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TECHNOLOGY RELEVANCE OF THE "UNCERTAINTY ANALYSIS IN MODELLING" PROJECT FOR NUCLEAR REACTOR SAFETY

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held in Paris on 10 and 11 May 2007

TECHNOLOGY RELEVANCE OF THE "UNCERTAINTY ANALYSIS IN MODELLING" PROJECT FOR NUCLEAR REACTOR SAFETY

Background

The OECD/NEA Nuclear Science Committee (NSC) endorsed the setting up of an Expert Group on Uncertainty Analysis in Modelling (UAM) in June 2006. This Expert Group reports to the Working Party on Scientific issues in Reactor Systems (WPRS) and because it addresses multi-scale / multi-physics aspects of uncertainty analysis, it will work in close co-ordination with the benchmark groups on coupled neutronics-thermal-hydraulics and on coupled core-plant problems, and the CSNI Group on Analysis and Management of Accidents (GAMA). The NEA/NSC has endorsed that this activity be undertaken with Prof. K. Ivanov from the Pennsylvania State University (PSU) as the main coordinator and host with the assistance of the Scientific Board. The objective of the proposed work is to define, coordinate, conduct, and report an international benchmark for uncertainty analysis in best-estimate coupled code calculations for design, operation, and safety analysis of LWRs entitled **"OECD UAM LWR Benchmark".**

At the First Benchmark Workshop (UAM-1) held from 10 to 11 May 2007 at the OECD/NEA, one action concerned the forming of a sub-group, led by F. D'Auria, member of CSNI, responsible for defining the objectives, the impact and benefit of the UAM for safety and licensing.

This report is the result of this action by the subgroup.

1. Introduction

1.1 The UAM framework and the scope

The deterministic accident analysis in the domain of Nuclear Reactor Safety (NRS) and the core design optimization, including reloading, in the domain of water cooled and moderated reactors constitute the framework and the scope for the UAM Project. Coupled system thermal-hydraulics / three-dimensional neutron kinetics techniques have been developed since the beginning of the Nineties and became operative towards the end of the same decade.

The calculation steps are the following:
- Provision of nuclear data.
- Fuel assembly burn-up calculations.
- Static and transient 3D nodal core calculations solving neutron diffusion or neutron transport equations with few neutron energy groups including thermal-hydraulic feedback.
- Monte Carlo calculations for steady state conditions.
- Coupled code calculations by thermal-hydraulic system codes and fully integrated 3D core models for accident conditions of the plant.

For safety analysis of NPPs the application of these techniques is essential for a full understanding of core and plant performance. In all steps the calculation methods have to consider the effect of uncertainties. Therefore, these steps also define the structure of the UAM project proposal, Ref. [1], in Fig. 1.

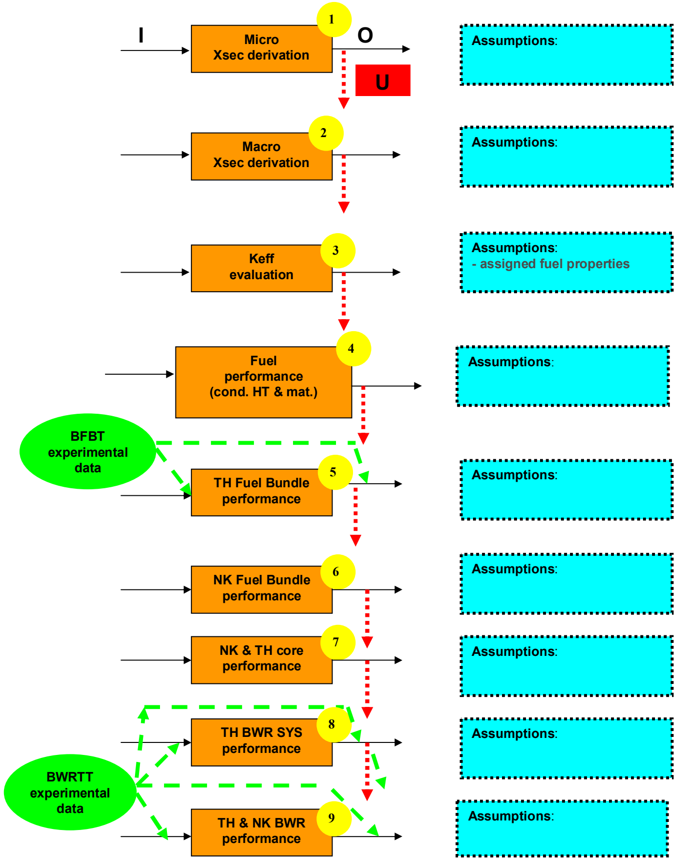


Fig. 1 – The 9 steps of the UAM Project

1.2 The Structure of the UAM Project

The UAM Project can be characterized as follows:

- 9 calculation steps (Fig. 1, ref. [1]),
- 6 uncertainty technology areas (see below),
- 3 phases (current status) each one consisting of 3 exercises, ref. [2],
- 6 years duration (starting in 2007).

Nine calculation steps are identified in the UAM planning document, Fig. 1, that are needed to obtain a qualified calculation from the coupled techniques. At least six different "uncertainty technology areas" can be distinguished and are relevant for the UAM project:

a) generation of neutron cross section, (NXS)
b) neutron-kinetics modelling, (NKM),
c) thermal-hydraulics modelling, (THM),
d) setting up input decks (or meshing) including 'user averaging' and 'user interference' with numerical techniques, (M&AVGN),
e) gathering input data, (GID),
f) coupling (all of the above), (COU).

Namely, each of the nine UAM steps is affected by uncertainties belonging to one or more of the areas listed above. Uncertainty area NXS has to consider approximation of experimental data and integration of nuclear theories. Uncertainty areas NKM, THM and GID have to consider the uncertainty of model parameter and modelling features. Different uncertainty types such as 'epistemic uncertainty' and 'aleatory uncertainty' are considered. Some uncertainty areas like M&AVGN and COU are ignored in most of the current uncertainty studies (or indirectly covered by qualification and engineering judgement).

Related to the uncertainty area of M&AVGN the following should be noted:

- averaging processes and numerical techniques are needed to build-up neutron kinetics and thermal-hydraulics models or computer codes and the related effects on uncertainties are embedded in areas a), b) and c);
- following the averaging process performed by code developers, the code user, when developing an input deck, still needs an additional averaging process; the selection of time steps and the acceptance of over-passing the Courant limitation constitute examples of user interference with numerics.

The uncertainties affect the results of each calculation step, but also have consequences on subsequent steps. The super-position of all effects of the 6 uncertainty areas might lead to unacceptable results of the overall process.

However, a number of 'intermediate' qualification and engineering judgement involvements are part of the process when applied to a technological purpose and aim at decoupling the effects of uncertainty.

1.3 Current Approach for Code Validation

The development of neutron-kinetics and thermal-hydraulics computer codes and their application for safety analysis was accompanied by an extensive programme of code validation.

The steps of nuclear data library generation and fuel assembly burn-up calculations were approved by analysis of critical assemblies, e.g. the NEA criticality handbook and collection of IRPhE, and the evaluation of measured nuclide inventories of assets taken from fuel rods irradiated in operating plants. The number of cases and the range of different fuel types, like UO_2 respectively MOX in PWR or BWR conditions, define the validation status of a code system. As a result specific trends or a relevant bias related to enrichment or moderation could be determined for specific applications.

The validation of thermal-hydraulic codes is based on separate effect test (SET) and integral effect tests (IET), summarized in validation matrices which identify the relevance of experiments to thermal-hydraulic phenomena [3, 4].

The validation of 3D reactor core models is mainly based on nuclear measurements at the plant. The power density distribution can be compared with measured values. In addition, the reactivity effect of control rods or changes of boron concentration can be determined experimentally. Results of calculations for steady state conditions are also part of benchmark comparisons of different code systems.

The validation of transient 3D reactor core models for the analysis of reactivity initiated accidents like a control rod ejection as well as the validation of coupled codes was supported by the OECD/NEA LWR core transient benchmark activities. In this framework typical accident conditions like a PWR control rod ejection, a PWR main steam line break (MSLB) [5] and a BWR turbine trip (TT) [6] have been defined as benchmark problems. The comparison of results from different international institutions and the detailed discussion of differences observed in the working groups have contributed to model improvements and to accepted reference solutions [7]. The variation of results demonstrated the accuracy achievable by such calculations for complex accident conditions. At this level of benchmark analysis, no quantification of the observed differences was achieved. The experience obtained during these studies generated the wish to improve the analysis by applying uncertainty and sensitivity methods.

2. The objectives

The objective of UAM is to consider the uncertainty sources and areas in a systematic way, by assessing the related propagation in each step.

The objective of the present document is to show the technology relevance of UAM, i.e. the expected UAM benefits for NPP design, safety and licensing and to provide guidance for further planning of the UAM activities.

3. The propagation of uncertainty inside the UAM and the uncertainty decoupling

3.1 Uncertainty propagation

The uncertainty transmission from one step to the following ones, without any control or decoupling (see below), apparently causes large and possibly unacceptable uncertainty bands for the end step, i.e. the NPP calculation (step 9 in Fig. 1). In order to substantiate this statement and to give an idea of the values of the errors involved with each step, sample sensitivity studies have been carried out.

Sensitivity studies consist of code runs performed within the domain of the six 'uncertainty technology areas' and deal with a coupled 3D neutron kinetics – system thermal-hydraulics NPP calculation, or aim at producing input data for the NPP calculation, or at demonstrating typical output uncertainty ranges. The sensitivity studies involve the identification of one or more target quantities (i.e. output uncertainty quantity) and the variation of a 'sensitive' parameter (i.e. input uncertainty quantity) in each step.

The sensitivity studies have been carried out according to the OECD/NEA/NSC benchmarks including MSLB [5, 8], BWR TT [6, 7], VVER1000CT [9], etc…and specific examples are established in Tab. 1 and in Appendix 1.

3.2 Overview of Current Status

The first uncertainty methodology presented was the Code Scaling, Applicability, and Uncertainty (CSAU) developed in the USA. The application of the CSAU methodology resulted in the calculation of the PCT during a LBLOCA design basis accident (DBA) event for a Westinghouse 4-loop pressurized water reactor (PWR) with the uncertainty to a 95% confidence level. The PCT was calculated using the TRAC thermal-hydraulic analysis code and was given as a single-valued number with uncertainty bands. The results of this work, first published in 1989, are a defining event for the nuclear safety community. Subsequently a CSAU analysis was performed for a small break loss-of-coolant (SB LOCA) transient on a Babcock & Wilcox PWR using RELAP5.

In the meantime, a number of uncertainty methodologies have been created in other countries, including the GRS method, the UMAE method and the AEA Technology method. These methods, although sharing a common goal with CSAU, use different techniques and procedures to obtain the uncertainties on key calculated quantities. More importantly, these methods have progressed far beyond the capabilities of the early CSAU analysis. Presently, uncertainty bands can be derived (both upper and lower) for any desired quantity throughout the transient of interest, not only for point values like peak cladding temperature.

With regards to the UAM project, uncertainty methods are available and have been applied for cases relevant to the nuclear technology, but they are not covering all the 6 areas identified above and, more important, no one of the available methods can currently deal with all the UAM steps. Detailed descriptions are available within the framework of OECD/CSNI BEMUSE [10] and within IAEA [11].

3.3 Current engineering practice for uncertainty decoupling

When considering the UAM flowchart (Fig. 1), the possibility should be envisaged to prevent the uncertainty propagation from one step to another, i.e. decoupling of uncertainty. An ‘uncertainty decoupling action’ is needed by the analyst and may imply error removal. The related procedures require specific calculations and comparison with actual data to ensure that the errors at a given step of the overall process are less than an assigned threshold. Examples of uncertainty decoupling are:

- Imposing $K_{eff} = 1$ in the steady state calculation of the 3rd step of UAM: imperfections in the process of deriving ‘macroscopic’ cross sections typically cause a K_{eff} value different from unity in steady state conditions. This cannot be propagated without the consequence of calculating a wrong power for the core. Therefore, the analysts ‘introduces’ proper corrections (e.g. insertion of control rod, change in boron concentration or even changes in cross sections values) without necessarily the need to check the consequences for these corrections.
- Assigning an average fuel geometry and material configuration, i.e. neglecting both local fuel conditions and changes in fuel geometry or material composition during a transient like control rod ejection. In this last case the fuel geometry may rapidly change creating a feedback that is ignored in a typical calculation.

The observation here is that the uncertainty decoupling can be (more or less arbitrarily) performed by analysts, possibly without checking of the consequences upon the actual predicted system performance. Therefore the detection and the assessment of the uncertainty decoupling actions constitute a target for the UAM.

No	Sensitivity study characterization	Considered problem	Objective and Notes
1	Effect of the material proprieties: Fuel Conductivity (FC)	NPP LOCA analysis	CATHARE2 simulation
2	Gap size as function of burnup using different options for Fuel Relocation (FR)	VVER-1000 analysis	TRANSURANUS simulation
3	User choice: 3-D Momentum Equation enabled (3D-ME)	Asymmetric transient: spatial variation of coolant proprieties at core inlet.	RELAP5-3D simulation of ROCOM (SETF) experiments
4	Selection of Nuclear Data Files (NDF)	Criticality calculations	MCNP5 code simulation of an RBMK reactor cell
5	Selection of Nuclear Data Files (NDF)	Criticality calculations	Deterministic transport and MCNP calculation for KRITZ and VENUS critical facilities
6	Solution methods for neutron transport equations	Static 2D and 3D calculations	Results for C5G7 benchmark
7	Effect of break mass flow and fuel temperature modelling on overall plant behaviour	MSLB analysis by coupled codes	ATHLET-QUABOX/CUBBOX TRAC CATHARE
8	Selection of Energy Mesh Grid for Averaging Flux and Current Spectra (AF&CS)	Microscopic Cross Section Generation	Microscopic cross section generation by the NJOY code

9	Convergence criteria for inner/outer iterations for neutron diffusion equation solution (CC)	Steady-State stand-alone 3D NK calculations	Setting up of convergence criteria for diffusion-equation based neutronic simulators
10	Detailed core delayed neutron fraction (BETA)	Transient 3D NK calculations	Sensitivities Calculations of a Rod Ejection Accident
11	3D NK –TH mesh overlay (MO)	Steady-State and Transient 3D NK TH coupled calculations	Sensitivities Calculations (PARCS, NESTLE) of a Rod Ejection Accident
12	Cross Section Parameterization (XSEC-PAR)	Steady-State and Transient 3D NK stand alone and TH coupled calculations	Sensitivities Calculations of a Rod Ejection Accident
13	ECCS Performances	NPP LOCA analysis	RELAP5 Calculations Delay of LPIS Accumulator set point
14	Containment	NPP LOCA analysis	RELAP5 Calculations Imposed BC or coupled with PS

Tab. 1 – Input identification for the sensitivity studies (Cont'd).

4. The UAM technology relevance

Within the area of nuclear reactor safety and design, the UAM project deals with the deterministic analysis relevant to core design and accident scenarios. The uncertainty methodologies are applied to all steps of the analysis ranging between the XS generation to the NPP system simulations addressing microscale, macroscale and multiphysics aspects. The technology relevance of UAM can be summarized as follows:

➔ Systematic **identification of uncertainty sources**.

➔ The **systematic consideration of uncertainty and sensitivity methods** in all steps of the realistic or best-estimate analysis calculations generates a **new level of accuracy**. This new approach leads to a **quantification of methods for applications** and to an **improved transparency of complex dependencies**.

➔ All calculation results are represented by **reference results and variances and suitable (i.e. requested by regulatory bodies) tolerance limits** considering the relevant uncertainties.

➔ The **dominant parameters are identified** for all physical processes and their effect on results is determined.

➔ Characterization of **ranges of uncertainty** making reference to the uncertainty technological areas.

➔ Identification of **uncertainty decoupling actions** and check of the related influence. In this case, methods for sensitivity and uncertainty analysis based on a global approach can be used to assess the level of coupling (correlation) between different physical parameters.

➔ Support of the **quantification of safety margins** (i.e. difference between calculated values and licensing safety values or technological failure values).

➔ The **experiences of validation are explicitly and quantitatively documented**. The results of validation are transferred to safety applications.

➔ **Recommendations and guidelines for the application** of the new methodologies are established on the basis of development of the new approach for each step of safety analysis.

Thus, the OECD/NEA/NSC (& CSNI) UAM constitutes an international project needed for the full exploitation of cross section generation, neutron kinetics, fuel performance, thermal-hydraulics codes including coupled codes within the water cooled nuclear reactor safety and design domains.

References

[1] **OECD/CSNI UAM** "Expert Group on Uncertainty Analysis in Modelling - Mandate and Work Programme", NEA/NSC/DOC(2006)17

[2] **OECD/CSNI** "Uncertainty Analysis in Modelling, Summary Record 28-29 April 2006", NEA/NSC/DOC(2006)15, Pisa, Italy

[3] **Aksan S.N., D'Auria F., Glaeser H., Pochard R., Richards C., Sjoberg A., 1993 –** "Separate Effects Test Matrix for Thermal-hydraulic Code Validation: Phenomena Characterization and Selection of Facilities and Tests, Vol. I" – OECD-CSNI Report OCDE/GD(94)82, Paris (F), pages 1-680

[4] **Annunziato A., Glaeser H., Lillington J.N., Marsili P., Renault C., Sjoberg A., 1996 –** "CSNI Integral Test Facility Validation Matrix for the Assessment of Thermo-Hydraulic Codes for LWR LOCA and Transients" – NEA/OECD Report OCDE/GD(97) 12, Paris (F)

[5] **Ivanov K., Beam T., Baratta A., Irani A., Trikouros N.**, "PWR MSLB Benchmark volume 1: Final Specifications", NEA/NSC/DOC(99)8, Paris (F), April 1999.

[6] **Solis J., Ivanov K., Sarikaya B., Olson A., Hunt K.W.**, "Boiling Water Reactor Turbine Trip (TT) Benchmark - Volume I, Benchmark Specification", NEA/NSC/DOC(2001)1, ISBN 92-64-18470-8

[7] **Ivanov K., Sartori E., Royer E., Langenbuch S., Velkov** K. – "Validation of Coupled Thermal-Hydraulic and Neutronics Codes for Safety Analysis by International Cooperations, Nuclear Technology", Vol. 157, No.2, p177, February 2007.

[8] **Bousbia Salah A., Vedovi, J., D'Auria F., Kostadin Ivanov K., and Galassi G. M.**, "Analysis of the Peach Bottom Turbine Trip 2 Experiment by Coupled RELAP5-PARCS 3D Codes", Nuclear Science and Engineering, Vol. 148, p.337-353, 2004

[9] **Ivanov, B.; Ivanov, K.**; "VVER-1000 Coolant Transient Benchmark: Phase-1 (V1000CT-1), Volume I: Final Specifications", OECD Paris, NEA/NSC/DOC, 2002

[10] **OECD/NEA BEMUSE** - "Best-Estimate Methods (Including Uncertainty Methods and Evaluation) Qualification and Application. First Meeting of the Programme Committee", NEA/SEN/SIN/AMA(2003)8, Issy-les-Moulineaux, France, February 12-13, 2003.

[11] **IAEA, 2007** – "Uncertainty Evaluation in Best Estimate Safety Analysis for Nuclear Power Plants" – IAEA Safety Series Report, to be issued, Vienna (A).

APPENDIX 1

QUANTIFICATION FOR UNCERTAINTY RELEVANCE

With the contribution by A. Del Nevo, C. Parisi and A. Petruzzi

Key results and information from the sample cases reported in Tab. I are provided in the table below. This includes eight main columns:

- (1): Case identification from Tab. I;
- (2): Step of the UAM process (from Fig. 1);
- (3): Identification of considered input uncertain parameter and range of variation;
- (4): Identification and quantification of selected uncertain parameters that propagate to the next step of UAM;
- (5): Identification and quantification of selected uncertain parameters that propagate to the final step of UAM;
- (6): Concerned technology area (section 1.2);
- (7): Planned exercise of UAM (Ref [2]);
- (8): Clarification notes and reference documents providing information about the calculation case discussed in the concerned row.

(1) No	**(2) Step (Fig. 1)**	**(3) Input Uncertain Parameter**		**(4) Uncertainty propagation in the next step**		**(5) Uncertainty propagation in the NPP calculation**		**(6) Uncertainty technology area**	**(7) Exercise of UAM (Ref. 2)**	**(8) Notes/Additional References**
		ID	**Variation (%)**	**ID**	**Variation (%)**	**ID**	**Variation (%)**			
1	4 5 7	FC	10%	--	--	PCT	~ 10%	THM	II-1 II-3 III-2 III-3	Connected with the step 4 (fuel performance) [A1]
2	4	FR	Diff. option	Gap dimension	~ - 40%	PCT	~ 8% *	THM	II-1	* PCT evaluation during LB-LOCA analysis with CATHARE2 code: sensitivity on the gap dimensions [A1, A2]
3	7	3D-ME	on	Distribution of the coolant proprieties at core inlet.	~ 100%	--	--	THM	III-3	[A3]
4	3	NDF	~ 5%	Power and flux distribution	~ 10%	PCT	~ 10%	GID	I-3	
5	1 2 3	NDF	diff. libraries	k_eff	0.2 – 0.6%			NXS	I-1 I-2 I-3	Bias of nuclear data libraries [A4, A5, A10, A11, A12]
6	2 3	NDF	Different solution methods	k_eff Power density	0.3-0.8 % 10 %			NXM	I-2 I-3	[A6, A10, A11, A12]
7	6 7		Different models	Break flow	~50 %	Power at recriticality time	250% 25 s	THM GID	III-1 III-3	[A10, A11, A12]
8	1	AF&CS	~ 10-50%	Averaged Cross Sections values	?	Reaction rates, k_eff	?	M&AVGN	I-1	[A7]
9	3 6 7	CC	~ 100% - 1000%	k_eff, power and flux distribution	~ 10%	PCT	~ 5%	M&AVGN	I-1, II-2, III-1, III-3	* Variation of convergence criteria can be order of magnitude, [A6]

10	6 7 9	BETA	10%	Power trend and magnitude	~ 10%	PCT, Energy release to the fuel	~ 10%	NKM	II-2, III-1, III-3	[A7]
11	7 9	MO	-	Power trend and distribution	~ 10-50%	PCT, Energy release to the fuel	~ 10-50%	COU	III-1, III-3	[A7]
12	7 9	XSEC-PAR	-	Power trend and distribution	~ 10-50%	PCT, Energy release to the fuel	~ 10-50%	M&AVGN	III-1, III-3	[A7]
13	8	ACC	1%	Time of core quenching	~ 10%	[a]	[a]	THM	III-2	Accumulator Pressure Set Point. [a] LOFT [A8] is assumed as NPP
14	8	CONT	Diff. option	PCT	~ 10%	[a]	[a]	THM	III-2	[a] ZION [A9] NPP

Tab. 2 – Input and output characterization for the performed sensitivity studies to show the technology relevance of the UAM Project

Two of the examples listed in the table above are discussed hereafter in more detail.

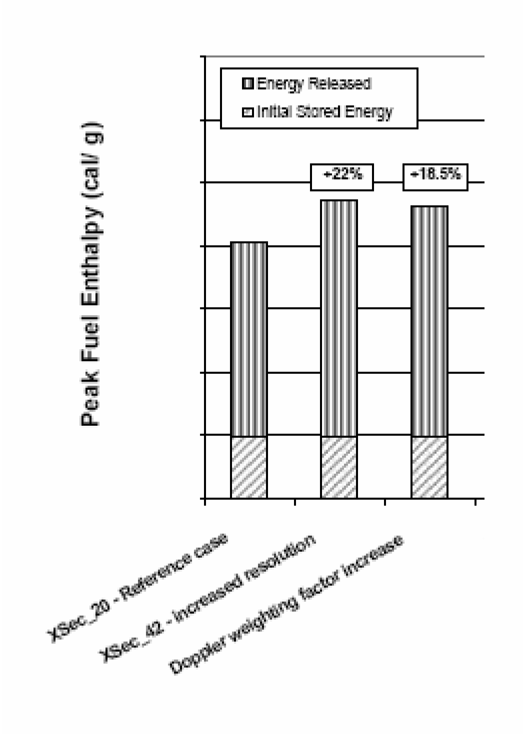
No. 2 in table above
With regards to the Uncertainty Technology Area THM (including the areas related with the Fuel Performance), if a LB-LOCA transient is considered the PCT decreases, increasing fuel thermal conductivity and gap conductance [A2]. It implies that the uncertainty in predicting the maximum cladding temperature is also connected with the accuracy of the material proprieties data provided. Nevertheless the gap conductance is also affected by the uncertainty because it is dependent on the thermal conductivity and the geometry. The geometry is connected to the material proprieties of the fuel and the cladding (e.g. thermal expansion) that change with the fuel burnup. For this reason, the unavoidable inaccuracies (due to user effect as well as epistemic leakage) [A3], of the results carried out using fuel performance code simulations (fuel pin mechanics, gas release and burn-up models), propagate the uncertainty in the estimation of the PCT performed using TH-SYS codes.

No. 11 in table above
The parameterization of few-groups homogenized Cross Sections can greatly influence the calculations of reactor core dynamics by the 3D neutron kinetics coupled thermal-hydraulics codes.

Generally, for a given core status (e.g., burnup, fuel, coolant and moderator temperatures distributions assigned), a set of neutron cross sections is produced by lattice codes calculations. Then this set of data (master library) is parameterized according to the same physical parameters such as fuel temperature, coolant density, etc.

The analyst should choose the most relevant feedback parameters and estimate the range of values that these parameters could have during the simulation of the selected reactor status and/or transient. The analyst has also to choose the right number of parameter points that can allow the best reproduction of the neutron cross section behaviour between the extremes of the selected range.
This has to be done balancing the increased accuracy in the trend simulation of a neutron cross section with the computational costs associated with an increased resolution.

An example of the effect of neutron cross section parameterization on the energy release to the fuel during a Rod Ejection Accident is shown in [A7]. Here a reference case was calculated using a cross section library with 20 points. A sensitivity calculation was performed increasing this value to 42, resulting in a variation of the energy release to the fuel of about +22%.



References

[A1] Del Nevo, G. M. Galassi, F. D'Auria, "Project BEMUSE:LB-LOCA Analysis and Sensitivity Evaluation of ZION NPP by CATHARE2/V2.5_1 Code", University of Pisa, DIMNP NT615(07) Rev.0, 2007, Pisa.

[A2] D. Elenkov, S. Boneva, M. Georgieva, S. Georgiev, "Research and Development for Licensing Fuel in Bulgaria", TRANSURANUS Users Meeting, 25-26 June 2007, ITU - Karlsruhe, Germany.

[A3] A. Frisani, A. Del Nevo, F. Moretti, F. D'Auria, T. Höhne, S. Kliem, U. Rohde, "RELAP5-3D©/V2.3.6 Assessment Against Boron Mixing Experiments Performed at ROCOM Test Facility", University of Pisa, DIMNP NT 589 (06) Rev.0, 2007, Pisa.

[A4] C. Parisi, F. D'Auria "MCNP5 Monte-Carlo Models and Calculations for an RBMK cell", University of Pisa, DIMNP NT 625 (07) Rev.0, 2007, Pisa.

[A5] "WIMS-D Library Update – Final report of a coordinated research project" IAEA Technical Report, Wien, 2006

[A6] C. Parisi, F. D'Auria "AER-FCM-101 VVER1000 Neutronics Benchmark – NESTLE and PARCS neutronic solutions", DIMNP NT 626 (07) Rev.0, 2007, Pisa.

[A7] C. Parisi, K. N. Ivanov, F. D'Auria, G.M. Galassi, "Study of the WWER Core Behaviour during a Rod Ejection Accident by Coupled Three Dimensional Neutron Kinetics / Thermal-Hydraulic Codes", MS Thesis, University of Pisa, March 2003.

[A8] A. Petruzzi, F. D'Auria (EDITORS), et al., "BEMUSE Programme. Phase 2 report <Re-Analysis of the ISP-13 Exercise, post test analysis of the LOFT L2-5 experiment" – OECD/CSNI Report NEA/CSNI/R(2006)2 – Issued June 19, 2006, JT03210882, Paris (F), © OECD 2006, pages 1-625 (2006).

[A9] M. Perez, F. Reventos, Ll. Batet, "Phase 4 of BEMUSE Programme: Simulation of a Large Break Loss Of Coolant Accident in ZION NPP", OECD/CSNI Report, Barcelona, May 2006

[A10] A. Seubert, S. Langenbuch, W. Zwermann, "Status and Development of Nuclear Design and Accident Simulation Methods", EUROSAFE 2005, Brussels, Nov 7-8, 2005

[A11] W. Zwermann, S. Langenbuch, "Monte Carlo Calculations with Nuclear Data Based on JEF-2.2 and JEFF-3.1 for the VENUS-7 Critical Benchmarks, Int. Conf. on Nuclear Data for Science and Technology, Nice, April 22-27, 2007

[A12] OECD/NEA-Report, "Benchmark on Deterministic Transport Calculations without Spatial Homogenisation, MOX Fuel Assembly 3D Extension Case", NEA/NSC/DOC(2005)16