Multinational Design Evaluation Programme (MDEP)

Technical Report

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VVER WORKING GROUP

Technical report on regulatory requirements and practices for uncertainty evaluations of deterministic safety analysis

Regulators involved in the VVER working group discussions:	HAEA, NDK, NNSA and NSC as its TSO, SEC NRS as TSO of Rostechnadzor
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Compatible with existing IAEA related documents:	Yes

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I. Introduction

Technical experts subgroup on accident and transient analysis (TESG A&T) was established in September, 2017 to foster the communication among the members on regulatory approaches for review and acceptance of A&T analysis in frame of the licensing procedure and to promote understanding of each country's regulatory decisions and basis for the decisions.

Safety assessment is a systematic procedure carried out in order to evaluate how the relevant safety requirements are met by the design of the plant. The requirements to evaluate the uncertainties of safety analysis are established in the regulatory documents of almost all countries using nuclear energy. Moreover, over the past three decades, ample experience has been amassed worldwide in the application of uncertainty estimation techniques for calculations conducted to demonstrate various aspects of nuclear power plant safety. However, the use of uncertainty evaluation techniques for NPP safety analyses is usually discussed in scientific publications, while actual application of these methods in NPP safety cases is still limited, despite the requirements in nuclear safety regulations, including IAEA safety standards, e.g. requirement 17 of GSR Part 4 (rev. 1) 'Safety Assessment for Facilities and Activities'. This may be caused by the lack of national regulatory documents guiding safety documentation authors on how they should fulfil the requirement obliging them to estimate errors and uncertainties in the results of safety analysis calculations.

Considering this, the TESG A&T has used the findings of an overview of national best practices in evaluation of safety analysis uncertainties to develop this report.

II. Regulatory requirements and guidelines

Paragraph 4.60 in the IAEA GSR Part 4 (rev. 1) 'Safety Assessment for Facilities and Activities' sets a general requirement: 'the uncertainties in safety analysis results shall be taken into account'. Guidance on the assessment of uncertainties in the results of simulation and modelling performed to support safety analyses for nuclear plants can be found in the IAEA SSG-2 (Rev.1) or IAEA Report 'Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation', Safety Reports Series No. 52, Vienna (2008). Many practical examples of uncertainty analysis of certain safety related issues can be found in publications and reports prepared by Nuclear science committee, Committee on the Safety of Nuclear Installations, Committee on nuclear regulatory activities of the NEA.

Russian regulations

Russian federal nuclear safety regulations establish a requirement to support the nuclear facility safety analyses with assessment of errors and uncertainties of the results. For nuclear power plants, this requirement is set in paragraph 1.2.9 of the 'Basic Safety Provisions for Nuclear Power Plants' (NP-001-15) [1], according to which '*safety analysis shall be backed up with an assessment of errors and uncertainties in its results*'.

Based on the results of the review of safety analysis of VVER-1200, SEC NRS initiated the development of the guidelines on uncertainty analysis of simulation and modelling. In 2020 the Russian Nuclear Safety Guide 'Guidance on Estimation of Errors and Uncertainties in the Results of Computational Safety Analyses for Nuclear Plants' (RB-166-20) was approved by Rostechnadzor. Detailed requirements to the verification and validation of computer codes are established in Ref. [3].

Finnish regulations

In Finnish regulations there is no strict requirement mandating the use of uncertainty approach (BEPU) but assumptions and their uncertainty shall be justified. In the YVL guides there is reference made to ref. 4. [Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation, Safety Reports Series No. 52, IAEA, Vienna (2008)].

Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant STUK Y/1/2018 Chapter 2, Section 3, paragraph 4 prescribes that: "*The analytical methods employed to demonstrate compliance with the safety requirements shall be reliable, verified and validated for the purpose. The analyses shall demonstrate the conformity with the safety requirements with high certainty. Any uncertainty in the results shall be considered when assessing the meeting of the safety requirements*". The related guidelines are:

- *YVL B.3 404 A description of the models and calculation methods employed in the analyses shall be presented. The models shall be described to a level of precision that allows for verifying the correctness of the model in relation to the plant design as well as assessing the applicability of the selected modelling solutions. The information presented in the description shall include an analysis model that describes the facility or a part thereof (such as the nodal distribution used in the model), a justification for the model parameters selected and the plant data used in the analyses or a reference to a source from where the plant data is available.*
- *YVL B.3 405 The validation of the physical models and computer code used for the analyses shall be substantiated by comparing their calculation results to separae effects tests or tests carried out on entire systems, or to disturbances that have occurred at nuclear power plants. Comparison with models that have already been validated may also be utilised.*
- YVL B.3 408, *The accepted methods to be used in the plant behaviour analyses are either the conservative analysis method supplemented with sensitivity studies or the best estimate method supplemented with uncertainty analysis.*
- YVL B.3 410, *Utilisation of the best estimate method shall be supplemented with an uncertainty analysis that is justifiable by statistical methods. Examples of such methods are given in* [Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation. IAEA Safety Reports Series No. 52. IAEA, Vienna 2008.].
- YVL B.3 411 *The initial conditions of the analyses and the chosen parameters used for the analyses shall be justified.*

- YVL B.3 411a. If the choice that is the least beneficial in terms of the acceptability of the end result is not unambiguous, analysis results covering the parameter's entire range of variation shall be presented.
- YVL B.3 602. The acceptance criteria set forth in chapters 6.2 and 6.3 are presented for the conservative analysis method. In applying a best estimate method with uncertainty analysis, the result is acceptable if there is a 95% probability with 95% confidence that the examined paraeter will not exceed the acceptance limit set for the conservative analysis method.

There is thus in principle 2 approvable approaches to the issue. Firstly, a conservative analysis supplemented with sensitivity studies. Another approach that is allowed in Finnish legislation is best estimate plus uncertainty. The IAEA SSG-2 (rev. 1), table 1 presents 4 approaches, of which 2 and 3 are to be used.

Hungarian regulations

In Hungary Nuclear Safety Code (NSC) requirement 3a.2.3.0400 prescribes that "*sensitivity analyses shall be performed to evaluate the uncertainty of assumptions, the data used and the calculation methods. Where the results of the analysis prove to be sensitive to the assumptions of the model, further analyses shall be carried out by using methods and procedures independent of the previously used methods and procedures*".

Other related requirements:

NSC 3a.2.2.3700. *When defining the design basis, reasonably conservative assumptions shall be applied to compensate for uncertainties*.

NSC 3a.2.2.6100. *During the analysis of DEC1 operating conditions, in order to compensate for uncertainties, either reasonably conservative assumptions shall be applied or the best estimate method and data shall be used, supplemented with the necessary uncertainty and sensitivity analyses*.

NSC 3a.2.2.7300. *In order to minimise uncertainties and ensure robustness of the safety of the nuclear power plant unit, demonstration of physical impossibility shall be preferred to demonstration of low probability when justifying practical exclusion*.

NSC 3a.2.2.6700. For the analyses of DEC events: [...] the reproducibility of the analysis shall be ensured also in cases where engineering judgement was taken into account during the analysis, and all uncertainties relating to the analysis and their effects shall be taken into account;

NSC 3a.2.2.4000. Among the assumed initiating events, all events listed below shall be considered:

- those related to the site of the nuclear power plant and its surroundings and are of natural origin,
- those that are the consequences of intentional human actions not purposefully directed against the nuclear power plant or of inadvertent on- or off-site human actions;

- technological failures resulting from the operation of the nuclear power plant or the failure of its systems, structures and components, or
- those resulting from human error.

Sensitivity analyses shall be performed to evaluate the uncertainty of assumptions, the data used and the calculation methods. Where the results of the analysis prove to be sensitive to the assumptions of the model, further analyses shall be carried out by using methods and procedures independent of the previously used methods and procedures.

Related Hungarian regulatory guides can be found at N3a.32. "Deterministic safety assessment for new NPPs". Between the sources of the uncertainties it is necessary to take into consideration the following:

- Uncertainties originating from the inaccuracy of the physical model,
- Initial and boundary conditions,
- Uncertainties originating from geometrical modelling,
- Approximate nature of the numerical solution,
- Effect of the hardware and compiler,
- User effect (nodalisation, time steps etc.),
- Scale effect.

Systematic and stochastic errors of the input parameters must be taken into account in the validation analyzes. The errors originated from the equations and models in the calculation model must be indicated in the calculation results. In hypothesis testing the confidence level must be presented. When presenting numeric values, the numeric value must be given with an error so that the last two decimal places of the valuable digits equal the two highest decimal places of the error. It is sufficient to report the error to two decimal places. Stochastic errors should be addressed in the analyses.

If during the validation the results of the code were compared with the results of experiments that are relevant to the unit, system, etc. examined in the safety analysis. the same nodalisation should be applied as during validation. In other cases, the adequacy of the nodalisation may be demonstrated by demonstrating that, in one or more relevant cases, the more detailed nodalisation does not provide a significantly different result from the analysis with that nodalisation in terms of the fulfilment of the acceptance criteria.

Uncertainties in the results of reactor physics calculations should be available and documented. The sources of computational uncertainties are the nuclear cross sections, the uncertainties of the geometric and composition data and the model uncertainties.

In reactor physics analyses valid parameter ranges and the method of parameterisation should be documented.

The results of both realistic conservative and best estimate analyses may be sensitive to the selection of certain input parameters. Sensitivity analyses should be used to screen out parameters whose choice significantly affects the results of the analyzes, in particular the fulfilment of the acceptance criteria. Both sensitivity analyses and uncertainty analyses should support the reliability of safety analyzes.

Chinese regulations

Chinese requirement "Code on the Safety of Nuclear Power Plant Design" (HAF 102-2016) is the requirement on the construction of critical structures, systems and components for nuclear power plant safety. And the requirements in HAF 102 must be met by procedures and organisational processes.

According to HAF 102-2016, the uncertainty analysis is required as shown in Section 5.8.1.4 "Uncertainties should be considered adequately in NPP design. Adequate margins should be demonstrated to avoid cliff edge effects and the early radioactive release or the large radioactive release. "

Chinese guidance "Deterministic Safety Analysis for Nuclear Power Plant" (2021) is released by NNSA in last year, to providing guidance for deterministic safety analysis for new designed NPP.

In this guidance, the scope and the method for uncertainty analysis is defined clearly. Section 2.3 " uncertainty analysis in deterministic safety analysis" shows that:

Several methods for performing uncertainty analysis have been published. They include:

a) Use of a combination of expert judgement, statistical techniques and sensitivity calculations;

b) Use of data from scaled experiments;

c) Use of bounding scenario calculations.

In Section 2.4 "the methods of deterministic safety analysis" shows that the method of best estimate plus uncertainty' approach is accepted for some design basis accidents and for conservative analyses of anticipated operational occurrences for licence application in China.

In Section 5.3 "best estimate deterministic safety analysis with quantification of uncertainties for anticipated operational occurrences and design basis accidents", the models, initial and boundary conditions, other input parameters and method for analysis

Turkish regulations

In Turkish legislation, there is no direct provision for assessment of errors and uncertainties of the safety analysis results. On the other hand, to be able to apply most recent requirements in the area of nuclear safety NDK developed a licensing approach in parallel with the methods suggested in INSAG-26 document utilising applicable IAEA, vendor country and third-party requirements to patch the gaps in current Turkish regulations. Therefore, IAEA requirements and Russian Federation legislation are taken as a basis during the evaluation of the uncertainties and errors in the safety analysis carried out for the Akkuyu NPP project.

Similarities and differences of national requirements and guidelines

In every TESG AT member countries the regulatory body experts are expect that uncertainties in the safety demonstration calculations are addressed. According to national regulations sensitivity studies should be a part of the safety demonstration, however the usage of uncertainty analysis methodologies are not directly required in Finland and Turkey. However, the recommendations to perform the uncertainty analysis are applied in regulatory practices of these countries through references to IAEA safety standards. Hungarian, Chinese and Russian regulations are require the uncertainty analysis but the certain methods for that should be chosen and justified by the Licensee. Most of the national regulatory bodies expect to see the usage of uncertainty analysis methodologies at least for transients and design basis accidents analysis, the Russian regulations are also required to use such methods for design extension conditions, including severe accidents. Corresponding recommendations are issued in Russian regulatory guide RB-166-20.

III. Uncertainty analysis practices

Despite the differences in national regulations including the different level of details of regulatory requirements and guidelines the TESG AT experts found the best practices in regulatory approach for uncertainty evaluations. These practices are summarised below, and recommended for consideration to safety analysis specialists both in utilities and regulatory bodies and their technical expert organisations.

Sources of uncertainty to be analyzed

It is encouraging that safety analysis of nuclear plants should have the results of analysis of uncertainties pertaining to the:

- approximations and simplifications adopted in calculation model development (uncertainties originating from the inaccuracy of the calculation model and its validation), including the uncertainties associated with the numerical solution method selected for the code's equation system;
- input data on geometry, initial state, scenario, boundary conditions adopted in the computational model, as well as nodalisation (control volumes, finite elements), and integration step chosen by the user for the system of equations used in the computer code;
- performance of systems controlling operating parameters of the reactor facility;
- properties of substances and materials used in the calculations; and
- closing relation coefficients implemented in the code.

The uncertainties resulting from the simplifications and assumptions

The analysis of the uncertainties resulting from the simplifications and assumptions adopted in computer code should be captured in order to demonstrate the correctness of the code's physical mathematical model, i.e. adequacy of equations, hypotheses, assumptions, nuclear data and substance/material properties. Domain of validity should also be assessed and presented.

Adopted nodalisation (control volumes, finite elements) and integration step selected for the equation system should be justified. Code validation and model description is usually not directly presented in the SAR. No strict requirement exists to this matter, however code validation and model description are required and usually presented in in the code verification and validation report.

A description of the models and calculation methods employed in the analyses shall be presented. The models shall be described to a level of precision that allows for verifying the correctness of the model in relation to the plant design as well as assessing the applicability of the selected modelling solutions.

When presenting numeric values, the numeric value could be given with an error so that the last two decimal places of the valuable digits equal the two highest decimal places of the error. It is sufficient to report the error to two decimal places.

The uncertainties associated with random events

The uncertainties associated with random events, such as equipment failures and human errors, should be taken into account in the development of the list of design basis initiating events and the list of beyond-design basis accidents for the nuclear plant based on PRA results.

Random events, such as equipment failures are not considered within uncertainty analysis assumptions, however they are to be postulated deterministically. Capacity of safety systems could be included into the variables that are chosen for uncertainty evaluation. This is to prove that cliff edge phenomena are avoided.

The uncertainties related to the performance of control systems

The uncertainties related to the initial state of the analysis, performance of control systems maintaining operational parameters of the reactor facility, the uncertainties in geometrical dimensions of reactor facility components and equipment, and the uncertainties in the substance / material properties data and in closing relation coefficients, recommended to be considered in a deterministic safety analysis for a nuclear plant.

Methodologies to assess the uncertainties

The effect of the uncertainties could be evaluated using different methods. No regulation requiring or guiding how mentioned uncertainties should be assessed. Methods to be used may be proposed for assessment. The choice of the method to evaluate the above uncertainties should be justified in the SAR. Detailed description of different methods could be found in Ref. [2, 4].

The NPP SAR should justify the choice of parameters affecting results of these calculations, as well as statistical characteristics of the parameters. The choice of parameters affecting safety calculation results should be justified by sensitivity analysis and/or by way of expert assessment.

In case of using statistical methods to evaluate safety calculation uncertainty, it is recommended to:

- simultaneously vary values of parameters affecting safety calculation results by using a pseudo-random number generator;
- run the required number of code calculations with varying parameter values, and
- perform a statistical review of the results of these calculations.

If a statistical review is undertaken to justify the tolerance interval boundaries for parameters critical for the nuclear plant safety, the number of code runs should be such as to enable the justification of these boundaries with at least 0.95 probability at 0.95 confidence level.

IV. Conclusions

The report describes national regulatory requirements and guidelines for Uncertainty Evaluations of Deterministic Safety Analysis of nuclear power plants. Similarities and differences of national regulations are discussed. Based on the licensing experience including the safety review of VVER-1200, the practices in regulatory approach for uncertainty evaluations are summarised and offered for consideration during safety justification of new designs.

V. References

1. Basic Safety Provisions for Nuclear Plants. NP-001-15. Approved by Rostechnadzor on 17.12.2015, valid since 16.02.2016. – M. SEC NRS, 2016.
2. Guidance on Estimation of Errors and Uncertainties in the Results of Computational Safety Analyses for Nuclear Plants. RB-166-20. Approved by the Federal Environmental, Industrial and Nuclear Supervision Service on July 30, 2020.
3. Safety Review Procedure for Computer Codes Used to Build Computational Models for Processes Affecting the Safety of Nuclear Facilities and/or the Safety of Types of Activities in the Field of Atomic Energy. Approved by Rostechnadzor order No. 325 dd. 10.07.2018.
4. Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation, Safety Reports Series No. 52, IAEA, Vienna (2008).