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AER WORKING GROUP D ON VVER SAFETY ANALYSIS – REPORT OF THE 2007 MEETING

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ABSTRACT

The AER working group D on VVER reactor safety analysis held its 16^{th} meeting in Paris, France during the period 08-09 May 2007. The meeting was hosted by the CEA France. It followed the final workshop on the OECD/DOE/CEA VVER-1000 Coolant Transient Benchmark held at 07 May. Altogether 11 participants attend the meeting of the working group D, 7 from AER member organizations and 4 guests from non-member organizations. 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 development and benchmarking for reactor dynamics applications
- Safety analysis methodology and results
- Future activities

New solutions for three different benchmarks were presented and discussed. These are the Second AER Dynamic Benchmark on control rod ejection at hot zero power (S. Kliem, FZD), the VVER-1000 Coolant Transient Benchmark (E. Syrjälahti, VTT) and the stationary AER-FCM101 Benchmark considering a VVER-1000 reactor (C. Parisi, UniPisa).

A. Keresztúri (AEKI) presented a statistical evaluation of the possibility to observe a fuel assembly misloading event. The second presentation of E. Syrjälahti was dedicated to the description how best-estimate coupled code calculations at VTT are supported by uncertainty and sensitivity analyses. K. Velkov (GRS) presented preliminary results of BIPR8KN/ATHLET calculations with a very detailed resolution of the calculation grid on the assessment of coolant mixing inside VVER-1000 assembly heads. Coolant mixing experiments at three different mixing test facilities, modeling different reactor types, were presented and compared by S. Kliem. A calculation study using the coupled code system KORSAR/GP on the consequences of the injection of a slug of unborated water into the reactor core was described by G. Ponomarenko (Gidropress).

A list of the participants and a list of the handouts provided at the meeting are attached to the report. The handouts can be obtained in electronic form from the chairman.

1. CODE DEVELOPMENT AND BENCHMARKING

1.1 The Second Dynamic AER Benchmark

S. Kliem [1] presented new solutions of the second Dynamic AER Benchmark. 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 feed back model and was defined in 1993. In 2005 this benchmark was included into the benchmark list of the EC financed integrated project NURESIM. The presented new solutions were obtained using the two nodal methods available in the DYN3D code. Comparisons were made with the solutions by other nodal codes obtained earlier and with a "reference" solution of the fine-mesh diffusion code CRONOS2. A lively discussion took place on the quality of the CRONOS2 solution. A number of shortcomings of the current solution were mentioned, as the use of a step-wise ejection of the control rod and a relatively big time step of 20 ms (S. Kliem) and the use of only six triangles per hexagon in the transient calculation (B. Ivanov). Especially the fact, that the CRONOS2 solution shows a power peak in the transient calculation, which is about 40 % higher than that of the DYN3D-HEXNEM2 solution while having a lower static reactivity worth of the ejected control rod, supports the assumption, that possibly the CRONOS2 solution is not fully converged. The participants agreed that a further discussion of the results of this benchmark is necessary. Some of the participants of the meeting (including N. Kolev, who produced the CRONOS solution) evinced their interest to calculate or to recalculate this benchmark until the next meeting.

1.2 The OECD/DOE/CEA VVER-1000 Coolant Transient Benchmark

The second phase of the OECD/DOE/CEA VVER-1000 Coolant Transient Benchmark (V1000CT) was discussed at the final workshop of the benchmark held before the working group meeting. This phase consists of three different exercises. The member organizations of the AER taking part in the single exercises and the content of the single exercises can be found in Tab. 1.

Results and further actions were discussed during the workshop. The reader is referred to the minutes of this workshop for further details. At the working group D meeting E. Syrjälahti explained some details of the solutions of exercise 2 and 3 obtained at VTT [2]. One of the crucial points of this benchmark is how to adopt the results of exercise 1 (coolant mixing inside the reactor pressure vessel) to the calculation of exercises 2 and 3. The pressure vessel in the VTT calculation was divided into six sectors. Turbulent mixing between these sectors was applied in the lower plenum. The corresponding coefficients are based on the measurements made available in the exercise 1 of the current phase of the benchmark. Using realistic neutron kinetic core data no recriticality was obtained after the reactor scram in the calculations of both exercises. This is in agreement with the results of the other participants. In the exercise 3 calculation, the minimum temperature in the overcooled loop was higher than in the exercise 2 (given boundary conditions). Here, the influence of the steam generator modeling reveals in the calculations of the different participants.

Tab. 1 Overview about Phase 2 of the V1000CT benchmark and the participants

Exercise	Content	Participants and codes (AER member organisations)
1	Computation of coolant mixing experiments to test and validate the reactor vessel mixing models against measured data. Coolant mixing effects from the vessel inlet nozzles until the outlet nozzles are included.	FZD (CFX-10) KI (BIPR8/ATHLET) INRNE (CATHARE) Budapest Technical University (CFX-10)
2	Analysis of a main steam line break (MSLB) transient for core and vessel only, using validated models for vessel mixing and coupled with 3D neutronics	FZD (DYN3D/ATHLET) VTT (HEXTRAN/SMABRE) INRNE (CATHARE/FLICA/CRONOS)
3	Best estimate analysis of the entire plant for two scenarios of the MSLB transient (realistic and pessimistic ones)	FZD (DYN3D/ATHLET) VTT (HEXTRAN/SMABRE) INRNE (CATHARE/FLICA/CRONOS)

1.3 The steady state benchmark AER-FCM-101 for a VVER-1000 core

At the WG D meeting in 2004, N. Kolev suggested that the VVER-1000 steady state benchmark AER-FCM-101 is used as a numerical test case to validate the accuracy of nodal codes to describe the core and reflector regions. The extrapolated fine-mesh solution by the French code CRONOS2 is given as a reference solution. This test case can also be used to validate methods applied to convert given diffusive reflector data into nearly equivalent albedo matrix data for the reflector.

C. Parisi presented first results of the solution of this benchmark using the 3D neutron kinetic core model PARCS [3]. This nodal code uses a triangular polynomial expansion method (TPEN) for the hexagonal geometry. The comparison to the reference solution showed a good agreement, the k_{eff} is underestimated by 31.5 pcm, the maximum deviation in nodal power is 3.3 % and in assembly power 1.2 %. The application of the 3D neutron kinetic core model NESTLE to this benchmark was not successful up to now, no solution could be obtained. C. Parisi will continue to work with both codes on the solution of this benchmark. Final results should be available at the next working group meeting.

2. SAFETY ANALYSIS METHODOLOGY AND RESULTS

A. Keresztúri [4] presented a study on the statistical evaluation of the on-line core monitoring effectiveness in order to limit the consequences of a fuel assembly misloading event. The goal is to recognize such an event already at a low power level, where the consequences are still considerable mild. A Monte-Carlo method was developed for the statistical evaluation of the assembly outlet measurements. "True" normal and abnormal measured distributions were calculated by the KARATE code system. The relative errors of the temperature measurements at different levels of the reactor power and the non-availability of up to 25 % of the measurements were taken into account. The assessment was done by two indication methods: the measured asymmetry factor and the comparison of calculated and measured temperature distribution. It was found that a fuel assembly misloading event can be effectively recognized down to 55 % reactor power even with only 75 % of the measurements available if the indication is based on the measured asymmetry factor. The effectiveness of the second method (comparison of calculated and measured temperature distribution) is much less and not satisfactory.

K. Velkov [5] gave a presentation about the modeling of thermal-couples measurements at the core exit of a VVER-1000 reactor. The aim of these studies was to find a correlation between the measured values of the thermal-couples at the core outlet with the real average assembly outlet temperatures. The problem arises from the very weak mixing of the assembly flow at the position where the thermal-couples are located, so that the measured temperature does not represent the average assembly outlet temperature. The studies were performed using the coupled code system BIPR8KN/ATHLET and measured data of one start-up experiment at the Kalinin-3 NPP (switch-off of one main coolant pump at full power). A very detailed nodalization of the whole system was applied inside the code, i.e. 1:1 modeling of the fuel assemblies and division of the downcomer and the lower plenum into several parallel channels with the consideration of cross flow between them. The assembly outlet region was resolved also in detail. The comparison with the measurement results confirmed the hypothesis, that the readings of the thermal-couples underestimate the real outlet temperature values. It was found that it is possible to reproduce the thermal-couples measurements in the calculations by the introduction of additional parallel flow channels in the assemblies with thermal-couples. It was noted that these findings are important for the Exercise 1 of the above described V1000CT Benchmark. N. Kolev from the benchmark team stated that the underestimation of the thermal-couples readings was taken into account by constant value of 0.5 K for all measurement positions.

E. Syrjälahti [6] reported about the introduction and application of uncertainty and sensitivity analysis tools at VTT. First of all an overview on the corresponding reactor analysis calculation system was given. The transition from the conduction of conservative calculations to best-estimate ones using coupled 3D neutron kinetic/thermal hydraulic system codes requires the assessment of the uncertainty of the obtained results. This uncertainty of the results is caused by uncertainties of the used models, of the initial and boundary conditions as well as of the properties of safety systems and instruments and the manufacturing tolerances. Special emphasis was given to a new developed statistical tool which works in combination with the HEXTRAN/SMABRE coupled code system and the 1D TRAB-CORE code. This tool allows an automatic DNB analysis with automatic parameter variation and the assessment of sensitivity measures. It is based on the well-known Wilks' formula. Results were presented on the application of the HEXTRAN/SMABRE code and the statistical tool to the post-test calculation of a start-up experiment at the Loviisa NPP. These results are used to validate the selected approach of the uncertainty and sensitivity assessment. A statistical approach to transient fuel behaviour codes calculations is also under development. It is not possible to apply the full-scope statistical approach described above for the HEXTRAN/SMABRE code due to the high amount of necessary variations. Some possibilities are outlined how to overcome this problem, like calculation of a limited number of worst global scenarios with also a limited number of single fuel rods only or the use of response surfaces or neural networks.

S. Kliem [7] presented the evaluation of experiments on coolant mixing carried out at three different test facilities representing three different reactor types. The background of this work is the quantitative assessment of the coolant mixing, which is an important mitigative mechanism against reactivity initiated accidents caused by local boron dilution. The facilities involved are the ROCOM test facility modelling a German KONVOI-type reactor, the Vattenfall test facility being a model of a Westinghouse three-loop PWR and the Gidropress test facility modelling a VVER-1000 reactor. The scenario of the start-up of the first main coolant pump with the injection of an underborated slug was investigated in all three facilities. The analysis of these mixing experiments showed comparable flow behaviour despite the constructive differences between the facilities. In accordance with velocity measurements, which were conducted using laser Doppler anemometry, the first part of the deboration is found on the opposite side in regard to the position of the loop with the slug. In this region, the maximum deboration (lowest boron concentration) is measured in all three cases. These maximum values are in the same order of magnitude for nearly identical initial slug volumes.

G. Ponomarenko [8] reported about calculations using the coupled 3D neutron kinetic/thermal hydraulic system code KORSAR/GP on a boron dilution scenario after the start-up of the first main coolant pump. The analysis was carried out for a modernized version of the VVER-1000 reactor with increased nominal power (up to 108 %) and increased cycle length. Further variation parameters were the length of the active fuel in the fuel assembly without increasing the length of the control rods, the number of control rods itself and the shutdown boron concentration. The code KORSAR/GP considers an incomplete coolant mixing inside the reactor pressure vessel. It has been validated on the above mentioned mixing experiments at the Gidropress test facility. For the given slug volume (about 9 m^3), the reactor remains sub-critical in all calculations if no additional safety system failure assumptions are applied. For cases, where the initial sub-criticality is reduced a significant over-criticality with corresponding power increase was obtained. In the most cases the over-criticality is restricted to values of about 1 β. Here, the integrity of the fuel is not violated. Extreme cases where the failure of additional safety systems is assumed showed the melting of the fuel in the most loaded channels. But even in these cases, the melting is restricted to a small number of fuel elements.

3. FUTURE ACTIVITIES

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

- Documentation of the 6th dynamic benchmark on an asymmetric main steam line break
- Solution of the international VVER-1000 Coolant Transient benchmark exercises of Phase 2 (V1000CT-2)
- Solution of the VVER-1000 steady state benchmark AER-FCM-101
- 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 Garching near Munich, Germany in the first week of April, 2008 in connection with the next workshop on the OECD Benchmark for uncertainty analysis in best-estimate modeling (UAM).

LIST OF PARTICIPANTS

From AER member organizations

01	S. Kliem	Research Centre Dresden-Rossendorf, Germany (FZD)
02	E. Syrjälähti	VTT Technical Research Centre of Finland (VTT)
03	A. Keresztúri	KFKI Atomic Energy Research Institute, Hungary (AEKI)
04	A. Kotsarev	Russian Research Centre "Kurchatov Institute", Institute of Nuclear Reactors, Russia (KI)
05	S. Nikonov	Russian Research Centre "Kurchatov Institute", Institute of Nuclear Reactors, Russia (KI)
06	N. Kolev	Institute for Nuclear Research and Nuclear Energy, Bulgaria (INRNE)
07	G. Ponomarenko	OKB "Gidropress", Russia (GP)

Guests

08	A. Kolychev	Kalininatomtechenergo, Udomlya, Tver region, Russia
09	B. Ivanov	The Pennsylvania State University, USA
10	K. Velkov	Gesellschaft fuer Anlagen und Reaktorsicherheit, Germany (GRS)
11	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] S. Kliem, U. Rohde: Second Dynamic AER Benchmark – Comparison of results (Intermediate state)

[2] M. Seppälä, A. Hämäläinen: V1000CT-2 benchmark calculation with HEXTRAN-

SMABRE

[3] C. Parisi: Preliminary results of the NESTLE and PARCS calculation of the AER-FCM101 benchmark

[4] A. Keresztúri, E. Temesvári, A. Molnár, L. Korpás: Statistical evaluation of the on line core monitoring effectiveness for limiting the consequences of the fuel assembly misloading event

[5] S. Nikonov, K. Velkov, S. Langenbuch: Modeling of Thermal Couples Measurements at Core Exit of Reactor VVER-1000

[6] E. Syrjälahti: Sensitivity analysis tool for VTT's reactor dynamic codes

[7] S. Kliem, B. Hemström, Y. Bezrukov, T. Höhne, U. Rohde: Comparative evaluation of coolant mixing experiments at the ROCOM, Vattenfall, and Gidropress test facilities

[8] G.L.Ponomarenko, Ju.G.Dragunov, M.A.Bykov: Hypothetical Accident with a Slug of Unborated Water and Safety of WWER-1000 at its modernization