools and joint benchmarking

Mr Hesketh's presentation began with a summary of the historical development of LWRs and their fuels. The presentation identified the following future driving factors: high fuel reliability; core uprating; improved operating margins; higher burnups; recycle of uranium, plutonium and possibly minor actinides and improved proliferation resistance. The key constraining factors for advanced LWR fuel development identified were: fission gas release modelling and understanding; fission gas release mitigation; cladding corrosion; the 5/.0 w/o criticality limit in fuel fabrication plants; pellet clad interaction at high burnups; clad stress (a specific issue in the UK) and accident behaviour at high burnups in LOCA and RIA. The presentation questioned whether the core internals constraints of next generation LWRs should be modified to accommodate longer plenum volumes and/or allow enhanced rating capabilities.

Mr Hesketh's presentation concluded by identifying the international experimental requirements for advanced LWR fuels: accident behaviour (LOCA & RIA); fission gas release; pellet clad interaction; compatibility of advanced fuels with cladding materials; rim effect behaviour and inert matrix fuels properties and irradiation behaviour. All these were areas where international collaboration would be required. Under simulation requirements, the areas identified where international collaborations might best contribute are: improved mechanistic understanding of fission gas release mechanisms and rim effect development and evolution and fragmentation behaviour in LOCA/RIA.

Dr Sartori's presentation began with an overview of the timescales on which fast reactors are expected to be introduced and the issues related to them: technology preparedness; industrial capabilities; economic viability; materials management; the timing of their introduction and the trade-offs between short and long term optimisation. Fast reactors have very clear benefits with respect to uranium resource utilisation efficiency and reduced radiotoxicity of the waste arisings.

Dr Sartori listed and summarised the international programmes currently in progress, some of which relate both to current and future reactor systems: Nuclear Power Plant Life Management (PLiM); Multi-national Design Evaluation Program (MDEP); Generation IV Systems; Generation IV International Forum (GIF); Global Nuclear Energy Partnership (GNEP); European Commission DG-research (7^{th} Framework); Sustainable Nuclear Energy Technology Platform (SNE-TP); International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), run by IAEA and the Fast Reactor Cycle Technology Development Project (FaCT) in Japan.

Dr Sartori concluded by remarking that nuclear power research profits greatly from international cooperation, knowledge sharing and data preservation, commonality of safety and licensing standards and mutual confidence building. He noted that issues such as resource management and availability and non-

proliferation are truly international ones that cannot be fully addressed by individual countries in isolation and that for these reasons the nuclear industry has become truly international.

Following the five presentations there was an opportunity for an open discussion, which was quite wide ranging and difficult to summarise, but here are some of the more important comments/observations made:

- Most importantly, no one questioned the validity of the R&D requirements for advanced reactors and fuels identified in the various presentations and no omissions were pointed out. This is taken as evidence that the presenters captured the essential R&D requirements that is taken to be the consensus view of the discussion panel.
- A suggestion was made from the floor that the NEA should organise a benchmark to compare competing reactor concepts and their fuel cycles to establish a consensus position on the best performing ones. In the ensuing discussion, this suggestion was dismissed as being impractical because of the impossibly many aspects to be considered and that there is no single viewpoint which different countries or utilities could agree upon. A comment from one of the panel members that within the industry and academia, we should avoid paying too much attention to ranking competing systems, as that could be viewed negatively by the wider community and possibly seized upon by opponents – it needs to be recognised that there are multiple viewpoints and that they are all valid.
- In a follow-up discussion, it was noted that there were many good reactor and fuel cycle options and that it would be a mistake to focus on too few, because the criteria for selection could not be universal. The reality is that LWRs are the most economic system and they will dominate until at least the second half of the century. Nevertheless, new systems are becoming more urgently required and mechanisms need to be found to speed up their development.
- A comment was made that rather than attempt to rank different systems in an absolute sense, it would be more relevant for the international community to develop a consensus as to what should be the evaluation criteria used. It is noted here that this is one of the aims of the IAEA's INPRO project.
- There was discussion as to the actual benefit in terms of uranium utilisation that can be achieved in fast reactors. The factor 50 noted in Dr Sartori's presentation was considered a realistic estimate that took account of the practical limitations resulting from neutron captures, that prevents all the U238 being fissioned. The more simplistic factor of 140 (the inverse of the U235 natural abundance) is unrealistic as it does not account for losses via neutron capture.
- The observation was made that the industry is very conservative and that radical new developments are generally avoided. This is evident in the very slight incremental changes seen in LWRs over the past 40 years and the fact that fast reactor designs have not changed fundamentally since the very first designs. It was suggested that researchers needed to develop radical new concepts and progress them beyond the theoretical stage. In the ensuing discussion, the point was made that this step was very expensive and this is what impeded it.
- The question was put to Dr Gehin why HTR was included in the Generation IV collaboration when it is a once-through cycle and therefore does not meet the sustainability goal. Dr Gehin answered that recycle was not the main priority when Generation IV started. Potential options for higher fuel utilization, including DEEP BURN, are being investigated.
- Finally, a comment was made that the industry is somewhat removed from the issue of spent fuel/high level waste geological repositories – that continued nuclear generation does not

necessarily require the back-end to be fully established first. The suggestion was that if industry was more closely tied to this issue, then progress towards establishing final repositories in different countries would speed up.