15th Symposium of AER on VVER Reactor Physics and Reactor Safety

Znojmo, Czech Republic

October 3-7, 2005

Organised by
ČEZ, a.s.. Dukovany NPP

Book of abstracts

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0.2 Twenty years of operation Dukovany NPP, *Z.Linhart, ČEZ, a. s., Dukovany NPP, Czech Republic*

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1.7 Reactivity and derivations of point kinetics equations for subcritical systems, J.Švarný, ŠKODA JS a.s., Czech Republic

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1.13 Influence of FA pin power minimisation on neutron-physical core characteristics, *P. Mikoláš, ŠKODA JS a.s., Czech Republic*

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2.2 VVER-1000 fuel cycle improvement, *Kosourov E.K., Pavlov V.I., Pavlovichev A.M., Spirkin E.I, INR, RRC «Kurchatov Institute», Moscow, Russia*

2.3 Outlook on Fuel Cycle Perspectives at VVER 440, *S. Štech, J. Bajgl, Dukovany NPP, Czech Republic*

2.4 Assessment of the effectiveness of implementing the axial profiling in VVER-440 assemblies, *Yu.A.Ananjev, K.Yu. Kurakin, FSUE OKB "GIDROPRESS", Podolsk, V.G.Artemov, A.S.Ivanov, FSUE NITI named after A.P.Alexandrov, Sosnovy Bor, Russia*

Coffee break

11:00 **Session 2B (chair: I.Stoyanova, Bulgaria)**

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2.6 Assessment of the control rod and burnable absorber assembly efficiency during operating in VVER-1000 core, *M.L.Yeremenko[1], Yu.P.Kovbasenko[1], O.V. Gorbachenko[2], A.I. Ignatchenko[2], [1] The State Scientific and Technical Center on Nuclear and Radiating Safety, Kyiv, [2] National nuclear power generating company «Energoatom», Detached subdivision «Zaporizhya NPP», Energodar, Ukraine*

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12:30 **Lunch**

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7.5 CFD analysis of the service shaft during the recovery work of the damaged cleaning tank in the Paks NPP, *G. Légrádi, A. Aszódi, BUTE Institute of Nuclear Techniques, Budapest, Hungary*

16:20 Sight-seeing tour – Group 2

18:30 Dinner

Wednesday, October 5

08:30

0.3 Determination of reactor thermal power using a more accurate method, *J. Papuga, ČEZ, a.s., Dukovany NPP, F. Madron, Chemplant Technology s.r.o., J. Pliska, I&C Energo, a.s., Czech Republic*

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(09:00) **Session 3A (chair: L.K.Shishkov, Russia)**

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3.2 Intrusion of resin into primary circuit, *O.Grežďo, V.Mráz, Jaslovské Bohunice NPP, Slovakia*

3.3 Database of Temelin NPP Operational States and Its Use for Neutron Codes Validation, *Monika Juříčková, NRI Řež, Czech Republic*

3.4 Elimination of dynamic code from the process of spatial effect correction in scram drop measurements, *Dementiev V.G., Shishkov L.K., Tsyganov S V., RRC "Kurchatov Institute", Moscow, Russia*

Coffee break

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3.7 Measurement of Reactivity Coefficients for Code Validation, Dr. Matthias Nuding, Dr. Thomas Lötsch, TÜV Energie Consult, Munich, Germany

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4.2 RELAP5-3D© calculation of steam outlet header rupture of VVER-1000 NPP at HZP, *Marek Benčík, Jan Hádek, NRI Řež, Czech Republic*

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F. Adorján, Hungarian Atomic Energy Authority Hungary
I. Karppinen, VTT Processes, Finland

4.4 Mild transients in VVER-440type reactors simulated by the coupled ATHLET/KIKO3D code system, *Gy. Hegyi, A. Keresztúri, I. Trosztel, KFKI Atomic Energy Research Institute, Budapest, Hungary*

Coffee break

15:30 Session 4B (chair: A.Keresztúri, Hungary)

4.5 Calculation Studies Of Transient Connected With Uncontrolled One Cluster Withdrawal With Subsequent Working Of Automatic Power Controller, *Y.N.Ovdiyenko, V.A.Khalimonchuk, The State Scientific and Technical Center on Nuclear and Radiation Safety, Kyiv, Ukraine*

4.6 3D- Modelling of neutron kinetics and thermal-hydraulic processes in VVER reactors by coupled computer code ATHLET/BIPR8KN in case of asymmetrical work of equipment, *Nikonov S.P., Kotsarev A.V., Lizorkin M.P., Danilin S.A., INR, RRC «Kurchatov Institute», Moscow, Russia*

4.7 Numerical methods in the KIKO3D three-dimensional reactor dynamics code, *István Panka, András Keresztúri, Csaba Hegedűs, KFKI Atomic Energy Research Institute, Budapest, Hungary*

Thursday, October 6

Topic 5 Criticality Safety

08:30 **Session 5A (chair: V.Chrapčiak, Slovakia)**

5.1 AER Working Group E in 2005, *Vladimír Chrapčiak, VUJE, Inc., Slovakia*

5.2 Preparation and verification of libraries for ORIGEN-S in SCALE4.4a, with cross-sections for WWER-1000 fuel, *Danail Hristov, Kozloduy NPP (VVER-1000), Bulgaria*

5.3 Criticality safety analysis of a WWER-440 fuel cooling pond at the Kozloduy NPP, *A. Kamenov, I. Stoyanova - Kozloduy NPP, I. Christoskov - Faculty of Physics, University of Sofi,a Bulgaria*

5.4 Calculations of criticality, nuclide compositions, decay heat and sources for VVER-440 fuel by new version of the SCALE 5 code, *Vladimír Chrapčiak, VUJE, Inc., Slovakia*

Coffee break

10:30 **Session 5A continues**

5.5 Monitoring of the fuel elements condition in a burnt fuel interim storage at the NPP SE-EBO, *Marek Mikloš, Vladimír Kršjak, Vladimír Slugeň, Slovak University of Technology, Faculty of Electrical Engineering and Information Technology, Department of nuclear physics and technology, Slovakia*

5.6 Computation of Selected Nuclides and Fission Gasses in UOX and MOX Spent Fuel by TRANSURANUS, *Juraj Breza, Department of Nuclear Physics and Technology, Slovak University of Technology Bratislava, Slovakia*

11:30 **Lunch**

12:30 – 18:00 Technical and historical excursion

- **Water mill in Slup**
- **Monastery "Loucký"**

19:00 Official dinner

Friday, October 7

Topic 6 Spent Fuel

09:00 **Session 6 (chair: V.Lelek, Czech Republic)**

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6.2 The IAEA international project on innovative nuclear reactors and fuel cycles (INPRO): Status, Development of approaches and Outlook, *M. Khoroshev, Y. Sokolov, I. Facer, IAEA, Vienna, Austria*

6.3 Transmutation Comparison of VVER-440 and PWR Assemblies, *Radoslav Zajac, Petr Dařílek, VUJE, Inc., Slovakia*
Juraj Breza Vladimir Nečas, Department of Nuclear Physics and Technology, Slovak University of Technology Bratislava, Slovakia

Coffee break

Topic 8 Discussion and Symposium Close

11:00 **Session 8 (chair: P.Siltanen)**

12:30 **Lunch**

13:30 **Departure**

0.1 Welcome address

Z.Linhart
ČEZ, a. s., Dukovany NPP
675 50 Dukovany
Czech Republic

ABSTRACT

0.2 Twenty years of operation Dukovany NPP

Z.Linhart
ČEZ, a. s., Dukovany NPP
675 50 Dukovany
Czech Republic

ABSTRACT

0.3 Determination of reactor thermal power using a more accurate method

Jaroslav Papuga, ČEZ, a.s., Dukovany NPP
František Madron, Chemplant Technology s.r.o.
Jiří Pliska, I&C Energo, a.s.
Czech Republic

ABSTRACT

Reactor thermal power is an important operational parameter in many respects such as nuclear safety, reactor physics or evaluation of turbine thermal performance. Thermal power of a pressurized water reactor is determined on the basis of the steam generator thermal balance. The balance can be made in several variants differing from one another by the selection of different measuring circuits whose data are used in the balancing. In principle, no one such variant gives the true value of the thermal power. Among the variant values, the one nearest to the unknown true value of reactor thermal power is probably the value calculated with the lowest uncertainty. The determination of such uncertainty is not easy and its value can make even several percent, which has significant economic consequences.

This paper presents the method of data reconciliation and its application to the data of the third unit of Dukovany NPP. The data reconciliation method allows to exploit all the information which process data contain. It is based on the statistical adjustment of the redundant data in such a way that the adjusted data obey generally valid laws of nature (e.g. conservation laws). Mass and energy balances based on the data not yet reconciled do not obey those laws because of measurement errors. For data reconciliation in Dukovany, a detailed model of mass and energy flows describing the 3rd unit from steam generators to alternator and condenser was set up. Laws of mass and energy conservation and phase equilibrium in water-steam systems are thus fulfilled. Moreover, the user can model momentum balances in pipelines and create other equations which are respected during calculation. The data reconciliation is done regularly for hourly averages.

As a result of data reconciliation, there is a new data set with more accurate data including the reactor thermal power. Data reconciliation also allows:

- determination of confidence intervals for results of mass and energy balancing
- calculation of non-measured quantities (e.g.thermal flows in loops of the primary circuit)
- detection and elimination of measurement gross errors
- design and optimization of measurement systems aimed at the minimization of uncertainty of measured and calculated values and minimization of measurement cost.

The paper describes the current state of implementation using overall plant information system and process data warehouse based on the Industrial SQL server database and the InTouch human machine interface. The paper also presents results of data reconciliation in 2004 and 2005.

1. Spectral and Core Calculation Methods

1.1 Information of AER WG A on improvement, extension and validation of paremetrized few-group libraries for VVER-440 and VVER-1000

J.Švarný
ŠKODA JS a.s., Orlík 266, 316 06 Plzeň
Czech Republic

ABSTRACT

Progress report - Information of WG A activities in 2005.

1.2 Two-dimensional test problem for neutron - physical calculation of the VVER cores

A.V.Tikhomirov
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Russia

ABSTRACT

Calculation of power in FA fuel rods is an important element in safety justification of the operating reactors and those under design. Traditionally in Russia and abroad the calculations of neutron physics characteristics (in particular, assembly-by-assembly and fuel rod-by-fuel rod power distributions) are performed in diffusion approximation.

This approach provides for rather quick and accurate calculation of neutron physics characteristics. However there is a number of problem areas (core periphery, CPS CR assemblies, etc.) wherein the diffusion approximation has a high methodical error.

On the one hand, the power in fuel in these areas is low and not important for safety justification with the conservative approach. On the other hand, increase in WWER competitiveness requires use of realistic approximation. Accuracy of calculations of fluence to the RPV, abrupt changes of power in fuel rods, adjacent to absorbing element, under their withdrawal from the core, as well as other important aspects are related directly to the issue considered in the report.

To improve the accuracy of calculation of power in the mentioned problem areas using the method of diffusion approximation the methodical corrections can be used and their usage is considered in the given report.

The work is performed within the frame of code package Sapfir_95&RC. In the comparative analysis the precision code MCNP5 is used that solves the equation of neutron transport by Monte-Carlo method.

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1.3 WWER radial reflector modelling by diffusion codes

P. T. Petkov
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S. Mittag
Forschungszentrum Rossendorf, Dresden, Germany

ABSTRACT

The two commonly used approaches to describe the WWER radial reflectors in diffusion codes, by albedos on the core-reflector boundary and by a ring of diffusive assembly size nodes, are discussed. The advantages and disadvantages of the first approach are presented first, then the Koebke's equivalence theory is outlined and its implementation for the WWER radial reflectors is discussed. Results for the WWER-1000 reactor are presented. Then the boundary conditions on the outer reflector boundary are discussed. The possibility to divide the library into fuel assembly and reflector parts and to generate each library by a separate code package is discussed.

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1.4 Investigation of some BURN-UP parameters influence on HELIOS spectral code calculated nuclide number densities

I.Stoyanova, K. Kamenov, A. Sofronieva
Kozloduy NPP
Bulgaria

ABSTRACT

A study of the 13 Actinides and 20 fission products number densities sensitivity to the burn up conditions is presented.

This work analyses the concentration changing of these isotopes of importance in “burn up credit” methodology as a function of burn up parameters.

The calculations are performed using the spectral code HELIOS-1.5.

It is found that 239Np, 241Am and 242Cm are very sensitive to the nuclide depletion conditions. The most sensitive to the nuclide depletion conditions among considered fission products are those having half-life T1/2 in the range (1÷40 days): 149Pm, 143Pr, 141Ce, 133Xe, 103Ru and 105Rh. In addition there is presented the estimation of the neutron flux spectrum influence only (the change in the concentration of the boric acid dissolved in the water moderator during the reactor campaign) on the calculated Actinides number densities. The calculations are performed for WWER-440 3.6wt% 235U enriched fuel assembly in the burn up range from 0 to 40MWd/kgU.

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1.5 On solution to the problem of criticality by alternative MONTE CARLO method

Jan Kyncl
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ABSTRACT

The contribution deals with solution to the problem of criticality for neutron transport equation. The problem is transformed to equivalent one in a suitable set of complex functions and existence and uniqueness of its solution is shown. Then the source iteration method of the solution is discussed. It is pointed out that final result of iterative process is strongly affected by the fact that individual iterations are not computed with sufficient accuracy. To avoid this problem a modified method of the solution is suggested and presented. The modification is based on results of the theory of positive operators and problem of criticality is solved by Monte Carlo method constructing special random process and variable so that differences between results obtained and the exact ones would be arbitrarily small. Efficiency of this alternative method is analysed as well.

1.6 Errors in the assessment of point kinetics parameters

Dementiev V.G., Sidorenko V.D., Shishkov L.K., Tsyganov S.V.
RRC "Kurchatov Institute", Moscow
Russia

ABSTRACT

The paper contains improved formulas for assessment of point kinetics parameters (effective delayed neutron fraction, prompt and delayed neutron lifetimes) as well as some ideas on the determination of these functionals in their calculation using the TVS-M-and BIPR-type codes.

This subject attracts interest because the point kinetics parameters cannot be practically measured, and the error of their determination may be estimated from the comparison of the results obtained using different codes, or analytically. It is proposed to use the first neutrons for estimating the dispersion of point kinetics parameters of some core internals, and then, using the analytical method, to determine the dispersion of point kinetics parameters of the whole reactor, obtained by "averaging" using the BIPR-type code.

1.7 Reactivity and derivations of point kinetics equations for subcritical systems

J.Švarný
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Czech Republic

ABSTRACT

In recent years, there has been an increasing interest in the analysis time dependent behavior of subcritical systems with external source (accelerator-driven systems -ADS). The so called "point kinetics equations" are widely used in the reactor analysis. Correct derivation of point kinetics equation on the deep subcritical level and close to criticality will help to understand "point kinetics approximation" in general.

At present is in MOBY-DICK macrocode used "quasi-critical" adjoint flux solution. It was found that the role of adjoint weighting function in sucritical system is not negligible.

To understand physical background of the role of weighting functions in point kinetics methodology is provided derivation point kinetics equations from the variational principle.

Extension of the present algorithm SKODA to the subcritical systems with external source is also derived.

For comparisons is given brief information about other approaches : Gandini perturbation methodology and Ott – (SIMMER code) methodology.

1.8 ANDREA: Advanced Nodal Diffusion code for REactor Analysis

J. Běláč, R. Jošek, L. Klečka, V. Starý, R. Vočka
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Czech Republic

ABSTRACT

The possibilities of intensive exploitation of both current and future high burnup fuels depend among others on the availability and performance of computational tools used for core reload optimization and core monitoring. For this reason a new macrocode is being developed at NRI which will allow coupling of the sub-channel analysis with neutronics calculations.

A major progress has been made in the core simulator development. Simulator itself is based on nodal expansion method, Helios lattice code is used for few group libraries preparation. Standard features as pinwise power reconstruction and feedback iterations on critical control rod position, boron concentration and reactor power are implemented. A special attention is paid to the system and code modularity in order to enable flexible and easy implementation of new features in future.

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1.9 Comparison of programme MOBY-DICK with nodal programmes on benchmark problems

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Czech Republic

ABSTRACT

In this paper we compare programme MOBY-DICK (which is diffusion difference programme) with nodal programmes on two-dimensional hexagonal benchmark problems for the VVER-type reactors (published by Chao and Shatilla).

Nodal results are partly from one's own programme NODRAM and partly from literature.

There is presented dependence on lattice pitch by difference programme and influence of boundary conditions.

1.10 Further validation of the KARATE code system

András Keresztúri, György Hegyi, Csaba Maráczy, Lajos Korpás*
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ABSTRACT

In the last years several projects aiming at introduction of new VVER-440 fuel types and resulting in more economic fuel cycles were initiated. Increased average enrichment, modification of the lattice pitch and fuel diameter, profiled enrichment, application of burnable absorber, modification of the absorber assembly coupler part could lead to higher burnup and maximum allowed reactor power. The above fuel modifications and the upgraded regimes requiring more accurate calculations have necessitated the further development and validation of the KARATE code system: application of new, more accurate nuclear data, corresponding renewal of the multigroup libraries and the parametrized few group constants. For the further validation the operational data of Paks (Hungary) NPP and zero reactor measurements were used. The global calculations, where burnup dependent node-wise power distributions, critical boron concentrations, reactivity coefficients of the core etc. are determined, have been validated against operational data of Paks NPP. Measured critical boron concentrations, reactivity coefficients, radial temperature and axial self power detector signal distributions were used for the comparison for 14 new cycles of each four units. The fine mesh diffusion calculations were benchmarked by using the measurements of ZR-6 critical facility lattices containing Gd burnable poison.

1.11 Benchmark calculations for hexagonal lattices with different methods

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ABSTRACT

Necessity to increase the safety conditions of exploitation of recently designed core of modern nuclear reactors causes stronger requirements to the precision of neutron-physical analysis.

To get more precise characteristics of nuclear reactor cells and assembly one can increase the accuracy of neutron-physical calculation analysis by taking account the spectral effects.

This paper deals with the analysis of the ZR-6 series of experiments using some components of the KARATE code system. The goal of our investigations is the comparison of measured and calculated parameters of perturbed hexagonal lattices containing Gd2O3 in Al2O3 matrix or water holes. The quoted results include: the criticality parameters Hcr, dρ/dh and the absorber rod efficiency: Δρ. The experiments are based on doubling time measurements.

The calculations have been compared not only to the measured data but to the Monte Carlo code results, too.

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1.12 Adaptation of macrocode MOBY-DICK for Gd- fuel loading

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Czech Republic

ABSTRACT

The paper briefly describes an adaptation of MOBY-DICK code to VVER-440 Gd-2 fuel. Described are changes in axial fuel geometry structure and other changes, which are necessary due to fact that the fuel length is different for different types of working and control assemblies. Also a change in pin pitch has to be considered by pin-to pin calculations. The functionality of code for "transient" loading patterns calculations is demonstrated on examples of 18th and 19th cycles of the Dukovany NPP Unit III.

1.13 Influence of FA pin power minimisation on neutron-physical core characteristics

P. Mikoláš
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Czech Republic

ABSTRACT

The influence of optimisation of radial fuel enrichment profilation in fuel assembly of Gd-2 type on lowering its pin power non-uniformity and subsequent lowering of FdH in WWER–440 cores are described in this paper.

2. Core Design, Operation and Fuel Management

2.1 AER working Group B Activities in 2005

Petr Dařílek
VUJE Inc., Trnava, Slovakia
Darilek@vuje.sk

ABSTRACT

Review of AER Working Group B Meeting in Czech republic, Horni Plana is given.

2.2 VVER-1000 fuel cycle improvement

Kosourov E.K., Pavlov V.I., Pavlovichev A.M., Spirkin E.I.
INR, RRC «Kurchatov Institute», Moscow
Russia

ABSTRACT

The problems of organization of fuel cycles with different operation time of stationary load for the reactor VVER-1000 are considered. The outcomes of matching of the characteristics for stationary load constructed on fuel cells of existing and improved designs are presented. Improved designs of a fuel cell are include increase of an altitude of a fuel stake, change of outside and axial diameters of a fuel pellet, change thickness of a cladding of a fuel cell. Effect of the layout solutions on improving of a fuel cycle VVER-1000 also is considered.

2.3 Outlook on to Fuel Cycle Perspectives at VVER 440

Svatobor Štech, Josef Bajgl
Dukovany NPP
Czech Republic

ABSTRACT

Current internal fuel cycle in NPP Dukovany 4x440MWe is shortly characterized with new types of fuel assemblies and advanced fuel cycles which have been introduced in the last years. The modernization activities accomplished until now might be extrapolated to the further period in fuel design - mechanic, thermal-hydraulic and neutronic respectively - with additional increase in fuel enrichments and burnups on the way to the 6-year cycle.

Reactor power uprating together with Unit thermal efficiency improvements could bring an increase in the electric output to the value nearly 500 MWe. The reasons are given for long-term cooperation with Fuel Supplier and Plant Designer in the area of fuel cycle as well as in Unit Design Basis. All innovations mentioned in the article including future fuel and fuel cycle changes might be a quite realistic perspective at the end of the first decade of the new century.

2.4 Assessment of the effectiveness of implementing the axial profiling in VVER-440 assemblies

Yu.A.Ananjev, K.Yu. Kurakin
FSUE OKB "GIDROPRESS", Podolsk

V.G.Artemov, A.S.Ivanov
FSUE NITI named after A.P.Alexandrov, Sosnovy Bor
Russia

ABSTRACT

The present report deals with consideration of fuel enrichment axial profiling in WWER-440 assemblies. The study is performed on improving the effectiveness of fuel utilization using the example of implementing the axial profiling in the assemblies of the second generation.

For simulation of fuel loadings the computer code package SAPFIR_95&RC is used that allows for correct consideration of specific features of assemblies design changes. The methodical approach to assessment of effectiveness of implementing the axial profiling is considered with the use of capabilities of the mentioned code package.

In conclusion the recommendations are given on using the fuel enrichment axial profiling in WWER-440 assemblies.

2.5 XS data recalculation with HELIOS-1.8 and Statistical investigation of C-PORCA and GEPETTO codes results based on in-core measurements

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ABSTRACT

As a part of the power up rate process at the NPP Paks some reactor physical model development and testing were fulfilled. The model development mainly focussed on the more flexible handling of assemblies with different initial material compositions in axial direction and the renewing of few group XS data storage. Parallel with this modification all of the few group XS data were recalculated by the newest HELIOS version.

To satisfy the correct and accurate off-line and on-line reactor physical analysis of reactor cores a comprehensive investigation of the relevant codes has been done. During this process the accuracy of applied models was determined and their appropriateness was also demonstrated.

The paper shows the main features of modifications and code developments and basic results of tests.

2.6 Assessment of the control rod and burnable absorber assembly efficiency during operating in VVER-1000 core

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ABSTRACT

This article describes the main result of calculation assessment the decreasing of control rod (CR) and burnable absorber assembly (BAA) efficiency during operating in VVER-1000 core. The results of this work can be used at calculations of the reactor control system efficiency and at definitions of CR and BAA residual efficiency.

The isotopic composition of VVER-1000 CR and BAA mainly was calculated with the German reactor cell code NESSEL. This code has been actively used for the last 6 years to prepare libraries of VVER neutron-physical constants depending on fuel burnup with their further use for modeling fuel campaigns (fuel loadings) of all without exception VVER reactors operating at Ukrainian NPPs.

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2.7 Practice of the engineering calculational code used for the WWER-1000 RPV fluence estimation

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ABSTRACT

This paper presents some results of using of the RETINA (KAB, Germany) code for fluence calculation. The paper describes the code accuracy assessment on a basis of the on-site experiments carried out at Ukrainian NPPs, calculation estimation of the accumulated fast neutron fluence at the WWER-1000 RPV of some Ukrainian NPPs. It briefly describes the current situation with fluence monitoring and developing requirements to calculational models for the fluence calculation.

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2.8 General structure and functions of the OPAL optimization system

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ABSTRACT

Presented version of OPAL – the in core fuel management system is under development also for core loading optimization of NPP Temelin (VVER1000 type reactor). Description of the algorithm of separate modules was presented in several AER papers.

The optimization process of NPP Temelin loading patterns comprises problems like preparation input data for NPP SW, loading searching, fixing and splitting of fuel enrichments, BP- assignment, FA rotation and fuel cycle economics.

In application for NPP Temelin has been used NPP Temelin code system (spectral code with macrocode).

The objective of fuel management is to design a fuel-loading scheme that is capable of producing the required energy at the minimum cost while satisfying the safety constraints. Usually the objectives are:

a) To meet the energy production requirements
(loaded fuel should have sufficient reactivity that covers reactivity defects associated with startup as well as reactivity loss due to the fuel depletion)

b) To satisfy all safety-related limits
(loaded fuel should preserve adequate power peaking limits (given namely LOCA), shutdown margins and no positive Moderator Temperature Coefficient (MTC).

c) To minimize the power generation cost.
($/kWh(e))

Flow of optimization process OPAL management system is in detail described and application for NPP Temelin cores optimization presented.

2.9 Innovation of genetic algorithm code GenA for VVER fuel loading optimization

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ABSTRACT

One of the stochastic search techniques - genetic algorithms – was recently used for optimization of arrangement of fuel assemblies (FA) in core of reactors VVER-440 and VVER-1000. Basic algorithm was modified by incorporation of SPEA scheme. Both were enhanced and some results are presented.

3. Core Monitoring, Surveillance and Testing

3.1 Summary of calculations and measurements for nuclear safety of damaged fuel on Paks Unit 2

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ABSTRACT

The NPP of Paks have achieved the important part of remove of fuels which were damaged because of the serious breakdown at 11 April 2003. The Hungarian Atomic Energy Authority have authorized the technology of the recovery. The verification of safety of technology was very important part of authorization. Within the scope of this it was had to demonstrate that the nuclear safety is could provide on high level under any circumstance. In this object six institutes included the NPP of Paks made analysis and gave expert opinions. The number of the nuclear safety analysis an opinions are more then 30.

In this paper I try to systematize and summarize the results of the analysis. Shortly it will be shown those conditions which guarantee the nuclear safety during removing of the damaged fuels. In a few words in the end of the paper further works will be listed.

3.2 Intrusion of resin into primary circuit

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ABSTRACT

During the refueling at the 1st unit of Bohunice NPP in 2005 a lot of sediment was found on the uper storage rack. This sediment was identificated as a filter resin. Resin was found in most of the fuel assemblies, pipes and tanks of the primary circuit and his auxiliary systems. Resin producer and WANO network was contacted in order to get information about similar events. Management of Bohunice NPP made a decision that primary circuit, fuel assemblies and auxiliary systems have to be cleaned. Subsequent cleaning extended outage by 31 days.

This paper summarizes causes, existing consequences and corrective actions. Accent was put on the hydraulic characteristics of the primary circuit measurement, power distribution core monitoring and the primary circuit water quality verification.

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3.3 Database of Temelin NPP Operational States and Its Use for Neutron Codes Validation

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ABSTRACT

Analogous to NPP Dukovany is made for NNP Temelin database of operational states. The database ETEBase is needed for the validation of various reactor computing codes, which will be developed during NPP Temelin life cycle and used for VVER-1000 core analyses. To obtain licenses in Czech republic for new neutron codes programs it is needed to publish technical report about validation and evaluated precision of the computer codes.

Benchmark data sets are processed from operational measurements data on Temelin VVER-1000 reactors. The input data from the NPP are verified; errors and inaccuracies are filtered out. Required data are chosen and processed, and then data are transferred to a form suitable for input data for neutron codes and for validation. Main data contained in benchmark dataset: effective time, boron concentration, thermal power, position of working group control clusters, inlet coolant temperature and flow rate of coolant water. Additional 3D-data are stored only for chosen time points (approx. 40 per cycle) – axial and radial power distribution in full and 60-degree core symmetry. Also datasets contain core description and list of outages during the cycle. At present, ETEbase contains processed data from these unit/cycles: 1-1, 1-2, 1-3 (partial of data), 2-1, 2-2.

3.4 Elimination of dynamic code from the process of spatial effect correction in scram drop measurements

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ABSTRACT

In the AER materials a great attention is given to the necessity of accounting for spatial effects of neutron flux redistribution in the VVER scram drop measurements. It has been emphasized that the said spatial effects manifest themselves the stronger the higher is the absolute value of negative reactivity inserted into the rector. It was proposed to determine the time dependence of the ratio of neutron density in the ion chamber's location to the neutron flux averaged over the reactor using the dynamic BIPR-8 KN or NOSTRA-type code.

At the same time it is known that in the insertion of a high negative reactivity (scram drop) the above ratio of neutron fluxes will remain unchanged during the whole measurement period. This fact makes it possible to use steady-state BIPR-type codes for this ratio.

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3.5 Further Development of the Dynamic Control Assemblies Worth Measurement Method for Advanced Reactivity Computers

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ABSTRACT

The dynamic control assemblies worth measurement (DCAWM) technique is a quick method for validation of predicted control assemblies worth. The DCAWM utilize space-time corrections for the measured out of core ionization chamber readings calculated by DYN 3D computer code. The space-time correction arising from the prompt neutron density redistribution in the measured ionization chamber reading can be directly applied in the advanced reactivity computer. The second correction concerning the difference of spatial distribution of delayed neutrons can be calculated by simulation the measurement procedure by dynamic version of the DYN 3D code. In the paper some results of DCAWM applied for NPP Mochovce are presented.

3.6 Modelling of power reactivity coefficient measurement

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ABSTRACT

Report describes results of modelling of power-reactivity coefficient analysis on power-level. In paper we calculate values of discrepancies arisen during transient process. These discrepancies can be arisen as result of experiment evaluation and can be caused by disregard of 3D effects on neutron distribution. The results are critically discussed.

3.7 Measurement of Reactivity Coefficients for Code Validation

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ABSTRACT

In the year 2003 measurements in the cold reactor state have been performed at the NPP KKI 2 in order to validate the codes that are used for reactor core calculations and especially for the proof of the shutdown margin that is produced by calculations only.

For full power states code verification and validation is quite easy because the calculations can be compared with different measured values, e. g. with the activation values determined by the aeroball system.

For cold reactor states, however, the data base is smaller, especially for reactor cores that are quite "inhomogeneous" and have rather high Pu-fiss- and 235U-contents. At the same time the cold reactor state is important regarding the shutdown margin. For these reasons the measurements mentioned above have been performed in order to check the accuracy of the codes that are used by the operator and by our organization for many years.

Basically, boron concentrations and control rod worths for different configurations have been measured. The results of the calculation show a very good agreement with the measured values. Therefore, it can be stated that the operator's as well as our code system is suitable for routine use, e. g. during licensing procedures.

3.8 Realisation of physics start-up tests with innovated I&C system at the Dukovany NPP

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ABSTRACT

The paper focuses on the innovation procedures of physics startup tests instrumentation used in DNPP (ANMS) connected with I&C system refurbishment of the Dukovany VVER 440/213 units. The following matters are further discussed in this paper :

- Changes in ANMS design, whose has been carried out due to reflect the situation connected with I&C innovation;
- Process of consistency checking during implementation stage;
- Results of the physics startup tests with upgraded ANMS in 2005 (introduction of Gd2 fuel on DNPP).

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3.9 Filtration neutron flux using Kalman filter

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ABSTRACT

To specified neutron flux there is necessary measurement, which includes complex operation and reflections which directs to a definite object. The partial or final object of procedure of measurement will be the best knowledge of true value of quantity, or a true rundown. With measuration on an ionization chamber, which are measurements a density of neutron flux, what is changing to an electrical signal. This electrical signal is have an equal bigness of neutron flux. We get a signal, what represents a value of neutron flux with a certain stream noise content

In fact, the density of neutron flux have certain stream noise content by reason of event character of detecting operation. To check off a vacant stream noise I used the Kalman filter. Kalman filter is a strong mathematic instrument, what is playing an important role in a leading methods and working up sensing signals. Kalman filter is a system of sets equation, which implemets predictive-corrective type of calculators, which are optimal in case, that they are minimize a covariance of error estimation.

One of these aspects of optimalised is that Kalman filter is including all of information, which are provided to it. It is works up all available measurements for an estimate a real value of signal, which we are intrerest to, with using knowledge methods and specific dynamics of contrivance. Kalman filter may by build-in a relationship of inverse kinematics and combinations all of measurements value and a knowledge of various dynamics methods for the best estimate of estate neutron flux.

(It will be included into the Proceedings only.)

4. Neutron Kinetics and Reactor Dynamics Methods

4.1 "AER working group D on VVER safety analysis" Report of the meeting in Garching, Germany, 6-7 April 2005

Compiled by
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ABSTRACT

AER Working Group D on VVER reactor safety analysis held its 14th meeting in the offices of GRS in Garching near Munich during the period 6-7 April 2005. The meeting followed the third workshop on the OECD/DOE/CEA VVER-1000 Coolant Transient Benchmark (V1000-CT) held at the same location on 4-5 April. Altogether 18 participants attended the Working Group D meeting, 12 from AER member organizations and 6 guests from non-member organizations. The coordinator for the working group, Mr. P. Siltanen (FNS) served as chairman. In addition to general information exchange on recent activities in the participating organizations, the topics of the meeting included:

- Code development and benchmarking for reactor dynamics applications.
- Safety analysis methodology and results.
- Dynamic benchmarks and solutions for the AER Benchmark Book.
- Future activities.

A list of participants and a list of handouts distributed at the meeting are attached to the report.

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4.2 RELAP5-3D© calculation of steam outlet header rupture of VVER-1000 NPP at HZP

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ABSTRACT

This presentation is devoted to one of the transients from the spectrum of steam line ruptures, which are analyzed for safety report purposes in Czech Republic. It is focused on the SG1 steam outlet header rupture of VVER-1000 at hot zero power conditions analysis with advanced thermal-hydraulic code RELAP5-3D©. The attention is addressed a reactor vessel and reactor core nodalization.

The steam line rupture followed by steam release results in a strong drain of energy from primary circuit, what presents significant decreasing of primary coolant temperature and pressure. Then the feedback reactivity coefficients in connection with reduction of coolant temperature on the reactor inlet would cause introduction of positive reactivity and they could result in reactor restart (after its shutdown).

Such transients are characterized by an unsymmetrical cool down of reactor and a strongly non-uniform distribution of power increase in the core. Safety analyses of such transients, where substantial changes of power distribution could occur, need apply thermal-hydraulic computational programs containing a 3D neutronic and 3D thermal-hydraulic model of the reactor.

The used computer codes, initial conditions for thermal-hydraulic calculations and basic results are presented. The DNBR value was monitored from the point of view of fuel integrity.

4.3 Severe accident analysis with the APROS SA code

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ABSTRACT

APROS is a commercially available system code for modelling one-dimensional, two-phase flow processes in nuclear power plants and other industrial facilities, which is developed by VTT in Finland. The new version of APROS is also capable to model containment processes and severe accident phenomena.

A detailed APROS model of a VVER440/213 type Paks unit was developed in the past years at the Institute of Nuclear Techniques of the Budapest University of Technology and Economics. The model contains the primary and secondary circuit with the emergency systems and the essential control and protection signals. The model was recently extended within the frame of a PHARE project, thus making capable to calculate containment response and beyond design basis accidents. Core degradation and melt, corium pool formation and relocation to the bottom of the reactor pressure vessel, and RPV failure can be investigated with this new model. The system code used for the modelling is the latest version of the APROS-SA software package.

The results of Station Black-out and large break LOCA scenarios were compared to previous MELCOR results.

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4.4 Mild transients in VVER-440type reactors simulated by the coupled ATHLET/KIKO3D code system

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ABSTRACT

The planned upgraded power and parallel application of new fuel type require the renewal of the relevant chapter of the safety analysis report. In this work the improved safety enhancement measures introduced in the plant, since 1997 are used. The fulfilment of the fuel design basis requirements, the acceptance criteria of the DBA analyses must be investigated during the normal and selected accidental conditions. The most dangerous scenarios have been investigated and presented in the former AER symposium. The "withdrawal of one control rod" and "initial phase of main steam line break" events are analysed by the ATHLET-KIKO3D program.

The postulated steam line break inside the containment at a VVER-440 type reactor was reanalysed. The initiating event is an asymmetric break of the main steam line at the end of the equilibrium fuel cycle with profiled Russian fuel working under upgraded power condition (1485MW). The break causes overpressure in the containment, which initiates a reactor shut down with one control rod stuck in its upper position, conservatively. In the scrammed reactor the break causes an overcooling of the primary circuit, the scram reactivity could be compensated and the reactor could become recritical. The calculation was continued until the highly borated water from the high pressure injection system terminated the supposed power excursion.

Another typical initiating event belonging to RIAs is the uncontrolled withdrawal of a control assembly at full power (1485 MW) with 2 cm/s velocity value from the most unfavourable position permitted by the operational rules (125 cm). During normal operation the regulating control assembly group consisting of 7 assemblies the only one, which is movable by the Unit Power Controller (UPC). Some electrical failure could cause, that nor the Emergency Reactor Protection (ERP) neither UPC can influence the motion of that group, but one of them starts to move upwards. As the transient slow enough the asymmetric power increase causes asymmetric coolant distribution.

4.5 Calculation Studies Of Transient Connected With Uncontrolled One Cluster Withdrawal With Subsequent Working Of Automatic Power Controller

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ABSTRACT

Calculation studies of transient related to uncontrolled one cluster withdrawal at the rated power are performed. As conservative factor subsequent working of automatic power controller and emergency protection failure are considered. Such assumptions lead to sizeable power distribution deformation in reactor core. For transient studies real fuel loading is chose and real algorithm automatic power controller working is modelled. Change of the main neutron and thermal-hydraulics parameters of WWER-1000 reactor core during transient are presented. Modeling was performed with the use of spatial kinetics computer code DYN3D used nodal method to calculate distribution of neutron flux in core.

4.6 3D- Modelling of neutron kinetics and thermal-hydraulic processes in VVER reactors by coupled computer code ATHLET/BIPR8KN in case of asymmetrical work of equipment

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ABSTRACT

The results of modelling of three-dimensional neutron kinetics and thermal-hydraulic processes in VVER-1000 reactor by program complex ATHLET/BIPR8KN is considered at asymmetrical work of the equipment. It is examined two processes. In the first one the cutting off a steamgenerator on the second circle of one of loops (closing feedwater valve and closing SIV) at low reactor power, resulting to given loop temperature rise, non-uniform distribution of temperatures on a core inlet is modelled. Comparison with experimental data is presented. In the second case - the break of a steam line, resulting to decrease of temperature on reactor inlet near one of the loop with following consequences - non-uniformity of temperatures in assemblies inlet, resulting to growth of power in one of reactor sectors due to feedback is considered.

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4.7 Numerical methods in the KIKO3D three-dimensional reactor dynamics code

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ABSTRACT

The goal of this paper is to give an overview of the numerical methods applied in KIKO3D three-dimensional reactor dynamics code. The focus is on the presentation of a new, more effective method (Bi-CGSTAB) for accelerating the large sparse matrix equation solution in the KIKO3D code. The validation of this method has been performed by solving of a VVER-1000 kinetic benchmark problem. The results has been obtained by the old (GMRES) and new (Bi-CGSTAB) methods are compared.

5. Criticality Safety

5.1 AER Working Group E in 2005

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ABSTRACT

The 10th meeting of the AER Working Group E "Physical Problems on Spent Fuel, Radwaste and Decommissioning of Nuclear Power Plants" was held in Modra - Harmonia, Slovakia, on 19-20 April, 2005.

The meeting was focused on the following topics:

- new SCALE 5
- burnup credit implementation
- storage and cask safety analyses
- PIE (Post Irradiations Experiments)
- IRT fuel
- miscellaneous

For the first time the Norwegian and American analysts took part in the meeting. Total number of participants was 23 from 8 countries.

5.2 Preparation and verification of libraries for ORIGEN-S in SCALE4.4a, with cross-sections for WWER-1000 fuel

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ABSTRACT

New WWER-1000 fuel libraries with cross-sections were created, they are intended to work with the ORIGEN-S module of the SCALE 4.4a system . The used model is described and main input data about geometry and material composition of WWER-1000 fuel assembly, densities, temperatures, masses and others are given too. Comparison by nuclide concentrations, between SCALE 4.4a with the 17x17 library for PWR and tvsm1000 library for WWER-1000, and the HELIOS-1.5 code is realized. Comparison by radioactivity and decay heat between the libraries 17x17 for PWR and tvsm1000 for WWER-1000 is realized for different nuclides and total.

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5.3 Criticality safety analysis of a WWER-440 fuel cooling pond at the Kozloduy NPP

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ABSTRACT

Results from a criticality safety analysis of the cooling pond of Unit 3 of the Kozloduy NPP in the case of zero boron content are presented and discussed. The dependence of the effective multiplication factor on burnup, cooling time and water density is explored. The uncertainty induced by different simplifications of the computational model, as well as by the assumed fuel composition at non-zero burnup values, is quantified. The analytical procedure is based on the SCALE-4.4a code system.

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5.4 Calculations of criticality, nuclide compositions, decay heat and sources for VVER-440 fuel by new version of the SCALE 5 code

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ABSTRACT

In this article are compared theoretical results by new version of the SCALE 5 code with experiments or other theoretical calculation for:

1. criticality - measurement on ZR-6 and LR-0
   - numerical benchmark No.1, 3 and 4 (CB1, CB3, CB4)
2. nuclide compositions - measurement in Kurchatov institute for 3.6%
   - measurement in JAERI (PWR 17x17)
   - numerical benchmark No.2 (CB2)
3. sources and decay heat - numerical benchmark No.2 – Source (CB2-S)

The focus is on modules KENO VI, TRITON and ORIGEN-S.

5.5 Monitoring of the fuel elements condition in a burnt fuel interim storage at the NPP SE-EBO

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ABSTRACT

Cladding of fuel elements is the best way how to keep the fission products inside the fuel elements and keep the environment safe. Hence is really important to make this cladding that it will be resistant toward many influences, witch have negative effects on their damage. The cladding plays big role in safety operation, transportation and also in final deposition of burnt nuclear fuel. Therefore is the research of these materials so important.
Cladding requirements:

- small neutron absorption,
- radiation, mechanical and chemical stability during transportation and also during his reside in reactor,
- high thermal conductivity.

There are many materials witch are used in reactors for cladding of fuel elements:

- magnesium alloys (for metallic uranium fuel and gas cooled reactors),
- zirconium alloys (for oxid uranium fuel, PWR,BWR,FBR),
- stainless steel (for ceramic fuel, HTR,FBR with Na),
- graphite (HTR,HTTR,GHTR,PBMR).

The corrosion, hydriding, growth and creep performance of current Zirconium alloys does not permit the more aggressive operating and environmental conditions required to achieve higher duty and longer cycle.

There are many systems for monitoring of fuel elements condition. The most used are:

- Sipping in core (“during” operation) ,
- Integral leaking test of fuel containers for transfer of fuel elements (C-30),
- Sipping in pool, ANF Ultratest, Sipping test on IEA R1 (during storage).

The detection of leakages in a wet burnt fuel interim storage by system Sipping in pool is excellent, but it took too much time, because when the interim storage is full, there are 14 200 spent fuel elements. Testing of one fuel element took about 30 minutes. So the time for finding one leaking fuel element is too long.
Thence, a quicker detector was found. This detector use special absorbent called Nifsil, which has big effective cross-section capture for ^{134}Cs and ^{137}Cs. Big advantages of these detectors are their easy and cheap manufacturing (also the Nifsil), easy and quick manipulation with detectors, compatibility for both (T-12, KZ-48) spent fuel containers and clear evaluation.

5.6 Computation of Selected Nuclides and Fission Gasses in UOX and MOX Spent Fuel by TRANSURANUS

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ABSTRACT

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5.7 Calculation of VVER-440 Nuclide Benchmark (CB2)

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ABSTRACT

The present paper is intended to show the results, obtained at the INRNE, Sofia, Bulgaria on the benchmark task, announced by L. Markova [1] at the sixth Symposium of AER, Kirkkonummi Finland 1996.

(It will be included into the Proceedings only.)

6. Spent Fuel

6.1 AER Working Group F in 2005

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ABSTRACT

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6.2 The IAEA international project on innovative nuclear reactors and fuel cycles (INPRO): Status, Development of approaches and Outlook

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ABSTRACT

During the last fifty years remarkable results have been achieved in the application of nuclear technology for the production of electricity. Looking ahead to the next fifty years it is clear that the demand for energy will grow considerably and also new requirements have to be fulfilled for the way nuclear energy will be supplied. UNCSD, WSSD, IPCC and others have emphasized the substantial growth in 21st century energy supplies needed to meet sustainable development (SD) goals. This will be driven by continuing population growth, economic development and aspiration to provide access to modern energy systems to the 1,6 billion people now without such access, the growth demand on limiting greenhouse gas emissions, and reducing the risk of climate change. .A key factor to the future of nuclear power is the degree to which innovative nuclear technologies can be developed to meet challenges of economic competitiveness, safety, waste and proliferation concerns.

There are two major international initiatives in the area of innovative nuclear technology: the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycle (INPRO) and the Generation IV International Forum. Following a resolution of the General Conference of the IAEA in the year 2000 an International Project on Innovative Nuclear Reactors and Fuel Cycles, referred to as INPRO, was initiated.

Based on scenarios for the next fifty years, requirements for the different aspects of the future of nuclear energy systems, such as economics, environment, safety, waste, proliferation resistance and infrastructure have been identified as well a methodology developed to assess innovative nuclear systems and fuel cycles. On the basis of this assessment, the need for innovations in existing nuclear technology, to be achieved via research, development and demonstration (RD&D), can be defined.

INPRO developed the above-mentioned requirements during its first step, called Phase 1A, which lasted from 2001 to middle of 2003. In the following second step, called Phase 1B (first part), INPRO organized 14 case studies (8 by national teams and 6 by individuals) to test and validate the methodology. INPRO has finished end of 2004 the first part of Phase 1B, by issuing an IAEA report (TECDOC1434) with an upgraded methodology based on the recommendations given in the case studies.

The paper summarizes the status of INPRO as well as the main results and provides an outlook on the future activities. The current status of the INPRO project and some examples of the development of the Manual for the users of INPRO Methodology are presented.

A method for assessing the operating risk of innovative designs using functional risk informed assessment of the conceptual design and potential consequences of incidents is described..

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6.3 Transmutation Comparison of VVER-440 and PWR Assemblies

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ABSTRACT

Regular assemblies of two reactor types - VVER-440 and PWR are compared. Comparison is based on RED-IMPACT (project in FP6 of EU) 'internal' numerical benchmark. Burnup of UOX and MOX assemblies is modelled by spectral code HELIOS 1.8. Infinite multiplication factor, isotopic composition, spectrums and parasitical absorption are compared. Reasons of differences are discussed.

7. CFD Codes Application

7.1 On Application of CFD codes to problems of Nuclear Reactor Safety

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ABSTRACT

The "Exploratory Meeting of Experts to Define an Action Plan on the Application of Computational Fluid Dynamics (CFD) Codes to Nuclear Reactor Safety Problems" held in May 2002 at Aix-en-Provence, France, recommended formation of writing groups to report the need of guidelines for use and assessment of CFD in single-phase nuclear reactor safety problems, and on recommended extensions to CFD codes to meet the needs of two-phase problems in nuclear reactor safety. This recommendation was supported also by Working Group on the Analysis and Management of Accidents and led to formation of three Writing Groups. The first Writing Group prepared a summary of existing best practice guidelines for single phase CFD analysis and made a recommendation on the need for nuclear reactor safety specific guidelines. The second Writing Group selected those nuclear reactor safety applications for which understanding requires or is significantly enhanced by single-phase CFD analysis, and proposed a methodology for establishing assessment matrices relevant to nuclear reactor safety applications. The third writing group performed a classification of nuclear reactor safety problems where extension of CFD to two-phase flow may bring real benefit, a classification of different modelling approaches, and specification and analysis of needs in terms of physical and numerical assessments. This presentation provides a review of these activities with the most important conclusions and recommendations.

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7.2 Recent results of CFD analysis of coolant mixing in the reactor pressure vessel

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ABSTRACT

From the point of view of accident transient analysis it is essential to know the coolant mixing in he reactor pressure vessel, since it determinates the distribution of the coolant temperature and boron concentration at the core inlet.

The so-called mixing factors (i.e. the weight of the coolant of a primary loop at the inlet of a given fuel assembly) are known from measurements and calculations for normal state of the reactor, i.e. with six operating MCPs. In case of less than six operating loops the mixing factors can be determined with experiments or numerical simulations.

In this paper CFD simulations are presented for the determination of the mixing factors in case of 3, 4 and 5 operating MCPs. The calculations were performed with the code CFX-5.7. According to the results for 5 or 4 operating loops the coolant flows into the downcomer in sectors poorly mixing with each other. However, assuming 3 operating loops this kind of symmetric flow disappears.

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7.3 CFD analysis of thermal stratification in the primary circuit

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ABSTRACT

The thermal stratification can lead an important role in the aging of the NPP piping because of the stresses caused by the temperature differences and the cyclic temperature changes. Therefore it is essential for the strength analyses to point out the affected pipes, and the thermal hydraulic parameters of the stratified flow.

For the investigation of the stratified flows the CFD codes provide an effective tool, with which the development and the breaking up of the stratification and the temperature distribution can be determined. The main difficulty of these CFD simulations is the uncertainty of the boundary conditions.

In this paper some results of CFX simulations are presented concerning the pressurizer surge line, the primary loops and the inlet of the HPIS. The results show that the thermal hydraulic analysis of certain pipes can be performed with CFD simulations, but for the determination of the boundary conditions further simulations with system codes, or measures are required.

7.4 Multidimensional modelling of temperature distribution in spent fuel pools of VVER-1000 and VVER-440 using FLUENT CFD code

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ABSTRACT

The paper presents results of CFD calculations of spent fuel storage pool at VVER-440 and VVER-1000 units. The calculations were performed by the Fluent 6.2 CFD code. Standard nuclear safety problems of spent fuel pools, such as k_{eff} calculation or spent fuel pool dry-out have been frequently discussed in many other papers. This paper pays special attention to more technical problems related to spent fuel pool operation in the Czech Republic NPP's Dukovany and Temelín. Following several problems had been identified during nuclear power plant operations and shutdown procedure validation:

- Inadequate water temperature and water level measurements
- Repeated cracking of pool stainless steel lining
- Lack of data for shutdown procedure validation

The first two items were supposed to have a common cause – significant non-uniformity of pool water temperature fields and related strong buoyancy effects. To verify this assumption and to solve above-mentioned problems we have analysed flow patterns in spent fuel pools and temperature fields at pool walls using the Fluent CFD code. Both steady state and transient calculations have been performed. This paper also includes basic comparison of flow pattern in spent fuel pools of VVER-440 and VVER-1000 and evaluation of typical large pool systems modelling features.

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7.5 CFD analysis of the service shaft during the recovery work of the damaged cleaning tank in the Paks NPP

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ABSTRACT

The recovery work of the cleaning tank that has suffered a serious incident in 10-11th of April, 2003 starts this year in the 2nd unit of the Paks Nuclear Power Plant. During this work the service shaft will be operated in a low-coolant-level operational mode. Since the operators of the damaged fuel removing equipment will work standing on a platform just above the surface of the coolant of decreased level, protecting them against unnecessary personal doses is a very important task.

From this viewpoint, the coolant of the service shaft plays double role. First, the few meters high layer of coolant between the working platform and the damaged fuel is an important part of the biological shielding for the workers. On the other hand, due to the considerable amount of radioactive contamination dispersed into the coolant, it is also a source of radiation. Therefore it is a very important question that how the distribution of the contamination shapes in the service shaft in different operational modes of the coolant cooling and purification systems.

Therefore, a complex 3D CFD model of the damaged cleaning tank, the service shaft with decreased level and the coolant cooling and cleaning systems has been developed at the Institute of Nuclear Techniques of the Budapest University of Technology and Economics. With this model a wide parameter study by performing 13 quite long transient calculations is under way. In this parameter study the effects of the following parameters on the contamination distribution are investigated:
-Mass flow of the cooling system of the service shaft – mass flow of normal and incidental operational mode,
-Mass flow of the coolant purification system of the 2nd unit – mass flow of normal and incidental operational mode,
-Difference between the inlet temperatures of the coolant entering from the cooling and the purification systems,
-Operation and effectiveness of the mechanical filter that is built into the protective flange of the cleaning tank.

In this paper the developed 3D CFD model for the code CFX-5.7, typical results and the evaluation of the parameter study are presented.