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Current use of Best Estimate plus Uncertainty methods on operational procedures addressing normal and emergency conditions

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Summary Report

Current use of Best Estimate plus Uncertainty methods on operational procedures addressing normal and emergency conditions

EXECUTIVE SUMMARY

This Report summarizes the results of the studies performed by the Joint Research Centre / Institute for Energy (JRC/IE) on a specific task dedicated to Thermal-hydraulics within the SONIS (Safety Of Nuclear InstallationS) 2008 program.

The aim of task 4 of the SONIS programme is to analyse European practice in verification and optimization of plant operational procedures for normal, abnormal and emergency conditions. More specifically task 4.1 analyses the effect of using new Best Estimate plus Uncertainty Methods (BEPU) in the re-licensing processes on plant operational procedures directly affecting the Thermal-Hydraulic (T-H) behaviour of the nuclear facilities.

Current trends in the industry to increase power production challenge the initial safety design limits of the plant which were performed generally using conservative tools and hypothesis. Advance numerical tools and methods allow demonstrating that safety margins are still respected. These tools are modern fully validated thermal-hydraulic codes, coupled thermal-hydraulic / neutron-kinetic (N-K) codes and methodologies that use realistic hypotheses rather than conservative ones and estimate also the uncertainty

Their effect on operational procedures for normal and emergency conditions and for Operating Limits and Conditions is investigated by asking directly the stakeholders of the European Union.

A questionnaire was sent to several stakeholders in the Nuclear Safety domain in the European Union and information was gathered on the new T-H tools for the re-licensing processes (for power uprates, SG replacements etc), their effect on the Operational Limits and Conditions (OLC) and on Operating Instructions and Procedures (OIP) of the Nuclear Power Plants (NPPs), the need of performing specific investigations in operational modes and of exchanging of information on new T-H tools/methodologies.

It was seen that almost all the interviewed countries confirmed the uprate of their Nuclear Power Plants (NPPs) with a maximum value of 10%. The major modifications were generally associated with the replacement of Steam Generators (SG) and introduction of new type of fuel.

All the participants also confirmed the impact on the original OLC (for example, changes were introduced in setpoints of reactor protection) but there no major effects on the original OIP were noticed (except some optimization of procedures).

In general, no additional operating or accidental modes were identified for development or improvement although two participants mentioned the current development of SAMG (Severe Accident Management Guidelines)

The full set of T-H safety analyses were done only at those NPPs where major modifications were performed and detailed simulations were executed manly for evaluation of set-points and modifications of accident management procedures.

The participants agreed that an exchange of experience by a forum/workshop with a qualitative comparison on of OLC and OIP of plants of different countries would be beneficial. They also agreed on different kinds of research activities mainly focussed on use of three dimensional and coupled T-H and N-K codes.

Finally, a large number of organizations participate directly in current research and international projects but the findings aren’t systematically applied to their plants.

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1 Introduction

1.1 Background

The European Union's Framework Programme 7 action SONIS (operated by the European Commission Joint Research Centre, Institute of Energy (JRC-IE) in Petten, the Netherlands.) aims at facing technical and organisational issues related to the safe operation of the existing European nuclear facilities in an integrated research approach, providing ready-to-use, validated methods, models and recommendations for procedures.

It's main goal are to support long-term EU policy needs on operational nuclear safety and security of the existing installations, and optimization of the advanced nuclear energy systems, through exploitation of the JRC competence in research in nuclear safety assessment methods and techniques and in developing the European Research Area by integrating the research efforts with the on-going efforts implemented by the nuclear utilities and plant designers, through development of suitable networks and collaborating with other EC bodies and International Organizations.

Its main topics are maintenance, surveillance and in-service inspection, engineering safety in the fire and seismic domain, thermal-hydraulics, and Human and Organisational Factors in Event Analysis and Root Cause Analysis.

The aim of task 4 of the SONIS programme is to analyse European practice in verification and optimization of plant operational procedures for normal, abnormal and emergency conditions.

More specifically task 4.1 analyses the effect of using new Best Estimate plus Uncertainty Methods (BEPU) in the re-licensing processes on plant operational procedures directly affecting the Thermal-Hydraulic (T-H) behaviour of the nuclear facilities.

This topic has been introduced in the 2008 working programme but will be thoroughly investigated through the duration of the SONIS project.

1.2 Objectives and scope of the report

This report aims at providing the results of the studies performed by the JRC-IE on a specific task dedicated to Thermal-Hydraulics within the SONIS 2008 program.

This report intends to provide view on the current practice of the European stakeholders on the use of BEPU tools in re-licensing situations and their effect on OIP and OLCs if applicable.

This topic is of interest to all the Regulatory Bodies of the European Union as almost each one is faced with re-licensing requests for NPP due to power uprate by the existing plants

Saftey is today demonstrated by new complex tools and methodologies that prove the presence of safety margins.

In order to determine the current situation, a questionnaire was sent to several stakeholders in the Nuclear Safety domain in the European Union and information was gathered on the new T-H tools for the re-licensing processes (for power uprates, SG replacements etc), their effect on the Operational Limits and Conditions (OLC) and on Operating Instructions and Procedures (OIP) of the Nuclear Power Plants (NPPs), the need of performing specific investigations in operational modes and of exchanging of information on new T-H tools/methodologies.

1.3 Structure of the report

Part I describes the impact of Modern T-H and N-K tools on OLC and OIP during major plant modifications at European Nuclear Plants.

The Regulation Approaches in the Nuclear Industry and Plant Modifications, Safety Margins and Licensing Process are introduced in chapter 2.

Chapter 3 describes the general features and requirements in using Advanced numerical Tools and Methods while chapter 4 addresses their impact on OLC and OIP

Part II describes the Feedback of International Stakeholders addressing OLC and OIP modifications at European Plants with a summary of the results.

Part I – Impact of Modern T-H and N-K tools on OLC and OIP during major plant modifications at European Nuclear Plants

2 Introduction

For a Nuclear Power Plant (NPP) to be operated in a safe manner the provisions made in the final design and subsequent modifications shall be reflected in limitations on plant operating parameters and in the requirements on plant equipment and personnel. Under the responsibility of the operating organization, these shall be developed during the design safety evaluation as a set of Operational Limits and Conditions (OLCs). A major contribution to compliance with the OLCs is made by the development and utilization of Operating Instructions and Procedures (OIPs) that are consistent with and fully implement the OLCs.

The IAEA's Safety Requirements for Operation [1] state that OLCs shall be developed to ensure that plants are operated in accordance with design assumptions and intent. In order to achieve this requirement the plant safety analysis report should be developed in such a manner as to identify clearly the OLCs that must be met to prevent situations from arising which might lead to accident conditions or to mitigate the consequences of accidents if they do occur.

The aim of this chapter is to illustrate the modern computational tools and approaches both at improving the plant operation and control of nuclear power plants and at supporting the design modifications on the existing nuclear power plant system. In relation to the last topic, dynamic analysis is a fundamental tool for evaluating the impact that modifications in components or systems have on the operation of the plant. Among these studies, those related to set-point adjustment, as well as those originated by important technological changes, are the most significant. Projects such as the replacement of steam generators, or power upgrading have required prior developments of the T-H models for to give a quite complete image of predicted plant behaviour already during the design phase.

2.1 Regulation Approaches in the Nuclear Industry

The concepts of Design Basis Accident (DBA) and the rules and the criteria for the related analysis constitute the fundaments of nuclear reactor safety that were fixed in the seventies (e.g. US NRC 10 CFR50 and Appendix K, [2]) when the large majority of NPP today in operation were designed. Owing to several weaknesses in the knowledge and understanding, conservatisms were introduced at each level of the safety analysis (e.g. acceptance criteria, conservative assumptions in models, conservative input conditions, etc.).

The availability of only conservative models did not allow the calculation of the actual 'distance' between a plant status and the acceptability criteria, even in the case of an accident. Thus, nor the 'safety margins' could be established in a quantitative manner, neither the optimization of a safety solution could be demonstrated.

This brought to huge research programs in thermal-hydraulics that went on since the seventies and basically were completed in the nineties. The knowledge and the understanding acquired in this period is at the basis of the ‘modern’ safety culture, independent upon the NPP design, and if the case, upon the number of operational years.

Therefore, a modern safety analysis, though accounting for the historical acceptance criteria and established rules, namely of the Regulatory Body in the Country, should also make reference to the fundamentals of the safety technology:

- The physical barriers to the release of the fission products;

- The safety functions realized by protective systems or features intended to preserve the integrity of barriers or to mitigate the effect of barrier failures.

This implies the consideration of:

a) the most advanced computational tools and techniques,

b) the interdisciplinary approach for accident analysis including the connection among different disciplines needed to address complex problems,

c) the recent guidelines issued by US Nuclear Regulatory Commission (NRC) and International Institutions like IAEA and OECD/CSNI and the European Utility Requirements (EUR).

A qualitative understanding of thermal-hydraulic phenomena in the seventies, including a qualitative understanding of the meaning of frequency and probability (i.e. with negligible operational feed-back at the time), was at the origin of the key associated concepts: DBA and Conservatism (code, boundary conditions, Acceptance Criteria, Appendix K approach).

The DBA was intended as a minimum set of enveloping scenarios whose positive-conservative evaluation, within the (overly) conservative Appendix K approach, could ensure that an adequate level of protection is provided by the designers. TMI-2 and also Chernobyl-4 were practical demonstrations that complex accidents out of the DBA list may occur. The needs from operator training and, above all, the progress in the techniques for deterministic and probabilistic accident analysis, i.e. an outcome of the research programs carried out during three-four decades, suggested a change in the conservative approach.

A recent issued OECD/CSNI report, ref. [3], identifies four classes of deterministic methods that can be seen as a historical progress for the licensing approach:

1. Very Conservative (Appendix K for LOCA);

2. Best Estimate Bounding;

3. Realistic Conservative;

4. Use of Best-Estimate Plus Uncertainty (BEPU).

A similar classification was proposed earlier (2003) by IAEA, e.g. ref. [4], in a well known table where “Type of Applied Code”, “Type of BIC (Boundary and Initial Conditions”, “Assumption on System Availability” and “Type of Approach”, are distinguished.

Without entering into detail of the four classes of methods nor of the IAEA table, two remarks apply:

- Drawbacks from the Applicant and from the Licensing Authority side are identified when the approaches 1) to 3) or the 'conservative' approaches of the IAEA table are pursued;

- BEPU constitutes the current trend (as also testified by ongoing projects like BEMUSE, IAEA CRP on uncertainty, or recently issued documents like US NRC RG 1.203 and IAEA Safety Series Report on uncertainty methods [5]).

The development of BEPU methods had, as specific reference framework, the deterministic accident analysis and the acceptance criteria valid within the conservative approach, i.e. items 1) to 3) above. The BEPU approach in ref. [5] is considered as "*… the biggest effort for a proper use of best estimate models in order to minimize unnecessary conservatism while accounting for uncertainties associated to simulation results.*".

The first framework for calculating the uncertainty was proposed by U.S. NRC and denominated Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology [6]. The first application of the CSAU methodology resulted in the calculation of the Peak Cladding Temperature (PCT) during a Large Break LOCA (LBLOCA) design basis accident event for a Westinghouse 4-loop pressurized water reactor with the uncertainty to a 95% confidence level. The peak temperature was calculated using the TRAC thermal-hydraulic system code and was given as a single-valued number with uncertainty bands.

In the meantime, a number of uncertainty methodologies were proposed in other countries. These methods, although sharing a common goal with CSAU, use different techniques and procedures to obtain the uncertainties on key calculated quantities. More important, 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 point values like peak cladding temperature.

The current challenge is the 'distributed' and 'comprehensive' application of the BEPU methods within the FSAR (Final Safety Analysis Report), making reference to the physical barriers to the release of the fission products and to the safety functions (i.e. the two dashed items above), other than considering the existing acceptability threshold (i.e. the heredity from the conservative approach). In this manner a homogeneous level of safety can be ensured for all the aspects that are part of the FSAR. Furthermore, nowadays the application of the BEPU methods must be consistent with the best-latest available information in various technological sectors (e.g. Computational Fluid Dynamics and 3D Neutron Kinetics).

2.2 Plant Modifications, Safety Margins and Licensing Process

Any changes to the design or operation of a nuclear power station must be the subject of a controlled process. However, in the context of regulatory involvement, only those modifications which have the potential to affect nuclear safety are normally within the responsibility of the nuclear regulator. To appropriately prioritise effort within the applicant and regulator organisations, it is necessary to categorise modifications dependent upon their safety significance.

In this context, nuclear safety modification means any alteration to structures, plants, components, operations, processes, safety analysis report, operating and emergency procedures, operating limits and conditions, technical specifications, both permanent or temporary, and includes any replacement, refurbishment or repairs to existing buildings, plants or processes that could influence nuclear safety. The following, for example, are excluded:

- Replacement of a component with one exactly the same.
- Normal maintenance
- Operational changes within the existing safety justification, without changing the operating procedures or other safety documentation.

Traditionally, the concept of "Safety Margins" has been introduced in recognition of the fact that uncertainties exist in the safety variable values at which damage occurs. It applies explicitly to either barrier or system losses. Therefore, in a nuclear power plant as many "Safety Margins" should exist as there are physical parameters that may challenge barriers or systems, the loss of which could cause a safety problem. Furthermore, for each barrier or system, "Safety Margins" exist for each damage mechanism that can lead to the loss of a barrier or a system.

Whether or not the loss of a particular system or barrier is a safety problem depends on the expected consequences. Since the ultimate goal of nuclear safety is to prevent unacceptable radiological releases to the public or to the environment, safety limits and margins should be considered at least for those systems and barriers, where a failure could potentially generate unacceptable radiological releases.

In order to understand whether changes in the design and / or operating conditions of an NPP could threaten the overall safety margin or raise risk to unacceptably level, modern computational tools and methods are strongly recommended to be used whenever possible.

3 Advanced numerical Tools and Methods

Nuclear power technology has been developed based on the traditional defence in depth philosophy for the design of the plant that was supported by deterministic and overly conservative methods for safety analysis.

In the 1970s conservative hypotheses were introduced for safety analyses to address existing uncertainties. Since then, intensive thermal-hydraulic experimental research has resulted in a considerable increase in knowledge, and the development of computer codes has improved their ability to calculate results that agree with experimental evidences.

The use of a conservative methodology may be so conservative that important safety issues may be masked. For example, the assumption of high core power may lead to high mixture level in the core in the case of Small Break Loss Of Coolant Accident (SBLOCA). Consequently, the calculated peak clad temperature may not be conservative as expected. Therefore, it may be preferable to use a more realistic approach together with an evaluation of the related uncertainties to compare with acceptance criteria. This type of analysis is referred to as a Best Estimate Plus Uncertainty (BEPU) approach and can provide more realistic information about the physical behaviour, identifying the most relevant safety issues and supplying information about the actual existing margins between the results of calculations and acceptance criteria.

In addition to the establishment of best-estimate calculations for design and safety analysis, understanding uncertainties is important for introducing appropriate design margins and deciding where additional efforts should be undertaken to reduce uncertainties. For this reason the sensitivity and uncertainty analyses are fundamental tools for providing quantitatively in a mathematically and physically well-founded way answers to typical scientific and engineering questions such as how much the model under consideration represents the physical phenomena, how far the calculated results can be extrapolated and etc…

3.1 Historical Background

Evaluation of Nuclear Power Plants (NPP) performances during accident conditions has been the main issue of the research in nuclear fields during the last 40 years. Therefore, several complex system thermalhydraulic codes have been developed for simulating the transient behavior of water-cooled reactors. In the early stage of the development, the codes were primarily applied for the design of the engineered safety systems. In 1978 the “Appendix K Requirements” [2] were issued, defining conservative model assumptions as well as conservative initial and boundary conditions to warrant conservative code results for critical safety parameters. On the other hand, the development and elaboration of accident management procedures, the application of Probabilistic Safety Analyses (PSA) and the operator training asked for so called “Best Estimate analysis” (BE), that means an accident simulation as realistic as possible. The main objective of best-estimate system codes was to replace the “Evaluation Models”, which used many conservative assumptions, by the best-estimate approach for more realistic predictions of Pressurized Water reactor (PWR) or Boiling Water Reactor (BWR) accidental transients that allow the reduction of safety margins. Best-estimate system codes are currently used for:

- Safety analysis of accident scenarios,

- Quantification of the conservative analyses margin,
- Licensing purposes if the code is used together with a methodology to evaluate uncertainties,
- Probabilistic Safety Analysis (PSA),
- Development and verification of accident management procedures,
- Reactors design,
- Analysis of operational events,
- Core management investigation.

Best-Estimate thermal hydraulic codes (e.g. RELAP, TRAC, CATHARE, ATHLET...) are, in general, based on equations for two-phase flow which are typically resolved in Eulerian coordinates. The two-phase flow field is described by mass, momentum and energy conservation equations for the liquid and vapour phases separately and mass conservation equations for non-condensable gas present in the mixture. The models are suitable for 1D system simulation even if for some NPP component (e.g. the vessel) some code has the capability to solve 3D system equations. Time discretization could be fully, semi or nearly implicit. Depending on the number of balance equations, different sets of constitutive equations are required to close the equation system. In comparison with the Homogeneous Equilibrium Model (HEM), which requires only two constitutive equations, namely the friction loss and the heat transfer relations at the wall, at least seven constitutive equations are required for the two fluid models with six balance equations describing the mass, energy and momentum transfer at the interface and the energy and momentum transfer of the water- and steam-phase at the wall. The constitutive equations have to describe the physical phenomena in a wide span of scale, ranging from down-scaled integral system experiments up to full size reactor geometry. This is one of the most challenging goals in code development and code validation. To develop and validate the scaling laws for individual phenomena, separate effects tests in different scale are necessary. In Fig. 1, the code development activities carried out in more than three decades are shown.

Due to the numerical approximations and the empirical nature of the included models in the thermal-hydraulic system codes, extensive activities related to validation of the codes have been pursued during the years. The validation has been performed using experimental data from specially designed scaled down test facilities. In addition, transient data from real NPPs were also considered due to the full scale and true geometry although those data concern only conditions under fairly mild transients (operational transients and start-up and commissioning tests). These activities have been planned and carried out in national and international contexts in four levels, mainly in the independent assessment area, involving the use of:

1. "Fundamental" experiments [7];
2. Separate Effects Test Facilities (SETF) [8];

3. Integral Test Facilities (ITF), including most of the International Standard Problems (ISP) [9];

4. Real Plant data

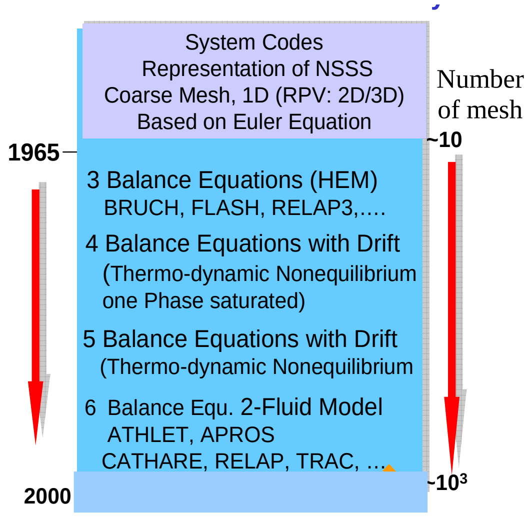


Figure 1: Code Development Activities in more than three decades.

However, notwithstanding the huge amounts of financial and human resources invested, the results predicted by the code are still affected by errors whose origins can be attributed to several reasons as model deficiencies, approximations in the numerical solution, nodalization effects, imperfect knowledge of boundary and initial conditions. In this context, the existence of qualified procedures for a consistent application of qualified thermal-hydraulic system code reveals necessary and implies the drawing up of specific criteria through which the code-user, the nodalization and finally the transient results are qualified.

The current situation related to the development, validation and use of system codes can be summarized as follows:

- A State of the Art Report in modeling LOCA (Loss of Coolant Accident) and non-LOCA transients and the Compendium on ECCS (Emergency Core Cooling Systems) Researches have been published in 1989, refs. [10, 11], by OECD/CSNI (Organization for Cooperation and Development / Committee on the Safety of Nuclear Installations) and US NRC. These reports broadly cover topics like plant features relevant to thermal-hydraulics, transients' description, phenomena identification, and code modeling capabilities, needs for data and present situation in the experimental area.

- The CSAU (Code Scaling Applicability and Uncertainty), published in 1990, e.g. ref. [6], constituted a pioneering effort made by NRC in the area of code uncertainty prediction.

- Code validation criteria and detailed qualification programs exist, although not fully optimized or internationally agreed. In particular:

- The Integral Test Facility CSNI Code Validation Matrix (ITF-CCVM) report, was initially published in 1987 and extensively updated in 1996, [9]. Tests for code validation were selected based on quality of the data, variety of scaling and geometry and appropriateness of the range of covered conditions. The decision was taken around 1984 to bias the validation matrix toward integral tests in order that code models were exercised and interacted in situations as similar as possible to those of interest to PWR and BWR. This was done because of the assumption that sufficient comparison with separate effects test data would be performed and documented by code developers.

- As the last expectation has proved unrealistic, a group of scientists was formed toward the end of '80s, to set-up the Separate Effect Test Facility CSNI Code Validation Matrix, SETF-CCVM, that was issued in 1994, ref. [8]. The development of the SETF-CCVM required an extension of the methodology employed for the ITF-CCVM [9], both in the scope and the definition of the thermal hydraulic phenomena and in the categorization and description of facilities. A significant result of the activity was the selection of sixty-seven phenomena assumed to cover all the thermal hydraulic situations of interest expected in PWR and BWR transients. The needed effort suitable for a comprehensive code validation was quantified: more than one thousand experiments should be part of a thermal hydraulic system code validation program. The impact of those findings in planning new researches was also evaluated, ref. [12].

- The codes have reached an acceptable degree of maturity although the reliable application is still limited to the validation domain;

- The use of qualified codes is more and more requested for assessing the safety of existing reactors, especially in the former Soviet Union and in the Eastern Countries, and for designing advanced reactors;

- The codes availability is increasingly growing especially in the countries belonging to the former Soviet Union, the Eastern Countries, Korea, China, etc.;

- Special topics like user [13] and computer-compiler effects upon code calculation results, nodalization qualification [14], accuracy quantification [15], relevance of International Standard Problems and lesson learned, use of best estimate codes in the licensing, have been widely discussed and main achievements are available to the international community;

- A special attention from the scientific community has always been given to the quantification of code uncertainty in predicting plant transients. Methodologies to evaluate the ‘uncertainty' have been proposed [16, 17] and tested in several international activities, like UMS (Uncertainty Method Study, [18]) and BEMUSE (Best-Estimate Methods – Uncertainty and Sensitivity Evaluation, [19, 20]) that allowed the comparison of uncertainty results obtained from different methodologies.

The following sections review the main features and limitations of the thermal-hydraulic system codes and the procedures adopted for the qualification of computational tools, i.e. not only the codes, through the ITF and SETF validation matrixes, but also the nodalization used to simulate the transient scenario in the NPP. Finally, taking into account the multi-disciplinary nature of reactor transients and accidents (which include thermal-hydraulics, neutronics, structural and radiological aspects), the needs, the status of development and the benefits of code coupling are pointed out.

3.2 Thermal-Hydraulics System Codes

The system thermal-hydraulic codes are based upon the solution of six balance equations for liquid and steam that are supplemented by a suitable set of constitutive equations. The balance equations are coupled with conduction heat transfer equations and with neutron kinetics equations (typically point kinetics). The two-phase flow field is organized in a number of lumped volumes connected with junctions. Thermal-hydraulic components such as valves, pumps, separators, annulus, accumulators, etc. can be defined in order to represent the overall system configuration. In the following sections, main problematic aspects - from the point of view of the user - of a thermal-hydraulic system code are highlighted.

System Nodalization

All major existing Light Water Reactor (LWR) safety thermal-hydraulics system codes follow the concept of a “free nodalization”, i.e. the code user has to build-up a detailed noding diagram which maps the whole system to be calculated into the frame of a one-dimensional thermal-hydraulic network. To do this, the codes offer a number of basic elements like single volumes, pipes, branches, junctions, heat structures, etc. This approach provides not only a large flexibility with respect to different reactor designs, but also allows predicting separate effect and integral test facilities which might deviate considerably from the full-size reactor.

As a consequence of this rather “open strategy”, a large responsibility is passed to the user of the code in order to develop an adequate nodalization scheme which makes best use of the various modules and the prediction capabilities of the specific code. Due to the existing code limitations and to economic constraints, the development of such a nodalization represents always a compromise between the desired degree of resolution and an acceptable computational effort. It is not possible here to cover all the aspects of the development of an adequate nodalization diagram, however, two crucial problems will be briefly mentioned which illustrate the basic problem.

Spatial convergence

As has been quite often misunderstood, a continuous refinement of the spatial resolution (e.g. a reduction of the cell sizes) does not automatically improve the accuracy of the prediction. There are two major reasons for this behavior:

(1) The large number of empirical constitutive relations used in the codes has been developed on the basis of a fixed (in general coarse) nodalization;

(2) The numerical schemes used in the codes generally include a sufficient amount of artificial viscosity which is needed in order to provide stable numerical results. A reduction of the cell sizes below a certain threshold value might result in severe non-physical instabilities.

From those considerations, it can be concluded that no a priori optimal approach for the nodalization scheme exists.

Mapping of multi-dimensional effects

Multi-dimensional effects, especially with respect to flow splitting and flow merging processes (e.g. the connection of the main coolant pipe to the pressure vessel), exist also in relatively small scale integral test facilities. The problem might become even more complicated due to the presence of additional bypass flows and a large re-distribution of flow during the transient. It is left to the code user to determine how to map these flow conditions within the frame of a one-dimensional code, using the existing elements like branch components, multiple junction connections or cross-flow junctions. These two examples show how the limitations in the physical modeling and the numerical method in the codes have to be compensated by an "engineering judgment" of the code user which, at best, is based on results of detailed sensitivity of assessment studies. However, in many cases, due to lack of time or lack of appropriate experimental data, the user is forced to make ad-hoc decisions.

Code Options: Physical Model Parameters

Even though the number of user options has been largely reduced in the advanced codes, various possibilities exist about how the code can physically model specific phenomena. Some examples are:

- Choice between engineering type models for choking or use of code implicit calculation of critical two-phase flow conditions,
- Flow multipliers for subcooled or saturated choked flow,
- The efficiency of separators,
- Two-phase flow characteristics of main coolant pumps,
- Pressure loss coefficient for pipes, pipe connections, valves, branches etc.

Since in many cases direct measured data are not available or, at least, not complete, the user is left to his engineering judgment to specify those parameters.

Input Parameter Related to Specific System Characteristics

The assessment of LWR safety codes is mainly performed on the basis of experimental data coming from scaled integral or separate effect test facilities. Typically in these scaled-down facilities, specific effects, which might be small or even negligible for the full-size reactor case, can become as important as the major phenomena to be investigated. Examples are the release of the heat from the structures to the coolant, heat losses to the environment, or small bypass flows. Often, the quality of the prediction depends largely on the correct description of those effects which needs a very detailed representation of the structural materials and a good approximation of the local distribution of the heat losses. However, many times the importance of those effects are largely underestimated and, consequently, wrong conclusions are drawn from results based on incomplete representation of a small-scale test facility.

Input Parameters Needed for Specific System Components

The general thermal-hydraulic system behavior is described in the codes by the major code modules based on a one-dimensional formulation of the mass, momentum and energy equations for the separated phases. However, for a number of system components, this approach is not adequate and consequently additional, mainly empirical models have to be introduced, e.g. for pumps, valves, separators, etc. In general, these models require a large amount of additional code input data, which are often not known since they are largely scaling dependent.

A typical example is the input data needed for the homologous curves which describe the pump behavior under single and two-phase flow conditions which in general are known only for a few small-scale pumps. In all these cases, the code user has to extrapolate from existing data obtained for different designs and scaling factors which introduces a further uncertainty to the prediction.

Specification of Initial and Boundary Conditions

Most of the existing codes do not provide a steady state option. In these cases pseudo-steady state runs have to be performed using more or less artificial control systems in order to drive the code towards the specified initial conditions. The specification of stable initial and boundary conditions and the setting of related controllers require great care and detailed checking. If this is not done correctly, a large risk that even small imbalances in the initial data will overwrite the following transient exists, especially for slow transients and small break LOCA calculations.

Specification of State and Transport Property Data

The calculation of state and transport properties is usually done implicitly by the code. However, in some cases, for example in RELAP5, the code user can define the range of reference points for property tables and, therefore, can influence the accuracy of the prediction. This might be of importance especially in more “difficult regions”, e.g. close to the critical point or at conditions near atmospheric pressure. Another example is constituted by the fuel materials property data: the specification of fuel rod gap conductance (and thickness) is an important parameter, affecting core dryout and rewet occurrences that must be selected by the user.

Selection of Parameters Determining time Step Sizes

All the existing codes are using automatic procedures for the selection of time step sizes in order to provide convergence and accuracy of the prediction. Experience shows, however, that these procedures do not always guarantee stable numerical results and, therefore, the user might often force the code to take very small time steps in order to pass through trouble spots. In some cases, if this action is not taken, very large numerical errors can be introduced in the evolution of any transient scenario and are not always checked by the code user.

Code Input Errors

In order to prepare a complete input data deck for a large system, the code user has to provide a huge number of parameters (approximately 15 to 20 thousand values for a NPP nodalization) which he has to type one by one. Even if all the codes provided consistency checks, the probability for code input errors is relatively high and can be reduced only by extreme care following clear quality assurance guidelines.

3.2.1 Qualification of Computational Tools

A key feature of the activities performed in nuclear reactor safety technology is constituted by the necessity to demonstrate the qualification level of each tool adopted within an assigned process and of each step of the concerned process. Computational tools include (numerical) codes, nodalizations and procedures. Furthermore, the users of those computational tools are part of the process and need suitable demonstration of qualification.

A consistent application (development, qualification and application) of a thermalhydraulic system code is depicted in Fig. 2. The code development and improvement process, block 1 in Fig. 2, is conducted by ‘code developers’ who make extensive use of assessment (block 4), typically performed by independent users of the code (i.e. group pf experts independent from those who developed the code). The consistent code assessment process implies the availability of experimental data and of robust procedures for the use of the codes, blocks 2 and 3 respectively.

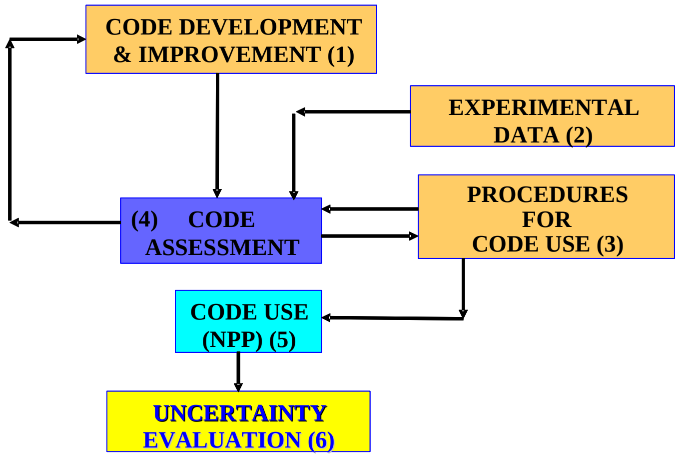


Figure 2: A consistent application (development, qualification and application) of a thermalhydraulic system code.

Once the process identified by blocks 1 and 4 is completed, a qualified code is available to the technical community, ready to be used for NPP applications (block 5). The NPP applications still require 'consistent' procedures (block 3) for a qualified use of the code. The results from the calculations are, whatever the qualification level achieved by the code, affected by errors that must be quantified through appropriate uncertainty evaluation methodology (block 6).

The following constitute the main requisites for a qualified use of the code [15]:

- Capability of the code to reproduce the relevant phenomena occurring for the selected spectrum of accidents;
- Capability to reproduce the peculiarities of the reference plant/facility;
- Capability to produce suitable results for a comparison with the acceptable criteria;
- Availability of qualified users.

Essentially the code must be able to reproduce two fundamental aspects, ref. [21]:

a. The NPP and the accident conditions: all the relevant zones, systems, procedure and related actuation logic is to be included in the calculation; this item also includes any external event, boundary and initial condition necessary to identify the plant but also the selected accident;

b. The phenomena occurring (expected) during the accident.

In order to ensure those capabilities, the code qualification process is needed and the following two phases can be identified:

1. Development phase: several models are created, developed and improved by the code development team; many checks are necessary to qualify each model and the global architecture of the code;

2. Independent assessment phase: the code is ready to be used but qualified calculations performed by organizations independent from the code-development team are needed to check independently the declared capabilities of the code.

It is relevant to note that in the development phase the code models can be changed and the code is not available to the final user. In the independent assessment phase, the final version of the code is distributed and the user is generally forbidden to change any element of the code models apart the normal available options as described in the user manual.

The activities performed during the development phase are:

- Verification: it consists in the review of the source coding relative to its description in the documentation. In other words, *code verification* involves activities that are related to Software Quality Assurance (SQA) practices and to activities directed toward finding and removing deficiencies in models and in numerical algorithms used to solve partial differential equations. SQA procedures are needed during software development and modification, as well as during production computing. SQA procedures are well developed in general, but areas of improvement are needed with regard to software operating on massively parallel computer systems. During the verification step, the correct working of models, interfaces, numerics is checked to ensure that the code, in all its components, is free of errors and produces results.

- Validation (or assessment): it consists in evaluating the accuracy of the values predicted by the code-nodalization against relevant experimental data for important phenomena expected to occur. In other words, *code validation* emphasizes the quantitative assessment of computational model accuracy by comparison with high-quality validation experiments, that is, experiments that are well characterized in terms of measurement and documentation of all the input quantities needed for the computational model, as well as carefully estimated and documented experimental measurement uncertainty. The validation process ensures the consistency of the results produced by the code; i.e. it proves that the code, as a whole system, is capable to produce meaningful results: not only the code-system works, but it works in the right direction.

The *independent code-assessment* is carried out by independent users of the code and has the aim to quantify the code accuracy, which is the discrepancy between transient calculations and experiments performed in ITF. The independent assessment of the code involves different aspects, like:

1. Qualification of the nodalization;

2. Qualification of the user;

3. Definitions of procedures for the use of the code;

4. Evaluation of the accuracy from a qualitative and quantitative point of view.

The above items are connected with the application of the code to experimental tests performed in ITF. The procedure for the qualification of the nodalization is described with more details in the next section.

Besides the demonstration of the code capability in reproducing an experiment performed in a test facility, the code must be checked also in performing NPP calculation. This constitutes the final step of the independent code assessment: the demonstration of the code capability at a different scale, i.e. the full scale of the NPP. A nodalization of a NPP is prepared and qualified. The check consists in a “similarity analysis” generally involving a Kv-scaled calculation. In this kind of calculation, the initial and boundary conditions of an experiment performed in a ITF are properly scaled and implemented in the NPP nodalization. The results of the NPP scaled nodalization must reproduce the relevant phenomena occurring in the experiment. Alternative ways to prove the code capability at the NPP scale are constituted by the comparison with other qualified NPP code results or, if available, with data obtained in NPP operational transients. As the procedure followed for this part of the code assessment is the same adopted for the qualification process of the nodalization, more details are given in the next section.

The contemporaneous acceptability of the accuracy (step of the process connected with experiments in ITF) and of the similarity analysis (step of the process connected with NPP) constitutes the positive demonstration of the code capability and the end of the code-assessment. The calculated accuracy is possibly included in the data base suitable for uncertainty evaluation (block 6 in Fig. 2, [17, 18]). If the accuracy is not in the range of acceptability or the code fails the similarity analysis, the code is considered not qualified and the code-development team shall be informed in order to develop new code models or to improve the existing ones.

3.2.2 ***Nodalization Qualification***

Assuming the availability of a qualified code and of a qualified user, it is necessary to define a procedure to qualify the nodalization in order to obtain qualified (i.e. reliable) calculation results. In this section a procedure for the nodalization qualification is discussed.

A major issue in the use of mathematical models is constituted by the model capability to reproduce the plant or facility behavior under steady state and transient conditions. These aspects constitute two main checks for which acceptability criteria have to be defined and satisfied during the nodalization-qualification process. The first of them is related to the geometrical fidelity of the nodalization of the reference plant; the second one is related to the capability of the code-nodalization to reproduce the expected transient scenario.

The checks about the nodalization are necessary to take into account the effect of many different sources of approximations, like:

1. The data of the reference plant available to the user are typically non exhaustive to reproduce a perfect “schematization” of the reference plant;

2. From the available data, the user derives an approximated nodalization of the plant reducing the level of detail;

3. The code capability to reproduce the hardware, the plant systems and the actuation logic of the systems reduces further the level of detail of the nodalization.

The reasons for the checks about the capability of the code-nodalization to perform the transient analysis deriving from following considerations:

1. The code options must be adequate;

2. The nodalization solutions must be adequate;

3. Some systems components can be tested only during transient conditions (e.g. ECCS that are not involved in the normal operation).

A simplified scheme of a procedure that can be adopted for the qualification of the nodalization is depicted in Fig. 3 [22]. In the following it has been assumed that the code has fulfilled the validation and qualification process and a "frozen" version of the code has been made available to the final user. This means that the code-user does not have the possibility to modify or change the physical and numerical models of the code (only the options described in the user manual are available to the user). With reference to Fig. 3, the qualification procedure of the nodalization is described step by step:

Step "a"

This step is related to the information available by the user manual and by the guidelines for the use of the code. This type of information takes into account the specific limits and assumptions of the code (specific of the code adopted for the analysis) and some guidelines deriving from the best practices for realizing the nodalization. From a generic point of view the following aspects should be carefully adopted:

- Homogeneous nodalizations;
- Strict observation of the user guidelines;
- Standard use of the code options.

Step "b"

User experience and developers recommendations are useful to set up particular procedure to be applied for a better nodalization. These special procedures are related to the specific code adopted for the analysis. An example is constituted by the "slice nodalization" technique adopted with the RELAP5 code to improve the capability of the code to reproduce transients involving natural circulation phenomena.

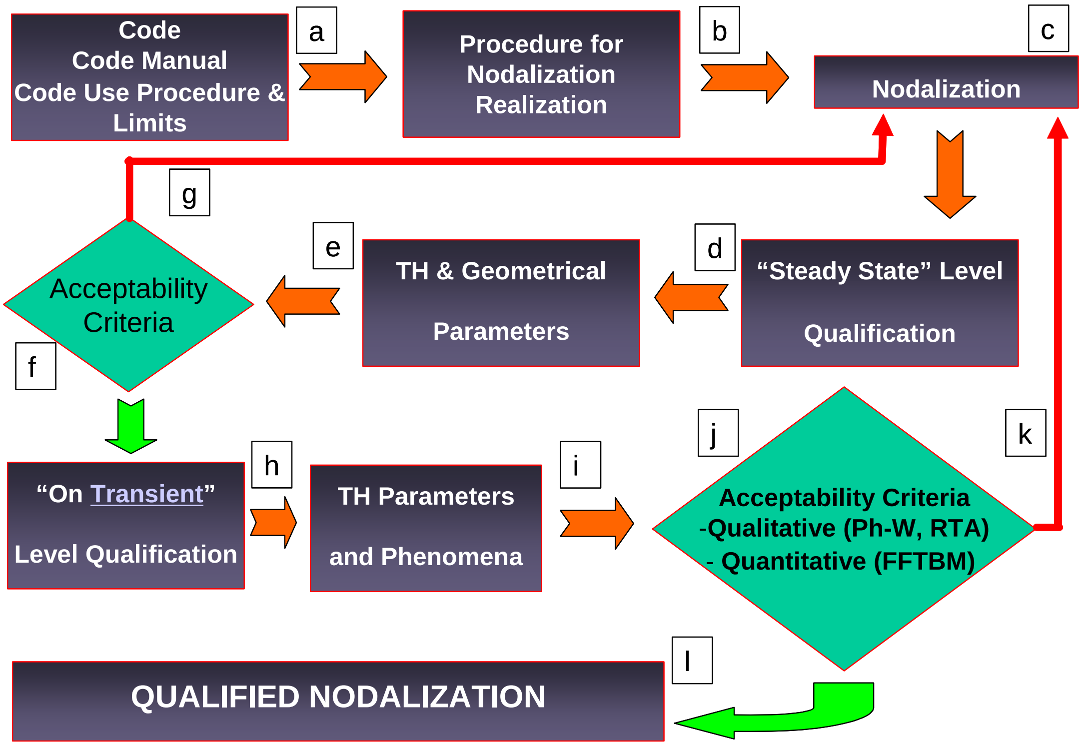


Figure 3: Flow sheet of nodalization qualification procedure.

Step "c":

The realization of the nodalization depends on several aspects: available data, user capability and experience, code capability. The nodalization must reproduce all the relevant parts of the reference plant; this includes geometrical and materials fidelity and reproduction of the systems and related logics. From a generic point of view the following recommendations can be done:

- Data must be qualified or in other words, data has to derive from:
    - qualified data facility (if the analysis is performed for a facility);
    - qualified test design;
    - qualified test data.
- The data base for the realization of the nodalization should be derived from official document and traceability of each reference should be maintained. However three different type of data can be identified:
- qualified data, from official sources;
- data deriving from non-official sources; these type of data can be derived from similar plant data, or other qualified nodalization for the same type of plant; the use of these data can introduces potential errors and the effect on the calculation results must be carefully evaluated;

- data assumed by the user; these data constitute some assumptions of the user (on the base of the experience or by similitude with other similar plants). The use of this type of data should be avoided. Any special assumptions adopted by the user or special solutions in the nodalization must be recorded and documented.

Step "d"

The "steady state" qualification level includes different checks: one is related to the evaluation of the geometrical data and of numerical values implemented in the nodalization; the other one is related to the capability of the nodalization to reproduce the steady state qualified conditions. The first check should be performed by a user different from the user has developed the nodalization. In the second check a "steady state" calculation is performed. This activity depends on the different code peculiarities. As an example, for RELAP5, the steady state calculation is constituted by a "null transient" calculation (i.e. the "transient" option is selected and no variation of relevant parameters occurs during the calculation).

Step "e"

The relevant geometrical values and the relevant thermal-hydraulic parameters of the steady state conditions are identified. The selected geometrical values and the selected relevant parameters are derived respectively from the input deck of the nodalization and from the steady state calculation for performing the comparison with the hardware values and the experimental parameters.

Step "f"

This is the step where the adopted acceptability criteria are applied to evaluate the comparison between hardware and implemented geometrical values in the nodalization (e.g. volumes, heat transfer area, etc…) and between the experimental and calculated steady state parameters (e.g. pressures, temperatures, mass flow rates, etc…). Some comments can be added:

- The experimental data are typically available with error bands which must be considered in the comparison with the calculated values and parameters;

- The steadiness of the steady state calculation must be checked.

Step "g"

If one or more than one of the checks in the step "f" are not fulfilled a review of the nodalization (step "c") must be performed. This process can request more detailed data, improvement in the development of the nodalization, different user choices. The path "g" must be repeated until all acceptability criteria are satisfied. A list of geometrical values and of thermal-hydraulic parameters to be checked is given in Refs [14, 22] together with acceptable errors.

Step "h"

This step constitutes the "On Transient" level qualification. This activity is necessary to demonstrate the capability of the code-nodalization to reproduce the relevant thermal-hydraulic phenomena expected during the transient. This step also permits to verify the correctness of some systems that are in operation only during transient events. Criteria, both qualitative and quantitative, are established to express the acceptability of the transient calculation. Two different aspects can be identified:

1. The code input deck concerns the nodalization of a ITF. In this case the code calculation is used for the Code Assessment. Checks include the code options selected by the user, the solutions adopted for the development of the ITF nodalization, the logic of some systems (e.g. ECCS). Typically many experimental results are available, thus a similar test can be adopted for performing the “On Transient” level qualification.

2. The objective of the code calculation is constituted by the analysis of a transient in a NPP. In this case it is necessary to check the nodalization capability to reproduce the expected thermal-hydraulic phenomena occurring during the transient, the selected code options, the adopted solutions for the development of the NPP nodalization and the logic of the systems not involved in the steady state calculation. Typically no data exists for the transients performed in the NPP. For this reason, data from experiments carried out in ITF can be used for performing the so-called “Kv-scaled” calculation. The Kv-scaled calculation consists in using the developed NPP nodalization for predicting an experimental transient (whose kind is similar to the one under investigation in the NPP) performed in a ITF. The NPP nodalization is prepared for the Kv-scaled calculation by properly scaling the BICs characterizing the selected transient in the ITF. In other words, power, mass flow rates and ECCS capacity are scaled adopting as scaling factor the ratio between the volume of the facility and the volume of the NPP. The capability of the nodalization to reproduce the same transient evolution and the thermal-hydraulic relevant phenomena is the needed request for satisfying the “On Transient” qualification level.

Step “i”

In this step the relevant thermal hydraulic phenomena and parameters are selected and a comparison between the calculated and experimental data is performed. The selection of the phenomena derives from the following sources:

- Experimental data analysis (engineering judgment is request);
- CSNI phenomena identification;
- Use of Relevant Thermal-hydraulic Aspects (RTA, engineering judgment is request).

Step “j”

This is the step where checks are performed to evaluate the acceptability of the calculation both from qualitative and from quantitative point of view. For the qualitative evaluation the following aspects are involved:

- Visual observation. This means that a visual comparison is performed between experimental and calculated relevant parameters time trends;

- Sequence of the resulting events. This means that the list of the calculated significant events together with their timing of occurrence is compared with the experimental events;

- Use of the CSNI phenomena. The relevant phenomena suitable for the code assessment and their relevance in the selected facility and in the selected test are identified. A judgment can be express taking into account the characteristics of the facility, the test peculiarities and the code results;

- Use of the RTAs. RTAs are typically identified inside the phenomenological windows (i.e. time windows where a unique relevant phenomenon is occurring) and are characterized by special parameters. These parameters can be time values, single values, integral values, gradient values and nondimensional values.

Quantitative checks are carried out by using the Fast Fourier Transform Based Method (FFTBM). This special tool performs the comparison between experimental and calculated time trends in the frequency domain for a list of selected parameters and calculates, for each of them, a numerical value by which the accuracy is quantitatively evaluated (no engineering judgment is involved in this process). The FFTBM makes also possible to obtain a numerical judgment of the overall results of the calculation. Criteria based on the values attained by FFTBM had been selected for accepting the transient calculation. A description of the FFTBM can be found in [23].

Step "k"

This path is actuated if any of the checks (qualitative and quantitative) is not fulfilled. The nodalization is improved by adopting different noding solutions, changing code options or increasing the level of detail using, if available, more precise data. Every time the nodalization is modified a new qualification process shall be performed through the loop "c-d-e-f-h-i-j-c".

Step "l"

This is the last step of the procedure. The obtained nodalization is used for the selected transient and the selected facility or plant. Any subsequent modification of the nodalization (e.g. necessary to better reproduce the experimental results) requires a new qualification process both at "steady state" and "on transient" level.

3.3 Development and Use of Coupled Computer Codes

Complex computer codes are used for the analysis of the performance of NPPs. They include many types of codes that can be grouped in different categories [24] like Three-Dimensional Neutron Kinetics codes (3D NK); Fuel behavior codes; Thermal-hydraulic codes, including system codes, subchannel codes, porous media codes and Computational Fluid Dynamic (CFD) codes; Containment analysis codes; Atmospheric dispersion and dose codes and Structural codes.

Historically, these codes have been developed independently, but have been mainly used in combination with system thermal-hydraulic codes. By increasing the capacity of computation technology, safety experts thought of coupling these codes in order to reduce uncertainties or errors associated with the transfer of interface data and to improve the accuracy of calculation. The coupling of primary system thermal-hydraulics codes with neutronics ones is a typical example of code coupling; other cases include coupling of primary system thermal hydraulics with structural mechanics, fission product chemistry, computational fluid dynamics, nuclear fuel behavior and containment behavior. Problems that need to be addressed in the development and use of coupled codes include ensuring adequate computer capacity and efficient coupling procedures, validation of coupled codes and evaluation of uncertainties, and consequently the applicability of coupled codes for safety analyses.

The major purposes of the development of coupled code are to be capable of representing the results of interactions between different physical phenomena in more detail. Since the calculation method of each code is not changed, reduction of computational time or necessary computer memory volume is not expected. Nevertheless, many additive benefits are expected as follows:

- Since the interface data are easily, automatically and frequently exchanged between codes, the results of calculation would be obtained faster than the combination of individual codes and also be more reliable;

- Since the development works are limited to the interface part, the cost and time for development can be minimized;

- Since the interface data between each code would be adjusted to meet the specifications (e.g. noding of the system or time increment of calculation) of each code at the development stage, additional assumptions or data averaging and reductions are not required when performing the calculation;

- Those that have the knowledge of the existing codes are not necessary to study the coupled code from the beginning, because the existing knowledge is applicable to the coupled code.

It is expected that those benefits can contribute to the improvement of activities carried out by both licensing authorities and industries. Expectations for licensing authorities can mainly be derived from the features of coupled codes such as more accurate calculation than the combination of individual codes. These are summarized as follows:

- Improvement of the understanding of the phenomena of interest for safety;

- Better assessment/demonstration of the conservatisms (versus historical approaches such as the use of point kinetics or evaluation models);

- Extension of the capabilities of the codes for safety analysis and training/simulators;

- Better assessment of uncertainties associated with the use of best estimate couplet codes.

Many benefits are expected with the use of coupled codes for industries. These are:

- Faster turnaround of calculation allows the users to perform more precise analysis and more sensitivity or case studies. This would contribute in more detail to understand the features of the plant, systems or components.

- More accurate calculation would contribute to remove unnecessary uncertainties and to identify margins available to use for the plant;

- Uncertainties due to user effects would be minimized because the existing knowledge of individual codes is applicable to the coupled codes.

The request to use qualified tools in licensing calculations constitutes one of the main problems to be addressed in the development of coupled computer codes and it is caused by the limited availability of data, which can be obtained from operating plants. To reduce the effort for the qualification of the coupled codes, code developers are requested to use only validated revisions of codes. In addition, the code developers are requested to:

- Design the coupling so that auditing is easy and feasible;

- Provide guidelines to minimize user effects;

- Allow provisions for reasonable conservatisms;

- Structure the code so that coupling is easy and feasible;

- Standardize the coupling procedures;

- Integrate as much as possible the existing approved calculation methodologies.

3.3.1 3D Neutron Kinetics Codes

The state of the art of currently used multi-dimensional neutron kinetics models for LWR calculations of core time-dependent spatial neutron flux distribution includes the utilisation of 3-D neutron diffusion equation based on two neutron energy groups and with six groups of delayed neutron precursors. This has been found adequate for steady-state applications and for those transient applications for which direct validation has been possible.

The energy resolution of the original interaction data is often reduced into only two energy groups: one group for thermalized neutrons and another group for all the rest. The geometry is homogenised into a mesh of macroscopic homogeneous regions, called nodes. The nodal calculation can handle the full-scale spatial dependence of the neutron flux and produce global power and reaction rate distributions that are coupled to a thermal hydraulics calculation. The result is either a static or a dynamic simulation of the reactor response under different operating conditions.

The major calculation features of the neutron diffusion models include the ability to perform eigenvalue (keff), transient flux, xenon transient, decay heat and adjoint calculations. Pin power reconstruction capabilities are also available to obtain pin power and associated intranodal neutron flux distributions from the calculated nodal fluxes.

The 3-D capability provides the basis for realistic representation of the complete reactor core, although provisions are included to represent appropriate symmetry sections, *i.e.* half- and quarter-core sections, for computational efficiency. Carefully selected boundary conditions for the symmetry planes are paramount for adequate calculation results in those cases. One-dimensional capabilities are also usually available for simulation of transients with predominant axial neutron flux variations. It should be noted that subcritical conditions are in some cases not directly calculated. Those conditions are, from mathematical perspective, obtained from partial differential equations representing a source term defined problem.

3.3.2 Computational Fluid Dynamics Codes

CFD (Computational Fluid Dynamics) is an advanced methodology for three dimensional computations of gas and liquid flows [25]. It has been used in non-nuclear industry for more than two decades.

All major CFD codes have in-built links to industrial-standard CAD/CAE packages to accelerate the mesh generation process, state-of-the-art solvers for running on parallel-architecture machines, sophisticated modules for post-processing of data (including the use of animations), and automatic links to FEM software packages for performing the associated stress analysis.

With the increasing cost-effectiveness and public acceptance of nuclear power, application of CFD in the nuclear area is rapidly increasing also, mostly for process design purposes. The use of CFD in safety-related analysis is in a starting stage, but development activities in this direction can be observed more and more.

The use of CFD for plant modifications can be found in the area resolving sump clogging issue. To determine the amount of debris transport, flow field in containment floor to the sump screen (or strainer) has been calculated by commercial CFD codes (like CFX, FLUENT, FLOW-3D, STAR-CD) with sophisticated turbulence models.

Another application includes the calculation of flow field in a pipe network in modifications where a component, such as a valve, is being replaced. It is important to show that the new component does not cause phenomena such as flow separation and vortex building which may cause unwanted phenomena such as vibrations. Another application area is investigation of thermal stratification and temperature fluctuations in mixing points, which can lead to thermal fatigue. The field of CFD has come a long way in one phase flows. However, in the field of two phase flows, a lot of development and validation work remains to be done. Since most of the scenarios of interest in reactor safety calculations are two phase transients, the future use of CFD in these areas will have a slow upward trend where every step of progress should be accompanied with experimental validation.

However, in regard to the application of CFD codes to nuclear safety there is a conceptual problem to overcome. Public perception of safety in regard to nuclear power is more highly charged than in other areas, and great care must be taken to ensure that results obtained using new methods can be trusted. From a regulatory standpoint, a significant part of the proof of plant robustness to Design-Basis Accidents (DBAs) has been supplied by application of systems codes, such as TRACE, RELAP5, ATHLET and CATHARE. Nominated versions of these codes have been made available to the regulatory authorities for inspection, and only the results of these versions are used in licensing procedures. However, the most highly developed CFD codes are of commercial origin, and the code vendors, all in competition with each other, do not want to give internal secrets away; consequently the source code is generally not available for inspection. Two possible alternatives to this problem are:

1. The code vendors come to a special confidential arrangement with regulatory authorities to give them access to the physical models within the code;

2. Put increased effort into open-source CFD development.

Furthermore, the following aspects can be mentioned in relation with the status of advancement and use of CFD code in the nuclear technology:

- Proper user guidelines on how to perform reliable CFD simulations have been formalised;

- Demonstration of the reliability of sample calculations in terms of validated cases has been established for single-phase applications;

- Two-phase CFD is a subject of active research but needs further development and validation;

- CFD does not aspire to replace system codes but complement them in certain (3-D) situations;

- Always in safety analysis, and hence also for the use of CFD in the licensing process, care must be taken to evaluate uncertainties to give margins of safety.

3.4 The Modern Approach: The Best Estimate Plus Uncertainty Methods

In the past, large uncertainties in the computer models used for nuclear power system design and licensing have been compensated using highly conservative assumptions. The Loss-Of-Coolant-Accident Evaluation Model is one of the main examples about this approach. However, the use of excessive conservatism results in significant economic penalty and not necessarily provides commensurate safety benefits. As a consequence, today the use of “best-estimate” code predictions rather than “conservative” estimates can be identified. This approach requires replacing subjective judgments about the adequacy of a code or of the degree of conservatism in the adopted assumptions with logical quantitative measures.

However, notwithstanding the important achievements and progresses made in recent years, the predictions of the best estimate system codes are not exact but remain uncertain because [18]:

1. The assessment process depends upon data almost always measured in small scale facilities and not in the full power reactors;

2. The models and the solution methods in the codes are approximate: in some cases, fundamental laws of the physics are not considered.

Consequently, the results of the code calculations may not be applicable to give exact information on the behaviour of a NPP during postulated accident scenarios. Therefore, BE predictions of NPP scenarios must be supplemented by proper uncertainty evaluations in order to be meaningful.

As already underlined the word ‘uncertainty’ and the need for uncertainty evaluation are connected with the use of BE codes instead of ‘conservative’ codes or assumptions in the code application. Moreover, nowadays the application of 3-D neutron-kinetics thermal-hydraulic coupled codes implies the choice of the BE approach and consequently the evaluation of uncertainty.

The selection of a best estimate analysis in place of a conservative one depends upon a number of conditions that are away from the analysis itself. These include the available computational tools, the expertise inside the organization, the availability of suitable NPP data (e.g. the amount of data and the related details can be much different in case of best estimate or conservative analysis), or the requests from the national regulatory body. In addition, conservative analyses are still widely used to avoid the need of developing realistic models based on experimental data, task that may reveal ‘un-realistic’ in the case of BDBA, or simply to avoid the burden to change approved code and/or the approaches or procedures to get the licensing.

A summary of drawbacks and benefits of the conservative and BEPU approaches is provided in Table 1. The conservative approach does not give any indication of the actual margins between the actual plant response and the conservatively estimated response. By contrast, the uncertainty evaluation provided in the best estimate approach is a direct measure of such margins. As a result the best estimate approach may allow for the elimination of unnecessary conservatism in the analysis and may allow the regulatory body and plant operating organization to establish a more consistent balance for a wide range of acceptance criteria. Moreover, a conservative approach does not give any indication about actual plant behaviour including time-scale for preparation of emergency operating procedures, or for the use in accident management and preparation of operation manuals for abnormal operating conditions.

Two independent approaches for estimating the uncertainties associated with the predictions of complex system codes can be identified:

- The propagation of code input errors (Fig. 4): this can be evaluated as being the most adopted procedure nowadays, endorsed by industry and regulators. It adopts the statistical combination of values from selected input uncertainty parameters (even though, in principle an unlimited number of input parameters can be used) to calculate the propagation of the errors throughout the code.

- The propagation of code output errors (Fig. 5): this is the only demonstrated independent working alternative to the previous one and has also been used for industrial applications. It makes full and direct reference to the experimental data and to the results from the assessment process to derive uncertainty. In this case the uncertainty prediction is not propagated throughout the code.

The first approach (Fig. 4), reviewed as the prototype for propagation of code input errors, is the so-called “GRS method” [26]), which includes the “CSAU method” (Code Scaling, Applicability and Uncertainty, [6]) and the majority of methods adopted by the nuclear industry. Although the entire set of the actual number of input parameters for a typical NPP input deck, ranging up to about 10^5 input parameters, could theoretically be considered as uncertainty sources by these methods, only a ‘manageable’ number (of the order of several tens) is actually taken into account in practice. Ranges of variations, together with suitable PDF (Probability Density Function) are then assigned for each of the uncertain input parameter actually considered in the analysis. The number of computations needed for obtaining the desired confidence in the results can be determined theoretically by the Wilks formula [27]. Subsequently, the identified computations (ca. 100) are performed using the code under investigation to propagate the uncertainties inside the code, from inputs to outputs (results).

Table 1: Drawbacks and benefits of the conservative and BEPU approaches.

	CONSERVATIVE	BEST ESTIMATE PLUS UNCERTAINTY
DRAWBACKS	Intentional conservatisms may not always lead to conservative results. For example, high power during Small Break LOCA (SBLOCA) may lead to over-prediction of swell level and over-prediction of core cooling, thus lower peak cladding temperatures, which is opposite to pessimistic expectations when evaluating the peak cladding temperature acceptance criterion	Practical application can be seriously time consuming – long and exhausting preparation of data, high number of calculations etc. This has also the impact on the requirements on the computation tools (high computer power, large data storage space)
	Degree of conservativism can change during a course of the event – specifically selected value of the parameter can be conservative in the beginning of the event but can change to even favorable value in another period of the event	Selection of uncertain parameters and definition of probabilistic distribution functions can be difficult due to the lack of information. Definition of uncertain parameters is also usually based on expert judgment leading to a possible user effect
	Intentional conservativism can results in misleading sequences of events and unrealistic time-scales	Extensive experimental and operational data are needed to reference applied values
	Conservative values of important parameters are typically selected based on engineering judgment (possible user effect) in combination with sensitivity calculations. Sensitivity calculations are usually limited in scope and typically do not include the investigation of the combined dependency, which means that each important parameter is tested individually without examining the possible influence when other parameters change. Moreover each of these parameters is tested for limited number of values (typically minimum and maximum is tested) so the most penalizing value can be easily omitted	
	When applying the best-estimate code in the conservative approach, the uncertainty and shortcomings of the code models are neglected assuming the intentional conservativism about the availability of the systems and components and about initial and boundary conditions is sufficient to compensate for it. This compensation is never analyzed and no evidence of sufficient conservativism over code models deficiencies is demonstrated	
BENEFITS	There is a long experience and well established procedures for conservative approach reducing the user effect	Prediction of 'realistic' response of the plant to the Postulated Initiating Event (PIE) is given
	There is a large amount of supporting materials – various FSARs, technical documents and reports with sensitivity calculations to provide the background information	Safety margins can be clearly determined
	Simple, clear and understandable procedures to demonstrate conservativism to convince the regulator.	Statistically sound evaluation of combined influence of input parameters is performed
		There are close links to experimental results justifying applied procedures

The main drawbacks of such methods are connected with:

a) The need of engineering judgment for limiting (in any case) the number of the input uncertain parameters;

b) The need of engineering judgment for fixing the range of variation and the PDF for each input uncertain parameter;

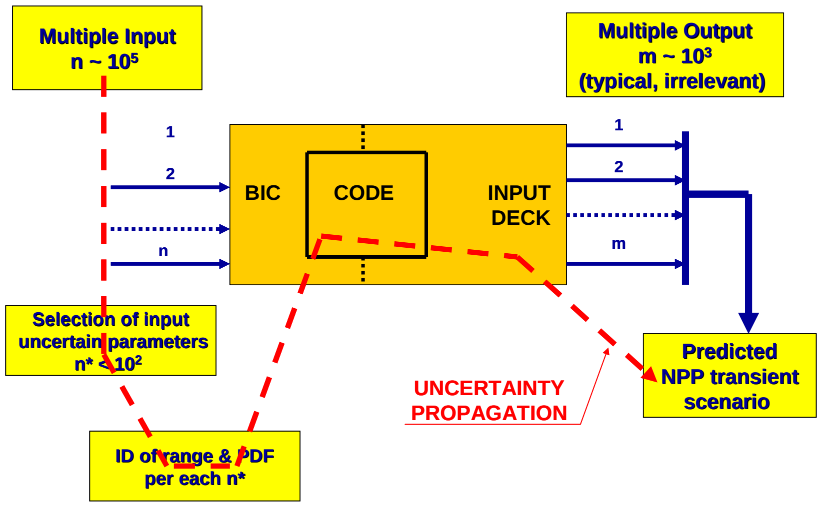


Figure 4: Uncertainty methods based upon propagation of input uncertainties.

c) The use of the code-nodalization for propagating the uncertainties: if the code-nodalization is wrong, not only the reference results are wrong but also the results of the uncertainty calculations;

d) The process of selecting the (about) 100 code runs is demonstrably not convergent, and the investigation of results from two or more different sets of 100 calculations shows different values for uncertainty.

The second approach reviewed as the propagation of code output errors, is representatively illustrated by the UMAE-CIAU (Uncertainty Method based upon Accuracy Extrapolation [28] 'embedded' into the Code with capability of Internal Assessment of Uncertainty [16, 17] in Fig. 5. Note that this class of methods includes only a few applications from industry. The use of this method depends on the availability of 'relevant' experimental data, where here the word 'relevant' is connected with the specific NPP transient scenario under investigation for uncertainty evaluation. Assuming such availability of relevant data, which are typically Integral Test Facility (ITF) data, and assuming the code correctly simulates the experiments, it follows that the differences between code computations and the selected experimental data are due to errors. If these errors comply with a number of acceptability conditions [28], then the resulting (error) database is processed and the 'extrapolation' of the error takes place. Relevant conditions for the extrapolation are:

- Building up the NPP nodalization with the same criteria as was adopted for the ITF nodalizations;

- Performing a similarity analysis and demonstrating that NPP calculated data are "consistent" with the data measured in a qualified ITF experiment.

The main drawbacks of this method are as follows:

a) The method is not applicable in the absence of relevant experimental information;

b) A considerable amount of resources is needed to establish a suitable error database, but this is a one-time effort, independent of subsequent applications of this method;

c) The process of combining errors originating from different sources (e. g, stemming from different ITF or SETF - Separate Effect Test Facility - different but consistent nodalizations and different types of transient scenarios) is not based upon fundamental principles and requires detailed validation.

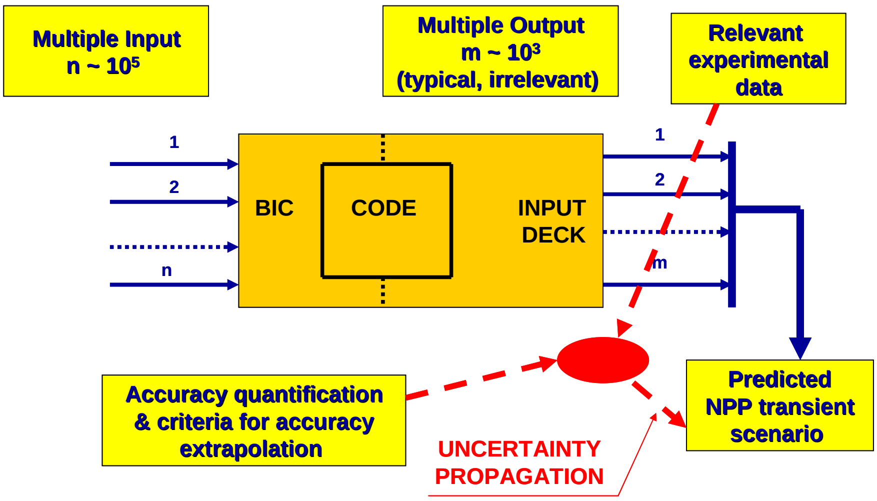


Figure 5: Uncertainty methods based upon propagation of output uncertainties.

4 USE OF ADVANCED NUMERICAL TOOLS FOR OLC AND OIP

Most modifications of plants, and all power uprates, include a number of component replacements. Examples of such components are steam separators, steam dryers, valves, moisture separator re-heaters, pumps and so on. The new components usually are different in detail from the ones that they replace, often due to the fact that the old design or its supplier does not exist on the market anymore. In introducing new components into an old plant one has to make sure that the new parts are compatible with old ones. In other words it should be proven that the new parts can coexist with the old parts without introducing unexpected disturbances (such as vibrations) into the plant normal operation. Experience shows that the large component suppliers of today prefer to take responsibility firstly for their component and not for the plant as a whole. At the same time it is the plant owner which always has the final responsibility for the safe and undisturbed operation of the plant. The plant owner should therefore carry out independent assessments of the suppliers analyses by doing new verification computations (can be done through consultants not contracted by the supplier). The use of best-estimate codes for this kind of verifying computations has therefore increased. Such computations are usually for process transient verifications and have to be as near to reality as possible. A rigorous quantification of errors in models used for process verifications is usually neither possible nor necessary. One should however ensure a minimum quality assurance of this approach in a few steps, such as the three steps outlined below. The first two steps are confidence building steps. They show the predictive capabilities of the model. The third step is to make use of the model to predict the future behaviour of the modified plant when subjected to various disturbance scenarios.

- A model of the relevant parts of the existing plant should be set up. By a time step/nodalisation study one should make sure that these factors do not have a significant effect on the solution (that one has a more or less converged solution). A check of the default setting of the models in the code should also be done to make sure that the default settings are in line with the physics of the phenomena to be simulated.

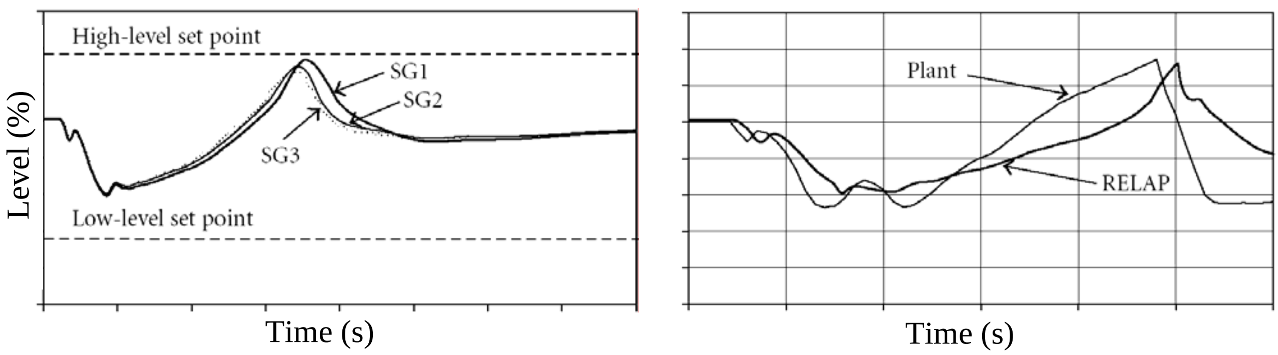
- The model should be used to simulate first the steady operation of the plant (for possible calibrating of some model parameters). Then a number of expected, or unexpected, transients that have occurred in the lifetime of the plant are simulated. The results should be compared with whatever relevant measurement data available from the plant in order to prove that the model first in a qualitative and then with an acceptable level of accuracy in a quantitative way reproduces the existing plant behaviour.

- The model should now be modified using data for the new components that replace the old components. New computations with the updated model are carried out. The results and the conclusions from them can be used to review the analyses by the hardware supplier. One should also look for phenomena which deviate unexpectedly from the known behaviour of the plant. The results of these predictions cannot be validated until after the modification is carried out, however until then they can serve as basis for the discussions involving assessment of the future behaviour of the modified plant.

4.1 Selected example

The selected transient took place in Unit 2 of Ascó NPP on August 6th 2000 [29, 30]. The transient started with the trip of the main Feed Water (FW) turbo-pump B when the plant was operating at steady-state nominal power. An automatic turbine run-back took place at 200% per minute until a load reference value of 70% was reached. Both the steam-dump and control rod systems actuated in order to compensate the load rejection. At the same time, the main FW control system required an increase in the speed of turbo-pump A and the opening of FW valves to avoid a decrease of SG levels. Subsequently, a first manual action was taken, consisting of an additional manual run-back of about 14%, with a new automatic steam-dump opening. SG levels decreased and reached their minimum values of 18%, 20%, and 26%. At this point, a second manual action took place, as a rapid increase in level was noticed in all SGs (150 seconds). The operator then manually closed the main feed water valve in loop 3 by about 20%. Under these conditions, the level of SGs 1 and 2 increased quite quickly and the automatic control produced a closing signal for the related valves (200 seconds). The flow increased unexpectedly in loop 3, still under manual control, and its level subsequently rose until it produced a reactor trip due to a high-level signal (Figure 6a). The unexpected flow increase was the concern of the operation team. A main FW turbo-pump trip usually leads to a turbine run-back and to renewed stability at lower power which allows the scram to be avoided. The behaviour of the plant seemed, a priori, abnormal. To start with the analysis, the available technical information was studied.

The run-back system was fully implemented in the model. FW turbo-pumps were included using the RELAP5 “pump” component and characterized by all necessary mechanical parameters. The follow-up actions arising from this analysis (Figure 6b) started from the fact that design adequacy is confirmed by the results of the calculations. No actions were taken on cause analysis, or design limits. Furthermore, no design or procedure modifications were needed. The only action taken was focused on informing on “lessons learned” about run-back effectiveness and on the fine behaviour of main FW valves.



a) SGs narrow range level b) SG3 narrow range level (plant and calculation results)

Figure 6: FW turbo-pump actual transient: an example of RELAP5 calculation.

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Part II – Feedback of International Stakeholders addressing OLC and OIP modifications at European Plants

6 INSTITUTIONS PARTICIPATING TO THE QUESTIONNAIRE

Thirty institutions among utilities, architect engineers and Technical Support Organisations (TSO) in the European Union have been contacted for participating to the questionnaire (Table 1). Eleven institutions confirmed their availability to participate the survey and filled the questionnaire (see Appendix). Among the institutions there are: 7 TSOs (ENPRO, FER, GRS, JSI, KFKI, LEI, NRI and University of Vienna), 1 Utility (Iberdrola), 1 Architect Engineer (Tractebel) and one Authority organization (SKI in Sweden).

TABLE 1: List of Institutions Contacted for Filling the Questionnaire.

#	INSTITUTION	COUNTRY	ACCEPTANCE
1	AREVA	France	
2	AREVA NP	Germany	
3	AVN	Belgium	
4	CEA	France	
5	CSN	Spain	
6	ENEL	Italy	
7	ENPRO	Bulgaria	YES
8	GIDROPRESS	Russia	
9	GRS	Germany	YES
10	IBERDROLA	Spain	YES
11	IJS	Slovenia	YES
12	IMPERIAL COLLEGE	UK	
13	IRSN	France	
14	KFKI	Hungary	YES
15	LEI	Lithuania	YES
16	NIKIET	Russia	
17	NRI	Czech Republic	YES
18	PSI	Switzerland	
19	SCN	Romania	
20	SKI	Sweden	YES
21	SOGIN	Italy	
22	SSTC NRS	Ukraine	
23	TRACTEBEL	Belgium	YES
24	TUDELFT	The Netherlands	
25	TVO	Finland	
26	UJD	Slovakia	
27	UNIVERSITY OF WIEN	Austria	YES
28	UNIVERSITY OF ZAGREB (FER)	Croatia	YES
29	UPC	Spain	
30	WESTINGHOUSE	Germany	

7 FEEDBACKS FROM THE SURVEY

The large part of the modifications mentioned by the institutions participating to the questionnaire are dealing with power uprate (maximum up to 10%) of several NPPs in their countries. In some cases the replacement of SG and/or introduction of new type of fuel have been also discussed. However minor modifications have been mainly introduced to theses plants like replacement of I&C. Table 2 summarizes the main plant modifications. It can be noted that different kinds of NPP were involved: PWR (Westinghouse and KWU), BWR (ABB and GE), WWER-440, WWER-1000 and also RBMK-1500.

In the major part of the cases, the same numerical tools adopted for the FSARs have been used for analyzing the impact of the modification on the plant system. However, in some cases new system codes or coupled codes have been used. The conservative deterministic approach was mostly adopted for the analysis, although some participants also mentioned the application of best estimate plus uncertainty methodology. In either way there were no direct benefit reported from the application of new numerical tools. At the same time half of the institutions confirmed the use of special set of T-H analyses that has to be performed in case of major plant modifications while the other half considered of no benefit the use of a specific set of analyses but the one adopted before for FSAR (see second and third column in Table 3).

The impact of the modifications on OLCs and/or OIPs of the plant differs from institution to institution depending on the exact plant modification performed. Most of the institutions reported only minor changes to the operating procedures but on the contrary in most cases the safety margins and set-points of the plant have been modified, e.g. former constant linear power limit was replaced by one decreasing with burn-up; the average temperature, the water level in the pressurizer, the pressure in the steam generators, the water level in the steam generators, etc… have been changed (see last column in Table 3).

In relation with the further exchange of experience on TH tools and methodology, most of the institutions (with only few exceptions) agreed on the benefit deriving from the discussion of modern T-H tools and methodologies and from the comparison of OLC and OIP between different countries. The suggested formats of such an exchange were forums and/or workshops (see second and third column in Table 4).

Finally, the participants mutually agreed that the most needed research is in the area of coupling of T-H and 3D N-K numerical tools and in the application of such tools for the analysis of local phenomena like the boron dilution or the pressurized thermal shock. Some participant also mentioned the importance of further development of the uncertainty evaluation methodologies and of their divulgation and use in the framework of the licensing process. Furthermore, some of the organizations are already participating in the current international projects and studies, like OECD projects SETH, PKL, PSB. The finding of these studies can be incorporated by either the plants (modifications of operational procedures) or the regulatory bodies (see last two columns in Table 4).

TABLE 2: Plant Modifications.

COUNTRY	PLANT	REACTOR TYPE	MODIFICATIONS
Belgium	Doel 1	Westinghouse PWR	SG replacement and power uprate (10%) in 2009
	Doel 2	Westinghouse PWR	SG replacement and power uprate (10%) in 2004
	Doel 3	Framatom PWR	SG replacement and power uprate (10%) in 1993
	Doel 4	Framatom PWR	SG replacement in 1997
	Tihange 1	PWR	SG replacement and power uprate (8%) in 1995
	Tihange 2	PWR	SG replacement and power uprate (10%) in 2001
	Tihange 3	PWR	SG replacement in 1998
Bulgaria	Kozloduy 5-6	WWER-1000	– New neutron monitoring system – Installation of an automated system for cold overpressure protection – Installation of hydrogen recombination systems
Czech Republic	Dukovany	WWER-440	– New fuel elements – Power uprate
	Temelin	WWER-1000	New fuel elements
Germany	Grohnde	KWU PWR	Power uprate
	Philippsburg 2	KWU PWR	Power uprate
	Neckarwestheim 2	KWU PWR	Power uprate
	Emsland	KWU PWR	Power uprate
	Isar 2	KWU PWR	Power uprate
	Unterweser	KWU PWR	Power uprate
	Brokdorf	KWU PWR	Power uprate
	Grafenrheinfeld	KWU PWR	Power uprate
Hungary	Paks	WWER-440	– New fuel elements – Power uprate
Lithuania	Ignalina 2	RBMK-1500	Second independent shutdown system

TABLE 2: Plant Modifications (Cont'd).

COUNTRY	PLANT	REACTOR TYPE	MODIFICATIONS
Slovenia/Croatia	Krsko	Westinghouse PWR	– SG replacement [2000] – Main FW and Condensate systems modifications [2000] – Power uprate by 6.3% [2000] – Replacement of the low pressure turbine [2006] – Uprated by additional 3% [2006] – New forced air cooling towers installed [2008]
Spain	Cofrentes	BWR	Power uprate
Sweden	Ringhals 1	ABB BWR	Power uprate
	Ringhals 2	Westinghouse PWR	– SG replacement – Power uprate
	Ringhals 3	Westinghouse PWR	SG replacement
	Forsmark 1-3	ABB BWR	– New types of fuel – Power uprate
	Oskarshamn 2	ABB BWR	Power uprate
	Oskarshamn 3	ABB BWR	– New types of fuel – Power uprate
	Barsebäck 1-2	ABB BWR	Power uprate

TABLE 3: Numerical Tools and Impact of the Modification.

COUNTRY	NUMERICAL TOOLS USED	USE OF SPECIFIC SET OF T-H ANALYSES	IMPACT OF MODIFICATION
Belgium	New methodologies and codes have been applied such as 3D Panther, COBRA-TRAC, CATHARE GB	Yes	Original OLC has been modified. Typical modifications are the average temperature, water level in the pressurizer, pressure in the steam generators, water level in the steam generators, primary mass flow rates, fuel enrichment. Some procedures have been modified taking into account the changes in operating conditions. For some units, set-points of safety valves and/or power operated relief valves have been adapted.
Bulgaria	Tools adopted for FSAR has been used. For Updated Safety Analysis Report such codes are used as RELAP5, MELCOR and DYN3D	Yes	Implementation of up-to-date high reliably equipment. Modification of pressure relief valves.
Czech Republic	Mainly the conservative deterministic approach is used. Although BE plus uncertainty is introduced lately. Such codes are used as RELAP5, ATHLET, MELCOR, COCOSYS, FLUENT, COBRA, DYN3D	No	Power up rate leads to the change of the RTS signals, e.g. output temperature trip. Change of I&C leads to confirmation of TRIP diversity.
Croatia	Tools adopted for FSAR has been used	No	OLC were updated: plant has been licensed for so-called “operating window”. This has enabled flexibility in the planning of the future fuel cycles.
Germany	Tools adopted for FSAR has been used. Partly an uncertainty analysis was required.	No	Reactor protection and instrumentation set-points were slightly changed. For example, pressurizer characteristics were changed for partial power and stretch-out operation. Optimizations of secondary and primary side bleed and feed procedures needed.

TABLE 3: Numerical Tools and Impact of the Modification (Cont'd).

COUNTRY	NUMERICAL TOOLS USED	USE OF SPECIFIC SET OF T-H ANALYSES	IMPACT OF MODIFICATION
Hungary	Tools adopted for FSAR has been used. Partly coupled code ATHLET-KIKO3D is used.	Yes	Changes in safety limits, e.g. the formerly constant linear power limit was replaced by one decreasing with burn-up. Hydroaccumulator initial pressure was decreased, while water volume was increased. Finally, no changes to the EOPs were needed
Lithuania	Such codes as RELAP5, ATHLET, CONTAIN, COCOSYS and QUABOX/CUBBOX-HYCA were adopted for RBMK-type and used	Yes	Modifications imply the small changes in previous OIP (for example: the availability of second independent shutdown system and new reactor shutdown and emergency core cooling algorithms were reflected in new versions of Technological Regulation)
Slovenia	– LOFTRAN code is used for non-LOCA transients – THINC-3 and FACTRAN are used for DNBR. New procedure has been used for DNBR calculations – BASH and LOCBART codes are used for LBLOCA – NOTRUMP is used for SBLOCA instead of obsolete code chains	No	Set-points and margins were changed due to new SGs and higher power: – Shutdown Margin – DNB parameters – RTS Instrumentation Trip Set-points Moderate impacts on operating procedures e.g.: emergency operating procedures (response to high steam generator level and low steam generator level)
Spain	Tools adopted for FSAR has been used	N/A	The limit of peak fuel pellet burnup has been extended
Sweden	Tools adopted for FSAR has been used	No (only review of initial list of transients)	For BWR with internal pumps modifications have been introduced to eliminate the loss of external power as a limiting transient by introducing energy storages. Minor changes to the operating instructions has to be done to address some phenomena as local boron dilution or positive moderator temperature coefficients of new fuel. In most cases, though, it is intended to avoid impact on the instructions by plant modifications.

TABLE 4: Needs and Benefits of TH-NK Research or International Study.

Country	Would there be a benefit exchange of experience on T-H tools/methodologies?	How to perform qualitative comparison on OLC and OIP between countries?	Specific needs for T-H and N-K research	Participation in current research or international study
Belgium	Existing currently exchange is enough	No benefit is foreseen from extensive comparison	N/A	N/A
Bulgaria	Yes. Mainly on 3D N-K and T-H codes.	Comparison between Russian-design plants would be beneficial. Better be organized in a form of seminars/workshops	Coupling of T-H and 3D N-K.	OECD PSB-WWER. But up to now no application of results at NPP
Czech republic	Would be useful in the area of T-H and CFD codes	N/A	Coupling CFD to system codes and 3D N-K codes	OECD-PSB, OECD-SETH, SETHII, OECD-PKLI and II, OECD/ROSA. Main aim is codes' validation
Croatia	Yes on both. Forums/workshops would be beneficial		Use of uncertainty methodologies	No direct participation. But Findings of the international projects are addressed through Periodic Safety Review of the plant.
Germany	No. Present information exchange is sufficient	No. Present information exchange is sufficient	Three-dimensional simulation of cooling system and containment, boron dilution, effect of non-condensable gases, ATWS, AM procedures, procedures at SG tube rupture, uncertainty analysis. Neutron transport methodology and sub-channel models	OECD SETH, OECD PKL//PMK/ROCOM, OECD ROSA, OECD BEMUSE, OECD SM2A. Results, possible consequences and recommendations are discussed in relevant committees, like Reactor Safety Commission and KTA

TABLE 4: Needs and Benefits of TH-NK Research or International Study (Cont'd).

Country	Would there be a benefit exchange of experience on T-H tools/methodologies?	How to perform qualitative comparison on OLC and OIP between countries?	Specific needs for T-H and N-K research	Participation in current research or international study
Hungary	Yes.	Yes. Mainly through workshops.	Quantification of uncertainty distributions for use in BE analysis	SETH-1, PKL and ROSA. Findings of those are not directly applied. However, a recent OECD GAMA action on evaluation of core exit temperature measurements could directly impact the selection of entry points to different procedures and transition points between EOP and SAMP.
Lithuania	Yes on both. Forums/workshops would be beneficial		Severe accident analysis for RBMK plants	PHEBUS. The findings of the project could not be directly incorporated into OLC and OIP due to different design of analyzed facilities
Slovenia	Yes. By forums.	Yes. By bilateral talks.	Usage of the latest ANS-2005 standards	No direct participation
Spain	No. Qualitative comparison leads a misunderstanding when is performed for a third party with no operating experience.		Further research in CFD codes, 3D T-H and N-K codes. Application of higher order numerical methods.	No participation
Sweden	Yes on both. Bilateral talks are very useful, especially for countries with rare types of reactors		Information of T-H behavior during shut-down or low power transients needs to be expanded. Local boron dilution and pressurized thermal shock fall into the same thermal hydraulic category of incidents. Also validation of detailed coupled T-H and N-K methods using plant data is important.	Participation in OECD/SETH-2, OECD/PKL-2 and the OECD/LSTF. The main reason is to have validated safety codes that can be used on Swedish containment and primary systems. The utilities update instructions regularly, e.g. the strategy to cool the core under small LOCA is under review

8 SUMMARY OF THE QUESTIONNAIRE

Brief summaries of the answers given by each institution to each questions of the questionnaire are provided here below. Detailed answers, as received by each institution participating to the questionnaire, can be found in the Appendix.

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification.

Almost all the participants confirmed uprate of several NPPs in their countries. Maximum uprate is 10%. Major modification is the replacement of SG. Some participants also mentioned introduction of new type of fuel.

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

Mainly the original T-H and N-K tools have been used. Just some organizations applied new tools / methodologies together with the original ones.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

All the participants confirmed the impact on the original OLC. For example, some changes were introduced in se-tpoints (e.g. in reactor protection).

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason?

In general no major effects on the original OIP have been noticed. Just some procedures have been optimized.

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

The participants replied that set of T-H analyses has been performed only at those NPPs where significant modifications have been performed.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

In general no additional operating modes to be improved / developed are foreseen. Two participants mentioned current development of SAMG (Severe Accident Management Guidelines).

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

The participants agreed that such comparison would be beneficial. Most of participants support the idea of exchange of experience by a forum/workshop.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Most of the participants are interested in different kinds of research activities. The main idea is the usage of three dimensional and coupled T-H and N-K codes.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

Detailed simulations have been performed manly for evaluation of set-points and modifications of accident management procedures.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

A large number of organizations participate directly in current research and international studies.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

The participants in general are interested in results of international projects but the findings aren't directly applied.

APPENDIX - Questionnaires

Bulgaria - ENPRO

Name of the Organization:

ENPRO Consult Ltd.

Entity (Electrical Utility, Architect Engineer, TSO)

Technical Support Organization

Reference Person (name, address, tel, e-mail):

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Country:

Bulgaria

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification

Kozloduy NPP, Units 5 and 6.

Modernization Programme of Units 5 and 6 of Kozloduy NPP. The Modernization Programme for Units 5 and 6 of Kozloduy NPP encompasses a set of improvements and modifications.

Basic objective of the Programme was to ensure long-term safe and effective operation of Units 5& 6 and to improve the equipment reliability and grid stability thereby serving as a basis for plant design lifetime extension. Some of the major improvements and activities are listed below:

- *Replacement of mechanical equipment;*
- *Operational conditions improvement;*
- *Updating the Safety Analysis Report*
- *Electrical equipment and systems modernisation;*
- *Implementation of I&C systems and new diagnostic systems;*
- *Enhancement of unit fire protection;*

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit).*

The original T-H and N-K computer codes were used for safety assessment of Units5&6, KNPP. Within the scope of elaboration of the Updated Safety Analysis Report of Unit5&6, KNPP, the latest versions of the T-H (e.g. RELAP5, MELCOR etc.) and N-K (e.g. DYD3D) computer codes available in ENPRO were used.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

A Measure of the Modernization program of Units 5&6, KNPP related to modifications of the OLCs is presented below:

Measure 10121 – Replacement of the neutron monitoring system (AKNP) with a more sophisticated system (AKNP-07-02) after its operation life runs out:

- *Implementation of up-to-date high reliably equipment*
- *Precise measurement of the new NF value in different operating modes*
- *Neutron flux value memory*

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason.

Some of the Measures (Items) of the Modernization program of Units5&6, KNPP related to modifications of the OIPs (not only) are presented below:

Measure 23251 – Installation of an automated system for cold overpressure protection (COP)

- *Modification of interlocks and pressure relief valves;*
- *Teleperm – controller*
- *Implementation of three independent channels.*

Measure 13011 – Installation of hydrogen detection and recombination systems

- *Implementation of alarms and monitoring of H2 concentration;*
- *H2 recombination and control of it's concentration below < 4%.*

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

A specific set of T-H analyses was developed within the scope of the Modernization program of Units 5&6, KNPP – Item 26122 "Updated Safety Analysis Report based on the format of PNAE G-01-36-95".

The same set (list) of T-H analyses would be used in the new version of the Updated SAR after implementing a specific modification in Units 5&6, KNPP (e.g. implementation of a new fuel type in the reactor core).

The list of T-H analyses was presented in the USAR documentation submitted to Kozloduy NPP.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

Yes, related to replacement of the type of the fuel.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment

The exchange of information and experience on the new modern T-H tools and methodologies would be highly beneficial especially from the point of view of interaction between 3-D neutron kinetics and system thermal-hydraulic codes. This is also relevant to the operating safety and the design/operation of the NPPs which are currently in operation (Units5&6, KNPP with WWER-1000 reactor type) and those which are intended to be constructed in the near future (Belene NPP).

b) performing a qualitative comparison on OLC and OIP of plants with those of other countries (having similar plant types)?

It would also be beneficial to make a qualitative comparison between the OLC and OIP of Kozloduy NPP Units 5&6 (WWER-1000) with those of other countries (Russia, Czech Republic etc.) with similar plant types.

If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

The most appropriate way of comparing the OLC and OIP of Kozloduy NPPs with those of other counties (with WWER reactor types) is to exchange information on seminars/workshops where the T-H tools/methodologies, applied in Bulgaria can be presented.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Activities in the field of coupling of the Thermal-hydraulic system and 3-D neutron kinetics codes are needed in the future cooperation programs. As a result, more detailed and accurate representation of NSSS structural components can be obtained for a variety of applications, including the analyses of BDBA. The integration of full three-dimensional (3D) neutron kinetics models of the reactor core into system transient codes allows for a best-estimate calculation of interactions between the reactor core behavior and plant response. In addition, the coupled T-H/3-D neutronics analyses could provide more detailed insight in the specification of operational procedures and could provide guidelines for optimization of EOPs.

Possible research activities are the following:

- *Implementation of the coupled code package RELAP5/DYN3D*
- *Implementation of the coupled codes package ATHLET/DYN3D*

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

A detailed analysis of the event "Total loss of electrical power supply" (SBO) for Kozloduy NPP, Units 5 and 6 was performed within the scope of the elaboration of USAR5&6. The effect of SG bleed and passive feed on the grace time for restoration of electrical power supply was investigated more precisely. Two main operator strategies are investigated. In order to be able to simulate the effect of the secondary bleed and the passive feed of the SGs, the feed water system was modelled in detail.

A detailed analysis of the initiating event "Total loss of feed water supply to the SGs, with application of Feed and Bleed on the primary side" was performed in the frame of Modernization program of Kozloduy NPP, Units 5&6.

The total loss of feed water (main, auxiliary and emergency) was analyzed, considering both possible paths (pressurizer discharge line and emergency pressurizer gas venting line) for bleeding capability, in order to assess the required discharge and injection capacities, the core cooling conditions during the Feed and Bleed mode, the time frame available for operator actions and the needed measurements and corresponding conditions for starting the Feed and Bleed mode. Several calculations have been performed with different groups of operator actions and different time for operator intervention.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

IAEA Research Contract № 13652 between the International Atomic Energy Agency (IAEA) and ENPRO Consult Ltd.

Coordinated Research Project on Evaluation of uncertainties in best estimate accident analysis. The main goals of this project are the following:

- *Development of input deck for the EREC PSB-WWER test facility for RELAP5/MOD3.3*
- *Qualification of the input deck in steady state and transient conditions;*
- *Evaluation of the uncertainties for a selected transient.*

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

No specific procedure is available for incorporation of findings of different international projects. Probably more close contacts and cooperation would improve the applicability of the developments and findings of such international projects.

Croatia - FER

Name of the Organization:

Faculty of Electrical Engineering and Computing

University of Zagreb

Entity (Electrical Utility, Architect Engineer, TSO)

TSO

Reference Person (name, address, tel, e-mail):

Tomislav Bajs

FER (ZVNE), Unska 3, 10000 Zagreb

Country:

Croatia

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification

PWR type NPP Krško (co-owned by Croatia and Slovenia) has replaced steam generators and did the power uprate of 6.3 % in year 2000. New fuel cycle strategy (18 month) has been licensed for the uprated power. In year 2006 following replacement of the low pressure turbine, NPP Krško output power has been uprated by additional 3%. In year 2008 new forced air cooling towers with cooling power of 167 MW were put in operation in the NPP Krško.

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

Combination of the original and new vendor codes and tools were used to license the plant for power uprate in year 2000. However, although new codes and tools were used, none of them could be considered as best-estimate or the state-of-the-art.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

OLC were updated to reflect new operating conditions. In order to have flexibility in the future cycles, plant has been licensed after power uprate for so-called "operating window". This has enabled flexibility in the planning of the future fuel cycles.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason.

There was no major effect on the OIPs. Some Emergency Operating Procedures (EOP) were optimised since new analysis provided more accurate data on uncertainties of certain setpoints.

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

This has not been developed. NPP Krško applies screening and evaluation of plant modifications in accordance with USA 10 CFR 50.59 rule. Need for new analysis or re-analysis is part of this process.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

Integration of the Shutdown Modes severe accidents into the Severe Accident Management Guidelines package is foreseen to complete the spectrum of accidental procedures.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?

Yes on both.

If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

Comparison forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries would be more beneficial.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Use of coupled codes (T-H and N-K) using uncertainty methodologies.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

Yes. Reason was verification of the setpoints/footnotes used in the EOPs.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

No.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

Finings of the international projects are addressed through Periodic Safety Review of the plant.

Germany - GRS

Name of the Organization:

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH

Entity (Electrical Utility, Architect Engineer, TSO)

TSO

Reference Person (name, address, tel, e-mail):

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Tel: +49 89 32004 408, E-mail: Horst.Glaeser@grs.de

Country:

Germany

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification.

8 PWRs applied for power up-rates and 6 were approved up to now. According to IAEA definition up-rates up to 7% are not considered as major modifications, what is the case for up-rates in Germany, see the following table:

Power plant	Thermal power (MW)	Uprate to (MW)	Year
Grohnde	3765	3850 3900	1990 1999
Philippsburg 2	3765	3850 3950	1992 2000
Neckarwestheim 2	3765	3850	1991
Emsland (in licensing process)	3765	3850 3950	1991 2003
Isar 2	3765	3850 3950	1991 1998
Unterweser	3733	3900	2000
Brokdorf	3765	3850	2006
Grafenrheinfeld (in licensing process)	3765	3950	2003

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

The original tools were applied. Partly, an uncertainty analysis was required for T-H and fuel rod failure analysis for a power up-rate.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

Reactor protection and instrumentation set-points were slightly changed. For example, pressurizer characteristics were changed for partial power and stretch-out operation. No major plant modifications.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason.

Optimizations of secondary and primary side bleed and feed procedures needed because of shorter times for preparation and successful actuation of these accident management procedures for beyond design basis accidents.

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

No major plant modifications, therefore, no specific check-list, usual check-list used.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

No.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

No, information exchange is sufficient.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

T-H: Three-dimensional simulation of cooling system and containment,
- *Boron dilution,*
- *Clogging and isolation material transport through the cooling system,*
- *Effect of non-condensable gases,*
- *ATWS, AM procedures, procedures at SG tube rupture,*
- *Uncertainty analysis.*

N-K: Neutron transport methodology,
- *Sub-channel models,*
- *High burn-up fuel behaviour,*
- *Uncertainty analysis.*

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

Extensive thermal-hydraulic calculations were performed for developing AM procedures, their instrumentation needs and initiators.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

OECD SETH, OECD PKL//PMK/ROCOM, OECD ROSA, OECD BEMUSE, OECD SM2A: Emergency procedures, validation, investigation of phenomena under conditions mentioned in question 8, alternative analysis methods.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

Results, possible consequences and recommendations are discussed in relevant committees, like Reactor Safety Commission and KTA (Kerntechnischer Ausschuss).

SPAIN - IBERDROLA

Name of the Organization:

IBERDROLA

Entity (Electrical Utility, Architect Engineer, TSO)

Utility

Reference Person (name, address, tel, e-mail):

NA

Country:

Spain

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification

Cofrentes NPP Extended Power Uprate to 111.85 of the original thermal power (2894 MWt).
Extension from eighteen month to two year cycles.

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

IBERDROLA has independent approved method. These methods were used to license the power uprate. No new methods were needed to uprate the power
No new methods to extend to two years cycles.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

The increase the power leads to highest thermal limits and increments PCT in LOCA . But enough margin to accommodate without changing licensing methods. Note that and increase in a 10% of limits is not a dramatic change.

For the extension to two years cycles has been necessary to extend the limit of peak pellet burnup. The vendors have evaluated the fuel design criteria extended slightly the maximum burnup. It was not necessary to go to high burnup.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason

N/A

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

In case of a design modification, the Spanish regulatory guideline 1.11 based on US 10CFR.50.59 rule has to be applied. There is a procedure in the plant to guarantee that every design modification or method and procedure change is evaluated in the same systematic way. For example, in case of the FSAR was affected, the specific chapter is addressed and if the 15th chapter is affected the accident or transient is identified.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

OLC and OIP have been established for all of modes. PSA analysis, for all of modes, has supported their performance.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

We don't see any benefit. Qualitative comparison leads a misunderstanding when is performed for a third party with no operating experience. An exhaustive comparison must be performed. There are specific organisms, INPO/WANO which support procedure evaluation and recommendations. Out of this framework is not recommended

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Computer Fluid dynamic codes
Three dimensional T-H and N-K codes
High order numerical methods

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

No. We don't have performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

NO, IBERDROLA is not participating.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

Following operational external experience.

Slovenia – JSI

Name of the Organization:

Jožef Stefan Institute

Entity (Electrical Utility, Architect Engineer, TSO)

TSO

Reference Person (name, address, tel, e-mail):

Prof. Dr. Borut Mavko

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Country:

Slovenia

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification

Plant: Nuclear power plant Krško, Westinghouse 2-loop PWR, operating since 1983

Type of modification: 6.3% power uprate with extended fuel cycle (gradually from 12 to 18 month) and steam generator replacement in 2000

- *new steam generators*
- *modification of Main Feedwater and Condensate Systems*
- *Auxiliary Piping Supports (small number of auxiliary piping supports had to be modified)*
- *Steam Dump Control System (lead/lag in controller)*
- *balance of plant instruments (small number)*

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

The analyses have been conducted as far as feasible using the same methodologies as used for the initial safety assessment of the plant. There have, however, been a limited number of changes to replace obsolete methods, to address new issues and to regain margin. All the safety analyses have been done in compliance with R.G. 1.70 Rev. 3.

The analyses were performed for the most conservative initial conditions from operating window and steam generator tube plugging (0 and 5%).

For most of the non-LOCA transients the computer code LOFTRAN was used intended to simulate the plant thermal kinetics, RCS, pressurizer, steam generators, and the feedwater system. It was the Westinghouse most used in the power uprate program.

For DNB analyses, the Revised Thermal Design Procedure (RTDP) has been used *(instead of the Improved Design Procedure). This universally accepted methodology statistically combines measurement and correlation uncertainties. The RTDP has not only contributed to accommodate the power increase, but has additionally permitted to increase the enthalpy rise hot channel factor (FDNH). The DNBR calculations were done by the computer codes THINC-3 and FACTRAN.*

The methodology used in the original large-break (LB) LOCA analysis was also used for the uprated design. It was performed with the combination of codes, including the codes BASH and LOCBART (same as in original licensing, but some newer versions were used). Also, besides chopped cosine power distribution, skewed to top power distributions has been considered.

For small-break (SB) LOCA the computer code NOTRUMP was used instead of obsolete code chains. Both LOCA analyses were based on Appendix K evaluation model.

For heatup events, the consideration of the ANS79 + 2σ decay heat curve helps to obtain acceptable results.

For steam line break core analysis the shutdown margin had to be reduced from 2% to 1.6% to compensate for the power increase and cycle length extension. The analysis was performed using the computer codes LOFTRAN, ANC and THINC3.

For the first time, a complete set of Krško specific ATWS transients has been analyzed.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

Yes, setpoints were changed due to new steam generators and higher power.

Examples of changed limiting conditions for operation (LCO):

- *Reactor core safety limits*
- *RTS Instrumentation Trip Setpoints (larger setpoint for power range neutron flux (positive and negative) rate, smaller setpoint for low-low steam generator level Rx trip)*
- *Shutdown Margin (reduced to compensate for the power increase and cycle length extension)*
- *DNB parameters (higher reactor coolant system (RCS) average temperature and flow, low pressurizer pressure)*
- *Engineered Safety Features trip setpoints (steam line pressure – low setpoint is higher, SG water level – high-high trip is higher and low-low trip is lower)*
- *Boron concentration for reactor coolant system, accumulators, boron injection tank and refuelling water storage tank (increased for 18-month cycle))*

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason

The impacts are moderate. The safety analyses provide inputs for updating the operating procedures. Examples are:

- *emergency operating procedures (response to high steam generator level and low steam generator level – the reason are new steam generators requiring modification of setpoints)*

- abnormal operating procedures (e.g. leakage of steam generator U-tubes due to new steam generators)
- operating procedures (e.g. power operation due to new operating window). The upper limit accounts for the maximum Thot fuel corrosion limit. The lower limit was based on the extrapolated turbine limit.)

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

N.A.

It should be noted that all critical safety analyses were repeated during Krško NPP modernization.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

N.A.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

Benefits could be:

a) the exchange of experience on modern T-H tools/methodologies for safety assessment by a forum/workshop

b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types) by bilateral talks

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

It should be noted that the analyses have been conducted as far as feasible using the same methodologies as used for the initial safety assessment of the plant (example Appendix K LOCA calculations). This means that even using existing modern tools would improve and optimize the operational safety.

For example, for large-break LOCA it was necessary to reduce the average power of the hot assembly to a lower value in order to be below acceptance criteria. The development of 3D system codes with built-in methods for uncertainty evaluation could significantly support optimization of operational safety.

In the case of heatup events the consideration of the ANS79 + 2σ decay heat curve helped to obtain acceptable results. The methodologies and codes using the latest ANS-2005 standard would help in this respect.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

No.

Our organization was involved in reference calculations for Krško full scope simulator verification, which was also part of modernization in 2000. Analyzed were the following transients and accidents: anticipated transients without scram, loss of coolant accident, loss of main feedwater, and steam generator tube rupture.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

No.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

N.A. The findings are incorporated indirectly through world practise.

Hungary - KFKI

Name of the Organization:

KFKI Atomic Energy Research Institute

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TSO

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Country:

Hungary

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification

Paks NPP:

- *power uprating*
- *introduction of new type of fuel*

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

There was no direct need to switch to new T-H and N-K tools/ methodologies in order to remain within the safety margins. However, new tools have been introduced in order to eliminate conservatisms/uncertainties related to earlier analyses. (E.g. the coupled ATHLET-KIKO3D was used for steam line break analysis instead of HEXTRAN and the same coupled tool was deployed in RIA analyses instead of SMATRA.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

Introduction of new fuel type implied changes in safety limits, e.g. the formerly constant linear power limit was replaced by one decreasing with burn-up.

Certain limits on operating parameters were changed for power uprating: e.g. hydroaccumulator initial pressure was decreased, while water volume was increased.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason.

The whole set of EOPs had to be reviewed prior to power uprating to check, whether the steps foreseen provide efficient response at increased power conditions. In certain cases computer code analyses have been performed (e.g. for upper head coolability during natural circulation cool-down or to check entry points for secondary/primary bleed). Finally, no changes to the EOPs were needed in these cases.

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

Yes. The list can be made available.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

Development of severe accident management procedures is under way.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?

Yes, especially on item b)

If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

Information on OLC/OIP should be compiled beforehand and then discussed by interested parties in a workshop.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Quantification of (especially) model uncertainty distributions for use in best-estimate safety analysis.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

Yes, with the reason to define entry points and action modes for different operator actions.

Examples:

- *upper head coolability during natural circulation cool-down*
- *to define entry points for secondary/primary bleed*
- *assessment of criteria to stop HP injection*
- *effectiveness of different means of primary bleed*
- *post-LOCA cool-down*

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions (e.g. OECD SETH, PKL etc). If yes, which ones and for what reason?

Yes: SETH-1, PKL and ROSA

To follow international development and to obtain data for code validation on specific conditions.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

SETH, PKL and ROSA findings are not directly applied. However, a recent OECD GAMA action on evaluation of core exit temperature measurements could directly impact the selection of entry points to different procedures and transition points between EOP and SAMP.

Lithuania - LEI

Name of the Organization:

LEI Lithuanian Energy Institute

Entity (Electrical Utility, Architect Engineer, TSO)

TSO

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Country:

Lithuania

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification.

Ignalina NPP with RBMK-1500 reactors.

Modifications: Development and implementation of Second Independent Shutdown System; development and implementation of new algorithms for emergency reactor shutdown and core cooling systems activation; Development of plant Emergency Operating Procedures and Severe Accidents Management Guidelines; Transportation and reuse of fuel from shutdown unit 1 to the operating unit 2.

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

Initial safety assessment was performed by RBMK designers in Russia with their own codes and tools. Later the original T-H (ATHLET, RELAP5, CONTAIN, COCOSYS) and N-K tools (QUABOX/CUBBOX-HYCA), which are developed for vessel type reactors, were adopted and used for safety assessment of RBMK-1500 reactors. Special subroutine for QUABOX/CUBBOX-HYCA code was used to model RBMK core.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

After the Chernobyl accident the maximal allowed power of Ignalina NPP was decreased from 4800 MW (design) to 4200 MW. Other latest modifications do not affect the Operational Limits and Conditions.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason?

Initially there were only event-based procedures available and used at Ignalina NPP. Later new OIP – Emergency Operating Procedures and Severe Accidents Management Guidelines were developed (EOP are implemented, SAMG are under implementation). The other modifications imply the small changes in previous OIP (for example: the availability of second independent shutdown system and new reactor shutdown and emergency core cooling algorithms were reflected in new versions of Technological Regulation).

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

The specific set of T-H analyses was defined and "check-list" was presented. This list is included in the project requirements for DBA analysis of RBMK-1500.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

Because Ignalina NPP Unit 1 was shutdown for decommissioning at the end of 2004 and Unit 2 is to be operated until the end of 2009, the new safety case for Ignalina NPP condition with both shutdown units should be developed. This is new, never analyzed, state (with both shutdown for decommissioning units) for Ignalina NPP.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

The exchange of experience on both: use of modern T-H tools and methodologies for safety assessment and qualitative comparison on of OLC and OIP with the similar plants types, are very useful. The best way to perform the exchange of experience is the forums and workshops, where the information is exchanged between different countries.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

The analysis of severe accident phenomena in RBMK-type plants.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

The detailed simulations of thermal-hydraulic transients in case of ATWS in RBMK were performed to define the set points for emergency reactor shutdown actuation (for development of Second Independent Shutdown System); Analysis of coolant flow stagnation in parallel fuel channels (for development of new algorithms for emergency reactor shutdown and core cooling systems activation); analysis of station blackout case and depressurization of primary circuit (for development of plant Emergency Operating Procedures and Severe Accidents Management Guidelines).

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

LEI is participating in network of excellence SARNET, in PHEBUS research program – to receive the experience, necessary for analysis of phenomena during severe accidents; some year ago participated in International Standard Program ISP-47 – to receive the knowledge in modeling of phenomena in Containment; etc.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

The findings of these international projects could not be directly incorporated into OLC and OIP due to different design of analyzed facilities, but the acquired knowledge and experience were used during development of plant Emergency Operating Procedures and Severe Accidents Management Guidelines.

Czech Republic – NRI

Name of the Organization:

Nuclear Research Institute Rez plc

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Research Institute, TSO

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Country:

Czech Republic

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification.

VVER 440/213I&C modification, power uprate, new fuel elements
VVER 1000/320 new fuel elements

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit):*

Our safety analyses methods are based on deterministic approach – we are use BE computer codes with conservative initial and boundary conditions. Nevertheless last 5 year we start with BE approach with uncertainties of input data and models and correlations in the code. We are use the GRS IRSN methods based on Wilks formula. For analyses we use next computer codes:

RELAP 5
Relap5-3D
ATHLET
MELCOR
COCOSYS
FLUENT
COBRA
DYN 3D

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

Power up rate leads to the change of the RTS signals, e.g. output temperature trip. Change of I&C leads to confirmation of TRIP diversity.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason

Yes, nevertheless not to much.

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

The reports are not open. Only some papers from seminars or conference.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

No

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

Exchange of experience will be very useful, mainly in the area of use of #D T/H computer codes, CFD codes and Best estimate methods.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

CFD codes, coupling CFD with system and 3D neutron kinetics code, coupling sub channel 3D neutron kinetics and system codes.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

e.g. LOCA

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions (e.g. OECD SETH, PKL etc). If yes, which ones and for what reason?

OECD-PSB, OECD-SETH, SETHII, OECD-PKLI and II, OECD/ROSA
Main aim is code validation.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

Use of validated codes, validated model and select important phenomena of OLC and OIP

Sweden – SKI

Name of the Organization:

Swedish Radiation Safety Authority

Entity (Electrical Utility, Architect Engineer, TSO)

Authority (There is no TSO in Sweden)

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Country:

Sweden

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification.

Ringhals 2 and 3. (3-loop Westinghouse PWR) Steam Generator Change
Ringhals 2. Power uprate
Ringhals 1, Barsebäck 1 and 2 (now closed), Oskarshamn 2 (BWR of ABB design with external pumps) Power Uprate
Forsmark 1-3, Oskarshamn 3 (BWR of ABB design with internal pumps) Power Uprate
Introduction of new types of fuel for all the BWRs
Exchange of insulation and strainers to avoid clogging for PWR and BWR with external pumps.
Modernization efforts on for instance I&C of older plants.

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

There have been no needs to change the T-H and N-K tools so far to maintain safety margins.

The power uprates could be carried because of hardware features such as of large initial safety margins, larger steam generators and fuel with much better performance. The change to more modern methods has been encouraged by the authority because they give better assessments of the actual safety and safety significance of incidents. Another important factor is that the best estimate licensing tools can be validated using plant transients.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

In some cases plant modification impacted the OLC. For BWRs with internal pumps, modifications have been introduced to eliminate the loss of external power as a limiting transient by introducing energy storages.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason.

For instance, power uprates may be a reason for change of the OIPs. Power uprates may affect procedures to control stability. Improved understanding of local boron dilution transients has caused the instructions to be changed. Modern fuel for BWR may have more positive moderator temperature coefficients which need to be handled by new instructions.

In most cases, though, it is intended to avoid impact on the instructions by plant modifications. For example, to avoid problems with strainer clogging, significant plant modifications were actually introduced in order to reduce modification needed on the instructions.

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

There is no specific check list. First the overall list of all transients is reviewed by the licensee who determines which will affected by the plant modification. Analyses will be made of the transients that will be affected. The list of affected transients is also reviewed by the authority. The full list of transients is based on the selection done by the NRC and some Swedish requirements. It is normally required that low power and shut down transients are addressed. For an operating plant, transients that have occurred in the plant will also be included. It must be proven that the final plant state is safe and that the fuel can safely be removed from the reactor.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

There should in principle be full coverage for all operating modes. The instructions and operating limits are continuously being reviewed and worked on based on operating experience and results of analyses and research.

Question 7: Would you see any benefit from:

a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?

Both these items are very beneficial, and both are also being used.

If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

Bilateral talks are very useful. In particular, since one of our reactor types is built in only two countries, bilateral talks between these two countries are especially important. For more general aspects of information exchange the second option is very useful.

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Information of T-H behaviour during shut-down or low power transients needs to be expanded. Such transients are often characterized by accumulation of water volumes at different temperatures. If such volumes inadvertently are moved around in the system, significant temperature gradients on vessel and internals may occur. Incidents have occurred that have exceeded technical specification and therefore have needed special safety considerations. Local boron dilution and pressurized thermal shock fall into the same thermal hydraulic category of incidents.

Information about circulation in a BWR-vessel when the pumps are stopped vessel is lacking. Validation of methods that address such behaviour is an important task. Such information is needed for assessment of internal circulation and heat removal from the vessel when the external grid is lost.

Validation of detailed coupled T-H and N-K methods using plant data is important. Such information should be used for addressing importance of asymmetries in the vessel. One large remaining issue is stability and the ability to correctly predict phenomena important for stability still needs improvements.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

It is a goal for the thermal hydraulic simulations to provide improvement of the operation procedures. Simulations of transfer to in-house load operation which have occurred are made to optimize plant response. Hypothetical local boron dilution has been simulated and also been addressed experimentally. Current understanding is that reactivity excursions because of local boron dilution are not wanted and this is reflected in the operating procedures. Current work is focused on feed water transients that may cause instabilities with the objective to improve the operation procedures for better plant response.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

The organization participates in OECD/SETH-2, OECD/PKL-2 and the OECD/LSTF. The main reason is to have validated safety codes that can be used on Swedish containment and primary systems. One objective is also, in particular for the systems which are well scaled, to demonstrate safety performance of systems under accident conditions.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

Important findings are communicated to and discussed with the utilities. The utilities update instructions regularly and the international input, operational experience and own analyses have importance for this work. An example that is worked on for the moment is the function of Core Exit Thermocouples under accident conditions. Also the strategy to cool the core under small LOCA is also under review.

Belgium – TRACTEBEL

Name of the Organization:

Tractebel Engineering

Entity (Electrical Utility, Architect Engineer, TSO)

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Country:

Belgium

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification

Doel 1: Steam generator replacement and power uprate (10%) in 2009
Doel 2: Steam generator replacement and power uprate (10%) in 2004
Doel 3: Steam generator replacement and power uprate (10%) in 1993
Doel 4: Steam generator replacement in 1997
Tihange 1: Steam generator replacement and power uprate (8%) in 1995
Tihange 2: Steam generator replacement and power uprate (10%) in 2001
Tihange 3: Steam generator replacement in 1998

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

In order to perform the safety studies, new codes and new methodologies have been used. In thermal - hydraulics a best estimate methodology (presented in several conferences) has been developed using the best estimate code Relap 5. For neutronics calculations the 3D Panther has been used. For the LBLOCA, methodologies and tools developed by vendors have been applied (COBRA-TRAC, CATHARE GB, …)

The Belgian Regulatory Body required first a qualification process of the codes, the models (imput deck) and the methods to address the uncertainties. For each safety analyses to be performed, a specific methodology report has been developed and approved by the Authority

A specific methodology has been developed in order to perform couples TH and NK calculations. Therefore a coupling of the Panther and the Relap codes has been performed.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

For all the above mentioned projects the original OLC has been modified. Typical modifications are the average temperature, water level in the pressuriser, pressure in the steam generators, water level in the steam generators, primary mass flow rates, fuel enrichment.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason

Some procedures have been modified taking into account the changes in operating conditions. For some units, setpoints of safety valves and/or power operated relief valves have been adapted. On plant one has to add an additional safety valve on the steam generator.

Additional modifications were also requested to guarantee the cooldown of the plant (auxiliary feedwater capacity).

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

For each of the above mentioned projects, a study program has been set-up and approved by the Authority. This program defines which studies of the FSAR have to analysed by means of a code calculation. For some transients, justifications of non reanalysis have been submitted to the Authority. This list is not public.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

Currently, best estimate TH calculations are performed in the framework of the Periodic Safety Review in order to support the PSA activity in normal condition and in shutdown conditions.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

a) There exist currently enough fora (cf OCDE NEA, user groups, conferences) where the experts can exchange on their experience

b) Since the OLC and OIP are very specific to the plant and the equipments, I do not see benefit from performing such extensive comparison. On specific aspects Utilities/Architect Engineers exchange already their experience

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Not answered.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

Not answered.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

Not answered.

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

Not answered.

Austria - University of Vienna

Name of the Organization:

University of Vienna, Institute of Risk Research

Entity (Electrical Utility, Architect Engineer, TSO)

University, (partly acting as TSO for the Austrian Government)

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Country:

Austria

Question 1: Please list which of nuclear power plant/s of your country has/have been subject to major modifications and the type of modification

Austria decided against the nuclear option following a referendum in 1978. The only NPP of Austria, the Gemeinschaftskraftwerk Tullnerfeld GmbH owned GKT Zwentendorf 700 MWe unit, has been constructed and hot-tested, but never taken into operation.

Question 2: Were the original T-H and N-K tools used for the initial safety assessment of the plant used again or was there a need to switch to new T-H and N-K tools/ methodologies to remain within the safety margins? If new tools were needed, please specify the name of the new tool/methodologies and the reason to use it *(e.g.: new version of code X necessary to remain within the LOCA limit)*

There was no need to repeat the original safety assessment, since the plant has not been taken into operation. Tools used at that time were e.g. Relap4 mod 3, mod 6, COBRA, DRUFAN and KWU proprietary TH and accident simulation codes. The German Kerkraftwerk Brunsbüttel KBB served as a reference plant.

Question 3: Did the major plant modification/s imply any major effects on the original (or previous) OLC. If it was the case, list one or more examples of the major impacted OLC with their trend/s.

Question not applicable – OLC was not awarded; no major plant modifications have taken place.

Question 4: Did the major plant modification/s imply any major effects on the original (or previous) OIP. If it was the case, list one or more examples of OIPs and which section was modified, and the reason

Question is not applicable – no major plant modifications have taken place.

Question 5: Have you developed a specific well defined set of T-H analyses (or developed a "check-list") that has to be regularly performed in presence of a specific major plant modification of type X? If yes, is this list publicly available?

Question is not applicable – no major plant modifications have taken place.

Question 6: Is there a specific operating mode of the plant (with its OLC and OIP) that currently needs to be improved or developed in your plants? If it is the case, then what is the reason? (*e.g. Absence of shutdown accident procedures)*

Question is not applicable – no major plant modifications have taken place.

Question 7: Would you see any benefit from:
a) the exchange of experience on modern T-H tools/methodologies for safety assessment
b) performing a qualitative comparison on of OLC and OIP of your country's nuclear power plants with those of other countries (having similar plant types)?
If it is the case then how would you like this comparison exercise to be performed (*e.g.*
a) by bilateral talks between counterparts RB of countries having the same plants,
b) by a forum/workshop where you would present your countries methodologies and obtain equivalent information of the other countries)

a.) would be beneficial, b.) is not applicable for the particular situation of Austria

Question 8: Can you identify any specific T-H and N-K research needs that could support improvement and optimization of operational safety?

Improvement/Introduction/Consideration of three dimensional effects in TH-System codes, like three dimensional modelling of downcomer and core region, coupling of TH-Sys codes with N-K codes for asymmetric transients like trip of a single MCP, variations of MRP trips or MSLB, and extensive testing and validation of the improvements to the codes.

Question 9: Have you performed detailed simulations of thermal-hydraulic transients to provide refined tuning of the operation procedures? If yes what kind of transients were performed and for what reason?

Multiple failure analysis was conducted to determine either mitigation procedures or determine by detailed simulation of the resulting sequences the compliance with safety requirements and in particular return to safe shutdown conditions.

Question 10: Is your organization participating directly in current research and international studies which are related operational procedures for normal, abnormal and emergency conditions *(e.g. OECD SETH, PKL etc)*. If yes, which ones and for what reason?

OECD/NEA GAMA Effectiveness of core exit temperature measurement (CET) in accident management (AM), to follow international developments in the field

Question 11: How does your country incorporate the findings of these international projects in the OLC and OIP of your own country's nuclear power plants?

Not applicable

European Commission

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Abstract

This Report summarizes the results of the studies performed by the Joint Research Centre / Institute for Energy (JRC/IE) on a specific task dedicated to Thermal-hydraulics within the SONIS (Safety Of Nuclear InstallationS) 2008 program.

The aim of task 4 of the SONIS programme is to analyse European practice in verification and optimization of plant operational procedures for normal, abnormal and emergency conditions. More specifically task 4.1 analyses the effect of using new Best Estimate plus Uncertainty Methods (BEPU) in the re-licensing processes on plant operational procedures directly affecting the Thermal-Hydraulic (T-H) behaviour of the nuclear facilities.

Current trends in the industry to increase power production challenge the initial safety design limits of the plant which were performed generally using conservative tools and hypothesis. Advance numerical tools and methods allow demonstrating that safety margins are still respected. These tools are modern fully validated thermal-hydraulic codes, coupled thermal-hydraulic / neutron-kinetic (N-K) codes and methodologies that use realistic hypotheses rather than conservative ones and estimate also the uncertainty

Their effect on operational procedures for normal and emergency conditions and for Operating Limits and Conditions is investigated by asking directly the stakeholders of the European Union.

A questionnaire was sent to several stakeholders in the Nuclear Safety domain in the European Union and information was gathered on the new T-H tools for the re-licensing processes (for power uprates, SG replacements etc), their effect on the Operational Limits and Conditions (OLC) and on Operating Instructions and Procedures (OIP) of the Nuclear Power Plants (NPPs), the need of performing specific investigations in operational modes and of exchanging of information on new T-H tools/methodologies.

It was seen that almost all the interviewed countries confirmed the uprate of their Nuclear Power Plants (NPPs) with a maximum value of 10%. The major modifications were generally associated with the replacement of Steam Generators (SG) and introduction of new type of fuel.

All the participants also confirmed the impact on the original OLC (for example, changes were introduced in setpoints of reactor protection) but there no major effects on the original OIP were noticed (except some optimization of procedures).

In general, no additional operating or accidental modes were identified for development or improvement although two participants mentioned the current development of SAMG (Severe Accident Management Guidelines)

The full set of T-H safety analyses were done only at those NPPs where major modifications were performed and detailed simulations were executed manly for evaluation of set-points and modifications of accident management procedures.

The participants agreed that an exchange of experience by a forum/workshop with a qualitative comparison on of OLC and OIP of plants of different countries would be beneficial. They also agreed on different kinds of research activities mainly focussed on use of three dimensional and coupled T-H and N-K codes.

Finally, a large number of organizations participate directly in current research and international projects but the findings aren’t systematically applied to their plants.

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