AER WORKING GROUP D ON VVER SAFETY ANALYSIS – REPORT OF THE 2009 MEETING

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ABSTRACT

The AER Working Group D on VVER reactor safety analysis held its 18th meeting in Řež, Czech Republic, during the period 18-19 May, 2009. The meeting was hosted by the Nuclear Research Institute Řež. Altogether 17 participants attended the meeting of the working group D, 16 from AER member organizations and 1 guest from a non-member organization. The co-ordinator of the working group, Mr. S. Kliem, served as chairman of the meeting.

The meeting started with a general information exchange about the recent activities in the participating organizations.

The given presentations and the discussions can be attributed to the following topics:

- Code validation and benchmarking
- Safety analysis and code developments
- Reactor pressure vessel thermal hydraulics
- Future activities including discussion on the participation in the OECD/NEA Benchmark for Kalinin-3 VVER-1000 NPP

A list of the participants and a list of the handouts distributed at the meeting are attached to the report. The corresponding PDF-files can be obtained from the chairman.

1. CODE VALIDATION AND BENCHMARKING

1.1 Dynamic AER Benchmarks

C. Parisi reported about the calculation of the second dynamic AER benchmark using the RELAP5-3D©/NESTLE and the PARCS codes [1]. This benchmark is part of a series of benchmarks defined in the frame work of the working group in order to verify the 3D neutron kinetic core models. It considers an asymmetric control rod ejection at low power with a simple adiabatic Doppler feedback model and was defined in 1993. During the last years the benchmark was recalculated by different organizations. C. Parisi compared the obtained results with the solution obtained by FZD using the DYN3D code with the HEXNEM2 method. The K_{eff}- value of the initial state is well predicted by PARCS while the RELAP5-3D©/NESTLE code overestimates this value slightly. The efficiency of the ejected control rod was overestimated by PARCS (6.0 %) and underestimated by RELAP5-3D©/NESTLE

(2.7 %). In contradiction to this, the maximum power reached during the transient calculation is in comparison to smaller in the PARCS calculation and considerably higher in the RELAP5-3D©/NESTLE calculation. The investigations on the reason of these discrepancies will be continued.

As agreed on the last meeting, A. Kotsarev prepared a draft version of the definition of 7th dynamic AER benchmark [2]. The purpose of this benchmark is to compare results of calculation by different codes of coolant mixing in the reactor pressure vessel (RPV) of a VVER-440 plant during the connection of one loop with reduced temperature and/or reduced boron concentration to several working loops. Each participant should use own models for the description of coolant mixing in the lower and the upper plenum in frame of using calculation code. Both upper plenum and lower plenum mixing are applied to temperature and boron acid concentration. It is expected that the different mixing options have a considerable influence on the response of the reactor core. The results on the coolant mixing obtained from a detailed CFD model (ANSYS CFX) can serve as reference solution.

Such a detailed model of the RPV was presented by B. Kiss in his presentation [3]. This detailed RPV model contains the main structural elements:

- inlet and outlet nozzles
- guide baffles of hydro-accumulators
- alignment drifts
- elliptical perforated plate
- planar perforated plates of brake tube chamber
- brake- and guide tube chamber
- simplified core
- perforation of reactor pit
- lower grid plate of protective tube unit

The new model of the RPV contains now about $19*10^6$ elements. The model was validated by calculating mixing factors obtained during corresponding start-up experiments at the NPP Paks. In further calculations the model was applied to different ECC injection scenarios.

1.2 OECD Kalinin-3 Benchmark

GRS and Kurchatov Institute proposed an international benchmark for coupled code systems based on a start-up experiment at the NPP Kalinin-3 (VVER-1000). It was accepted by OECD/NEA. K. Velkov (GRS) prepared an overview on the transient and the available data [4]. This presentation was given by S. Kliem. After this presentation the participants of the working group meeting discussed the participation in this benchmark. The following organizations expressed their readiness to participate in the benchmark (Table 1). Preliminary information on the code systems to be used is also included into the table.

Table 1 Overview on the envisaged participation in the Kalinin-3 Benchmark

N°	Organization	Code system
1	Kurchatov Institute / GRS	BIPR8/ATHLET
2	NRI Řež	DYN3D/ATHLET RELAP5-3D©
3	KFKI AEKI Budapest	KIKO3D/ATHLET
4	FZ Dresden-Rossendorf	DYN3D/ATHLET
5	VTT Espoo	HEXTRAN/SMABRE
6	Gidropress Podolsk	TRAP_KS/KORSAR/GP
7	UniPisa	RELAP5-3D©/NESTLE TRACE/PARCS

It was agreed that every organization should make its own registration to this benchmark. This is a precondition to obtain the benchmark specification and the experimental data.

1.3 Validation of DYN3D/ATHLET

J. Hadek reported about the post-test calculation of the test „Opening of steam dump to atmosphere (SDA)", which was performed at Temelín NPP Unit 2 at April 4th, 2004 [5].
The initial state for this experiment was 20 % of nominal power at the beginning of the 2nd fuel cycle.
The test was conducted according to the following scenario:

- Main steam header (MSH) pressure was increased with a trend of 1.7 MPa/h (0.028 MPa/min). Increasing and decreasing of pressure made with SDC (steam dump to condenser).
- When the SG1 pressure was equal to 6.9 MPa then the SDA1 (steam dump to atmosphere on the first main steam line) was opened.
- When the SG1 pressure was lower then 6.8 MPa then the regime of SG1 pressure regulation with constant value of 6.5 MPa (with accuracy of 0.15 MPa) was started.
- After measurement termination the final MSH pressure equal to 6.15 MPa was reached with a trend of - 2.5 MPa/h (- 0.04 MPa/min).

This test was calculated by the DYN3D/ATHLET coupled code system. The standard ATHLET input deck for VVER-1000 (Temelín NPP) available at NRI was used, where all 4 loops of reactor coolant system are fully modeled (primary and secondary side) as well as the pressurizer system. A two-group neutronic library created by NRI Rez - division ENERGOPROJEKT Prague by help of the lattice code HELIOS was used for the calculation. The aim of the analysis consisted in the validation of DYN3D/ATHLET reactor core model (including coolant mixing) on the basis of coolant loops temperatures, and fuel assemblies outlet temperatures. Further it had to be proven that the main thermal-hydraulic parameters show identical trends.
The conclusions from the calculation were as follows:

- The trends of main calculated parameters (coolant loops temperatures, primary and secondary circuit pressures) and test results are very close.
- The calculated and measured thermal hydraulic phenomena like energy release from secondary circuit, asymmetrical cooling of primary circuit, pressurizer water level decreasing, primary pressure decreasing, and asymmetrical energy distribution agree sufficiently.
- A certain degree of mixing can be observed in the measured fuel assembly outlet temperatures which is not reproduced by the simplified mixing model included in

the DYN3D/ATHLET code.

- The DYN3D/ATHLET code is a very suitable tool for calculations of MSLB in VVER-1000 NPP.

2. SAFETY ANALYSIS AND CODE DEVELOPMENTS

A. Pinegin (KI Moscow) spoke about problems of the definition of DNBR if several hot cells are present in the core [6]. In the current approach at KI very rare events are taken into account in the same manner as all events. All deviations are considered to be independent what is responsible for a high degree of conservatism in the determination of the minimum DNBR value. For reducing this conservatism it is needed to specify statistical hypotheses for deviations of technological parameters and use them in order to consider the probability of appearance of different events (deviations). It is foreseen to include the GRS statistical tool SUSA into the determination of the minimum DNBR value.

A. Hämäläinen gave an overview about recent code developments at VTT Nuclear Energy in Finland [7]. Work on improvement of the neutronic part of the well-known APROS code has started recently. The HEXTRAN nodal model was implemented into the APROS code (fully reprogrammed) and is now in the test phase, standard use is expected for 2010. It is also planned to develop a rectangular version. A nodal multigroup option for e.g. GEN-IV applications is also foreseen. It will be based on the DIF3D code. APROS disposes of a homogeneous two-phase 3-D flow model, which was developed about 20 years ago. The main application area is the PWR pressure vessel outside the core. Now a turbulence model is under development and a coupling to the APROS 6-equation model is in preparation. Internal coupling of TRAB-3D and the SMABRE codes has been accomplished. The first application of this new coupling should be the flow in the core of an HPLWR. In this way, problems in case of occurrence of flow reversal can be avoided. The PORFLOW code which is based on a 3D porous medium model is under development. The objectives of PORFLO development are: independent code, easier to set up and faster to compute than typical CFD simulations. As one of the first applications the BFBT Benchmark (void distribution in a single BWR fuel assembly) was calculated. Currently calculations of the secondary side of a horizontal steam generator with about 10^5 nodes are underway. The results will be compared to FLUENT calculations. Serpent - a Monte Carlo reactor physics burn-up calculation code is under development at VTT since 2004. The code should be used for burn-up calculations and for the generation of homogenized few-group cross section data.

A. Keresztúri gave a very detailed overview on safety analyses for licensing of a new fuel containing Gd burnable poison [8]. The introduction of burnable poison can have economic advantages because it is possible to use higher enriched fuel. The introduction of burnable poison into the fuel is accompanied by modifications in the geometry of the fuel assemblies. Using such fuel has an impact on a number of safety features. In the presentation it was demonstrated that a sound basis of licensing methodology, safety analysis, and necessary computer codes for the VVER fuel modernization is available also in case of application of burnable poison. It was further shown that the significantly modified assembly-wise, pin-wise and intra-pellet power distributions - due to the presence of burnable poison - must be taken into account in the analyses, especially in the hot channel, intra-assembly mixing and fuel behavior calculations. This was demonstrated in the comparison of the results of an ATWS analysis (withdrawal of the 6^{th} control assembly group) using the KIKO3D/ATHLET code for fuel with and without burnable poison.

G. Alechin (GP Podolsk) presented calculations using the TRAP-KS code system of the transient with tripping of one feed water pumps with accelerated power reduction, which caused a reactor scram by a decrease in the reactor period at Khmelnitski-2 NPP, at the end of the 3rd fuel cycle. A comparison of the results between TRAP-KS, DYN3D (calculations conducted by the Ukrainian authority) and measurements for indications of neutron flux monitoring equipment AKNP-I was carried out. Comparison of results between TRAP-KS, DYN3D and measurement shows a satisfactory agreement for time dependence of calculated relative neutron power (within 1-2% N_{rated}). The agreement for decreasing of relative neutron power level in ion-chamber channels between TRAP-KS calculation and measurement after group dropping took place is also satisfactory (within 2% N_{rated}). The speed of increase of relative neutron flux in the calculation is higher in comparison with measured data due to the fact that the minimum design value of reactor period in calculation is less than at the plant. The possibility of a reactor scram, caused by reactor period decreasing below 10 s was confirmed in calculations for neutron flux monitoring system. Calculations with respect to change of algorithm of determination the reactor period in order to avoid the reactor scram were performed for the Khmelnitski-2 NPP (increasing of time delay of the algorithm).

3. REACTOR PRESSURE VESSEL THERMAL HYDRAULICS

At the two 1:5 scaled test facilities ROCOM at FZD and Gidropress test facility in Podolsk experiments have been performed on flow distribution during steady-state operation and on coolant mixing during start-up of the first main coolant pump. S. Kliem presented a comparison of experimental results obtained at both test facilities for similar flow conditions [10]. The analysis of the slug mixing experiments showed comparable flow behavior. The first part of the tracer is found on the opposite side in regard to the position of the starting-up loop. In this region, the maximum tracer is measured in both facilities. These maximum values differ by about the same value as the initial slug sizes in the experiments. In stationary experiments at both test facilities a clear sector formation at the core inlet could be observed. Coolant from the loop with the perturbation arrives nearly unmixed in both cases at single measurement positions at the core inlet. In one of the Gidropress experiments an additional counterclockwise swirl was found which is responsible for moving the sector. In the corresponding ROCOM experiment such an additional swirl is fully absent. Contrary to that, a shift of the sector is found in a four-loop ROCOM experiment at reduced flow rate.

4. FUTURE ACTIVITIES

The following topics are either in progress or are of potential interest in the future activities of the working group D:

- Solution of the VVER-1000 steady state benchmark AER-FCM-101
- Calculation of OECD/NEA Benchmark for Kalinin-3 VVER-1000 NPP

- Clarification of the cause and significance of mesh refinement effects on solutions to control rod ejection benchmarks
- Methodology for safety analyses
- Safety criteria for high burnup fuel
- Uncertainty and sensitivity analysis for safety analyses
- Hot pin and hot channel approximations in safety analyses
- Transient fuel behaviour models and approximations for use with 3D core models
- Representation of reflectors, including wide range data
- Wide range representation of two-group cross section data
- Application of two-group neutron kinetics data
- Application of 3-D thermal-hydraulic calculations for coolant flow and mixing in the reactor vessel
- Utilization of data from physical start-up experiments

It was tentatively agreed to hold the next meeting of working group D in April, 2010 in Paris, in connection with the next workshops on the OECD benchmark for uncertainty analysis in best-estimate modeling (UAM) and the Kalinin-3 benchmark.

LIST OF PARTICIPANTS

From AER member organizations

01	J. Hádek	Nuclear Research Institute Řež plc., Czech Republic (NRI)
02	R. Meca	Nuclear Research Institute Řež plc., Czech Republic (NRI)
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05	M. Benčik	Nuclear Research Institute Řež plc., Czech Republic (NRI)
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07	P. Dvořák	ČEZ a.s., JE Temelin, Czech Republic
08	J. Bajgl	ČEZ a.s., JE Dukovany, Czech Republic
09	S. Kliem	Research Centre Dresden-Rossendorf, Germany (FZD)
10	A. Hämäläinen	VTT Technical Research Centre of Finland (VTT)
11	A. Keresztúri	KFKI Atomic Energy Research Institute, Hungary (AEKI)
12	A. Kotsarev	Russian Research Centre “Kurchatov Institute”, Institute of Nuclear Reactors, Russia (KI)
13	A. Pinegin	Russian Research Centre “Kurchatov Institute”, Institute of Nuclear Reactors, Russia (KI)
14	G. Alechin	FSU EDO Gidropress Podolsk, Russia (GP)
15	B. Kiss	Budapest University of Technology and Economics, Institute of Nuclear Techniques, Hungary (BUTE)
16	C. Strmensky	VUJE Trnava Inc., Slovakia

Guests

17	C. Parisi	University of Pisa, Department of Mechanical, Nuclear and Production Engineering, Italy

LIST OF HANDOUTS

The following handouts of presentations were made available to the participants during the meeting, at least in electronic form. The corresponding PDF-files can be obtained from the chairman.

[1] C. Parisi: GRNSPG/UNIPI results for the second AER benchmark by the RELAP5-3D©/NESTLE and PARCS codes
[2] A. Kotsarev: Definition of the 7th Dynamic AER benchmark – pressure vessel coolant mixing at connection of loop in a NPP with VVER-440 (edition 0)
[3] B. Kiss, I. Boros, A. Aszódi: First benchmark calculations with VVER-440 RPV CFD model
[4] K. Velkov: OECD/NEA Benchmark for Kalinin-3 VVER-1000 NPP
[5] J. Macek, R. Meca, J.Hádek: Validation of Thermohydraulic Computing Model of VVER-1000/320 Temelin NPP for Calculations with Coupled ATHLET/DYN3D Codes
[6] M. Lizorkin, D. Oleksyuk, A. Pinegin: Features of definition DNBR at presence of several hot cells in a core
[7] A. Hämäläinen: Overview on current code developments at VTT
[8] A. Keresztúri, Gy. Hegyi, I. Trosztel, G. Hordósy, I.Panka, A. Molnár, Cs. Maráczy: Safety analyses for licensing of Gd dopped fuel
[9] G. Alekhin, M. Bykov, I. Petkevich: Analysis of indications of neutron flux equipment under the conditions with actuation of accelerated preventive protection for WWER-1000 reactor
[10] S. Kliem, T. Höhne, U. Rohde, M. Bykov, E. Lisenkov: Comparative evaluation of coolant mixing experiments at the ROCOM and the Gidropress test facilities
[11] I. Tinka: High burnup and Gd dependent fuel criteria