INFORMATION ABOUT AER WG A ON IMPROVEMENT, EXTENSION AND VALIDATION OF PARAMETRIZED FEW-GROUP LIBRARIES FOR VVER 440 AND VVER 1000

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ABSTRACT

Joint AER Working Group A on „Improvement, extension and validation of parameterized few-group libraries for VVER-440 and VVER-1000“ and AER Working group B on „Core design“ eighteenth meeting was hosted by ŠKODA JS a.s. in Plzeň (Czech Republic) during the period of 4th to 6th May 2009. There were present altogether 16 participants from 6 member organizations and 13 presentations were read.

Objectives of the meeting of WG A are: Issues connected with spectral calculations and few-groups libraries preparation, their accuracy and validation.

Presentations were devoted to some aspects of few group libraries preparations and to the benchmark dealing with VVER-440 follower modeling in calculations.

Gy. Hegyi gave some new information about NURESIM-NURISP EU project (ZR-6), ***R. Zajac*** spoke about the development of data libraries for codes BIPR-7 and PERMAK, ***P. Dařílek*** compared FA's with Gd during burning process and ***Yu. Bilodid*** described further development of plutonium-based burnup history modeling in DYN3D burnup calculations.

G. Hordósy presented results of control rod follower induced local power peaking computational benchmark and ***J. Švarný*** described Monte Carlo VVER-440 control rod follower benchmark computations.

Future activities are also shortly described in the end of the paper.

INTRODUCTION

Joint AER Working Group A on „improvement, extension and validation of parameterized few-group libraries for VVER-440 and VVER-1000“ and AER Working group B on „core design“ held their eighteenth meeting in Pilsen during the period of 4th to 6th May 2009. There were present altogether 16 participants from 6 member organizations (four states

- List of participants is attached in the end of this paper) and 13 presentations were read (List of presentations is also attached).

Objectives of the meeting of WG A: Issues connected with spectral calculations and few-groups libraries preparation, their accuracy and validation.

PRESENTATIONS

Six papers were presented in frame of WG A:

Gy. Hegyi gave some new information about **NURESIM-NURISP EU project (ZR-6).**

In the beginning of his speech he repeated the purpose of **NURESIM - European Platform for Nuclear REactor SIMulations** that has been divided into five parts namely:
SP1 – Core physics, SP2 – Thermohydraulics, SP3 – Multiphysics, SP4 – Sensitivity and Uncertainty Analysis, SP5 – Integration and NURESIM - Users Group (AREVA TRACTBEL, FORTUM, FZK)

The whole project is subdivided into three parts, where:
NURESIM: target with first significant possibilities
NURISP: consolidation + extension
NURENEXT: confirmation +rationalization +extension

NUclear Reactor Integrated Simulation Project is characterized by the following features: **Continuity -** based on the NURESIM platform & its results, **New developments -** multiscale & multiphysics coupling, **New field** – fuel, **Application -** to LWR(Gen-…) to PWR, BWR, VVER and Links with other EU projects.

Work having been done by APOLLO2.7 (Pij, Sn, MOC solvers)
20 measurements (pitch/enrichment/temperature) - Multiplication coefficient (keff), Temperature Coefficient Measurement

Measured power distribution in the M7 macrocell and its simulation by the APOLLO2.7 code.
- Necessity of further benchmarking

- Evaluation uncertainties coming from XS inaccuracy
Experiments are important - can reflect: both the XS uncertainties and the applied methods.

Zr-6 critical facility - highly accurate, integral data for hexagonal lattices.
Validation of calculation tools by "clean" experiment - code should be validated on the basis of reliable experiment.

The multiplication factor is crucial parameter **but** one should keep in mind its integral nature.

K_{eff} accuracy <=> crit. boric acid concentration; length of cycle, etc.

$\Delta\rho/\Delta C_B$ => sensitivity & safety

The role of database (ENDF/B, JEF, etc).

NURISP:
Aiming at ***experimentally*** validated "best estimate" simulation tools:

- Preparing a consistent set of numerical & ***experimental*** benchmarks (provided for verification, demonstration of the codes and databases)

Work plan:

T1.3 (CEA, FZD, PSI, INRNE, KFKI, NRI):
Development of calculation schemes and generation of XS libraries with APOLLO2 at the nodal level for **CRONOS2** and **DYN3D** diffusion and SP3 analysis in PWR and **VVER**.
T1.4 (PSI, INRNE, KFKI, NRI):
The goal is to verify and demonstrate the developed codes and tools in NURESIM CP platform on the different realistic kind of cells and lattices in power reactors (PWR, **VVER** and BWR).
Within the objective of pin-by-pin neutronics fidelity for safety margins analysis.

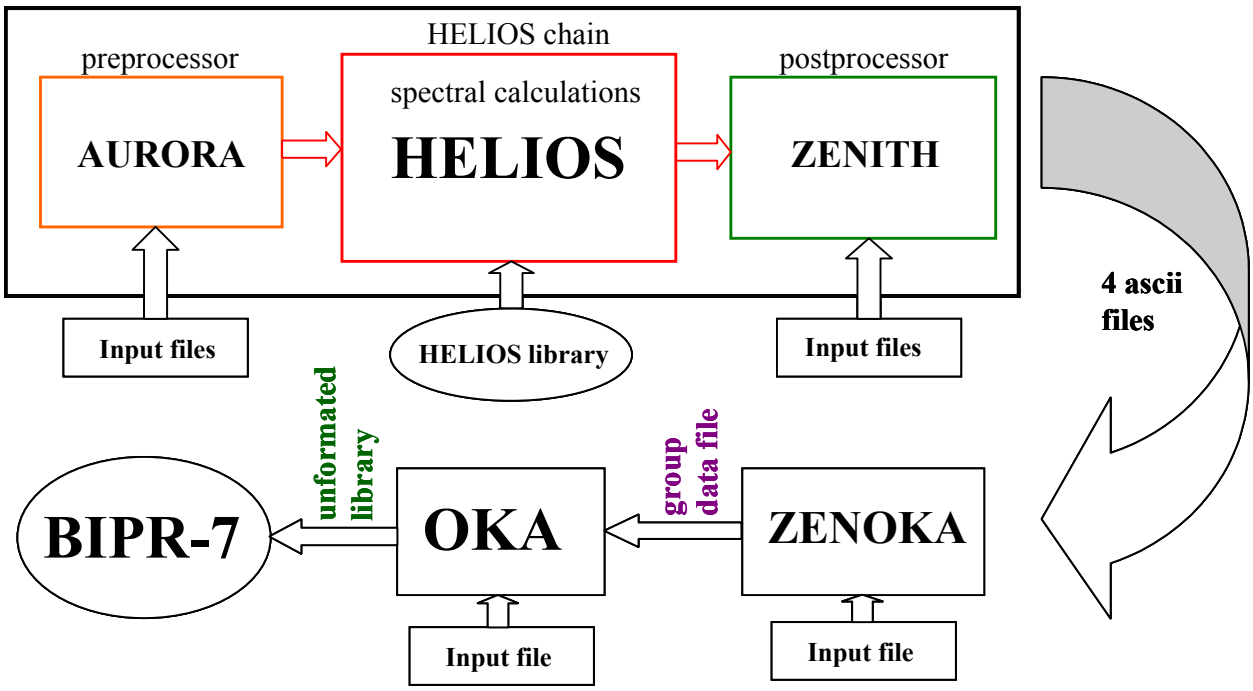
Comments:

•The (old as NO new) measurements are important
•Efforts are always needed in the field of validation
•This EU project draws some attentions
(A.E.: M.N. Zizin: **Vol.**103, No. 4, pp. 232-235)

Why all this is needed and what can be accomplished:
• An increase in the quality of the substantiation of the safety of our reactors being constructed abroad; increasing competition makes it necessary to be ready for more stringent requirements (in our country safety can be validated with existing programs)
• Formulation of the foundations for developing systems for computer-aided design of nuclear reactors; increasing the accuracy of calculations of reactor parameters and decreasing their uncertainty, decreasing the margins which are incorporated for inaccurate simulation;
• Preserving knowledge in the field of reactor simulation;
• Increasing the productivity of the labor of specialists;
• Preserving independence from applied foreign programs;
• Image: if we are a great energy power, then it is impossible not to have our own national reactor simulation project;
• Full use of the capabilities of supercomputers, microprocessor systems, and GRID networks; providing the reactor community with reference solutions;
• Means for preparing the next generation of reactor researchers;
• Possibility of independent expert analysis of various aspects of the safety of reactor facilities.

R. Zajac spoke about the development of **data libraries for codes BIPR-7 and PERMAK.**

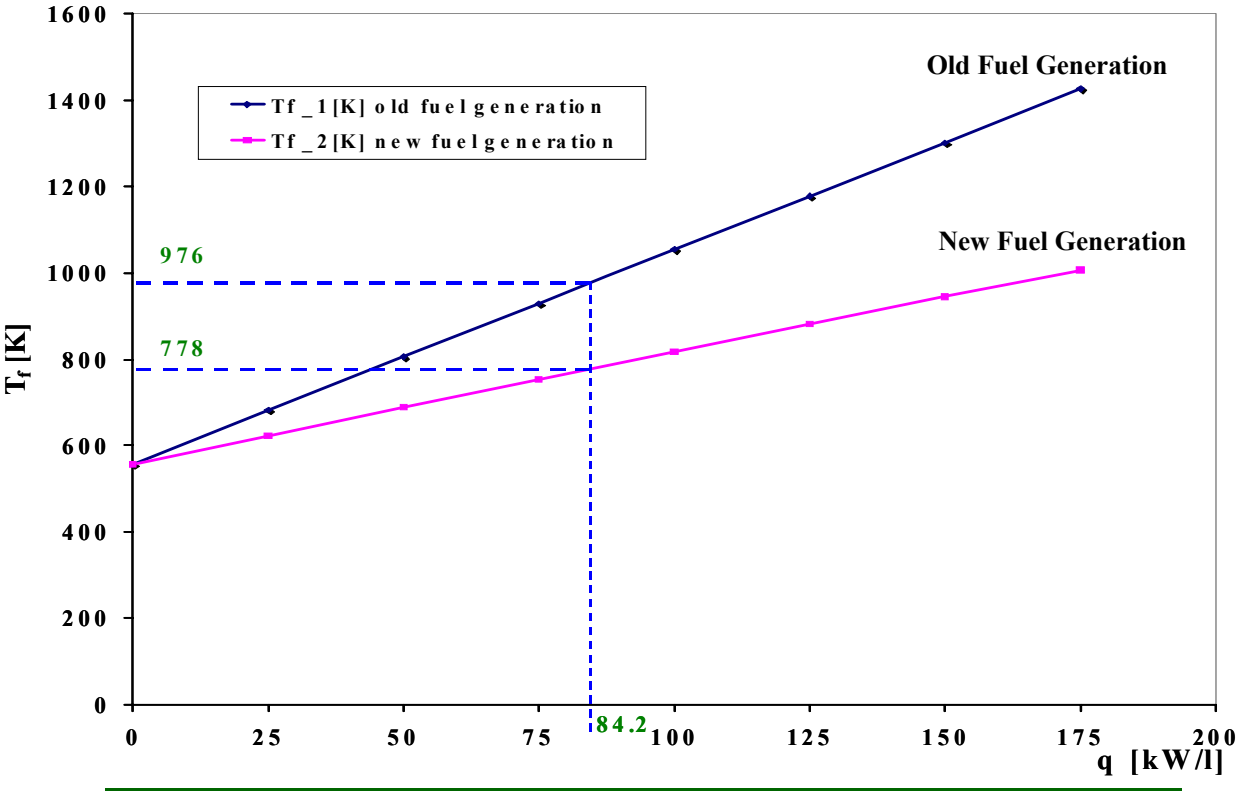
Based on the following Chart of data calculation, he described the whole process of macroscopic cross sections calculation:



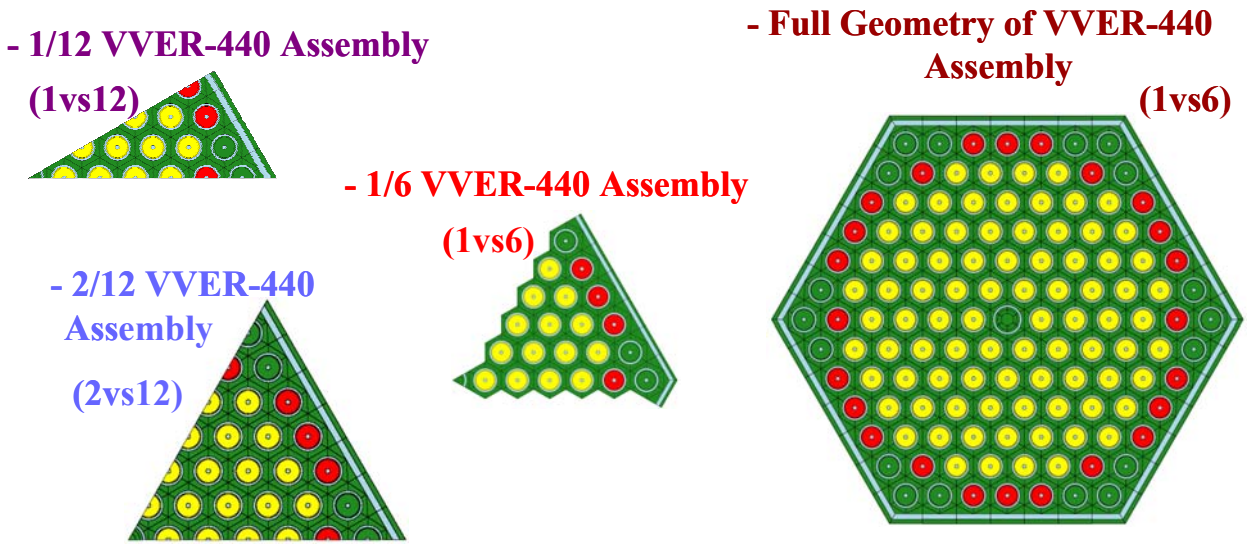
Next, he outlined Basic Modifications in the Data Calculation Process, namely:

1. The change of Fuel Temperature:

Old fuel generation: 976 K
New fuel generation: 778 K



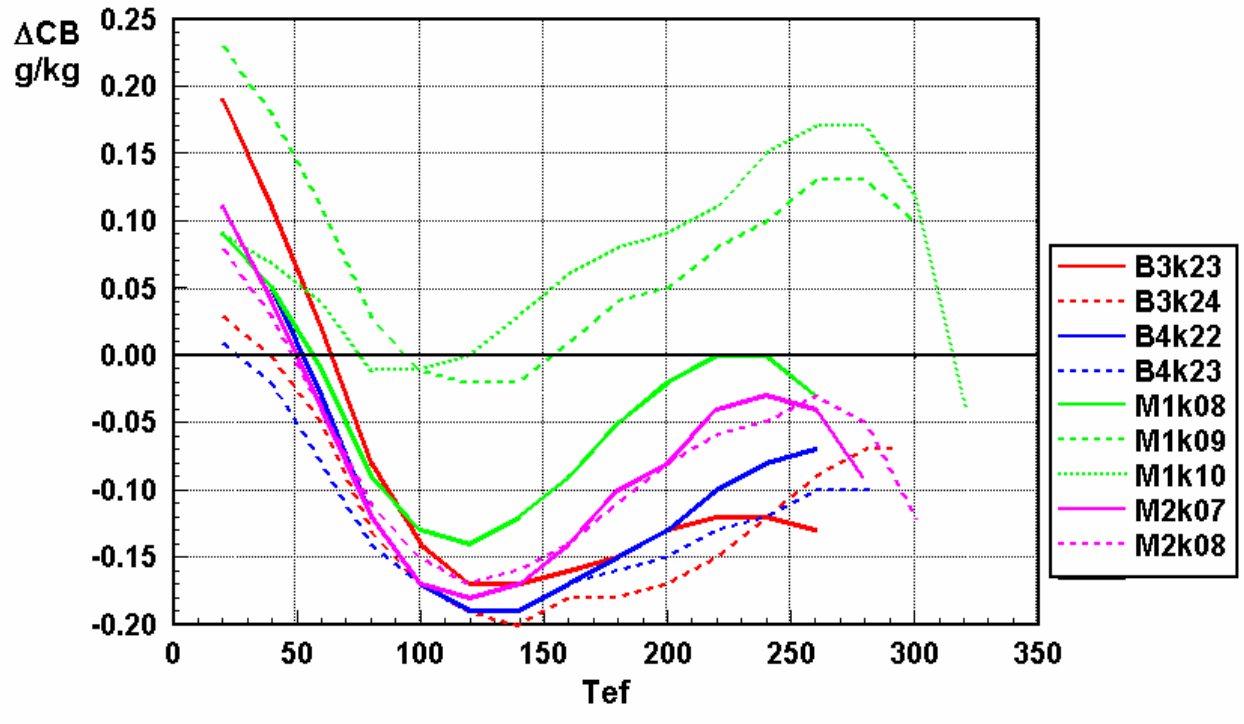
2. The change of VVER-440 Assembly Geometry:



Influence on K_{inf}, Sm, Xe, and Pu239 concentrations.

Comparison of **Power Reactivity Coefficient** [10^{-5}/MW], T_{in} = const., BOC, N= 90 % N_{nom} was performed.

The difference between calculation and experiment (**Boric acid concentration**) was evaluated.



The **Ratio** between β_i and β_{eff} was determined.

Conclusion:

- Both basic changes (modification of temperature and geometry) lead to the precision data calculations for computer codes BIPR-7 and PERMAK.

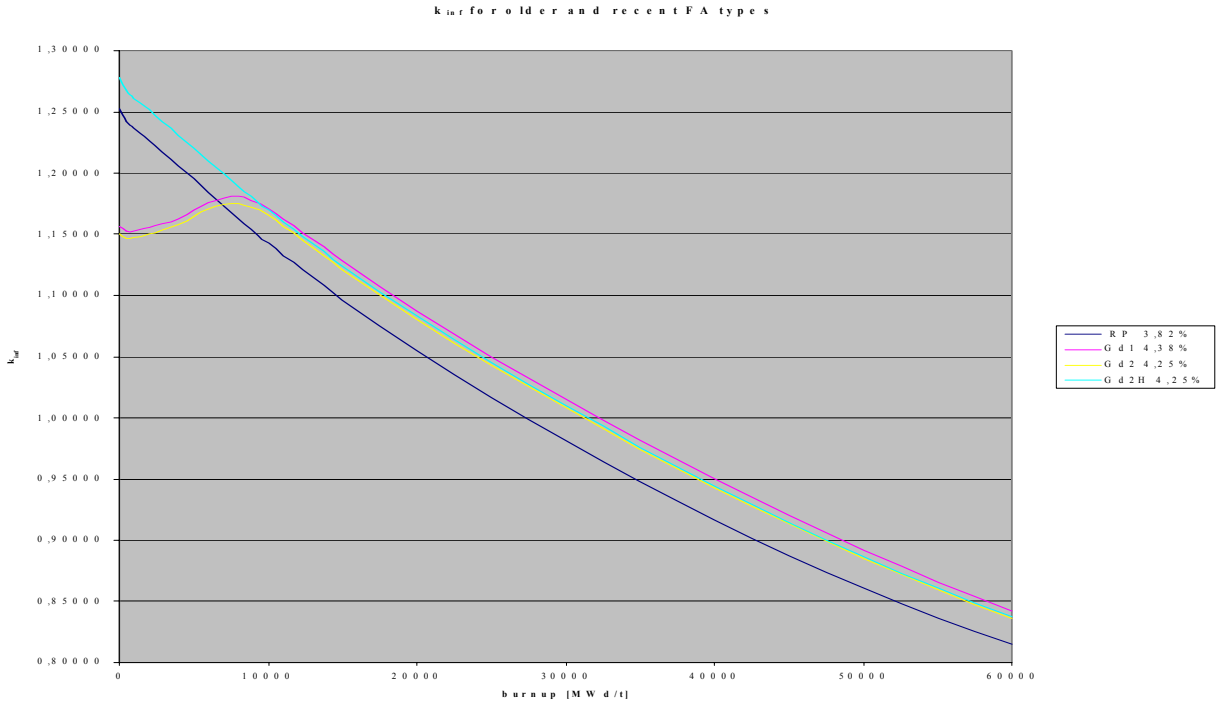
- Computer codes and NPP reactimers include no uniform data. This fact causes big differences between calculations and experiments.

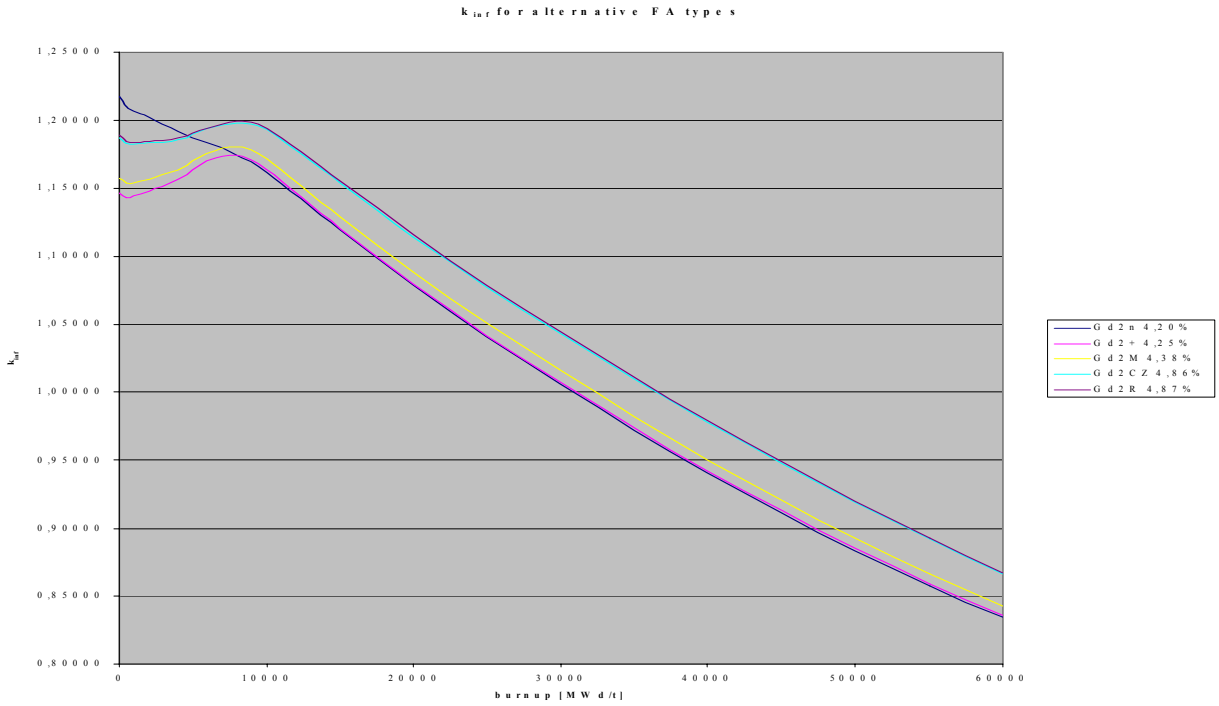
P. Dařílek compared **FA's with Gd during burning process.**

In the beginning of his speech he described recent FAs with Gd for VVER-440 (summarized in the following table)

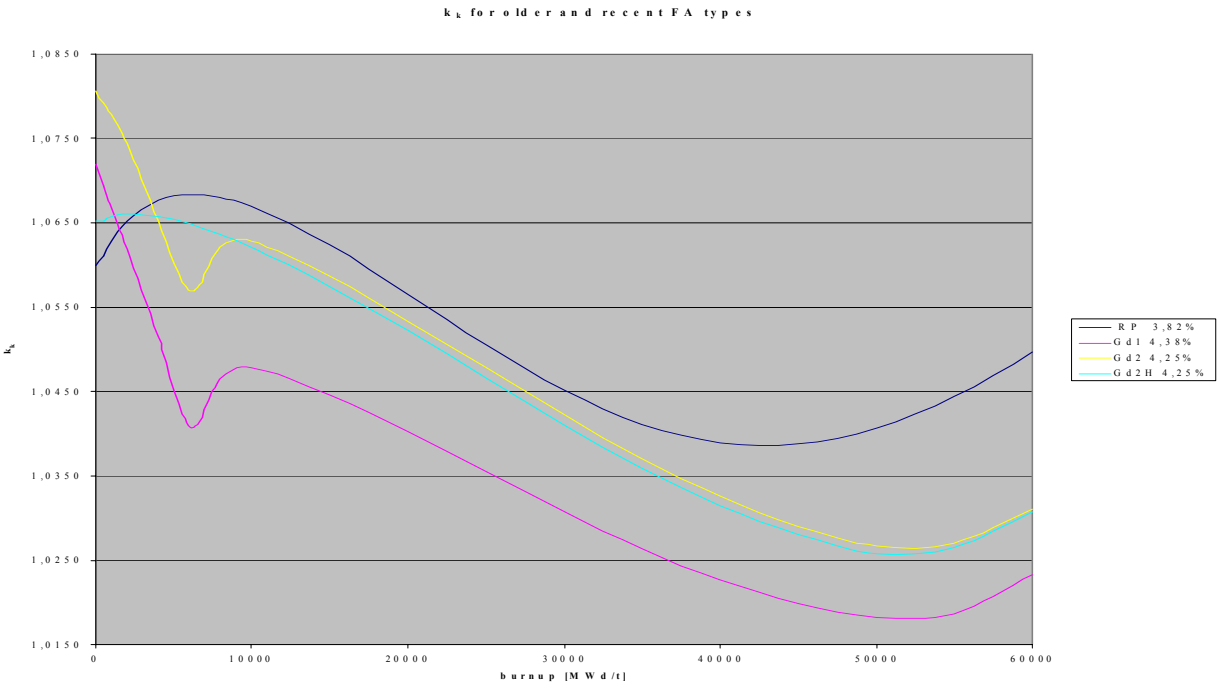
FA group	FA label	mean enrichment[wt%]	pins (enrichment[wt%]/number)					remark
			basic	1-st prof.	2-nd prof	3-rd prof	4-th with Gd	
older	RP	3,82	4,0/84	3,6/24	3,3/18	_	_	no Gd
	Gd1	4,38	4,6/84	4,0/30	3,6/6	_	4,0/6	
recent	Gd2	4,25	4,4/84	4,0/30	3,6/6	_	4,0/6	
	Gd2H	4,25	4,4/84	4,0/36	3,6/6	_	_	hypotet - no Gd
alternatives	Gd2n	4,2	4,4/81	4,0/24	3,6/18	_	4,0/6	
	Gd2+	4,25	4,6/54	4,0/30	3,7/30	3,3/6	4,0/6	
	Gd2M	4,38	4,6/84	4,0/30	3,6/6	_	4,0/6	
	Gd2CZ	4,86	5,0/84	4,0/30	4,4/6	_	4,0/6	
	Gd2R	4,87	4,95/102	4,6/18	_	_	4,0/6	

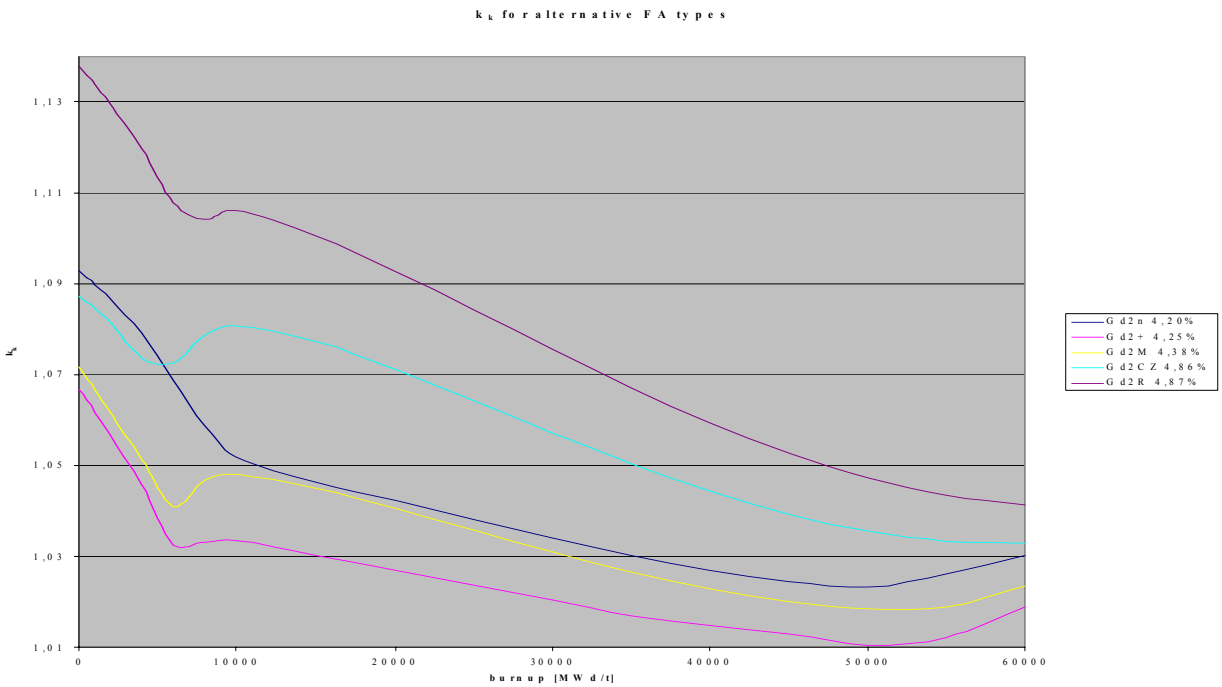
and compared multiplication ability of these FAs:





Then, he compared in-assembly power peaking:





He also presented visualization of burning process – (2-dimensional, pin powers characterized by colours, burn-up 0 ÷ 60000 MWd/t,1 MWd/t ~ 0,2 s, 1 FA ~ 12 s) for the following FAs: RP – 3,82 wt%, Gd2+ – 4,25 wt% and Gd2n – 4,2 wt%.

What Next?

FA group	FA label	mean enrichment[wt%]	k_{inf} local max.	k_k	
				max 1	max 2
older	RP	3,82	no	1,068	_
	Gd1	4,38	yes	1,072	1,048
recent	Gd2	4,25	yes	1,081	1,063
	Gd2H	4,25	no	1,066	_
alternatives	Gd2n	4,2	no	1,093	_
	Gd2+	4,25	yes	1,067	1,033
	Gd2M	4,38	yes	1,072	1,048
	Gd2CZ	4,86	yes	1,087	1,081
	Gd2R	4,87	yes	1,138	1,106
proposals	Gd2P a	4,55	yes	1,069	_
	Gd2P b	4,71	negligible	1,078	1,064

Yu. Bilodid described **further development of plutonium-based burnup history modeling in DYN3D Burnup Calculations.**

Standard **XS treatment scheme** was described in the beginning of his presentation.

Building of XS library from macroscopic XS for reference points and approximation coefficients.
Calculation of actual XS for each node by interpolation of reference XS from library using actual parameters of node.

On an example of moderator density history effect he explained Physical reason for history effects.

Reason of burnup history effects:
Deviations in parameters (T_m, ρ_m, T_f, C_B) influence neutron spectrum, spectrum influences nuclides creation/depletion processes in fuel during burnup.
Result: Change in nuclides content causes change in macroscopic XS.

The linear proportionality between change of homogenized XS and change of Pu^{239} concentration can be assumed. History effects can be taken into account by:

$$\Sigma_{actual} = \Sigma_{base} \cdot (1 + k_h \cdot \Delta Pu^9 / Pu^9_{base})$$

Σ_{base} – diffusion parameter, calculated in standard way;

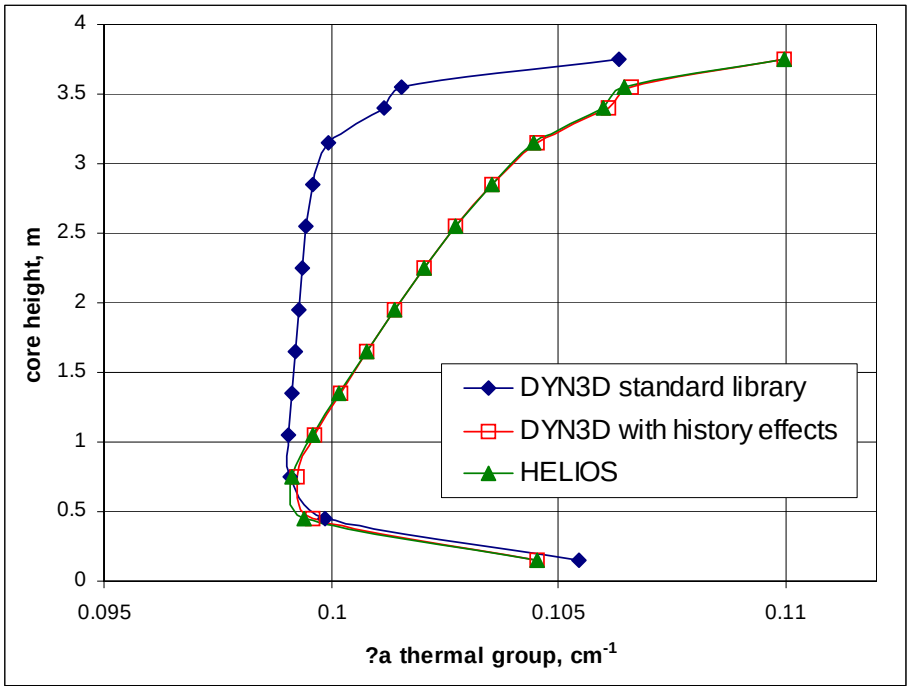
Pu^9_{base} – Pu^9 concentration in standard depletion;

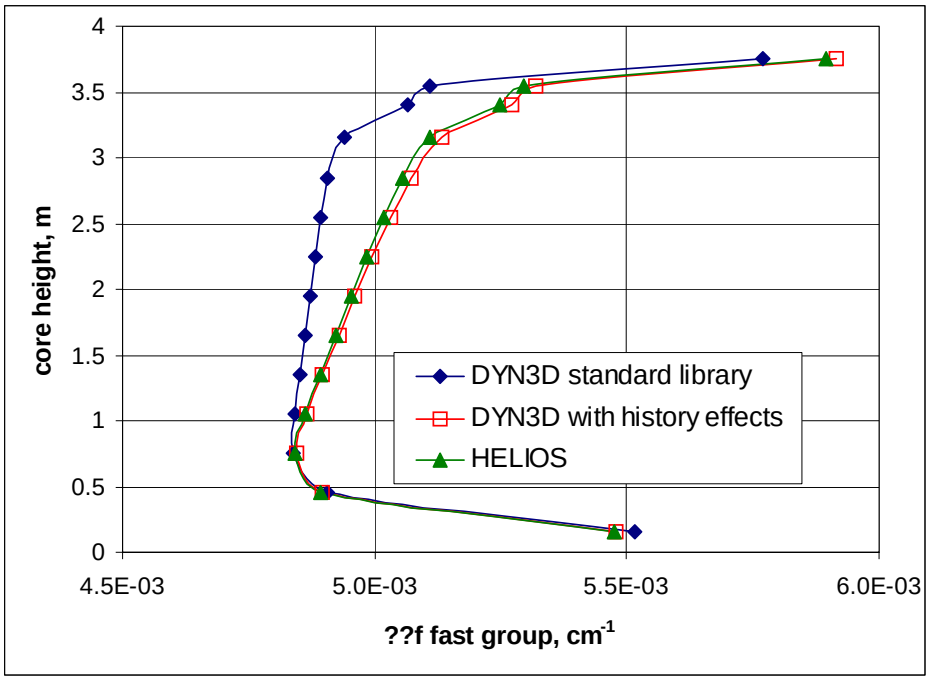
ΔPu^9 – difference between reference Pu^9 concentration and actual Pu^9 concentration in node;

k_h – proportionality coefficient, defined for each XS type.

Standard XS treatment scheme and also history - sensitive XS treatment scheme was described.

Results of test calculations:





Clear improvement in the cross section accuracy, due to the use of history parameters in DYN3D was observed.

Pu^{9} calculation in DYN3D was also comprehensively described.

Conclusion:

- Method is actually working.
- Implementation of historical correction shows a clear improvement of the cross section estimation.
- Sufficient accuracy has to be ensured for the DYN3D estimation of the actual Pu^{9} concentrations.

Next steps:

Improving accuracy of Pu^{239} calculation and validation.

J. Švarný described **Monte Carlo VVER-440 control rod follower benchmark computations.**

Validation of the coupler part of the VVER-440 control fuel assembly (CFA) calculations by measurements was performed within WG3 organized on series of experiments which were realized in the NRI Řež plc. in the years 2004 and 2005.

Calculation analysis of all participants of WG3 was performed mainly by Monte Carlo codes type MCNP.

For validation of design codes (MOBY-DICK, KARATE system, BIPR – PERMAK, …) the mathematical (code-to-code) benchmark was defined by ÚJV Řež. The benchmark configuration consists central VVER-440 CFA surrounded by one ring of VVER-440 fuel assemblies slightly extended to form a simple hexagon with outer radial reflecting conditions.

This code-to-code calculation model was adopted to the Skoda JS a.s. for MCNP-4C. It was found that detailed modeling of region of steel inserts, plenum and structures just above of the plenum have credible impact on critical points of pin power distribution. According sensitivity analysis provided, the optimal *model geometry* (see Fig. 1) of the coupler part of CFA have been used in present Škoda calculations. In this paper are presented four calculations (Variants) of two cases which are defined in original ÚJV Řež reference (Case 1 – central CFA without hafnium plates and Case 2 – central CFA with hafnium plates):

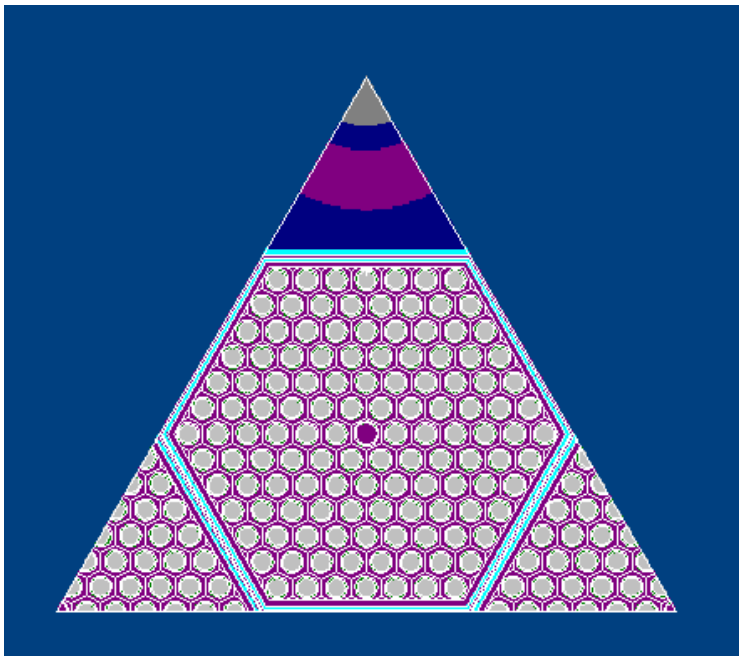


Fig. 1

Var. 1: (Equal to the Case 1 from the reference) CFA without hafnium plates inserted into depth 87.44 cm (distance of top of CFA fuel stack from top of other fuel stack was 102.68 cm).

Var. 2: CFA without hafnium plates fully withdrawn (distance of top of CFA fuel stack from top of other fuel stack was 10 cm).

Var. 3: CFA without hafnium plates partially inserted (distance of top of CFA fuel stack from top of other fuel stack was 56.33 cm).

Var. 4: (Equal to the Case 2 from reference) CFA with hafnium plates inserted into depth 87.44 cm (distance of top of CFA fuel stack from top of other fuel stack was 102.68 cm).

Axial cross-sections of calculated variants were also provided. (On top and bottom of this models are vacuum boundary conditions.)

Detailed description of structures of each variant was given in the presentation. Škoda model mostly agrees with reference ÚJV Řež model.

Identical characteristics

a) Identical Monte Carlo code MCNP-4C using the same cross-section library JEFF-3.1 temperature dependent (*.34c and *.37c and lwtr07.31t).

b) Identical dimensions of all structures (including dimensions of reflectors) except coupler part of control fuel assembly.

c) Identical outer boundary conditions.
d) Identical atom densities of fuel and structures (including reflectors) except coupler part of control fuel assembly.
e) Identical dimensions of steel inserts and plenum.
f) Close Monte Carlo calculation options: 50 000 particles per cycle and from 250 to 300 active cycles.

Different characteristics

g) Different dimensions and atom densities of coupler part of CFA (except steel and hafnium plates which were identical with the reference).
h) The central hole of pellet was smeared.
i) Spacer grids of CFA were included into fuel cladding.
j) In plenum were not used zirconium inserts but diluted steel of spring.

Results of the calculation:

A/ Criticality calculation

Variants	**k_{eff}**	**σ**	**k_{eff}(ÚJV)**	**σ**
Var. 1				
15 mil. nps	1.0494	0.00016	1.0492	0.00017
Var. 1				
36 mil. nps	1.0492	0.00012		
Var. 2	1.0597	0.00019		
Var. 3	1.0553	0.00017		
Var. 4	1.0481	0.00021	1.0480	0.00015

B/ Power distribution calculation

According to the code-to-code CFA benchmark (for better statistics) was calculated average pin power of pin rows („row pin power“) of outer fuel assembly and each row consists pins positioned parallel with CFA side. Row 0 represents 12 pins is in the central part of outer fuel assembly and row 6 comprises 7 pins close to the CFA.

Axial distribution was calculated in core height segmentation by 10 cm from the bottom of fuel stack to 140 cm and from this to 230 cm by 5 cm.

Var. 1 is a basic calculation with no Hf inserts and CFA partially inserted (CFA fuel stack 139.3 cm)

The relative errors of row pin power axial distribution in critical region (lower then 160 cm height) is not greater than 1%.

Comparison with ÚJV Řež calculation shows good agreement between both calculations is reached.

Precision of the Monte Carlo calculation was tested by alternative

Precision of the Monte Carlo calculation was tested by alternative calculation with 90 000 particles per cycle and 350 active cycles.

Var. 2 and **Var. 3** check alternative CFA insertion depth and its impact on the row pin power distribution.

Var. 4 is identical with Case 2 from reference, and calculation results were also presented.

Conclusions:

The Škoda calculation model of the coupler part of the CFA provides nearly the same results in k_{eff} and row pin power distribution like reference calculation performed in ÚJV Řež.

Positive impact of Hf plates on pin power distribution was preserved also in code-to-code benchmark calculation.

An alternative insertion depth of CFA into core was calculated. In the height 200 cm is impact of coupler on row pin power distribution approximately the same as in the height 150 cm (also due to burnup axially uniformly distributed.

G. Hordósy presented **results of control rod follower induced local power peaking computational benchmark.**

Introduction

In the beginning he named deficiency of the original coupler design:

- Relatively large amount of water in the coupler between the absorber and fuel part of the control assembly can cause sharp power peaking in the fuel rods next to the coupler.

- The power peaking can be especially high after control rod withdrawal when the coupler reaches low burnup level region of the adjacent assembly.

- Affection of linear heat generation rate and power ramp.

Modernized coupler: Hf plate in the critical region.

The problems of the calculations:

- Very complicated structure (geometry and composition data) of the coupler.
- Complicated spectral and 3D spatial effects of the neutron transport.
- Need for the verification of core design methods by 3D Monte Carlo calculations:

Definition of a VVER-440 control rod coupler benchmark without and with Hf plates.
- MCNP reference solution.
- KARATE-440 verification.

Description of linear pin power calculation in the KARATE-440 code system:

MULTICELL: Fuel assembly transport calculations with environment.
- Nodal and pin-wise 2-group cross section libraries. (Nodal XS, Fuel Pin Cell XS and Non-multiplying Cell XS.)

COLA: 1D transport calculation to evaluate albedos for reflector and absorber regions.
- Determination and parameterization of the two-group albedo matrices for regions not containing either boron steel or Hf, for regions containing boron steel and for regions containing Hf.

In the burnup calculations the ^{174}Hf, ^{176}Hf, ^{177}Hf, ^{178}Hf, ^{179}Hf and ^{180}Hf isotopes are treated. The burnout of the Hf isotopes is characterized by the time integral of the incoming partial current at the proper surface of the control assembly.

Determination of correction factors using MCNP 3D results.

GLOBUS: Nodal diffusion calculations (K_{eff}, C_B iteration, Flux and power distributions inside the core).

SADR: Pin-wise FD calculations with inhomogeneous flux boundary conditions.

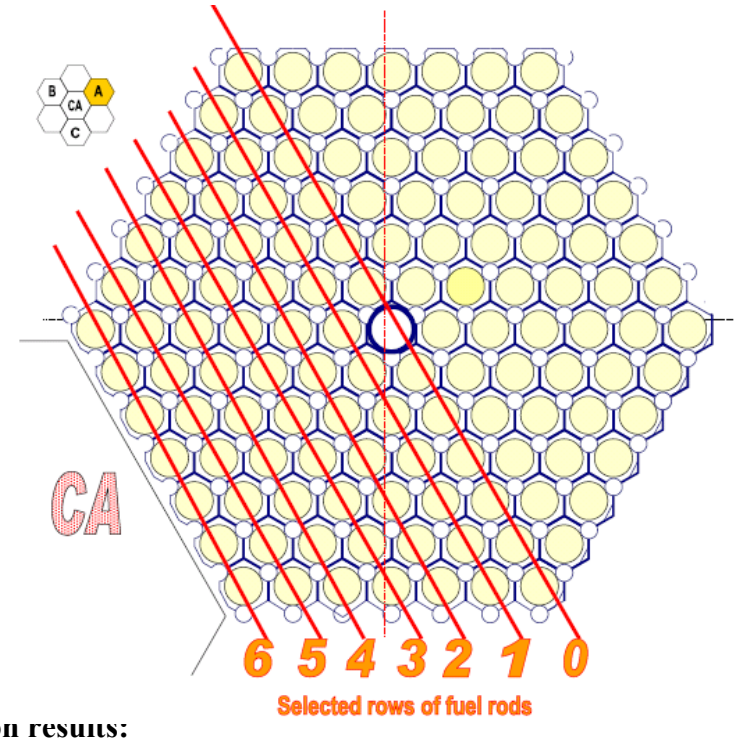
Outline of the benchmark:

- Horizontal cross section: hexagonal
- 1 central control assembly partially inserted
- 8 equivalent surrounding working assemblies
- Isotopic content of fuel rods: burnt, flat distribution
- Temperatures, water density: Flat distribution
- End of Cycle nominal conditions
- Reflecting boundary conditions on radial surfaces
- Axial homogenized reflector regions

Case 1 – central control assembly without hafnium inserts
Case 2 – central control assembly with hafnium inserts

Results to be reported:

- Keff .
- Power distributions of selected fuel rows normalized in row 0.



Calculation results:

	No Hf plate		Hf plate	
	Keff	σ	Keff	σ
MCNP Řež JEFF 3.1	1.0492	1.7E-04	1.0480	1.5E-04
MCNP AEKI JEFF 3.1	1.05025	1.1E-04	1.04898	1.1E-04
KARATE ENDF/B-VI	1.04462	-	1.04357	-

0.5% lower KARATE ENDF/B-VI k_{eff} than MCNP AEKI JEFF 3.1
Different library and coupler model.
Library effect: 0.4% lower.

Working Fuel Assembly	k_{inf}	σ
MCNP AEKI JEFF 3.1	1.0678	1.1E-04
KARATE ENDF/B-VI	1.0636	

Only two fuel rows are affected in the local power peaking.
The application of Hf plate removes the local power peak.

Comparison of ***power distributions***:
- MCNP REZ JEFF 3.1 axial mesh: 5 & 10 cm
- MCNP AEKI JEFF 3.1 axial mesh: 5 & 10 cm
- KARATE-SADR ENDF/B-VI axial mesh: 6.05 cm

SADR results:
- Bottom reflector peak is visible (finer mesh).
- The under prediction of local power peak by SADR at H=148 cm: 4%

Provided the relative error is the same at the linear power limit of 325 [W/cm], the absolute deviation is **13** [W/cm], which is in accordance with the **39** [W/cm] uncertainty of KARATE-SADR corresponding to **3σ**.

Summary & Conclusions

The coupler mathematical benchmark was solved by the KARATE code system using the same methods and approximations as in case of NPP applications.

Without Hf plate in the fuel pin row next to the problematic coupler section pronounced local power peak arises.

Though the under prediction of the peak by KARATE-SADR is 4%, it is in accordance with the applied uncertainty of KARATE-SADR.

The **application of Hf plate solves** the **problem**.

CONCLUSION

Six papers were presented on WG A, which are devoted to different topics.

Effort was concentrated on special effects like historical dependencies of cross sections.

Proper determination of boundary conditions is still an important issue. A benchmark was focused to this point. Conclusions from the solutions of this benchmark will result to improvement, if necessary, in industrial codes.

List of presentations

WG A

1.1 ***Gy. Hegyi:*** **Some new information about NURESIM-NURISP EU Project (ZR-6)**

1.2 ***R. Zajac:*** **The development of data libraries for codes BIPR-7 and PERMAK**

1.3 ***Gy. Hegyi, G. Hordósy, Cs. Maráczy and E. Temesvári:*** **Computations on Control Rod Follower Induced Local Power Peaking Computational Benchmark**

1.4 ***J. Švarný:*** **Monte Carlo VVER-440 Control Rod Follower Benchmark Computations**

1.5 ***P. Dařílek:*** **FA's with Gd - Analysis of Burning Process**

1.6 ***Yu. Bilodid:*** **Further development of plutonium-based burnup history modeling in DYN3D**

WG B

2.1 **T. Parkó: New version of neutron physical data display tool**

2.2 **I. Pos: Test calculations by C-PORCA code**

2.3 **P. Mikoláš, J. Švarný, V. Razým, M. Dostál and P. Krupař: VVER-440 Fuel Cycles Possibilities Using Modified FA Design**

2.4 **V. Krýsl, P. Mikoláš, D. Sprinzl and J. Švarný: Pin Power Distribution on a Small Testing Core Calculated by MOBY-DICK Code**

2.5 **J. Breza: Gd benchmark calculations in VUJE with upgraded library**

2.6 **K. Katovský: AER Gd-2 Fuel Benchmark Calculation Using Various MOBY-DICK Code Options**

2.7 **J. Bajgl: Remarks to the 5-year fuel cycle at Dukovany NPP**

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