

Multinational Design Evaluation Programme (MDEP) Technical Report

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VVER WORKING GROUP

Regulatory approaches related to accidents and transients analyses

Regulators involved in the VVER working group discussions:

AERB, HAEA, NNSA (and NSC as its TSO), Rostechndzor (and SEC NRS as its TSO), STUK and NDK

Regulators which support the present report:

AERB, HAEA, NNSA (and NSC as its TSO), Rostechndzor (and SEC NRS as its TSO), STUK and NDK

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Content

I. Introduction

II. Scope and Objectives

III. Comparative summary of main findings

- 3.1. Regulatory requirements for accident and transient analyses
- 3.2. Computer codes used for modelling of accidents and transients
- 3.3. Issues concerning safety demonstration of passive systems
- 3.4. Approaches for regulatory review of safety analyses

IV. Conclusions and Recommendations

V. References

- Appendix A – Regulatory requirements for accident and transient analyses
- Appendix B – Computer codes used for modelling of accidents and transients
- Appendix C – Issues concerning safety demonstration of passive systems
- Appendix D – Approaches for regulatory review of safety analyses

I. Introduction

In May 2017, during the 8th VVER Working Group (VVERWG) meeting [1], the representative from STUK expressed the interest of the VVER's community to understand the regulatory approaches used in different countries related to transients and accidents analysis of NPPs.

It was suggested to establish the technical experts subgroup on accident and transient analysis (TESG A&T) to have further discussions between regulators to better understand differences in regulatory approaches and practices as well as to identify commendable practices in this area.

In September 2017, during the 1st TESG A&T meeting [2], it was agreed that subgroup would conduct a discussion and prepare a technical report on the following topics:

- Regulatory requirements for accident and transient analyses;
- Computer codes used for modelling of accidents and transients;
- Issues concerning safety demonstration of passive systems;
- Cooling in spent fuel pool with internal and external hazards;
- Approaches for regulatory review of safety analyses.

The topic concerning cooling in spent fuel pool was postponed and could be reconsidered in the next phase of the TESG activity.

This report has been prepared on the basis of answers given by VVERWG members to the questionnaire elaborated by HAEA, SEC NRS and STUK. The following regulatory bodies have participated in the work: HAEA (Hungary), NSC as TSO of NNSA (China), SEC NRS as TSO of Rostechnadzor (Russian Federation), STUK (Finland), NDK (Turkey).

II. Scope and Objectives

This document is intended to summarize main findings from discussions between TESG A&T participants concerning the regulatory requirements and existing practices in the field of accidents and transients analyses. At the time of preparation of this technical report not all parties possess design data of the plant. The work done to this technical report allows to share the regulatory experience, to highlight the items where approaches of regulators in member countries are similar and also to identify and discuss the differences. More detailed summaries of the topics and the given answers are presented in the appendixes of this document.

The focus of the information presented in this technical report, as well as the activity of the VVERWG TESG A&T is on all accident and transient issues, including severe accidents. The group sees it valuable not to limit the scope of discussions but acknowledges the valuable work done in the VVER TESG SA and does not go into details concerning severe accidents.

III. Comparative summary of main findings

3.1. Regulatory requirements for accident and transient analyses

The common approach of representatives of member states assumes that the questions relating to accidents and transient analysis have to be under control of national regulators and conform to domestic laws, requirements and guidelines. In all member states the design of NPP shall be justified with deterministic and probabilistic safety analysis of accidents and transients that might occur on an NPP.

According to the participants the safety analysis report is mandated to be in line with the plant design and to be kept up to date to reflect changes in the plant design. Plant safety is assessed periodically. Length of the period varies between member states.

Accident and transients are categorized in all member states. However, there are differences in terminology and in exact definitions, like beyond design basis accidents and design extension conditions.

More detailed information can be found in appendix A.

3.2. Computer codes used for modelling of accidents and transients

The confidence of computer codes that are used to carry out the modeling of nuclear facilities is crucial for the safety. The results of such calculations not only provide baseline for safety assessment but also underlie the documentation according to which nuclear facility to be operated.

Computer codes used for safety analyses should be well-established, verified and validated. The detailed requirements to the content of the documents confirming verification and validation are presented only in China and Russia.

Every TSEG A&T member country have adopted a legal background for regulatory assessment of computer codes which are used for safety analysis calculations. In Finland and Hungary regulatory assessment of computer codes is performed by regulatory bodies as part of the review of safety analysis. In Russia and China codes used for modeling are thoroughly reviewed and assessed by TSOs in accordance with special regulatory procedures. It should be ensured that the codes are evaluated and updated, as necessary, to reflect the latest knowledge.

According to regulatory requirements of TSEG A&T members validation should be aimed on identification of all the uncertainties associated with the evaluation model. Validation should be made using the evaluated experimental data, uncertainty of measurements and scaling factors should be taken into consideration. It also should be demonstrated that the uncertainties of evaluation model are taken into account in the safety analysis.

STUK recommendation on level of confidence and reliability (95/95) and Chinese recommendations on phenomena identification methodology could be seen as a good practice worth to consider in other countries.

More detailed information can be found in Appendix B.

3.3. Issues concerning safety demonstration of passive systems

Based on the responses testing is one aspect used to prove correct and expected behavior of the passive safety systems. The analysis should in general take into account proven behavior of the systems. There are not many specific requirements for passive safety systems. Passive and active safety systems are treated generally quite similarly in regulations.

Consideration of gradual degradation of passive safety systems in analysis is not defined in detailed manner in regulations in accordance with the responses. However, passive safety systems are mostly dimensioned so that minor degradation does not threaten performance of these systems.

Behavior of passive safety systems in connection to other passive or active systems is seen as an important issue to be taken care of and justified not to have not detrimental effect on plant safety.

More detailed information can be found in Appendix C.

3.4. Approaches for regulatory review of safety analyses

The reviews by regulatory bodies are essential for justified regulatory decision on NPP safety. The review needs enough competent human resources, appropriate review methods and regulatory framework.

Based on the responses every regulator and/or TSOs have experts for safety analysis, who are trained. The experts in the member countries keep themselves updated, e.g., by information exchange on international and domestic levels.

For reviewing safety analyses Russia and Finland have written guidance for review process with the different level of details. The depth of the reviews are mainly based on the requirements and regulatory guides. Hungary has in addition a checklist to assisting the review based on regulatory guides. To ensure that all requirements are met, Finland is using a requirement management system and Hungary is planning to use such soon. All member countries are preparing documentation to summarize the results of the reviews.

The use of independent analysis in the frame of licensing is part of all member country's practice. The scope of the independent analysis is selected by technical assignment from regulator to TSO in Russia, by expert judgment in Finland and in Hungary there is a regulatory guide for the scope.

More detailed information can be found in Appendix D.

IV. Conclusions and Recommendations

The current report presents the overview of the existing national regulatory practices concerning accident and transient analysis for licensing.

The information presented in the report will assist in appreciation of national practices used in the field of accidents and transients. It could be used as a basis for the further activity of the VVER TSEG on A&T for elaboration of the common position regarding the items of common interest for regulators of the MDEP VVER member countries.

V. References

1. Summary Record of the 8th Meeting of the VVER Working Group, MDEP, May 2017, Boulogne-Billancourt, France.
2. Summary Record of the 1st Meeting of the VVER Technical Expert Subgroup on Transients and Accidents Analysis, MDEP, September 2017, Boulogne-Billancourt, France.
3. Summary Record of the 2nd Meeting of the VVER Technical Expert Subgroup on Accidents and Transients Analysis, MDEP, February 2018, Boulogne-Billancourt, France.
4. Summary Record of the 3rd Meeting of the VVER Technical Expert Subgroup on Accidents and Transients Analysis, MDEP, June 2018, Helsinki, Finland
5. Summary Record of the 4th Meeting of the VVER Technical Expert Subgroup on Accidents and Transients Analysis, MDEP, December 2018, Helsinki, Finland
6. Summary Record of the 5th Meeting of the VVER Technical Expert Subgroup on Accidents and Transients Analysis, MDEP, March 2018, Boulogne-Billancourt, France

Appendix A

Task force 1 Regulatory requirements for accident and transient analyses (inc. acceptance criteria, expectation for analyses documentation, failure criteria, PRA)

Main goal of this questionnaire is to foster better understanding of different approaches to accident and transient analyses and to familiarize with different terminology used by the regulators.

Contents

Regulatory requirements for accident and transient analyses	7
Regulatory framework concerning accident and transients analysis	7
What are the most relevant legal and regulatory documents that govern the analysis of A&T? List those in hierarchical order	9
Has there been a change in the scope of requirements in the recent years. If yes, highlight the major changes. Please also state reason for changes.....	11
Are there requirements to update earlier analyses from time to time, or due to specific reasons? For example, in the context of Periodic Safety reviews.....	13
Is the concept of design basis used? If yes, how is it defined?	14
How is the scope of events/faults for analysis (postulated initiating events) defined? Is the list of such events fully prescribed? Is it up to the designer/vendor/licensee to propose a list of events based on some principles?	16
Is the scope of analyzed events based on probabilities or engineering judgment?	18
Is there a need to demonstrate diversity for each safety function?	18
Can events/faults be screened out of the DBA based on low consequences? What are the screening criteria?	19
Are there explicit requirements for ensuring conservatism of analysis, for example by incorporating certain penalties into the methodology?	20
Are there any specific requirements for “safety margins”? Elaborate if so.	21
How are claims on operator action treated within the DBA?	22
Describe briefly, preferably in a table format, your regulations on accident categories and failure criteria and acceptance criteria to be applied in those.	23
Describe your expectations how far the analyses should be made in the analyzed cases. Describe the end states of the analysis (safe state/controlled state/steady state etc..)	25
Describe main requirements concerning safety injection filtering devices (sumps).	26

REGULATORY REQUIREMENTS FOR ACCIDENT AND TRANSIENT ANALYSES

This paper summarizes the results of comparison of regulatory approaches of Finland, Hungary, Russia and Turkey to the requirements for accident and transient analysis of nuclear power plants. The paper was prepared in the framework of Technical expert subgroup on accidents and transients analysis of MDEP VVER Working Group. The objective of the analysis was to identify commonalities among the Regulatory Authorities and their TSOs within the requirements concerning accident and transient analysis in order to:

- promote understanding of each country's regulatory requirements to accident and transient analyses;
- identify areas where harmonization and convergence of regulations, standards, and guidance could be further developed;
- enhance communication among the members;

Regulatory framework concerning accident and transients analysis

General "Top level" requirements concerning safety are given in the following laws:

- Nuclear Energy Act 11.12.1987/990 in Finland,
- Hungarian Atomic Act (Law 1996/CLVI) in Hungary
- Federal law No. 170 on Use of Atomic Energy in Russia

In all participating member countries laws define the need to justify safety. In Hungary and Finland more detailed requirements than those mentioned in Acts are given in Decrees. Nuclear Energy Decree 12.2.1988/161 in Finland and Govt. Decree 118/2011 in Hungary. In Turkey the general framework in nuclear regulatory area is under development. IAEA Safety Fundamentals and Requirements and Russian regulation is used as licensing basis for the Akkuyu NPP.

In Russia detailed safety requirements are set in Federal Regulations and Rules (NP) and there are a number of Safety Guidelines (RB) with recommendations on how to follow the requirements.

Regulations have been updated recently in all member countries that replied to the questionnaire. It is not always clear whether changes have been due to the lessons learned from the Fukushima accident, or whether they are natural development of requirements for other reasons. Russian requirements have been improved lately in the area of assessment of errors and uncertainties in A&T analyses, internal and external hazards and different operational modes. Hungarian regulator has improved regulations for example concerning external hazards and severe accident management. There are detailed presentations available concerning post Fukushima development in regulations for example in IAEA Convention on Nuclear safety reports and these findings are not duplicated here.

Periodical safety assessments

According to the participants the safety analysis report is mandated to be in line with the plant design and to be kept up to date to reflect changes in the plant design. In Hungary and Finland periodic reviews are required. In Turkey it is seen that periodic safety review in accordance with IAEA suggestions will be established.

According to Russian federal law "On the use of nuclear energy" if a nuclear facility is operating on the basis of a permit (license) issued for a period of more than 10 years, the utility performs a periodic safety assessment of the nuclear installation. The first periodic safety assessment is performed 10 years after the start of their operation, followed by a periodic safety assessment every 10 years until the end of their operation.

Concept of design basis

All parties use the concept of design basis where similarities exist. Terminology differs between the member countries.

In Russia the following concept is used. There are normal operation and deviations of normal operations. Deviations of normal operations include transients and two categories of accidents. Design Basis Accidents and Beyond Design Basis Accidents. BDBA include scenarios and events that are not in DBA and also severe accidents.

Hungarian approach separates DBA and DEC. Categories in DEC are DEC 1 and DEC 2. DEC 1 includes complex accidents without severe fuel damage and DEC 2 includes severe accidents.

In Finland design envelope includes Design basis categories (DBC1-DBC4), design extension categories (DEC A, B, C) and Severe accidents.

In Turkey the Regulation on Specific Principles for Safety of Nuclear Power Plants design basis accident and beyond design basis accidents are defined. In addition severe accident is defined based on pre-Fukushima accident IAEA approach.

Usually initiating events are grouped based on the physical evolution of the events. Each group should include event sequences that lead to a similar challenge to the safety functions and barriers and need similar mitigating systems to drive the plant to a safe state.

Defining cases to be analyzed

There are different approaches between member countries. However, in all member countries it is the responsibility of the licensee to develop and present plant specific list of cases to be analyzed.

Approximate list of initiating events for DBA and a list of BDBA are given in Russian Federal Regulations and Rules NP-006-16. RB-150-18 provides recommendations on the development of the list of BDBA to be taken into account in the design of nuclear power plants with VVER-type reactors. Hungarian regulations in NSC Volume 3a gives detailed list of events that are minimally required to be assessed including screening criteria for the events. STUK does not give recommended list of events in regulations.

In accordance with replies all parties note that selection of scope of the analyzed events is based on combining deterministic and probabilistic method. Also engineering judgment is mentioned. HAEA presents in regulation 3a.2.2.3500 that selection can be also done based on deterministic approach. Screening of events or faults based on estimated frequency is allowed in Russia for internal events with probability less than 10^{-6} per year. For external accidents of natural origin screening criteria 10^{-4} is used in Russia. For BDBA scenarios no screening criteria is given. Screening based on frequency is allowed in Hungary for internal events with frequency less than 10^{-6} per year. For external events of natural origin 10^{-5} per year is used as criteria. For Events resulting from human activities as external events the screening criteria is set to 10^{-7} . In Finland there is no screening criteria set in regulations.

All parties noted that there is need to demonstrate diversity for safety functions to provide evidence that design takes into account common cause failures in safety systems adequately. Answers do not go into details by defining what level of diversity is required and where.

Conservative analysis

All parties replied that conservative approach is acceptable, although in certain cases also best estimate approach can be used. In Finland and Hungary best estimate approach is to be supplemented with uncertainty analysis. Conservatism of analysis can be either justified by the results of uncertainty analysis or by using certain known conservative assumptions, such as the

30 minute rule of assumptions of single failures. Russian regulations do not define safety margins. In Hungary and Finland there is requirement that sufficient margins need to be demonstrated. The reasoning behind safety margins is avoidance of cliff edge scenarios.

Operator actions

Operator actions to mitigate the situation can be credited, but justifications are expected by the regulator concerning the timeframe that is used in the analysis.

Defense in Depth

Different requirements concerning accident categories and failure and acceptance criteria are summarized in table format in the questionnaire. All tables present 5 levels of defense in depth. There exists similarities in the tables, but certain acceptance criteria, naming of Defense in Depth levels and failure assumptions are found to be different. In Hungary and Finland for example single failure is cumulated with maintenance assumption in certain plant states.

Filtering elements

All answers mention regulations exist concerning safety injection filtering devices to some extent. Filtering devices shall provide sufficiently clean water to safety injection system. In responses from Hungary and Finland requirements concerning possibility to clean the filters is mentioned.

What are the most relevant legal and regulatory documents that govern the analysis of A&T? List those in hierarchical order

SEC NRS Response

1. Top-level requirements have been given in Federal law No. 170 on Use of Atomic Energy (Examination of safety justification and the computer programs used for safety justification observance of requirements of federal norms and rules of use of atomic energy).
2. Requirements (What should be done to prove the NPP safety).
 - NP-001-15 (Mane Provisions on NPP Safety Justification)
 - NP-006-16 (Requirements on structure and contents of SAR for nuclear power plant with VVER reactor type);
 - NP-082-07 (Rules of nuclear safety for reactor facility of nuclear power plants)
3. Guideline (How safety justification (accident analyses) could be done to fulfill the requirements)

IAEA Deterministic Safety Analysis for Nuclear Power Plants. No SSG-2.

HAEA Response

The top level requirements are given by the Hungarian Atomic Act (Law 1996/CLVI), although this document only requires from the licensees to justify and prove nuclear safety through safety assessments but does not specify the means and methods to achieve that.

On the second level the legal requirements for A&T assessment is given by the 118/2011 Gov. decree and its annexes, which are the 10 volumes of the Nuclear Safety Codes. The main text of the government decree requires the licensee to assess A&T through deterministic and probabilistic safety assessments, but the requirements for means and methods are described in details within the NSCs:

- NSC Volume 3 sets the requirements for the design of operating NPPs,
- NSC Volume 3a for the design of the new NPPs,

- NSC Volume 4 sets the requirements for the operation of NPPs (which also contains several dedicated requirements on where and how to use safety assessments to support various operating activities),
- NSC Volume 5 sets the requirements for the design and operation of research reactors
- NSC Volume 6 sets the requirements for the design and operation of interim spent fuel storage facilities

One more NSC Volume worth mentioning as relevant to A&T analysis, which is Volume 10. NSC 10 contains the official regulatory definitions and terminology used and accepted by the HAEA, such as DBA, DEC, safety functions, core damage, safe shutdown state, etc.

Based on the detailed requirements set by the NSCs, HAEA issues several regulatory guides in order to provide recommendations of the preferred methodologies. In the field of A&T analysis, the following guides are the most relevant:

Based on the NSC Volume 3 requirements

- A3.11 Reg. Guide Probabilistic Safety Assessment of Operating NPPs
- 3.32 Reg. Guide Deterministic Safety Assessment of Operating NPPs

Based on NSC Volume 3a requirements

- N3a.11 Reg. Guide Probabilistic Safety Assessment of New NPPs
- N3a.32 Reg. Guide Deterministic Safety Assessment of New NPPs
- N3a.33 Reg. Guide Deterministic Severe Accident Assessment of new NPPs

There are Regulatory Guides in preparation to provide regulatory recommendations on the safety assessment of low- and mid-level nuclear waste repositories (drafted version is under review right now), which will cover both DSA and PSA aspects of such facilities and there are plans to develop such RGs for the interim spent fuel storage facilities as well.

[STUK Response](#)

Top level requirements for methodology have been given in **Nuclear Energy Act 11.12.1987/990**, **Nuclear Energy Decree 12.2.1988/161** and **STUK regulation STUK Y/1/2016**, which are binding.

Detailed technical requirements are given in **YVL guides**.

The publication of a YVL Guide shall not, as such, alter any previous decisions made by STUK. After having heard the parties concerned STUK will issue a separate decision as to how a new or revised YVL Guide is to be applied to operating nuclear facilities or those under construction, and to licensees' operational activities. The Guide shall apply as it stands to new nuclear facilities and to other use of nuclear energy.

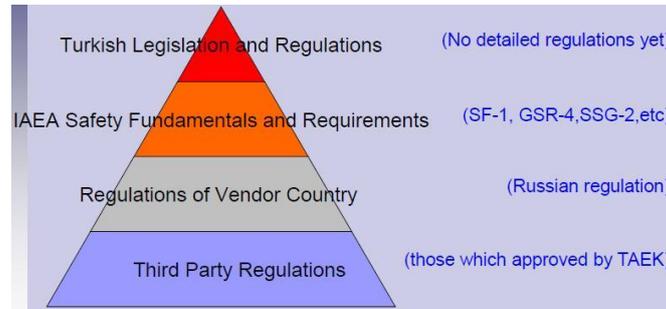
When considering how the new safety requirements presented in the YVL Guides shall be applied to the operating nuclear facilities, or to those under construction, STUK will take due account of the principles laid down in Section 7a of the Nuclear Energy Act (990/1987): The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified considering operating experience, safety research and advances in science and technology.

According to Section 7 r (3) of the Nuclear Energy Act, the safety requirements of the Radiation and Nuclear Safety Authority (STUK) are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations. If the licensee can convincingly demonstrate that the proposed procedure or solution will implement safety standards in accordance with this Act, the Radiation and Nuclear Safety

Authority (STUK) may approve a procedure or solution by which the safety level set forth is achieved.

NDK Response

A simple “licensing basis” approach is applied for the nuclear power plant project in Turkey in their licensing process (please see the brief figure).



According to this approach, list for hierarchical order is below.

Turkish Legislation:

1. Decree on Licensing of Nuclear Installations
2. Regulation on Design Principles for Safety of Nuclear Power Plants
3. Regulation on Specific Principles for Safety of Nuclear Power Plants
4. Regulation on Nuclear Power Plant Sites
5. Regulation on Radiation Protection in Nuclear Facilities
6. Guide on Specific Design Principles

IAEA Safety Fundamentals & Requirements:

1. IAEA No. GSR Part 4 (Rev. 1) Safety Assessment for Facilities and Activities
2. IAEA No. SSG-2 Deterministic Safety Analysis for Nuclear Power Plants

Regulations of Vendor Country – Russia:

1. Federal law No. 170 on Use of Atomic Energy
2. NP-001-97 (OPB-88/97) General regulations on Ensuring Safety of Nuclear Power Plants
3. NP-006-98 (PNAE G-01-036-95) Requirements to Contents of Safety Analysis Report of Nuclear Power Plant with VVER Reactors
4. NP-082-07 Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants

Note: A new statutory decree has been published in 02.07.2018 which describe the new and general framework in nuclear regulatory area.

Has there been a change in the scope of requirements in the recent years. If yes, highlight the major changes. Please also state reason for changes.

SEC NRS Response

The most recent changes related to the A&T were made in post-Fukushima period and mostly reflected lessons learned from Fukushima accident, but not only. The major changes related to A&T were implemented in NP-001-15 and in NP-006-16:

1. Safety analyses shall be accompanied with assessments of errors and uncertainties of obtained results;
2. Safety analyses should include internal and external events and cover all operational modes and all locations of nuclear fuel;

3. Probabilistic criteria for DBA IE screening ($f < 10^{-6}$);
4. Explicit requirement to provide special technical means to manage long lasting blackout and loss of alternative heat sink resulted on accident analyses to confirm the efficiency of such technical means.

HAEA Response

Post-Fukushima modifications and amendments have been added to the Hungarian legal and regulatory framework mainly in the field of external hazard assessment and severe accident management (e.g.: to install connection point/port for mobile equipment) in accordance with the international practice and the new requirements of the European Union, e.g.: to assess the possible combinations of external hazards due to correlations and causations, to assess and prevent cliff-edge effect, etc.

There were modifications because of 2013/59/EURATOM directive and adoption/compliance with IAEA documents (i.e. SSR-2/1, SSR-2/2) and WENRA requirements. The new governmental decree was issued in April 2018.

Partly related to the Fukushima accident there were a lot of changes mainly related to hazards assessments and PSA. The ones that are mainly not related to hazard assessment or PSA but was adopted from WENRA are the following:

“The classification of SSCs shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment.”

“A set of DECAs shall be derived and justified as representative, based on a combination of deterministic and probabilistic assessments as well as engineering judgement.”

“The selection process for DEC A shall start by considering those events and combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to severe fuel damage in the core or in the spent fuel storage. It shall cover:

- Events occurring during the defined operational states of the plant;
- Events resulting from internal or external hazards;
- Common cause failures.
- Where applicable, all reactors and spent fuel storages on the site have to be taken into account. Events potentially affecting all units on the site, potential interactions between units as well as interactions with other sites in the vicinity shall be covered.”

“A stuck control rod shall be considered as an additional aggravating failure in the analysis of design basis accidents.”

STUK Response

Government decree has been substituted by STUK regulation. No major changes in the level of the requirements.

Reasons for changes are in COUNCIL DIRECTIVE 2014/87/EURATOM (amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations), COUNCIL DIRECTIVE 2013/59/EURATOM (laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom).

NDK Response

Turkish regulatory documents which govern the A&T have not been changed recently. But the licensing basis for the Akkuyu NPP has been approved in 14.11.2014 and it reflects the all new provision of IAEA & Russian documents by this date.

Are there requirements to update earlier analyses from time to time, or due to specific reasons? For example, in the context of Periodic Safety reviews.

SEC NRS Response

There is no explicit requirement to renew the results of safety analyses periodically. However, the main requirement (p. 1.2.8 NP-001-15)

«NPP safety important Discrepancies between information contained in the NPP SAR and the NPP design, as well as its implementation are not allowed. The compliance of the NPP SAR with the real NPP status shall be maintained by the NPP operational organization within the entire service life of the NPP».

All safety related modernizations should be confirmed by safety analyses.

Safety analyses should comply with state of art.

The renewal of the license and the periodic reassessment of safety in themselves are not the cause to perform the new safety analyses. That means accident analyses presented in SAR should comply with actual status of NPP unit.

HAEA Response

Yes, however there is no requirement for the time period. If there is a modification the licensee have to prepare and update the analyses accordingly. Also the licensee must review the analyses periodically, especially when new safety information is received.

3.2.3.0600. During the whole lifetime of the nuclear power plant the suitability of all interventions or modifications of nuclear safety related systems, structures and components that deviate from the authorized conditions shall be demonstrated with deterministic safety analysis or a combination of deterministic and probabilistic safety analyses.»

3.2.3.0700. The design basis, the extended design basis and their substantiation shall be periodically reviewed at the completion of the design, as well as during the whole lifetime of the nuclear power plant, when significant new safety information is received and based on the results of deterministic and probabilistic calculations modifications shall be implemented, if necessary. The identified defects shall be evaluated and the necessary corrective actions shall be taken in time.

STUK Response

Analyses are required to be updated from time to time i.e. during periodic safety assessment, operating licence renewal or if major modifications at NPP are carried out, for instance during power uprating.

In accordance with the STUK Regulation Y/1/2016 3: *The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear power plants, the results of safety research, plant modifications, and the advancement of calculation methods.*

NDK Response

Even there is no explicit requirement for periodic safety review, the Decree on Licensing of Nuclear Installations state that “any modification in the facility which might affect safety is

subjected to the approval of the Authority”. This sentence refers to the need for re-evaluation of the new situations in the plant and required necessary safety analysis.

In the Decree another article also ask “if new scientific findings in the field of the nuclear safety, operating experiences and national interests necessitate certain modifications in the nuclear facility then the applicant may be asked to make these changes”. This also can require the new safety analysis when it deems necessary.

In addition to these, under the new regulation works, periodic safety review in accordance with IAEA suggestions will be established in near future.

Is the concept of design basis used? If yes, how is it defined?

SEC NRS Response

Yes. There are two main majorities of accidents:

Design Basis accidents (DBA) and Beyond Design Basis Accident (BDBA);

DBA – postulated list of initial events, including internal and external events, which deteriorate of normal operation and could not be excluded due to inherent safety of reactor design. Single failure criteria, dependent criteria and hidden failures should be considered in safety analyses.

Results of DBA analyses should demonstrate the compliance with design limits.

Graded approach: more strong criteria correspond to more often events.

Approximate list of DBA is installed in Federal norms and rules NP-006-16, App.). The final list of DBA should be developed by operational organization and presented in SAR. Design specific aspects as well as operational experience should be considered in the formation of the list.

BDBA- majority of accidents caused by the initiating events excluded from consideration for the design basis accidents or which involve some failures of safety systems components in addition to a single failure or human errors.

Approximate list of BDBA is installed in Federal norms and rules. The final list of BDBA is developed by operational organization and should be presented in SAR. The final BDBA list should include representative severe accidents 1/2/16 NP-001-15.

In practice, the list of BDBA includes:

- complex sequences (DBA with multiple failures);
- ATWS scenarios;
- rare events excluded from DBA;
- failure of digital automation safety control systems

The final list of BDBA should be representative to develop the guideline for BDBA management. Results of BDBA analyses should confirm the restriction of radioactive release.

Two probabilistic target should be confirmed: CDF <10⁻⁵; LRF <10⁻⁷.

HAEA Response

Design basis is defined in the terms of frequencies of operating conditions. Conditions that deviate from normal operation shall be divided into design basis and beyond design basis operating conditions. For new NPPs the following limits are used:

Operating condition	Description	Frequency of event (f [1/year])
DBA1	normal operation	-

Operating condition	Description	Frequency of event (f [1/year])
DBA2	anticipated operational occurrences	$f \geq 10^{-2}$
DBA3	low frequency design basis accidents	$10^{-2} \geq f \geq 10^{-4}$
DBA4	very low frequency design basis accidents	$10^{-4} \geq f \geq 10^{-6}$
DEC1	complex accidents without melting of the fuel in the core and the spent fuel pool	$10^{-6} \geq f$
DEC2	severe accidents resulting in a significant fuel melting	

STUK Response

Yes. Postulated initiating events are divided into different design bases conditions (DBC). Division is based on the frequency of an event on the following way:

- Normal operation (DBC 1)
- Anticipated operational occurrences (DBC 2), $f > 10^{-2}/a$
- Postulated (design basis) accidents
 - Class 1 (DBC 3), $10^{-2}/a > f > 10^{-3}/a$
 - Class 2 (DBC 4), $f < 10^{-3}/a$
- Design extension conditions (DEC)
 - DEC A - includes conditions in which a common cause failure (CCF) in a safety system is assumed during anticipated operational occurrence (DBC 2) or class 1 accident (DBC 3)
 - DEC B - includes complex sequences
 - DEC C - rare external events
- Severe accidents $f < 10^{-5}/a$

All above mentioned are within design envelope.

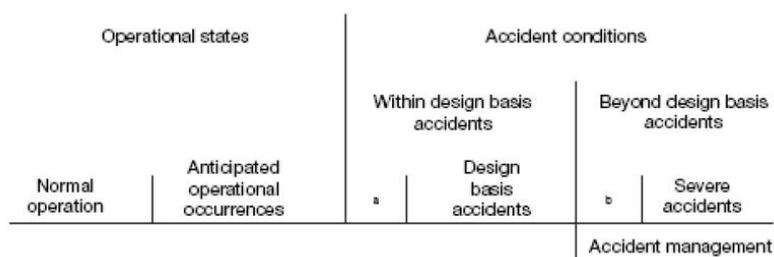
NDK Response

Yes, concept of design basis has been defined in Turkish Regulation. In the Regulation on Specific Principles for Safety of Nuclear Power Plants there are definitions for design basis accident and beyond design basis accidents.

DBA is defined as “accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits”.

BDBA is defined as “conditions more severe than a design basis accident”.

In addition to these, there is also “severe accident” definition which is accidents caused significant core damage & radiological consequences” in a.m. regulation. It is based on old IAEA approach.



How is the scope of events/faults for analysis (postulated initiating events) defined? Is the list of such events fully prescribed? Is it up to the designer/vendor/licensee to propose a list of events based on some principles?

SEC NRS Response

Approximate DBA and BDBA lists are presented in the regulatory requirements (NP-001-15 appendixes №7 and № 9). However the licensee should develop and present in SAR the final lists of DBA and BDBA, considering design specific aspects, and operational experience.

HAEA Response

The responsibility to define postulated initiating events lies with the licensee but the NSC volume 3a give a very detailed list on events that are minimally required to be assessed and also sets the screening criteria for these events.

3a.2.2.4500. Among the various DBA1-4 operating conditions, at least the following internal events shall be taken into account in the design of the nuclear power plant:

- *Normal operating conditions belonging to DBA1;*
 - *operation at full power,*
 - *uploading process,*
 - *hot standby condition,*
 - *hot shutdown condition,*
 - *cold shutdown condition,*
 - *refuelling condition,*
 - *operation with a disconnected loop, if allowed,*
- *Anticipated operational transients belonging to DBA1 operating condition:*
 - *increase or decrease in temperature at a rate allowed by the Operational Limits and Conditions,*
 - *abrupt increase or decrease in load to an extent allowed by the Operational Limits and Conditions,*
 - *increase or decrease in load at a rate allowed by the Operational Limits and Conditions,*
 - *switchover to house load from rated output, with steam blow-off,*
 - *over voltage or the instability of the electric grid,*
 - *operation under the limiting conditions allowed by the Operational Limits and Conditions,*
- *DBA2 operating conditions:*
 - *inadvertent movement of the control rod assembly with a subcritical reactor,*
 - *inadvertent movement of the control rod assembly with operation at full power,*
 - *incorrect positioning of control rod assemblies or rod groups,*
 - *inadvertent dilution of boric acid,*
 - *partial decrease of the primary flow rate of the coolant,*
 - *inadvertent closing of the main steam isolation valves;*
 - *total loss of load or turbine failure,*
 - *loss of the main feedwater flow of the steam generator,*
 - *uncontrolled decrease or increase of the main steam flow rate,*
 - *failure of the main feedwater system of the steam generator,*
 - *loss of off-site voltage for less than 2 hours,*
 - *turbine overload,*
 - *temporary pressure decrease in the primary cooling loop,*

- *pressure decrease in the secondary circuit caused by inadvertent opening of the safety valve of the steam generator or another single failure,*
- *unjustified startup of the emergency core cooling system,*
- *failure of the primary circuit chemical and volume control system,*
- *slight loss of coolant, in particular, the rupture of a pulse line,*
- *loss of the main heat sink,*
- *DBA3 operating conditions:*
 - *loss of coolant in the primary circuit, in particular, a small pipe break,*
 - *small pipe break in the secondary circuit,*
 - *forced reduction of the coolant flow,*
 - *placing a fuel assembly in an incorrect position,*
 - *pulling out a control rod assembly in operation at full power,*
 - *unjustified operation of the safety valve of the pressurizer,*
 - *rupture of the volume control tank,*
 - *rupture of a tank for retaining gaseous wastes,*
 - *rupture of a liquid waste collection tank,*
 - *rupture of one steam generator tube or a pipe connected to the primary cooling circuit of the nuclear reactor that is partially outside of the containment, or damage to a heat exchanger tube, without a preliminary iodine peak,*
 - *loss of off-site voltage for 72 hours,*
 - *instability of the active core,*
 - *delayed intervention by systems fulfilling a reactor shutdown function required during DBA2 operating conditions,*
- *DBA4 operating conditions:*
 - *rupture of the main steam pipeline,*
 - *rupture of the main feedwater pipeline,*
 - *sticking of the main circulating pump,*
 - *ejection of any control rod assembly,*
 - *loss of coolant in the primary circuit, including the rupture of the largest diameter pipeline of the primary circuit with a discharge through 200% of the cross-section,*
 - *accident related to the handling, moving and storage of nuclear fuel;*
 - *rupture of a steam generator tube with a preliminary iodine peak,*
 - *rupture of more than one steam generator pipe or lift-off of the primary collector.*

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.*

NSC Volume 3a defines the screening criteria for new NPPs in the following way:

3a.2.2.5000. The following can be excluded from the scope of postulated initiating events:

- *internal initiating events occurring due to the failure of systems, structures or components or human error or both if their frequencies are less than 10⁻⁶/year;*
- *an event resulting from external human activities characteristic to the site, the frequency of which is less than 10⁻⁷/year, or if the hazard factor is at a distance that it can be demonstrated that it is not expected to have an effect on the nuclear power plant unit; and*
- *all initiating events triggered by a recurring external effect of natural origin, with a frequency of less than 10⁻⁵/year.*

STUK Response

Designer/vendor/licensee proposes the cases to be analyzed at each category during construction permit phase in PSAR.

NDK Response

There is no explicit list for DBA & BDBA in the Turkish Regulation. It is up the licensee to propose & justify the list in terms of its completeness & applicability to the design. In accordance with this, Russian NP-006-98 Section 15.1 & 15.2 and Annex 15 has been used in Akkuyu NPP Unit 1 PSAR.

Is the scope of analyzed events based on probabilities or engineering judgment?

SEC NRS Response

The scope of analyses is based on deterministic and probabilistic analyses and in some cases also on engineering judgments

HAEA Response

The selection of the scope must be based on deterministic methods or the combination of deterministic and probabilistic methods.

3a.2.2.3500. *For the design, all postulated initiating events that may influence the safety of the nuclear power plant shall be identified. The ones of these events to be incorporated into the design basis shall be selected by a deterministic method or the combination of deterministic and probabilistic methods.*

STUK Response

The scope of analyses covers deterministic and probabilistic analyses and in some cases also engineering judgments.

NDK Response

The scope of analyses is based on deterministic and probabilistic analyses and in some cases also on engineering judgments.

Is there a need to demonstrate diversity for each safety function?

SEC NRS Response

Yes, according to p. 3.1.9 NP-001-15, Measures on protection of safety systems and elements as well as systems and elements of special technical features for BDBA management from common cause failures shall be considered in the design by implementing the diversity, redundancy and independency principles in the design.

It is confirmed by the results of the deterministic analyses in the ch. 15 of SAR.

HAEA Response

Yes, but it not specifies exactly that it needed for each safety function. NSC 3a sub-chapter 3a.2.3 sets the requirements on how to demonstrate the safety of the design, although it does not specify to demonstrate diversity, just sets the requirements on the means and methods of safety assessments and the justification of design requirements. Diversity of such systems is a basic design requirement, therefore it has to be assessed and proved by the licensee, e.g.:

3a.4.5.2100. *In the case of systems included in Safety Classes 2 and 3, the following verification analyses shall be carried out:*

...

- *analysis of common cause failure possibilities, in particular, the ones relating to specification, design, fabrication, software and hardware, environmental impacts, maintenance problems, the use of an identical system or component in different levels of defence in depth, architecture, separations and sufficient diversity,*

...

STUK Response

Yes. This is covered by the analyses of design extension conditions (DEC A) - includes conditions in which a common cause failure (CCF) in a safety system is assumed during anticipated operational occurrence (DBC 2) or class 1 accident (DBC 3).

NDK Response

Yes. Turkish Regulation on Design Principles for Safety of Nuclear Power Plants defined diversity as “the presence of two or more redundant systems or components to perform an identified function, where the different systems or components have different attributes so as to reduce the possibility of common cause failure, including common mode failure.”

In Article 7 states that “Safety systems and functions are designed to be testable during operation, under realistic load and performance conditions if possible. High values are chosen for reliability targets of safety systems, design principles such as fail-safe, single failure, diversity and physical separation are applied and complete independence of safety systems from normal plant systems are achieved.”

And in Article 8 states that “provisions are made during design so that no single event can lead to a loss of a safety function by affecting more than one structure, system or component. Internal events such as fire or power loss and external events such as earthquake, flood and airplane crash as well as faults during manufacturing and assembly are taken into account.” In this manner diversity has been applied for safety systems and functions.

Can events/faults be screened out of the DBA based on low consequences? What are the screening criteria?

SEC NRS Response

According to p.1.2.15 NP-001-15, Internal events with probability of 10⁻⁶ or less per year may not be included into the list of the design basis accidents to be provided in SAR.

There are no probabilistic criteria screening BDBA scenario. The reason for BDBA screening are the following: similar strategy for BDBA management.

HAEA Response

NSC 3a only allows to screen out external human hazards if the distance is high enough that it can be demonstrated that it cannot have an effect on the NPP:

3a.2.2.5000. *The following can be excluded from the scope of assumed initiating events:*

- *internal initiating events occurring due to the failure of systems, structures or components or human error or both if their frequencies are less than 10^{-6} /year;*
- *an event resulting from external human activities characteristic to the site, the frequency of which is less than 10^{-7} /year, or if the hazard factor is at a distance that it can be demonstrated that it is not expected to have an effect on the nuclear power plant unit; and*
- *all initiating events triggered by a recurring external effect of natural origin, with a frequency of less than 10^{-5} /year.*

In practice on the regulatory guide level, we recommend to list all such events within the design bases but if it can be concluded with a high certainty the event/fault in question cannot interrupt the normal operation of the NPP it can be screened out further/detailed analysis. Such screening has to be documented in detail and justification shall be provided for it. Regulatory Guide 3a.11 and N3a.11 highlights the fact that the sum of such minor events and faults may interrupt the normal operation if they happen simultaneously (e.g.: due to an external hazards as earthquake), therefore they have to be reassessed from that perspective as well.

STUK Response

No. All scenarios need to be analyzed.

NDK Response

In the Guide on Specific Design Principles frequency definitions for normal operation, design basis accidents and beyond design basis accidents are given. It can be found in the Table related with defense in depth levels below.

In Turkish Regulation on Nuclear Power Plant Sites, Article 12 also states that “human induced events of which probability of occurrence is 10^{-7} /year or more should be investigated”. And also in Article 15 it is stated that “human induced potential external events which might result in radiological consequences or have a probability of occurrence 10^{-7} /year or more should be regarded as design basis.

As last, Guide on Specific Design Principles Article 9 states that

External Events-Earthquake

ARTICLE 9- (1) The following design principles shall be taken into consideration in relation with earthquakes:

- Nuclear power plants shall not be located on sites directly situated on active faults.
- *S2 earthquake, against which structures, systems and components important to safety are designed, shall be taken as the site-specific earthquake with a return period of 10,000 years, which corresponds to 0.5% probability of being exceeded in 50 years.*
- *Vertical ground acceleration shall be at least 2/3 of the horizontal ground acceleration.*

Are there explicit requirements for ensuring conservatism of analysis, for example by incorporating certain penalties into the methodology?

SEC NRS Response

Yes, according p. 1.2.9 NP-001-15, Deterministic analyses of design bases accidents shall be done on the basis of the conservative approach.

Realistic (non-conservative) analysis of the beyond design basis accidents shall be presented in SAR, according p. 1.2.16 NP-001-15.

Conservatism of analysis is provided by the choice of the initial and boundary conditions and could be confirmed by the results of the uncertainty analyses.

HAEA Response

Yes. Conservatism in general is required for DBA assessment by NSC 3a, and there are requirements on how to achieve such conservatisms, e.g.:

- *No operator action can be taken into consideration in the first 30 minutes of the transient*
- Input values must be based on conservative assumptions
- Single failure criterion shall be applied during DBA assessment and analyses
- Etc.

3a.2.3.1500. In the analyses of events resulting in DBA2-4 and DEC1 operating conditions, the operator interventions may only be considered on the basis of a conservatively defined time requirement. In the case of operator interventions assumed within a 30-minute timeframe, the analysis also determining the uncertainties shall demonstrate that the assumed operator activities can be performed within the available time.

3a.2.3.1600. In the analyses of events resulting in DEC1 and DEC2 operating conditions, the best estimate method shall be used. The inoperability of any system, structure or component shall be presumed if damage to it is likely as a result of the initiating event or the course of the breakdown.

The NSCs allows the licensee to use best estimate approaches and assumptions even for the justification of DBC but in such cases uncertainty analysis shall be performed as well the licensee has to prove the appliance with the requirements through the “95/95” approach:

- The licensee has to demonstrate that the fulfillment of the requirements can be proved with a 95% probability on a 95% confidence level. (This method is only described in detail in Reg. Guide N3a.32, the NSCs themselves only gives the permission to justify the DBC via BEPU analyses.)

STUK Response

Yes. Two optional methods:

- Conservative method complemented by sensitivity analysis:
Input values (power, temperature and pressure, etc.), performance of safety systems (failure criteria) are selected conservatively.
- Best-estimate method complemented by uncertainty analysis:
Approval of statistical methods and acceptance criteria used in uncertainty analysis are required.

NDK Response

Yes. In the Regulation on Specific Principles for Safety of Nuclear Power Plants Article 11 states that “physical and mathematical models used in the design are verified by experiments, operational tests and data analyses. In the safety analysis of the plant, conservative data and models are used. Realistic data and models are used only when their validity and suitability are proven.”

Are there any specific requirements for “safety margins”? Elaborate if so.

SEC NRS Response

Requirements on safety margin is not defined. Compliance with acceptance criteria should be demonstrated.

HAEA Response

Yes, there are numerous specific requirements for the licensee to ensure safety margins, e.g.:

3a.2.2.6700. For the analyses of TAK events:

- *it shall be demonstrated that sufficient margins are available for avoiding the cliff edge effect;*

3a.2.3.0500. The analyses used for the demonstration of safety shall be documented in a such way and to such a depth that they may be repeated, independently reviewed and modified to an extent necessary for the evaluation of modifications throughout the lifetime of the nuclear power plant; furthermore, the extent of conservatism applied and the extent of margins available based on the analysis may be reviewed and re-evaluated.

3a.2.3.2800. The Preliminary and Final Safety Analysis Report shall be compiled on the basis of the following content requirements:

- *description of the safety analyses carried out to evaluate the safety of the nuclear power plant and to demonstrate the fulfilment of the safety criteria and the release limits for radioactive materials in the case of TA1-4 and TAK1-2 operating conditions as well as demonstrating that appropriate safety margins are available in the case of TA1-4 and TAK1 operating conditions,*

3a.3.2.1300. During design, it shall be demonstrated through the analysis of degradation processes limiting the lifetime that

- *despite the effect of ageing, the strength properties of structural materials correspond to the maximum loads calculated for TA1-4 and TAK1 and TAK2 operating conditions by taking into account the safety margins specified for the operating condition, of the system component concerned performs a safety function in the given operating condition;*

STUK Response

Safety margin to acceptance criteria have to be demonstrated.

NDK Response

In Turkish Regulation on Design Principles for Safety of Nuclear Power Plants Article 12 states that “Reactor is designed to be protected against reactivity induced accidents with a conservative margin of safety.”

How are claims on operator action treated within the DBA?

SEC NRS Response

The NPP design shall include the technical and organizational measures dedicated to prevent DBA and to mitigate their consequences and to ensure non-exceedance of the limits established for DBA by means of inherent safety and by means of safety systems (p. 2.1.11 NP-001-15).

The activation of the safety systems shall be carried out automatically. The acceptability of start of safety systems by the operator has to be proved in SAR (p. 3.1.11 NP-001-15).

Operational organization is responsible for development and observance of the emergency instructions and procedures in which operator actions are defined for DBA, BDBA and SA conditions.

These procedures and instructions should be based on the signs of the events, on the plant conditions and on the prognoses of accident development. Operator actions should be directed on restoration of the safety function and on the mitigation of their consequences (p. 4.1.5 NP-001-15).

Switch off the safety systems after their automatically start should be prevented by means of design within 10-30 minutes, but it should not prevent the operator’s actions provided by emergency instructions and procedures (p. 3.4.4.2 NP-001-15).

HAEA Response

The NSCs 3a require the following approach:

3a.2.3.1500. In the analyses of events resulting in DBA2-4 and DEC1 operating conditions, the operator interventions may only be considered on the basis of a conservatively defined time requirement. In the case of operator interventions assumed within a 30-minute timeframe, the analysis also determining the uncertainties shall demonstrate that the assumed operator activities can be performed within the available time.

STUK Response

If adequate emergency operational procedures are available, it is acceptable for the operator to follow these guidelines but only after certain time period, which normally is 30 minutes.

NDK Response

In Turkish Regulation on Design Principles for Safety of Nuclear Power Plants Article 12 it is stated that “In case of deviation from normal operation, reactivity feedback effects and process controls that restore normal operating conditions are supported by provisions for shutdown, continued core cooling and protection against the release of radioactive materials. Further protection is provided in design through automatically actuated engineered safety systems. At the onset of any deviation, systems are automatically actuated to provide the operating staff with sufficient time to assess systems, review possibilities and decide on a subsequent course of action. The design makes provision for diagnostic aids and symptom based emergency procedures for use in these circumstances. “

And in Regulation on Specific Design Principles for Safety of Nuclear Power Plants Article 31 it is stated that “The goal in managing an accident that exceeds the design basis would be to return the plant to a controlled state in which the nuclear chain reaction is essentially terminated, continued fuel cooling is ensured and radioactive materials are confined. The results of an analysis of the response of the plant to potential accidents beyond the design basis are used in preparing guidance on an accident management plan. Nuclear plant staff are trained and retrained in the procedures to follow if an accident occurs that exceeds the design basis of the plant.”

Describe briefly, preferably in a table format, your regulations on accident categories and failure criteria and acceptance criteria to be applied in those.

SEC NRS Response

D&D	Plant state	Organizational & Technical Means	Acceptance criteria for fuel rods	Failure assumption
Lev.1	Normal operation Prevention of AO	normal operation systems	Operational limits and conditions	
Lev.2	AO, prevention of DBA	10 ⁻²	Prevention of CHF	Single failure criteria tolerance
Lev.3	DBA, prevention the transition to BDBA and SA	10 ⁻⁴ < f < 10 ⁻² safety systems	Maximum design limit of fuel damage*	Single failure criteria tolerance
Lev.4	BDBA management Prevention of SA and mitigation of their consequences	f < 10 ⁻⁴ safety systems + special technical means dedicated for BDBA management	No needs on protection measure beyond the boundary of planning zoon The radius of planning zone < 25km	Multiple failures
Lev.5	Emergency planning			

*Maximal design limit on fuel damage:

- Cladding temperature < 1200°C;
- The degree of cladding oxidation lower than limit defined on the base of experiments;
- The proportion of reacted zirconium should not be more than 1% of the total mass of fuel claddings;
- The maximum fuel temperature should not be above the melting point.

HAEA Response

Level of defence in depth	Relevant operating condition	Means	Acceptance Criteria	Failure Assumptions	
1	DBC 1	Conservative design, implementation and operation to a high standard; maintaining the main operating parameters between the prescribed limits	For operating NPPS: Alpha void fraction = 0 (No boiling is allowed to occur in the reactor)	Conservative assumptions on failures; N+1(+1) (1 random failure, 1 system under maintenance)	
2	DBC 2	Control and safety protection systems; other surveillance methods	DNBR; Dose on the population <1 mSv		
3a	DBC 3	Safety systems, emergency operating procedures	cladding temperature <1200°C; cladding oxidation equivalent <17% (18% for the new reactor type) Averaged enthalpy in the cut set of the fuel <963 J/g for fresh fuel, 691 J/kg for fuel with a burn-up higher than 50 MW/kg; Chain reaction is under control		Dose on the population <1 mSv; Effective dose <10 mSv or 100 mGy thyroid dose
3b	DBC 4	Added safety features for the elimination of complex accidents, emergency operating procedures, on-site emergency response measures		Dose on the population <5 mSv; Effective dose <10 mSv or 100 mGy thyroid dose	No-single failure; Reasonably conservative assumption for failures
4	DEC 1	Supplementary safety features to limit fuel melting, accident management guidelines, on-site emergency response measures	CDF <1.00E-04/ry (1.00E-05/ry for New NPPs); LERF <1.00E-05/ry, earthquake not included (1.00E-06 for new NPPs, earthquake included) LERF ≈ CLI from the EUR for operating NPPs; LERF = CLI from the EUR for new NPPs	Best estimate assumptions	
5	DEC 2	On- and off-site emergency response measures; intervention levels			

STUK Response

	Plant state	Means	Acceptance criteria	Failure assumptions
Level 1 Prevention	Normal operation	Normal operation, Control systems	T _{clad} =T _{coolant} No melting 0,1 mSv	N+1
level 2 Control of AOOs	Anticipated operational occurrences, f >10 ⁻²	Control & limiting systems, Safety systems	N _{rods} , CHF < 0,1% N _{rods} / P _{noCHF} ≥ 95% (95% DP) 0,1mSv	N+1/N+2 for type 2 demonstration
Level 3 Control of accidents	Postulated Accidents 1:(10 ⁻² /a) > 10 ⁻³ /a 2:f < 10 ⁻³ /a	Safety systems (incl. Necessary supports)	1:650 °C, 1 mSv 2:10 fuel failure, 1200 °C, 5 mSv, 110 DP	N+2
	DEC A: CCF+AOO/Class 1 accident	Diverse systems and functions	No severe fuel dmg, 120% DP, 20 mSv	N+1
	DEC B: Failure combination DEC C: Rare external events	Varied means (depending on event)	No severe fuel dmg, 120% DP, 20 mSv	N+0
Level 4 Containment of severe release in a severe accident	Severe accidents	Severe accident systems	CDF < 10 ⁻⁵ /a LRF < 5*10 ⁻⁷ /a 100TBq CS-137	N+1
Level 5 Mitigation of consequences		Off-site emergency preparedness		

NDK Response

Regulation on Radiation Protection in Nuclear Facilities and Guide on Specific Design Principles define the levels and related dose limits as given in the table.

Level of defence (GSDP)	Plant state (GSDP)	Frequency (GSDP)	Criteria for maintaining integrity of barriers (IAEA SSR 2/1)	Criteria for limitation of radiological consequences (Draft NDK reg)
Level 1	Normal operation		No failure of any of the physical barriers except minor operational leakages	0.1 mSv
Level 2	Anticipated operational occurrence	$f > 10^{-2}/y$	No failure of any of the physical barriers except minor operational leakages	0.1 mSv
Level 3	Design basis accident	$10^{-2}/y < f < 10^{-4}/y$	No consequential damage of the RCS, maintaining containment integrity, limited damage of the fuel	5 mSv
Level 4a	Design extension conditions without significant fuel degradation	$10^{-4}/y < f < 10^{-6}/y$	No consequential damage of the RCS, maintaining containment integrity, limited damage of the fuel.	20 mSv
Level 4b	Design extension conditions with core melt (severe accident)	$10^{-6}/y < f < 10^{-7}/y$	Maintaining containment integrity both in an early as well as late phase, and practical elimination of fuel melt when the containment is disabled or by-passed	Radiological acceptance criteria ensuring that only emergency countermeasures that are of limited scope in terms of area and time are necessary.
Level 5	Accidents with releases requiring implementation of emergency countermeasures	$f < 10^{-7}/y$	Containment integrity severely impacted, or containment disabled or bypassed	Off site radiological impact necessitating emergency countermeasures.

Describe your expectations how far the analyses should be made in the analyzed cases. Describe the end states of the analysis (safe state/controlled state/steady state etc.)

SEC NRS Response

It should be demonstrated that at the end of the analysis power units is in safe, controlled state and this state can't be significantly changed due to minor changing of the one from the parameters and besides, there are no inevitable the threats of an exit from a controlled safe state. **p. 15.1.5 NP-006-16.**

Definition of the controlled safe state:

State when the main safety functions (subcriticality, cooling and localization) are fulfilled and this state cannot be significantly changed due to minor changing of the one from the parameters and besides, there are no inevitable the threats to get out of controlled safe state not related to equipment failures.

HAEA Response

In the case of DBA the analyses shall cover 72 hours based on the NSC requirement which states that the safe shutdown state shall be achieved in 24 hours and maintained for 72 (at least):

3a.2.2.5900. It shall be ensured by design solutions that following DBA2-4 operating conditions, the nuclear power plant unit reaches a controlled condition, then a safe shutdown condition as soon as reasonably possible. A controlled condition shall be achieved within 24 hours, at the latest, and a safe shutdown condition shall be achieved within 72 hours, at the latest.

NSC Volume 10 gives the regulatory definition of a safe shutdown state:

32. Safety shutdown condition: A state of the nuclear power plant unit following DBA2-4 and DEC1 operating conditions when the unit is brought into a subcritical state by active or passive safety systems or operator intervention, and the control of reactivity, heat removal from the active core and the spent fuel pool, keeping of release limits as well as load parameters within permitted values are ensured.

In practice DEC 1 is analyzed until it can be justified that the reactor reached safe shutdown state and can maintain it indefinitely or the given time is sufficient to safely remove the core from the reactor, therefore no core damage can occur.

In the case of DEC 2 the analysis should cover sufficient time period to prove that the calculated releases are within the acceptance criteria.

[STUK Response](#)

YVL B.3 402 Anticipated operational occurrences and accidents shall be analyzed starting from the initiating event and ending in a safe state.

YVL B.3 601 In the analyses of anticipated operational occurrences, postulated accidents and design extension conditions, it shall be shown that the reactor can be shut down and maintained in shutdown state, and that the plant can be brought to a controlled state and, thereafter, to a safe state. In addition, it shall be shown that the plant can, in the long term, be brought to a state where fuel removal from the reactor is possible.

For severe accident analyses, it is to be shown that the release as a whole is maintained within the acceptance criteria.

[NDK Response](#)

There is no explicit definition in Turkish or Russian Regulation (NP-006-98) for end state of these type of analysis.

Describe main requirements concerning safety injection filtering devices (sumps).

[SEC NRS Response](#)

NPP design shall ensure measures to exclude negative impact of the primary circuit thermal isolation on operability of safety systems (p. 3.3.6)

The design of the sumps have to ensure the availability of safety system, using these sumps, and also water purification, pumped to safety systems, from mechanical pollution, including the pollution caused by washout of pipe insulation in case of pipe break (p. 98, 99 NP-006-16).

[HAEA Response](#)

The requirement is set in NSC Volume 3a:

*3a.4.3.1400. The emergency core cooling system shall be so designed that it is capable of removing the residual heat for the necessary time. To achieve this, among others, the recirculation of the coolant discharged from the primary circuit to the reactor shall be ensured. During the design of the recirculation system, special attention shall be paid to the adverse effects of the solid and chemical contamination that gets into the discharged coolant. **In order to avoid that such contamination damages the recirculation system and the reactor, appropriately sized filtering equipment (sump filters) shall be installed. The suitability of the***

filtering equipment shall be demonstrated by experiment. During the design of the filters, the following shall be taken into account:

- *the ratio of contamination passing through or bypassing the filter should be sufficiently low as not to jeopardize the operation of the recirculation system and the efficiency of the cooling of the reactor;*
- *the drop in pressure caused by contamination and caught by the filtering equipment shall not prevent the operation of the recirculation system or significantly reduce its efficiency;*
- *it shall be ensured that the filtering equipment can be cleaned by reverse flow or gas injection in order to avoid its clogging.*

STUK Response

YVL B.1 requirement 5111:

The emergency core cooling system shall be designed to remove the decay heat produced in the reactor for as long as necessary. To achieve this, provisions shall be made to allow the recirculation of the leaked water back into the reactor. In the course of design, due consideration shall be given to any solid or chemical impurities that may be released into the water and impede water recirculation or impair reactor cooling. To control impurities, the coolant recirculation system shall be provided with filtering structures whose intended function and adequate performance is verified by tests. These tests shall be carried out in chemically representative conditions using representative aged insulation and coating materials. The design of the filtering structures shall take into account the following:

- The amount of impurities passing through the filters shall be low enough so as not to interfere with the operation of the coolant recirculation pumps or reduce the efficiency of reactor cooling;
- The pressure loss caused by the impurities trapped by the filtering structures shall not prevent the coolant recirculation system from performing as designed;
- It shall be possible to clean the filtering structures by means of a reversed coolant flow or gas blowdown if the pressure loss across the filters suggests a risk of excessive clogging.

NDK Response

There is no explicit definition in Turkish for sump design. But Russian Regulation has provisions such as:

In NP-001-97 4.4.4.6., it is stated that “provisions and methods shall be envisaged in the design for detecting leakage of primary coolant exceeding the value established in the design and, if possible its location. The automatic control of coolant radioactivity and control of discharges and releases of radioactive substances and control of radiation situation in NPP rooms, in the safe area and surveyed area during NPP operation including accidents and in a period of NPP decommissioning shall be incorporated in the design.”

And also NP-006-98 has special article in Section 12.2.3.4. Water collectors of sprinkler system pumps.

The following shall be indicated:

- what factors were taken into account in selection of the design and number of water collectors of sprinkler system pumps;

- that the water collector design includes protection against contamination, for example, filtering elements (maze-type multi-layer meshes, grids) and excludes loss of water under any operational mode of the NPP Unit;
- that water inventory in the water collector, design of its filtering elements and water intake equipment provides for simultaneous operation of all pumps of the sprinkler and other safety systems that are connected to this water collector without disruption of water supply taking into account delay in water return to the water collector from ALA premises within the whole post-accident period.

An experimental confirmation of operability of tank-sump (tank-sumps) or ponds in case of a breakage of thermal insulation from pipelines during an accident shall be presented. The amount of broken away insulation shall be justified. It is required to demonstrate how homogeneous composition of the solution in water collectors is maintained.

Additional information from SEC NRS

Presentation the results of accident analyses: what should be confirmed and justified by the results of accident analyses; the structure and content of the presented results.

DBA

The final DBA list (the scope of the analyses) should be complete and representative to demonstrate the compliance with design limits and safety criteria due to the inherent safety and due to safety systems. The list of the acceptance criteria should be presented in SAR. The results of the DBA analyses should confirm the capacity of the safety systems and to confirm the possibility to bring the power unit to safe controlled state when the main safety function (subcriticality, fuel cooling and localization of FP) is fulfilled.

Presentation of the results should include for each representative scenario at least:

- the cause of the IE;
- time schedule of the accident progress with the most important events and phenomena;
- description of the accident development with reference to the plots of the most significant parameters to demonstrate the compliance with design limits and safety criteria.

The minimal scope of the parameters should be presented and analyzed in SAR is defined in NP-0016-16 (appendix 8).

For the containment besides the compliance with the design limits (pressure and temperature) it should be demonstrated that deflagration of the flammable gases are excluded

BDBA

The BDBA list should be complete and representative to demonstrate the restriction of the radioactive consequences according to the installed requirements and to develop on the base of the results of BDBA analyses the guideline for BDBA management.

Presentation of the results of safety analyses for BDBA (without core melting) has the same structure and content as for DBA.

For the containment besides the compliance with the design limits (pressure and temperature) it should be demonstrated that detonation of the flammable gases are excluded.

For BDBA with core melting the scope of the investigating parameters enlarged to cover the behavior of the damaged core, reactor vessel, and containment, as for in-vessel and for ex-vessel

stage of severe accident. The minimal list of the parameters should be additionally presented and analyzed in SAR is presented NP-006-16 (appendix 8).

Subcriticality of the damaged core should be confirmed.

Additional information from NDK

Presentation the results of accident analyses: what should be confirmed and justified by the results of accident analyses; the structure and content of the presented results.

In Regulation on Specific Design Principles for Safety of Nuclear Power Plants Article 13 it is stated that “It is shown by documentation that basic safety matters are resolved to an adequate level before the construction of the plant; and that the remaining issues are to be resolved before plant operation.”

And in Russian Document NP-006-98

Section 15.5.3 defines that “Analysis of computation results - The information shall be presented for all phases of transient process or accident. The indication of process termination may be arrival at the stationary mode when the operation is carried out according to the design solution for normal operation or when at least one SS channel operates in the steady-state mode at the parameters of cool down equipment.”

And it is also stated in 15.5.4 that as a conclusion “Conclusions shall be made on the basic analysis results including identification of the most severe modes and bases for a conclusion statement regarding safe operation of the power unit under design basis accident.”

Appendix B

Task force 2 Verification, validation and uncertainty qualification of the computer codes used for safety analysis of VVER-reactors

Contents

COMPUTER CODES USED FOR MODELLING ACCIDENTS AND TRANSIENTS	31
1. Regulatory process for computer codes assessment	31
2. Verification and validation of computer codes.....	34
3. Uncertainty and sensitivity qualification in the safety analysis.....	36
4. Quality assurance of development, verification, validation and implementation of computer codes.....	38
5. Summary	39
APPENDIX A – QUESTIONS AND ANSWERS	40
1. REGULATORY PROCESS FOR CODES ASSESSMENT	40
1.1. Has the regulatory body established or adopted regulations and guidelines to specify the principles, requirements and associated criteria for verification, validation and uncertainty qualification of computer codes used for safety analysis of nuclear installations, upon which its regulatory judgements, decisions and actions are based?. 40	
1.2. Does the regulatory body perform review and assessment of relevant information for determining whether the computer codes are verified, validated and comply with applicable regulatory requirements? If so, please describe the procedure emphasizing the decision making process.....	43
1.3. Has the regulatory body made arrangements for analysis of code usage experience and for dissemination of the lessons learned?.....	45
1.4. Has the regulatory body established formal or informal mechanisms of communication with code developers and users on all verification, validation and uncertainty qualification related issues? How regulatory body ensure the traceability and transparency of the code assessment process?	46
1.5. Does regulatory body or its TSO has sufficient computer codes and qualified staff to carry out independent audit calculations?	47
1.6. How does the regulatory body review changes to the previously accepted computer codes?	47
2. VERIFICATION AND VALIDATION OF COMPUTER CODES	48
2.1. What are the main requirements for the verification of computer codes?	48
2.2. How does the regulatory body ensure that the computer code was properly verified?. 49	
2.3. Has the regulatory body established requirements to demonstration of the capability and credibility of a computer code for use in specific analysis application? What kind of tests should be performed for the validation process?.....	50
2.4. Has the regulatory body provided recommendations on identification of key phenomena, key parameters, and the range of parameter values associated with the range of code applicability?	50
2.5. Has the regulatory body provided recommendations on methods for extrapolation of validation results (the scalability of the integral effects tests)?	51

2.6. Has the regulatory body provided recommendations for quantification of accuracy of the code calculation?	52
3. UNCERTAINTIES QUALIFICATION	56
3.1. Has the regulatory body established requirements and guides to specify the uncertainty analysis process?	56
3.2. Has the regulatory body established acceptable level of <i>confidence and the reliability of the tolerance interval</i> ?	59
3.3. Has the regulatory body established requirements and guides on methods for uncertainty analysis of experimental data used for validation of computer codes?	59
3.4. Has the regulatory body established requirements to the statistical tools (like SUSA, DAKOTA etc.) used for uncertainty evaluation of the calculation results?	60
3.5. Has the regulatory body established requirements for <i>pseudorandom number generators used for uncertainty analysis</i> ?	61
4. USER EFFECT ISSUES	61
4.1. How does the regulatory body ensure that the qualification of computer codes users is sufficient to perform safety analysis?	61
4.2. Are theory and user manuals subjected to regulatory review and assessment?	62
5. OTHER QUESTIONS	62
5.1. Has the regulatory body made provision for establishing, maintaining and retrieving adequate quality assurance of computer codes used for safety analysis?	62
5.2. Does the regulatory body have adequate arrangements for creation, maintaining and using of databanks with verification and validation data?	62
5.3. Who makes the safety analyses? Who does the inspection of the analyses?	63

COMPUTER CODES USED FOR MODELLING ACCIDENTS AND TRANSIENTS

This paper summarizes the results of comparison of regulatory approaches of China, Finland Hungary and Russia to the computer codes used for safety analysis of nuclear power plants. The paper was prepared in the framework of Technical expert subgroup on accidents and transients analysis of MDEP VVER Working Group. The objective of the analysis was to identify common positions among the Regulatory Authorities and their TSOs reviewing the VVER accidents and transients simulation tools in order to:

- promote understanding of each country’s regulatory decisions and basis for the decisions for review and assessment of computer codes used for A&T;
- identify areas where harmonization and convergence of regulations, standards, and guidance can be achieved or improved;
- enhance communication among the members.

This would also help vendors and utilities to gain deeper understanding of regulatory expectations, related to the innovative computer programs which are being used for safety analysis of new reactor designs.

1. Regulatory process for computer codes assessment

In all participating member countries 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. Deterministic analyses of transients and accidents should be carried out using well-established

computer tools aiming to confirm that the overall plant design is capable of meeting the acceptance criteria. Any review of safety analyses performed by Regulatory Authority or its TSO should cover the methodologies and tools used for the safety cases.

Table 1. Main regulations of TESG AT member countries related to computer codes used for safety analysis

Country	Requirements	Guidelines
China	HAF 102, “Code on the Safety of Nuclear Power Plant Design” (2016). Requirements on the construction of critical structures, systems and components for nuclear power plant safety, and the requirements that must be met by procedures and organizational processes. It contains the general requirements for code used in safety analysis in HAF102.	HAD102-17-2006 “Safety Assessment and Verification for Nuclear Power Plants” principle requirements for the verification and validation of the computer codes. Guide “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, this guide is used for providing guidance for the development and application of Chinese independent and development, domestic nuclear power plant safety software. In this guide, the specific requirements are raised, including the scope of the safety analysis computer software, the method of the evaluation model development and assessment process, the verification and validation of the safety analysis computer software development, the quality assurance of the safety analysis computer software development, the application of the evaluation model and so on.
Finland	STUK regulation Y/1/2016 states that analytical methods employed to demonstrate compliance with the safety requirements shall be reliable, verified and qualified 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.	YVL B.3 states the guidelines for analysis methods
Hungary	Nuclear Safety Codes Volume 3a, Design requirements for new nuclear power plant units, which is the annex of the Government Decree No. 118/2011 (VII.11.) on the nuclear safety requirements of nuclear facilities and on related regulatory activities – requirements for verification, validation and uncertainty analysis	Recommendations related to V&V and uncertainty qualification: <ul style="list-style-type: none"> • N3a.32. Deterministic safety assessment for new NPPs • N3a.33. Severe accident analysis for new NPPs • N3a.46. Independent safety assessments for new NPPs
Russia	<ul style="list-style-type: none"> • Federal Law “On the use of atomic energy” (№ 170-FZ) - computer codes must undergone the review process performed by TSO of Regulatory Authority • Procedure on Review of Computer Programs used for development of the evaluation models of nuclear installations (approved by the act of Rostechnadzor № 325 July, 30th 2018) • RD-03-34-2000 “Scope and content of V&V Report” • NP-001-15 «The General safety provisions for NPP» - computer codes should be 	<ul style="list-style-type: none"> • RB-061-11. “Recommendations on Verification, Validation and Review of computer programs in the Field of Neutron Kinetic Calculations” • RB-074-12. “Recommendations on Comparison of the Calculated and the Measured Reactivity during Nuclear Safety Analysis of VVER-type Reactors” • RB-040-09. “Calculation Methodologies for Hydrodynamic and Thermal Characteristics of Components and Equipment of the Water-Cooled Nuclear Power Installations”

Country	Requirements	Guidelines
	verified, validated and approved by Regulatory Authority, safety analysis to be supported with an evaluation of errors and uncertainties of obtained results <ul style="list-style-type: none"> <li data-bbox="368 488 831 568">• NP-006-16 “Requirements to the content of the SAR for NPP unit with VVER-type reactor” 	<ul style="list-style-type: none"> <li data-bbox="863 360 1345 421">• RB-152-18 «Comments on The General safety provisions for NPP» <li data-bbox="863 432 1345 577">• RB-150-18 “Recommendations on the development of the list of beyond design basis accidents to be taken into account in the design of nuclear power plants with VVER-type reactors” <li data-bbox="863 589 1345 674">• Recommendations for assessment of calculation errors and uncertainties (is being developed) <li data-bbox="863 685 1345 741">• Analytical tests for verification of Risk Assessment (PSA) codes (is being developed)

Procedure

Every TEGS AT member country have adopted a legal background for regulatory assessment of computer codes which are going to be used for safety analysis calculations. In Finland and Hungary regulatory assessment of computer codes performed as part of the review of safety analysis.

In Russia, according to Federal Law “On the use of atomic energy”, computer programs have to undergo the review procedure before the beginning of Licensing process for nuclear installations. The review process is performed by SEC NRS (as TSO of Russian regulatory authority) which hosts the Expert Council for computer programs acting since 1991. The results of the review are presented in computer program certificate.

In China NNSA has established the formal mechanism of software assessment. Nuclear and Radiation Safety Center (NSC), as NNSA’s TSO undertakes and develops this work, and communicates specific matters with code developers. In December 2017, the institute of the check calculation and independent verification (the assessment center of the nuclear safety analysis software) has been established by NSC, the duties of the new institute include the checking calculations and tests verification for the nuclear and radiation safety field, the verification and validation of the nuclear and radiation safety analysis model etc. For imported code, NNSA requires the approval of the code from the nuclear safety regulatory of export country and the explanation about the applicability in the project.

Periodical review

TEGS AT members believes that it should be ensured that the codes are periodically evaluated and updated, as necessary, to reflect lessons learned and the latest knowledge. More specifically, if a prior analysis relied on an old evaluation model, the licensee should be asked to update their model in case of more advanced models are available (such approach can be a part of the regulatory driven innovation in safety analysis technologies). It also should be ensured that evaluation model adequately reflects all modifications of nuclear installation, such as new fuel types and power uprates of NPP, new safety systems, ageing issues etc.

In Russia every 10 years the results of computer codes verification and validation are reassessed to ensure that the computer code and its V&V are still up to date and comply with modern safety requirements, recommendations and latest V&V research, as well as relevant modern experimental data taken into account by code developers. Moreover, the code usage experience is also subject for the review and reassessment.

NNSA has not arranged for the analysis of code usage experience or dissemination of the lessons learned which are responsible for the applicants. But in “People’s Republic of China Nuclear Safety

Law”, the 35th Article is required “the relevant departments of the State Council should establish the nuclear safety experience feedback system, and deal with the information of nuclear safety report in time to realize information sharing”. Meanwhile, in section 4.3.9.1.2 and section 4.6.7 of the guide “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006), the users are requested to be trained and adequate experience as the principle requirements.

In Finland state of the art code development issues are, in among other instances, dealt within SAFIR research program (The Finnish Research Program on Nuclear Power Plant Safety). STUK actively promotes TSOs to follow the developments in mentioned areas.

Review and Assessment of changes

The impact of any changes to the computer program would have to be evaluated. Despite in TSEG AT member countries there are no formal procedure for evaluation of such changes, it happens on case by case basis.

In China according to the requirements of the section 3.8 of the guide “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, for some evaluation models submitted for review with minor modifications to existing evaluation models, the differences of the codes should be focused by the review experience, and the reviewers should consider the following four attributes of the evaluation model when determining the extent to which the full model development process may be reduced for a specific application, as described in the following subsections:

- novelty of the revised evaluation model compared to the currently acceptable model,
- complexity of the event being analyzed,
- degree of conservatism in the evaluation model,
- extent of any plant design or operational changes that would require reanalysis

Audit calculations as part of the review process

In some cases, the regulatory body may decide to perform a limited number of audit calculations to check that the Licensee has justified a particular aspect of safety correctly. Audit calculations help to Identifying possible weaknesses in the Licensee’s safety case and provide the basis for gaining confidence in regulatory decision-making process. Regulatory authorities should aim to have capabilities to perform deterministic and probabilistic audit calculations and relevant experience with other regulatory bodies worldwide.

STUK has capabilities to perform comparative in-house analyses with short notice to support decision making. For example, APROS code with Loviisa NPP and Olkiluoto 3 models is available. Finnish TSOs have capability and personnel to perform analysis.

NNSA is also capable to perform audit calculations on their own. HAEA ask TSOs to perform audit calculations. SEC NRS experts perform independent calculations within the frame of regulatory review upon request of Rostechndzor. About 50 computer programs are available to SEC NRS experts. In most cases, these computer programs are alternative to computer programs used by Applicant or Licensee.

2. Verification and validation of computer codes

Verification

The following concept of computer program verification as basic step of code development established in regulatory requirements. Only properly verified codes can be used for deterministic safety analysis. The verification of the code is the responsibility of the code developer. The code developer shall document the verification.

In Hungary the licensee shall present the process which is used to approve the manual. The manual is made by the organization which performed the analysis and it describes they derived the data for the models from the safety reports. The regulatory body does not perform tests, only inspects the manual.

In Russia analytical tests as a means of a code verification are obligatory for a code model quality assessment. SEC NRS also performs test runs of computer programs as part of the review. It should be demonstrated that the calculational results presented in V&V report were obtained with the “frozen” version of the computer program submitted for the regulatory review.

In China, chapter 4 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)” states that verification and validation are a support process in the life cycle of evaluation models, including demands analysis, design, coding implementation, testing, evaluation, operation and maintenance activities. Verification and validation (V&V) activities are divided into demands V&V, design V&V, coding implementation V&V, testing V&V, evaluating V&V, running V&V, and maintaining V&V and so on. According to the section 4.1.2 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the procedure of evaluation model development and evaluation the applicability assessment also constitute a process of verification and validation as a whole. And the quality of evaluation models is controlled by quality assurance program or software quality assurance plan, described in section 5.1.1.1 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”. In addition, the above evaluation report should be provided during the review of code applicability.

In Finland, it is the task of the code user to make sure that the code used is properly verified. STUK makes its review to assess that the regulation given in section 3 of Radiation and Nuclear Authority Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2016) is fulfilled: the analytical methods employed to demonstrate compliance with the safety requirements should be reliable, verified and qualified for the purpose. For codes used by TSO, STUK or other stakeholders, documented verification is part of the code development process.

Validation

General requirements for the validation of the computer codes are mostly the same in each TESS AT member country. The validation report shall present evidence that for the specific reactor type and for the different processes computer program provides accurate results (based on comparison with validation basis). The comparison of the experimental data and calculations shall be based on proper statistical methods. The validation report shall define the scope (operational states, environmental conditions etc. where the code is valid). Validation of the computer codes is performed using:

- analytical tests;
- experimental data which describe the particular processes and phenomena (local experiments);
- data obtained at the experimental facilities structurally similar to the real nuclear facilities (integral experiments);
- experimental data obtained at the real nuclear power plants;
- comparison with models that have already been validated;
- standard problems and/or numerical benchmarks with sufficiently accurate results being obtained.

PIRT

One of the main steps of validation is the selection of evaluation model phenomena and parameters which significantly affect the code output, and their statistical characterization. There are no regulatory recommendations on identification of key phenomena, key parameters, and the range of parameter values associated with the range of code applicability in Russia, Finland and Hungary (case by case basis). However, in China In chapter 3 and chapter 6 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the principle guide requirements related to the recommendations on identification of key phenomena, key parameters and the range of parameter values have been provided. Before the release of this guide, in a number of projects (such as AP1000, EPR, HPR1000 and etc.) review, NNSA has raised many review questions on the applicability of the key phenomena, key parameters, and the range of parameter values, and put forward the review requirements about some of the problems.

Extrapolation of validation results beyond the validation domain

One of the hardest issue of assessment of computer codes adequacy is related to the extrapolation of validation results beyond the validation domain. Code validation on integral experiments should be supported with the justification of that the experimental facility is similar to the nuclear power plant, and the impact of the dissimilation between the experimental facility and the real one should be evaluated through scaling factors analysis (the scalability of the integral effects tests). Regulatory requirements on demonstration of scaling capabilities of computer program are established in each TESH AT member country. For instance, in section 3.4.3 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the requirement is that it is necessary to further prove that the conclusion related to the code capacity could be extended to the prediction of specific transient behavior of a special nuclear power plant through the comparison of the calculated results and the experimental data. In section 3.5.4.3, the requirement is that extended applications which exceed the original basis, application scope, accuracy of any physical model need to be justified. In addition, in section 5.8.4.7, the requirement is that the extrapolation method of model, correlation and the criteria need to be described and proved. However, regulatory guidelines in TESH member country on how to fulfil these requirements are still missing. The US NRC Regulatory Guide 1.203 “TRANSIENT AND ACCIDENT ANALYSIS METHODS” can be recommended as a best practice for the Evaluation Model Development and Assessment Process.

3. Uncertainty and sensitivity qualification in the safety analysis

Every TESH AT member country has established regulatory requirements according to which uncertainty and sensitivity analysis shall be performed and taken into account in the results of the safety analysis and the conclusions drawn from it.

In Hungary for uncertainty analysis the following requirements must be followed:

- *3a.2.2.3700. When defining the design basis, reasonably conservative assumptions shall be applied to compensate for uncertainties.*
- *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.*
- *3a.2.2.7300. In order to minimize 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.*

- *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;*
- *3a.2.3.0400. 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.*

The HAEA recommendations are the following:

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 modeling
- Approximate nature of the numerical solution
- Effect of the hardware and compiler
- User effect (nodalization, time steps etc.)
- Scale effect

Russian regulations require to perform uncertainty analysis for results of safety analysis calculations. For NPP such requirement constrains NP-001-16, according to it:

- The deterministic analysis of the design accidents should be done on the base of the conservative approach
- analysis of the beyond design accidents should be done on the base of the realistic approach
- The evaluation of the calculation result errors and uncertainties should be taken into account in the framework of the NPP safety analysis

Moreover, the uncertainty analysis has to be demonstrated in V&V report of each computer code (according to RD-03-34-2000). Russian regulatory recommendations on the UQ methods are currently under development.

In Finland YVL B.3 410 states: Utilization 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.]. Also, for conservative analyses need to assess uncertainties. 424. In severe accident analyses, application of the best estimate method need not be complemented with an uncertainty analysis as required in para 410.

In China section 4.5 of “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006) established the principle requirements for sensitivity and uncertainty analysis. In section 6.3.6 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the requirements for sensitivity and uncertainty analysis are more detailed, including the sources of uncertainty, the effect of the uncertainty by the range and the probability distribution of parameters, the uncertainty evaluation of the calculated results under acceptable confidence, the codes uncertainty derive from the data comparison between integral effects experiment and the different scales of separate effects experiment, the comparison between calculation and experimental data of all the important parameters, the evaluation of uncertainty in the phenomena or physical processes of different periods of time, the uncertainty evaluation

derived from the experimental results obtained by a small scale tests used to analyze large scale objects and so on.

Level of confidence and reliability

Uncertainty analysis utilizes static methods which required to set up levels for confidence and reliability. Hungary and Russia haven't regulatory requirements for acceptable level of confidence and reliability. In China, the acceptable level of confidence and the reliability of the tolerance interval are different in different objects, which can be determined by the review experience. HAD102 requires that the safety analysis results should ensure that the nuclear power plants run according to the design with a high confidence level. In Finland when applying a best estimate method with uncertainty analysis, the result is acceptable if there is a 95% probability with 95% confidence that the examined parameter will not exceed the acceptance limit set for the conservative analysis method (YVL B.3 602).

Statistical tools

There are a number of statistical tools (like SUSA, DAKOTA etc.) used for uncertainty evaluation of the calculation results. There are no specific regulatory requirement concerning such tools. However, the suitability of analysis methods for their purpose shall be justified.

Evaluation of the experimental data

Uncertainty of the experimental data used for validation of computer codes may come from the measurement error, the experimental distortion and the other aspects of the experiment. If the uncertainty of the experimental data is too large relative to the evaluation model, these data or correlations cannot be used. The uncertainty of experimental data itself (including measurement error, experimental distortion and so on) needs to be clearly reported in the experimental documents, and its uncertainty is appropriate.

4. Quality assurance of development, verification, validation and implementation of computer codes

Quality management for development, verification, validation and implementation of computer codes is required and its use should be inspected.

China In chapter 5 of "Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)" has detailed requirements on quality assurance program, document control, configuration management, tools evaluation, corrective actions, third-party assessments, plans for the development and assessment process, the evaluation model development documents and so on.

In Hungary for the codes used for analysis at least the following steps shall be presented:

- The code description made for the developers and users
- The verification and validation activities and documentation of these
- Validation and code description development and documentation, issue of new versions

For new code version there must be new code description (the developer shall ensure this by proper quality assurance). The quality assurance system shall define the documentation requirements. The documentation related to quality assurance shall be accessible by the regulatory body. The licensee shall present the approval process of the safety analysis reports. The regulatory body shall assess and review the code manuals.

According to Finnish regulations (YVL A.4 305) the licensee shall define the requirements for any tasks and work important to safety carried out by suppliers at the nuclear facility, as well as supervise and approve such tasks and work. This is assessed with inspections and audits.

In Russia quality assurance of development, verification, validation and implementation of computer codes should be established by the Licensee in Quality Assurance Program (NP-090-11). According to Procedure on Review of Computer Programs, code user qualification should be assessed by the code developer or by the organization, which holds the computer code certificate.

Good practices on user effect reductions can be found at the following NEA reports:

- NEA/CSNI/R(1998)22 Good Practices for User Effect Reduction
- NEA/CSNI/R(1994)35 User effects on the transient system code calculations.

5. Summary

A comparative analysis of regulatory practices of TESG AT member countries shows general resemblance of the approaches for computer codes assessment. A computer code's capabilities, limits and accuracy, including the area of applicability to which the code provides adequate simulation results, should be gained from validation and accepted by the Regulatory body experts. Validation should be made using the evaluated experimental data (uncertainty of measurements and scaling factor should be taken into account). It should be ensured that the codes are periodically evaluated and updated, as necessary, to reflect lessons learned and the latest knowledge.

According to regulatory requirements of TESG AT members validation should be aimed on identification of all the uncertainties associated with the evaluation model. It should be demonstrated that the uncertainties of evaluation model are taken into account in the safety analysis of nuclear installations.

TESG AT members agreed that uncertainty analysis can only be made in the framework of well-defined, verified and validated model having well-defined safety objectives. In practice, the choice of UA method is case dependent (UA could be made in different ways). Some countries have general regulatory requirements to perform UA for all safety analysis calculations. It seems unnecessary to perform statistical UA for every single calculation in the SAR.

Main steps of any statistical uncertainty analysis methods (such as Identification of the evaluation model parameters for certain accident scenario, carrying out of the variant calculations, the statistical evaluation of the code calculation results) could be supported with regulatory recommendations. STUK recommendation on level of confidence and reliability (95/95) and Chinese recommendation on PIRT methodology could be considered as a good practice of such recommendations, worth to implement in other countries.

Chinese detailed requirements for quality assurance for development V&V and usage of computer codes can also be considered as a good practice.

Potential areas for enhancing regulatory effectiveness and efficiency could be focus on development of:

- recommendations on specific UA methods;
- criteria for assessment and recommendations on verification and validation for new generation of computer programs (based on machine learning techniques, CFD-DNS etc.).
- recommendations on Extrapolation of validation results beyond the validation domain;
- criteria for evaluation of special computer tools used for UA (such as DAKOTA, SUSAN, PANDA etc.) are missing;
- recommendations on user effect reduction.

APPENDIX B.1 – QUESTIONS AND ANSWERS

1. Regulatory process for codes assessment

1.1. Has the regulatory body established or adopted regulations and guidelines to specify the principles, requirements and associated criteria for verification, validation and uncertainty qualification of computer codes used for safety analysis of nuclear installations, upon which its regulatory judgements, decisions and actions are based?

SEC NRS Response

According to the Federal Law “On the use of atomic energy” (№ 170-FZ), calculational models used for safety analysis should be built with computer programs which have undergone the review procedure established by Regulatory Authority. Review procedure is to be carried out by technical support organization of Regulatory Authority.

Main goal of the review is to determine the availability of the computer program to perform simulation of different processes and/or conditions of nuclear installations with the reasonable accuracy.

Computer codes verification and validation report (in further text - V&V Report) is considered as the fundamental document, which demonstrates the declared characteristics of computer program submitted for certification. Main goal of the certification is to determine the availability of the computer program to perform simulation of different modes, processes and/or conditions of nuclear installations with the reasonable accuracy. The requirements to V&V Report scope and content are specified in [RD-03-34-2000](#). Moreover, provisions of the following Safety Guides are recommended to be applied in the course of verification and validation:

- RB-061-11. “Provision on Verification, Validation and Review of Software in the Field of “Neutron Kinetic Calculations”;
- RB-074-12. “Provision on Recommendations to Comparison of the Calculated and the Measured Reactivity during Nuclear Safety Analysis of WWER-type Reactor Installations”;
- RB-040-09. “Design Ratio and Calculation Methodologies for Hydrodynamic and Thermal Characteristics of Components and Equipment of the Water-Cooled Nuclear Power Installations”;
- RB-075-12. “Calculation Ratio and Calculation Methodologies for Hydrodynamic and Thermal Characteristics of Components and Equipment of Nuclear Power Facilities with Liquid-Metal Coolant”.
- Recommendations for assessment of calculation errors and uncertainties (is being developed)
- Analytical tests for verification of Risk Assessment (PSA) codes (is being developed)

Federal nuclear safety regulations require safety analysis to be supported with an evaluation of errors and uncertainties of obtained results. For nuclear power plants, this requirement is set out in para 1.2.9 in NP-001-15 “The General safety provisions for NPP”.

Computer code calculation errors are to be argued in a code verification and validation report, which should satisfy the requirements provided in RD-03-34-2000. However, RD-03-34-2000 does not explain how exactly the code-calculated parameter errors should be evaluated; nor does it suggest how the error values obtained as a result of code validation should be used in a safety analysis of a nuclear facility (NF).

Figure 1 outlines the process for building a rationale for the values of code output errors.

Code calculation error implies code output deviation from the measurement data obtained in the course of code validation experiments. Considering any measurements results to feature uncertainty, one of the code calculation error components is *the uncertainty of experimental measurements* (section 3 in this paper). Meanwhile, RD-03-34-2000 requires obligatory justification of the sufficiency of experimental data used for code validation, in addition to measurement uncertainty evaluation.

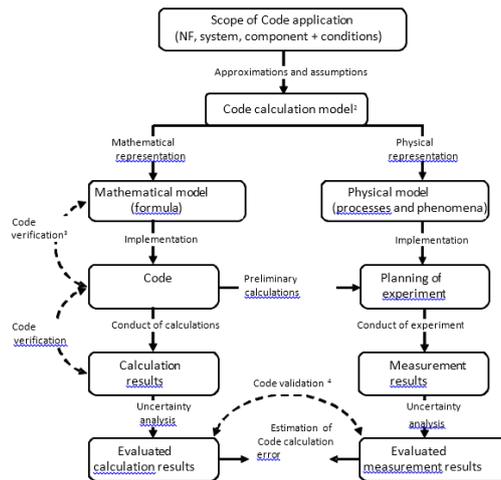


Figure 1. Process of code output error evaluation

Another code calculation error component is *uncertainties caused by the assumptions and simplifications adopted to build the code calculation model, including the uncertainties associated with model nodalisation (control volumes, finite elements, etc.), selection of numerical solution technique and integration step for the code's set of equations, and the adverse effect of an unskilled code user*. According to RD-03-34-2000, these uncertainties analysis and minimisation area prerequisite for code model correctness; however, the paper does not discuss the methods and ways to analyse and minimise these uncertainties. In fact, the detailed discussion covers only two error components of code output: uncertainty of code calculation model parameters that have statistical nature, and uncertainty of experimental measurements. Code calculation uncertainty caused by the uncertainty of code calculation model parameters is defined as code output variation due to uncertainty of model parameters having statistical nature such as physical and chemical properties of materials, geometry, empirical equation coefficients underlying the code calculation model, etc. This uncertainty is presented as an interval (range) with certain probabilistic characteristics.

HAEA Response

The HAEA established requirements and regulatory guides as well for verification, validation and uncertainty qualification. The related requirements can be found in the Nuclear Safety Codes Volume 3a, Design requirements for new nuclear power plant units, which is the annex of the Government Decree No. 118/2011 (VII.11.) on the nuclear safety requirements of nuclear facilities and on related regulatory activities. The most important related requirements is the following:

3a.2.3.0100. The design and analysis tools, models and model parts used for the demonstration of the fulfilment of the general safety requirements of the design basis as well as the input data shall be verified and validated. The validation of the analysis tools shall be presented on the basis of appropriate internationally available data, i.e. experimental results. The verification of the

analysis models shall also be performed by a person or workgroup independent of the person or workgroup performing the analysis or design.

The following regulatory guides are providing detailed and specific recommendations related to V&V and uncertainty qualification:

N3a.32. Deterministic safety assessment for new NPPs

N3a.33. Severe accident analysis for new NPPs

N3a.46. Independent safety assessments for new NPPs

[NSC Response](#)

HAF 102, “Code on the Safety of Nuclear Power Plant Design” (2016). Requirements on the construction of critical structures, systems and components for nuclear power plant safety, and the requirements that must be met by procedures and organizational processes. It contains the general requirements for code used in safety analysis in HAF102.

In the guide “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006) released by NNSA, there are the principle requirements for the verification and validation of the computer codes used for safety analysis, such as in section 4.3.2.9 of the guide, there is a general requirement for the computer codes used for the design basis accidents analysis. And in section 4.3.3.3.3, for the severe accidents analysis codes, the entire prospective phenomenon should be analyzed sufficiently. Also, in section 4.6, there are some principle descriptions for the assessment of the used computer codes. Section 5.8 mentioned that for the initial used codes, the verification and validation are still not completed. In that case, the accuracy of the codes should be verified by the authorized codes.

In Dec. 20 2017, NNSA has released the new guide “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, this guide is used for providing guidance for the development and application of Chinese independent and development, domestic nuclear power plant safety software. In this guide, the specific requirements are raised, including the scope of the safety analysis computer software, the method of the evaluation model development and assessment process, the verification and validation of the safety analysis computer software development, the quality assurance of the safety analysis computer software development, the application of the evaluation model and so on.

[STUK Response](#)

STUK does not licence/certify computer codes used for accident analyses.

STUK regulation Y/1/2016 states:

3. The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear power plants, the results of safety research, plant modifications, and the advancement of calculation methods.

4. The analytical methods employed to demonstrate compliance with the safety requirements shall be reliable, verified and qualified 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.

YVL B.3 states the following guidelines for analysis methods:

The suitability of analysis methods for their purpose shall be justified.

A description of the models and calculation methods used in the analyses shall be given. The models shall be described to a level of detail that facilitates conducting of verifying analyses. The

information to be presented shall include the analysis model representing the plant or its component (e.g. the division into nodes applied in the model), justification of the selected model parameters as well as the plant data used for the analyses or a reference to the source of the available plant data.

The validation of the physical models and computer code used for the analyses shall be substantiated by comparing their calculation results to separate 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 utilized.

The plant and fuel type specific experimental correlations used in the calculation methods shall be justified by presenting the measurement data from which the correlations have been derived. If the correlation is commonly known and the measurement data are publicly available, a bibliographic reference is sufficient.

If reliable calculation methods are not available, the acceptability of the technical solution in question shall be justified by means of experiments.

1.2. Does the regulatory body perform review and assessment of relevant information for determining whether the computer codes are verified, validated and comply with applicable regulatory requirements? If so, please describe the procedure emphasizing the decision making process.

SEC NRS Response

Russian Federal Safety Regulations in the field of atomic energy require the computer programs used for the safety analysis of nuclear facilities and (or) activities in the field of atomic energy use to be certified.

The computer programs certification is performed to approve the possibility of computer program application for modelling and simulation of processes, which may have impact on nuclear facilities safety.

Computer programs certification shall be carried out in compliance with Procedure on Review of Computer Programs established by the Russian Nuclear Regulatory Authority (Rostechnadzor).

According to Procedure on Review of Computer Programs the computer programs certification is performed by SEC NRS and the Expert Council (hereinafter to be referred to as “the Council”) operating since 1991 under the aegis of Rostechnadzor. The Council has technical expert boards for the following thematic areas of simulation and modeling:

- neutron kinetic;
- thermal hydraulics, heat transfer and multiphysics;
- radiation protection and radiation safety;
- structural mechanics;
- civil structures;
- chemical and physical processes;
- probabilistic safety analysis calculations.

The Expert Council and its technical expert boards include leading specialists from sixty scientific and technical organizations operating in the nuclear power industry, as well as representatives of the regulatory authority body and its TSO, national research centers, leading higher education institutions and institutions of the Russian Academy of Sciences. According to Rostechnadzor directive the Council activity management is performed by SEC NRS, which has the status of Rostechnadzor scientific and technical support organization.

HAEA Response

The HAEA does not have a separate certification/licensing process for codes, however the licensee has to attach the “code description” which consists of: Name, version, modeling considerations, conditions, structure of code and models, input and output, validation report, therefore the codes used are reviewed as part of SAR.

NSC Response

In the history of nuclear power plant projects review, NNSA has paid attention to the verification and validation of computer codes. Before the new guide “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)” released, the review of the safety analysis codes is mainly in accordance with “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006). In the review of specific projects, according to the review experience, the applicants are required to provide detailed information including: the information for the code version (such as the version differences between the used codes and the codes verified and validated by the previous tests as well as the influence due to the differences), the practical experience of the codes (e.g. for the imported codes, should be demonstrated the approval by other countries’ nuclear safety regulatory authorities), the codes assessment report (such as the applicants should demonstrate the impacts on the key physical phenomenon by the design, whether these impacts could be enveloped by the previous tests verification), and so on. If necessary, NNSA will organize a special review for the code applicability.

STUK Response

Analyses and their used codes are reviewed as part of SAR.

YVL B.3 states: *The preliminary safety analysis report shall present the calculation methods for transient and accident analyses and their validation, as well as the preliminary transient and accident analyses demonstrating the acceptability of the systems’ technical solutions.*

The final safety analysis report shall present the calculation methods for transient and accident analyses and their validation, as well as the final transient and accident analyses demonstrating the acceptability of the systems’ technical solutions.

1.3. Has the regulatory body made arrangements for analysis of code usage experience and for dissemination of the lessons learned?

SEC NRS Response

Every 10 years the results of computer codes verification and validation are reassessed to ensure that the computer code and its V&V are still up to date and comply with modern safety requirements, recommendations and latest V&V research, with relevant modern experimental data taken into account by code developers. Moreover, the code usage experience is also subject for the review and reassessment.

HAEA Response

There are no arrangements taking into consideration the experiences and lessons learned.

NSC Response

At present, NNSA has not arranged for the analysis of code usage experience or dissemination of the lessons learned which are responsible for the applicants. But in “People’s Republic of China Nuclear Safety Law”, the 35th Article is required “the relevant departments of the State Council should establish the nuclear safety experience feedback system, and deal with the information of nuclear safety report in time to realize information sharing”. Meanwhile, in section 4.3.9.1.2 and section 4.6.7 of the guide “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006), the user is requested to be trained and have adequate experience as the principle requirements.

STUK Response

Not as such, but code development issues are, in among other instances, dealt within SAFIR research programme (The Finnish Research Programme on Nuclear Power Plant Safety).

1.4. Has the regulatory body established formal or informal mechanisms of communication with code developers and users on all verification, validation and uncertainty qualification related issues? How regulatory body ensure the traceability and transparency of the code assessment process?

SEC NRS Response

All relevant information regarding code assessment process, regulatory requirements and recommendations on V,V & UQ, as well as main results of codes assessment is publicly available on the SEC NRS website. Decisions made by the Expert Council and its Technical boards are logged with records to be approved by the Chairman of the Expert council and registered in SEC NRS document management system. The Expert Council can initiate a broader discussion on V, V & UQ issues for particular computer codes. Such discussions can be brought up in the form of special workshops or conferences followed by the certain action courses recommendations. For instance, in 2017 the Expert council thermal hydraulic board initiated a series of workshops related to uncertainty analysis methods. The results were published in the SEC NRS journal and can be used for preparing the draft of regulatory guideline on uncertainty analysis methods.

HAEA Response

The HAEA is not communicating directly with the developers, however there are inspections aiming to supervise the users at the licensee. The code assessment process is not transparent, however the main resolutions and decisions can be accessible at the website of the HAEA.

NSC Response

NNSA has established the formal mechanism of software assessment. Nuclear and Radiation Safety Center (NSC), as NNSA's TSO undertakes and develops this work, and communicates specific matters with code developers. In December 2017, the institute of the check calculation and independent verification (the assessment center of the nuclear safety analysis software) has been established by NSC, the duties of the new institute are the checking calculations and tests verification for the nuclear and radiation safety field, the verification and validation of the nuclear and radiation safety analysis model, the verification of the functions and reliability of the nuclear and radiation safety important devices and systems and so on, meanwhile, the new institute also takes part in the activities of international communication and popular science propaganda.

For the traceability of the codes assessment, the requirements are mentioned definitely in the chapter 5 "the quality assurance of the safety analysis computer software development" of the guide "Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)". For instance, the requirement in section 5.1.3.6 is for maintaining the traceability of the changes in software design demands. The requirement in section 5.1.3.8 is that the documents generated during the stages of development and evaluation should be auditability, traceability and conformity with the standards. The evaluation models developed should be provable credibility, testable and maintainable, and must be verified and validated. In section 5.1.3.10, the requirement is the designs and the codes should be traceable, also demands, design and code tests should be traceable.

NNSA ensures the effective communication with applicants through the mechanism of software assessment, ensures the transparency of the review reach an agreement. However, due to the protection of intellectual property rights, the process of code assessment has not been opened to the public.

STUK Response

STUK actively promotes TSOs to follow the developments in mentioned areas for example through SAFIR programme (The Finnish Research Programme on Nuclear Power Plant Safety).

1.5. Does regulatory body or its TSO has sufficient computer codes and qualified staff to carry out independent audit calculations?

SEC NRS Response

According to the article 3.38 of IAEA Safety Guide GS-G-1.2 “Review and Assessment of Nuclear Facilities by the Regulatory Body” Regulatory Body (or its Technical Support Organisation) staff can conduct independent (audit) calculations during review of safety analysis to make sure of:

- lack of weaknesses in safety analysis;
- conservative approach in safety analysis;
- proper choice of initial data, specialized data libraries (for example evaluated neutron data) and data basis (for example on material properties).

Almost 50 computer codes are available to SEC NRS experts for checking calculations during the safety review.

HAEA Response

There are not enough competence and resources to carry out independent calculations by the regulatory body itself. For the existing NPP units our TSOs carry out calculations for specified cases. For the new NPP it will be a challenge because of the low number of TSOs in the area of safety assessments already used by the licensee.

NSC Response

NSC is NNSA's TSO. The number of staff of the institute of the check calculation and independent verification (the assessment center of the nuclear safety analysis software), which belongs to NSC, are 45. The staff have experiences of independent checking calculation for many years.

At present, NSC has acquired supervision software authorized by U.S. NRC and a variety of design software, coverage the majors of radiology analysis, neutron physics calculation, fuel behavior analysis, thermal hydraulic analysis, structure analysis, accident analysis, radiological consequences analysis, probabilistic safety analysis and so on. NSC is also currently developing new supervision software.

STUK Response

STUK has capabilities to perform comparative in-house analyses with short notice to support decision making.

For example, APROS code with Loviisa NPP and Olkiluoto 3 models is available.

TSOs have capability and personnel to perform analysis.

1.6. How does the regulatory body review changes to the previously accepted computer codes?

SEC NRS Response

If changes were caused by the significant modifications of nuclear installations (like new fuel types, power uprate etc.), regulation RD-03-34-2000 requires the additional validation of the evaluation model in order to demonstrate that the code adequately simulates the nuclear installation behavior with the modifications taken into account. If the computer code changes are minor (user interface modification or bug fixing) the code developer demonstrates code changes not to affect the code evaluation model.

HAEA Response

The HAEA does not license computer codes separately, so it happens case by case.

NSC Response

According to the requirements of the section 3.8 of the guide “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, for some evaluation models submitted for review with minor modifications to existing evaluation models, the differences of the codes should be focused by the review experience, and the reviewers should consider the following four attributes of the evaluation model when determining the extent to which the full model development process may be reduced for a specific application, as described in the following subsections:

- novelty of the revised evaluation model compared to the currently acceptable model,
- complexity of the event being analyzed,
- degree of conservatism in the evaluation model,
- extent of any plant design or operational changes that would require reanalysis.

For the import software, it is concerned with the experience of practice and the approval of other countries’ nuclear safety regulatory authorities. If necessary, NNSA will organize the special code applicability review.

STUK Response

As STUK does not license codes it is up to the code user and to the licensee to make sure that the changes are correctly applied. Supportive documentation (code manuals) shall be kept up to date.

2. VERIFICATION AND VALIDATION OF COMPUTER CODES

2.1. What are the main requirements for the verification of computer codes?

SEC NRS Response

RD-03-34-2000 requires V&V report to contain demonstration of the code meeting its requirements specifications. Analytical tests as a means of a code verification are obligatory for a code model quality assessment. They demonstrate the computer code to correctly implement the equations and closing correlations used in the code, the adequacy of evaluation model nasalization and selection of integration step, etc. However, the difference between the code output and the analytical tests results cannot be adopted as a modeling error in a safety analysis of a nuclear facility because analytical tests do not describe a real nuclear facility.

HAEA Response

Only properly verified codes can be used for deterministic safety analysis. The verification of the code is the responsibility of the code developer. The code developer shall document the verification. The licensee shall present the process which is used to approve the manual. The manual is made by the organization which performed the analysis and it describes they derived the data for the models from the safety reports.

NSC Response

In chapter 4 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, for the new development computer codes, verification and validation are a support process in the life cycle of evaluation models, including demands analysis, design, coding implementation, testing, evaluation, operation and maintenance activities.

STUK Response

Radiation and Nuclear Authority Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2016) states in section 3:

3. The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear power plants, the results of safety research, plant modifications, and the advancement of calculation methods.

4. The analytical methods employed to demonstrate compliance with the safety requirements shall be reliable, verified and qualified 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.

YVL B.3 404 states: The models shall be described to a level of detail that facilitates conducting of verifying analyses. The information to be presented shall include the analysis model representing the plant or its component (e.g. the division into nodes applied in the model), justification of the selected model parameters as well as the plant data used for the analyses or a reference to the source of the available plant data.

YVL B.3 405 states: The validation of the physical models and computer code used for the analyses shall be substantiated by comparing their calculation results to separate 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 utilized.

2.2. How does the regulatory body ensure that the computer code was properly verified?

[SEC NRS Response](#)

SEC NRS performs the computer code tests.

[HAEA Response](#)

The regulatory body does not perform tests, only inspects the manual.

[NSC Response](#)

Verification and validation (V&V) activities are divided into demands V&V, design V&V, coding implementation V&V, testing V&V, evaluating V&V, running V&V, and maintaining V&V and so on. According to the section 4.1.2 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the procedure of evaluation model development and evaluation the applicability assessment also constitute a process of verification and validation as a whole. And the quality of evaluation models is controlled by quality assurance program or software quality assurance plan, described in section 5.1.1.1 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”. Also the above evaluation report should be provided during the review of code applicability.

[STUK Response](#)

It is the task of the code user to make sure that the code used is properly verified. STUK makes its review to assess that the regulation given in section 3 of Radiation and Nuclear Authority Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2016) is fulfilled:

3. The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear power plants, the results of safety research, plant modifications, and the advancement of calculation methods.

4. The analytical methods employed to demonstrate compliance with the safety requirements shall be reliable, verified and qualified 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.

Due to the use of independent analysis some grading can be applied.

For codes used by TSO or STUK, documented verification is part of the code development process.

2.3. Has the regulatory body established requirements to demonstration of the capability and credibility of a computer code for use in specific analysis application? What kind of tests should be performed for the validation process?

SEC NRS Response

According to RD-03-34-2000, the verification and validation of the computer codes is performed using:

- analytical tests;
- experimental data which describe the particular processes and phenomena (local experiments);
- data obtained at the experimental facilities structurally similar to the real nuclear facilities (integral experiments);
- experimental data obtained at the real nuclear facilities.

HAEA Response

The validation report shall present that for the specific reactor type and for the different processes how accurate is the code compared to the experimental data. The comparison of the experimental data and calculations shall be based on proper statistical methods. The validation report shall define the scope (operational states, environmental conditions etc. where the code is valid).

NSC Response

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STUK Response

YVL B.3 403: The suitability of analysis methods for their purpose shall be justified.

No specific requirements for specific analysis application.

YVL B.3 405 states: The validation of the physical models and computer code used for the analyses shall be substantiated by comparing their calculation results to separate 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 utilized.

2.4. Has the regulatory body provided recommendations on identification of key phenomena, key parameters, and the range of parameter values associated with the range of code applicability?

SEC NRS Response

Formal regulatory recommendations are currently under development, however the following approach is considered acceptable by the SEC NRS:

The experience of its implementation for transients and accidents analysis of VVER-1200 is collected and reviewed.

Selection of evaluation model parameters which significantly affect the code output, and their statistical characterisation

The PIRT methodology ('identification and ranking of processes and phenomena') is used to make a list of n parameters of the code model X_{1i}, \dots, X_{ni} which uncertainty affects the code output for parameter Y_i , ($i=1, m$). These parameters has a statistical nature (e.g., physical and chemical properties of materials, geometry, coefficients of empirical equations underlying the code model, etc.).

Deviation from nominal values for a geometrical parameter and for the range of physical and chemical properties of materials the distribution type is taken as indicated in a reference book

for these parameters. If the distribution type is not specified, then, according to the maximum entropy principle, either a normal distribution (if the information source gives a mathematical expectation and parameter dispersion) or a uniform distribution (if the mathematical expectation and parameter dispersion are not given in the reference book) is adopted.

For empirical equation coefficients included in the code model, it is recommended to determine the variation range and distribution type by comparing measurement data obtained in local experiments with the results of code modelling of these experiments.

As a result, a matrix of code model parameters X_1, \dots, X_n is set for each parameter Y_1, \dots, Y_m ; the uncertainty of these parameters affects the code output. The matrix example is in Table 1.

Table 1. Matrix of code model parameters whose uncertainty affects code output for parameter $Y_i, (i=1,m)$

Name of code model parameter	Range of code model parameter	Type of distribution function of code model parameter
X_{i1}	X_{i1_min}, X_{i1_max}	uniform
.....
X_{in}	X_{in_min}, X_{in_max}	normal

HAEA Response

The recommendation is that the variable parameter, the range of uncertainty and the method used shall be presented in the report.

NSC Response

In chapter 3 and chapter 6 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the principle guide requirements related to the recommendations on identification of key phenomena, key parameters and the range of parameter values have been provided. Before the release of this guide, in a number of projects (such as AP1000, EPR, HPR1000 and etc.) review, NNSA has raised many review questions on the applicability of the key phenomena, key parameters, and the range of parameter values, and put forward the review requirements about some of the problems.

STUK Response

No such recommendations are given. The suitability of analysis methods for their purpose shall be justified.

2.5. Has the regulatory body provided recommendations on methods for extrapolation of validation results (the scalability of the integral effects tests)?

SEC NRS Response

While performing the local experiments for the code validation, the comparison of the code-calculated and measurement data can help to estimate a variation range of the code model parameters which uncertainty affects code output, and to verify the adequacy of the selected phenomenon modelling. However, deviation of code-calculated parameters from the measurement data obtained in local experiments cannot be taken as a code-calculated parameter error in safety analysis of nuclear facilities because local experiments focus only on certain processes and/or phenomena which may occur during a nuclear facility operation.

Error values for code-calculated parameters obtained from integral experiments during the code validation may be used in safety analysis of nuclear facilities. However, as demanded by RD-03-34-2000, it is necessary not only to prove that the experimental facility structure is similar to the real nuclear facility, but also to evaluate the impact of the dissimilation between the experimental facility and the real one through scaling factors analysis consideration.

The error values for code-calculated parameters obtained during the experiments conducted at commissioned nuclear facilities are eligible for safety analysis. However, the number of such experiments and the number of associated measurements are very limited; furthermore, no such experiments can be performed at the nuclear facilities of a new design.

According to RD-03-34-2000, code verification and validation also practices comparison of code output with an output of a similar code already certified by Rostechnadzor ('cross-verification'). In the case of a cross-verification it is necessary to take into account the error values of the parameters calculated by a similar certified code. Parameter error values justified by a cross-verification against a similar certified code may be used in a safety analysis of a nuclear facility only if the similar certified code has been validated against the experiments conducted at a real nuclear facility and integral experimental facilities.

HAEA Response

There are no recommendations related to extrapolation.

NSC Response

In section 3.4.3 of "Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)", the requirement is that it is necessary to further prove that the conclusion related to the code capacity could be extended to the prediction of specific transient behavior of a special nuclear power plant through the comparison of the calculated results and the experimental data. In section 3.5.4.3, the requirement is that extended applications which exceed the original basis, application scope, accuracy of any physical model need to be justified. In addition, in section 5.8.4.7, the requirement is that the extrapolation method of model, correlation and the criteria need to be described and proved.

STUK Response

No such recommendations in regulations. Validation needs to be justified.

2.6. Has the regulatory body provided recommendations for quantification of accuracy of the code calculation?

SEC NRS Response

Russian regulations require to estimate the simulation and modelling errors. The exact way to perform errors estimation is not defined in regulatory basis, however the recommendations on the error evaluation are currently developed and the following approach is considered acceptable by SEC NRS.

Estimation of error of steady-state calculations

Let Y be a parameter with a calculation error to be argued in a code verification and validation report (steady-state calculation). Y_{meas} is a result of the experimental measurement of Y parameter; Y_{calc} is a result of the code calculation of Y parameter.

It was mentioned in the Background that to argue the value of code output error, it is necessary to evaluate deviation of these calculations from the experimental data which have been used in code validation. Furthermore, this shall be done considering the uncertainty of both the code model parameters and the measurements made in the course of experiments.

Code output deviation from the measurement data is determined from the formula.

$$E = Y_{meas} - Y_{calc} \cdot \quad (3)$$

Quantitative assessment of code calculation uncertainty for parameter Y due to the influence of parameter uncertainty in code model is determined using formula.

$$\sigma_{calc} = \sqrt{\frac{\sum_{j=1}^n (Y_{i,calc,j} - \bar{Y}_{i,calc})^2}{n-1}}, \quad (4)$$

where:

$$\bar{Y}_{i,calc} = \frac{\sum_{j=1}^n Y_{i,calc,j}}{n}. \quad (5)$$

The method for assessing the number of n variant code calculations is described in detail in Ref. [40]. The main idea of the method is that variant code calculations should be run until the stabilisation of the value of σ_{calc} (within the accuracy of a second or third digit after the point).

Estimation of error of non-steady-state calculations

Let Y be a parameter with calculation error to be argued in a code verification and validation report (non-steady-state calculation). Let us denote the result of the code calculation of parameter Y as $Y_{calc}(t)$, and the result of parameter Y measurement as $Y_{meas}(t)$.

Evaluation of code model parameter uncertainty at each time point t provides interval estimate of code output (Y_{calc_min} , Y_{calc_max}). Evaluation of measurement uncertainty at each time point t provides interval estimate of measurement results (Y_{meas_min} , Y_{meas_max}).

Considering para 1.2.9 in Ref. [1] that requires to demonstrate the conservative approach adopted for safety analysis of a nuclear facility, code calculation error E for parameter Y at each time point t shall be evaluated by comparing those boundaries of the interval estimates of the calculation and the measurement which have maximum space between them. Maximum value of quantity E over the entire calculation/measurement time should be taken as estimate of code calculation error.

Figure 3 shows the correlation between code output and measurement data at a certain point in time.

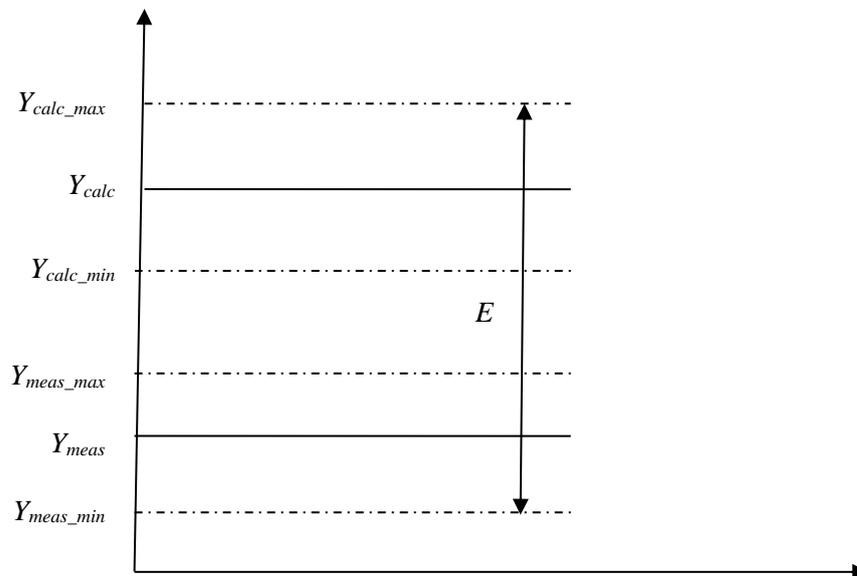


Figure 3. Correlation between code output and measurement data

Code validation often features a situation when time dependences of code-calculated parameter and the measurement data obtained in a code validation experiment are in good qualitative agreement, but the calculated and experimental dependences are offset at time axis relative to one another. To calculate the quantity E correctly, a one-to-one correspondence is

set between the non-steady-state output and the measurement data from a code validation experiment. To this end, both for code output and for experimental measurement data, the time intercept stretching from nil to the point of calculation/experiment completion is partitioned into a similar number of intervals within which both the code calculations and the experiment have qualitatively similar processes. (so called 'phenomenological windows').

[HAEA Response](#)

The validation report shall evaluate the accuracy of the estimations. The validation report shall present that for the specific reactor type and for the different processes how accurate is the code compared to the experimental data. The comparison of the experimental data and calculations shall be based on proper statistical methods.

[NSC Response](#)

In section 4.6.4 of "Safety Assessment and Verification for Nuclear Power Plants" (HAD102-17-2006), regarding the outputs of the computer codes, it should be confirmed that the predictions of the code have been compared with:

- Experimental data for the significant phenomena modelled. This would typically include a comparison against 'separate effects' and larger 'integral' experiments.
- Plant data, including tests carried out during commissioning or start up and operational occurrences or accidents.
- Other codes which have been developed independently and use different methods. This is particularly important in modelling severe accident phenomena.
- Standard problems and/or numerical benchmarks with sufficiently accurate results being obtained.

In section 5.8.8.7 of "Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)", there are the similar requirements to gain confidence in the predictive capability of an evaluation model when applied to a plant-specific event, it is important for assessment reports to achieve the following purposes:

- Assess calculation device capability and quantify accuracy to calculate various parameters of interest (in particular, those described in the PIRT).
- Determine whether the calculated results are attributable to compensating errors by performing appropriate scaling and sensitivity analyses.
- Assess whether the calculated results are self-consistent and present a cohesive set of information that is technically rational and acceptable.
- Assess whether the timing of events calculated by the evaluation model agrees with the experimental data.
- Assess the capability of the evaluation model to scale to the prototypical nuclear plant. (Almost without exception, such assessments also address the experimental database used in developing or validating the evaluation model.)
- Explain any unexpected or (at first glance) strange results calculated by the evaluation model or component devices. (This is particularly important when experimental measurements are not available to give credence to the calculated results. In such cases, rational technical explanations greatly support credibility and confidence in the evaluation model.)

Meanwhile, in section 5.8.8.8 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, it requests that whenever the calculated results disagree with experimental data, assessment reports must also achieve the following purposes:

- Identify and explain the cause for the discrepancy; that is, identify and discuss the deficiency in the device (or, if necessary, discuss the inaccuracy of experimental measurements).
- Address how important the deficiency is to the overall results (that is, to parameters and issues of interest).
- Explain why a deficiency may not have an important effect on a particular scenario.

In section 4.6.4 of “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006), regarding the outputs of the computer codes, it should be confirmed that the predictions of the code have been compared with:

- Experimental data for the significant phenomena modelled. This would typically include a comparison against ‘separate effects’ and larger ‘integral’ experiments.
- Plant data, including tests carried out during commissioning or start up and operational occurrences or accidents.
- Other codes which have been developed independently and use different methods. This is particularly important in modelling severe accident phenomena.
- Standard problems and/or numerical benchmarks with sufficiently accurate results being obtained.

In section 5.8.8.7 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, there are the similar requirements to gain confidence in the predictive capability of an evaluation model when applied to a plant-specific event, it is important for assessment reports to achieve the following purposes:

- Assess calculation device capability and quantify accuracy to calculate various parameters of interest (in particular, those described in the PIRT).
- Determine whether the calculated results are attributable to compensating errors by performing appropriate scaling and sensitivity analyses.
- Assess whether the calculated results are self-consistent and present a cohesive set of information that is technically rational and acceptable.
- Assess whether the timing of events calculated by the evaluation model agrees with the experimental data.
- Assess the capability of the evaluation model to scale to the prototypical nuclear plant. (Almost without exception, such assessments also address the experimental database used in developing or validating the evaluation model.)
- Explain any unexpected or (at first glance) strange results calculated by the evaluation model or component devices. (This is particularly important when experimental measurements are not available to give credence to the calculated results. In such cases, rational technical explanations greatly support credibility and confidence in the evaluation model.)

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- Address how important the deficiency is to the overall results (that is, to parameters and issues of interest).
- Explain why a deficiency may not have an important effect on a particular scenario.

STUK Response

No detailed recommendations.

3. UNCERTAINTIES QUALIFICATION

3.1. Has the regulatory body established requirements and guides to specify the uncertainty analysis process?

SEC NRS Response

Russian regulations require to perform uncertainty analysis for results of safety analysis calculations (for NPP such requirement constrain NP-001-16, moreover the uncertainty analysis has to be demonstrated in V,V&UQ report of each computer code (according to RD-03-34-2000). Regulatory recommendations on the UQ methods are currently under development, the following approach covered acceptable by the SEC NRS experts.

Analysis of code calculation sensitivity to the code model parameters variation.

Analysis is conducted to determine Y_i , ($i=1,m$) parameter sensitivity to variation of each code model parameter X_{1i}, \dots, X_{ni} .

Let us look at model parameter X_{ki} , ($k=1,n; i=1,m$). Code calculation is run to evaluate the parameter Y_i both with a non-deviated value of the model parameter X_{ki} (let us denote it as $Y_i(X_{ki})$), and with the minimum and maximum values of parameter X_{ki} variation range determined in Stage 1 (let us denote them as $Y_i(X_{ki_min})$ and $Y_i(X_{ki_max})$).

Indicator H_{ki} of parameter Y_i sensitivity to model parameter X_{ki} variation is determined from formula

$$H_{ki} = \max(Y_i(X_{ki_min}), Y_i(X_{ki_max})) / Y_i(X_{ki}). \quad (1)$$

$H_{ki} \approx 1$ suggests low sensitivity of parameter Y_i to variation of model parameter X_{ki} value. If $H_{ki} < 0.95$ or $H_{ki} > 0.95$, this points to high sensitivity of parameter Y_i to variation of model parameter X_{ki} value.

The resultant sensitivity matrix gives an idea of how uncertainty of each model parameter from the list produced in Stage 1 affects code output. The way this matrix looks is illustrated in Table 2.

Table 2. Sensitivity matrix for parameter Y_i , ($i=1, m$)

Name of code model parameter	Indicator of parameter Y_i sensitivity to variation of code model parameter
X_{1i}	H_{1i}
.....
X_{ni}	H_{ni}

Defining model parameters to which code output has low sensitivity

On completion of Stage 2.1, those model parameters parameter Y_i has low sensitivity to are excluded from the list of code model parameters produced in Stage 1. The result of this action is the final list of p parameters of the code model X_{1i}, \dots, X_{pi} that have significant influence on the code output for parameter Y_i , ($i=1,m$).

Analysis of effect of code model parameter uncertainty on code output.

Setting various values to model parameters, and testing their mutual independence. Pseudorandom number generator is used to vary n values of code model parameters X_{1i}, \dots, X_{pi} . Distribution laws and parameter X_{1i}, \dots, X_{pi} ranges to be used in this variation are identified before.

Correlation matrix R is constructed to assemble the model from p parameters. Correlation coefficients lying between the i – th and j – th model parameters ($i, j = \overline{1, p}$) are terms (r_{ij}) of this square matrix. According to correlation coefficient definition, matrix R is symmetrical, with units on its principal diagonal.

Normally, mutual independence of code model parameters X_{1i}, \dots, X_{pi} is verified by an approach used in the mathematical statistics to test the assumption of mutual independence of random quantities. In this method, random quantity $\left(\frac{2p+11}{6} - n\right) \ln(\det R)$ that has distribution χ^2 with degrees of freedom $\frac{1}{2}p(p-1)$ is compared with a reference tabular value (see e.g., statistical tables in Ref. [20]) of quantity $\chi_{\gamma, \frac{1}{2}p(p-1)}^2$, where γ is confidence level. In safety analysis calculations for nuclear facilities, confidence level is usually taken to be 0.95 or higher.

If $\left(\frac{2p+11}{6} - n\right) \ln(\det R) > \chi_{\gamma, \frac{1}{2}p(p-1)}^2$, then the assumption of mutual independence of random quantities is not accepted, and code model parameter values are varied once again.

If $\left(\frac{2p+11}{6} - n\right) \ln(\det R) < \chi_{\gamma, \frac{1}{2}p(p-1)}^2$, then values of code model parameters are considered to be mutually independent with γ confidence.

Stage 3.2. Conducting variant calculations using the code, and statistical processing of their results

After Stage 3.1 completion variant code calculations are run for parameter Y_i , $i = \overline{1, m}$. All parameter values in the code model are simultaneously changed in each run. Minimum number of variant code calculations of parameter Y_i depends on the method employed to evaluate the impact of code model parameter uncertainty on the code outcome.

If an impact of code model parameter uncertainty on code output is evaluated using a method which is based on the Wilks' formula, the minimum number of variant runs is determined in the following way.

Let $Y_{i,calc,1} < Y_{i,calc,2} < \dots < Y_{i,calc,p-1} < Y_{i,calc,n}$ be the results of n variant code calculations of parameter Y_i , in increasing order.

It is recommended to use the Wilks' formula to determine the minimum number of variant calculations of parameter Y_i required to construct, with the confidence level b and probability a (proportion of population in the tolerance interval), the two-sided tolerance limits for this parameter with the lower boundary $Y_{i,calc,1}$ and upper boundary $Y_{i,calc,n}$.

$$1 - a^n - n(1-a)a^{n-1} \geq b. \quad (2)$$

Table 3 shows the minimum number of code runs with various combinations of the confidence level b and probability a .

Table 3. Minimum number of calculations required to set two-sided tolerance limits for code-calculated parameters, n

b	a		
	0.9	0.95	0.99
0.9	38	77	388
0.95	46	93	473
0.99	64	130	662

HAEA Response

For uncertainty analysis the following requirements must be followed:

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

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.

3a.2.2.7300. In order to minimize 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.

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.

3a.2.3.0400. 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.

The recommendations are the following:

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 (nodalization, time steps etc.)
- Scale effect

NSC Response

In section 4.5 of “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006), there are the principle requirements for sensitivity and uncertainty analysis. In section 6.3.6 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the requirements for sensitivity and uncertainty analysis are more detailed, including the sources of uncertainty, the effect of the uncertainty by the range and the probability distribution of parameters, the uncertainty evaluation of the calculated results under acceptable confidence, the codes uncertainty derive from the data comparison between integral effects experiment and the different scales of separate effects experiment, the comparison between calculation and experimental data of all the important parameters, the evaluation of uncertainty in the phenomena or physical processes of different periods of time, the uncertainty evaluation derived from the experimental results obtained by a small scale tests used to analyze large scale objects and so on.

STUK Response

YVL B.3 410 states: *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.].*

Also for conservative analyses need to assess uncertainties.

3.2. Has the regulatory body established acceptable level of confidence and the reliability of the tolerance interval?

SEC NRS Response

In Table 3 it is seen that with $\alpha=0.95$ and $b=0.95$, the number of variant code calculations required to determine the two-sided tolerance limits for the calculated parameter is $n=93$. Currently Russian nuclear regulations do not contain any requirements or recommendations pertaining to the confidence level and probability values that should be used in evaluation of model parameter uncertainty effect on code calculation uncertainty. However, the IAEA guide SSG-2 points out that the general practice is to use $\alpha=0.95$ and $b=0.95$ in safety analysis calculations, and normally, values for these quantities are specified in relevant regulations.

HAEA Response

There are no defined levels for confidence and reliability. The sensitivity and uncertainty analysis shall confirm the reliability of the safety analysis.

NSC Response

For different analysis objects, the acceptable level of confidence and the reliability of the tolerance interval are different, which can be determined by the review experience. In section 4.1.2.9 of "Safety Assessment and Verification for Nuclear Power Plants" (HAD102-17-2006), the requirement is that the safety analysis results should ensure that the nuclear power plants run according to the design with a high confidence level, and it should be able to meet all design acceptance criteria during commissioning and throughout the life cycle.

STUK Response

If BEPU is used, the result is acceptable if there is a 95% probability with 95% confidence for the examined parameter not to exceed the acceptance criterion.

YVL B.3 602 The acceptance criteria set forth in chapters 6.2 and 6.3 are written 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 parameter will not exceed the acceptance limit set for the conservative analysis method.

3.3. Has the regulatory body established requirements and guides on methods for uncertainty analysis of experimental data used for validation of computer codes?

SEC NRS Response

Russian regulations have general requirements that the uncertainty of measurements in the experiments used for code validation should be taken into account when determining the error of calculations. Moreover, the State corporation «Rosatom» decree requires that every measurement in the field of atomic energy use should be accompanied by uncertainty evaluation.

To obtain a sound value for code calculation error, code validation should only use the experimental data that measurement uncertainty has been evaluated.

General approaches to the evaluation of measurement uncertainty are described in Russian national standard GOST 54500 “Uncertainty of measurement”, which is based upon ISO/IEC Guide 98-1:2009. According to these documents, the assessment comprises the following steps:

- Select quantity to be measured.
- Identify parameters influencing the measured quantity.
- Relying on existing information, select distribution function for these parameters.
- Identify functional dependence of measured quantity on these parameters.
- Identify distribution type for the measured quantity and its statistical characteristics (mathematical expectation, mean root square deviation, etc.).
- Construct interval that would contain measured quantity value with the prescribed probability.

Code review experience shows that quite often code validation uses experiments devoid of measurement uncertainty information. In these cases, expert judgements, which are very subjective, are used to evaluate measurement uncertainty.

[HAEA Response](#)

There are no requirements and guides on methods for uncertainty analysis of experimental data used for validation.

[NSC Response](#)

In section 3.4.6 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the principle requirement is that the uncertainty of the experimental data may come from the measurement error, the experimental distortion and the other aspects of the experiment. If the uncertainty of the experimental data is too large relative to the evaluation model, these data or correlations cannot be used. Also in section 4.6.5, the uncertainty of experimental data itself (including measurement error, experimental distortion and so on) needs to be clearly reported in the experimental documents, and its uncertainty is appropriate.

[STUK Response](#)

No.

3.4. Has the regulatory body established requirements to the statistical tools (like SUSAN, DAKOTA etc.) used for uncertainty evaluation of the calculation results?

[SEC NRS Response](#)

The statistical tools (like SUSAN, DAKOTA etc.) used for uncertainty evaluation of the calculation results should definitely be verified. However, there are no specific requirement for such tools in Russian regulations.

[HAEA Response](#)

During the statistical evaluation special attention has to be paid to probabilistic distributions and correlations between different parameters.

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
- [...]

The first two uncertainties could be treated by statistical methods.

NSC Response

At present, NNSA does not establish requirements to the statistical tools themselves. But in section 2.3.5 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the basic requirement is raised that the uncertainty of the calculation codes, initial conditions and boundary conditions should be combined by statistical methods, and the relationship among the uncertainty should be considered. Meanwhile, the sensitivity analysis should be carried out, especially for those with “cliff edge effects”.

STUK Response

No requirement concerning tools but YVL B.3 403 The suitability of analysis methods for their purpose shall be justified.

3.5. Has the regulatory body established requirements for *pseudorandom number generators used for uncertainty analysis*?

SEC NRS Response

There are no specific requirement for such tools in Russian regulations.

HAEA Response

There are no requirements related to pseudorandom number generators.

NSC Response

There is no specific requirement for pseudorandom number generator so far. But in section 6.3.6 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, there are some principle requirements, such as “in fact, the real statistical distribution of each key parameter is not available, engineering data and information are needed to verify each statistical distribution. The reasonable statistical distribution of key parameters can be estimated by reasonable data or engineering analysis results, in addition, the relevant supporting documents should be provided for above process.”

STUK Response

No. If used, it needs to be justified as part of the code.

4. USER EFFECT ISSUES

4.1. How does the regulatory body ensure that the qualification of computer codes users is sufficient to perform safety analysis?

SEC NRS Response

According to Procedure on Review of Computer Programs, code user qualification should be assessed by the code developer or by the organization, which holds the computer code certificate.

HAEA Response

The regulatory body participates in audits organized by the licensee and also inspects the users at the licensee.

NSC Response

In section 4.6.7 of “Safety Assessment and Verification for Nuclear Power Plants” (HAD102-17-2006), there are the principle requirements for the code users: 1) they should be trained and enough to understand the codes; 2) they should have enough experience to use the codes, and fully understand the usage and limitation of the codes; 3) appropriate codes manual are needed; 4) if possible, they should had analyzed the standard problems with this codes before safety analysis.

STUK Response

YVL A.4 305. The licensee shall define the requirements for any tasks and work important to safety carried out by suppliers at the nuclear facility, as well as supervise and approve such tasks and work.

This is assessed with inspections and audits.

4.2. Are theory and user manuals subjected to regulatory review and assessment?

SEC NRS Response

No.

HAEA Response

The regulatory body shall assess and review the manuals.

NSC Response

A theoretical manual is required as application material to NNSA. Whether the user manual is submitted depends on the actual review demand.

STUK Response

Not as such but calculation methods shall be presented. This may include user manual.

5. OTHER QUESTIONS

5.1. Has the regulatory body made provision for establishing, maintaining and retrieving adequate quality assurance of computer codes used for safety analysis?

SEC NRS Response

In Russia quality assurance of development, verification, validation and implementation of computer codes should be established by the Licensee in Quality Assurance Program (NP-090-11).

HAEA Response

For the codes used for analysis at least the following steps shall be presented:

- The code description made for the developers and users
- The verification and validation activities and documentation of these
- Validation and code description development and documentation, issue of new versions

For new code version there must be new code description (the developer shall ensure this by proper quality assurance). The quality assurance system shall define the documentation requirements. The documentation related to quality assurance shall be accessible by the regulatory body. The licensee shall present the approval process of the safety analysis reports.

NSC Response

In chapter 5 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the detailed requirements have been raised on quality assurance program, document control, configuration management, tools evaluation, corrective actions, third-party assessments, plans for the development and assessment process, the evaluation model development documents and so on.

STUK Response

On general level configuration management is required and its use is also inspected. These requirements do not limit to computer codes used for safety analysis.

5.2. Does the regulatory body have adequate arrangements for creation, maintaining and using of databanks with verification and validation data?

SEC NRS Response

Rosatom and Rostechnadzor have the cooperation agreement for development of national digital database of experimental facilities which can be used for safety research as well as for validation of computer codes.

HAEA Response

The HAEA does not use databanks for V&V.

NSC Response

According to the requirement in section 3.4.2 of “Development and Application of Computer software for safety analysis of nuclear power plants (Trial edition)”, the database assessment should generally contain the following:

- Separate effects experiments needed to develop and assess empirical correlations and other closure models;
- Integral systems tests to assess system interactions and global code capability;
- International standard problems;
- Benchmarks with other codes that have been fully evaluated and put into practical engineering applications (optional);
- Plant transient data (if available) during commissioning and operation (if available);
- Simple test problems to illustrate fundamental calculational device capability.

It should be noted that (4) and (6) in the above list are not intended to be substitutions for obtaining appropriate experimental and/or plant operation data for evaluation model assessment.

According to the requirements on performing scaling analysis and identifying similarity criteria in section 3.4.3, the need is to demonstrate that the experimental database is sufficiently diverse that the expected plant-specific response is bounded and the evaluation model calculations are comparable to the corresponding tests in non-dimensional space. Section 3.4.4 also requests to identify existing data and then carry out new tests to complete the database.

According to the requirements of section 5.8.4.6, the source, database, accuracy, scaling analysis ability and availability information of transient conditions for specific nuclear power plants of evaluation model and correlations, should be included in the quality assessment report.

STUK Response

Adequate databank for example OECD NEA databank validation matrices. Also SAFIR data available.

5.3. Who makes the safety analyses? Who does the inspection of the analyses?

SEC NRS Response

Utilities are usually choose the organization to perform safety analyses of NPP based on procurement procedure. Some of the common criteria for candidates: holding the licence for design of NPP, competent staff, validated simulation tools, previous experience, etc.

HAEA Response

The safety analyses are made by the licensee, its TSOs, and the supplier. Inspection by the licensee, TSOs (independent analysis) and the regulatory body.

NSC Response

-

STUK Response

Plant vendor and licensee perform the analyses for licensing purpose.

Licensee may make additional comparative analyses for its own use to get better understanding of the plant and its functioning. Licensee also reviews the analysis reports made by plant vendor and others and then presents the analyses for review and assessment (inspection) to regulator.

STUK and its TSOs make independent verification analyses for the key initiating events affecting the acceptability of the plant's systems.

Appendix C

Task force 3 Performance & analyses of passive systems

Main goals of this questionnaire is to share the views of the participating regulators on the approaches to passive systems.

Contents

ISSUES CONCERNING SAFETY DEMONSTRATION OF PASSIVE SYSTEMS.....	65
Proof of the performance of passive safety systems	66
Reflecting commissioning test results in analyses	66
Expectations concerning using full scale or scaled experimental facilities.....	66
Success criteria in PRA to passive systems.....	66
Degradation of passive systems.....	66
Performance of passive system working in connection with other passive or active systems.....	66
What kind of evidence from the commissioning tests you would expect to receive/have received that proves the performance of the safety systems as used in the safety analysis?	66
Are the commissioning test results used to update analyses?	69
Is there performance tests (mockup/factory test/test facility) expected to be done to certain components/phenomena by the licensee or vendor? Does the regulator (or regulators TSO) make or plan to make such testing for code validation or safety system characterization?	69
How is success criteria set in PRA to passive systems?.....	70
Is the possible performance degradation of passive systems assessed in analysis? Is the conservative approach assuming the worst possible performance is used, or is there some sort of performance degradation curve? Requirements concerning this issue?	71
Are there any requirements or practical approaches to prove the performance of passive system working in connection with other passive or active system?	71

ISSUES CONCERNING SAFETY DEMONSTRATION OF PASSIVE SYSTEMS

This paper summarizes the results of comparison of regulatory approaches of Finland, Hungary and Russia to the requirements concerning passive systems. The paper was prepared in the framework of Technical expert subgroup on accidents and transients analysis of MDEP VVER Working Group. The objective of the analysis was to identify commonalities and differences among the Regulatory Authorities and their TSOs within the requirements concerning passive systems in order to:

- promote understanding of each country's regulatory requirements to accident and transient analyses;
- identify areas where harmonization and convergence of regulations, standards, and guidance could be further developed;
- enhance communication among the members;

Proof of the performance of passive safety systems

In all responses to the questionnaire it was evident that during commissioning specific activities are carried out to show correct and expected behaviour of the safety systems. Both active and passive systems are tested. In discussions it was highlighted that direct and full checking would be preferable but this is not always possible. Therefore sometimes indirect or partial checking is carried out and justified. Testing is done based on predefined plans.

Reflecting commissioning test results in analyses

In the responses participants express the need to update safety assessment report if there is deviations arising from commissioning tests. Recalculation of safety analysis may be needed if deviations are significant.

Expectations concerning using full scale or scaled experimental facilities

In Russia vendor has made several tests with full scale and scaled experimental facilities. The expectation presented in the answers is that systems operability should be demonstrated by using proven solutions. Part of the proof that is expected is thorough testing separately before commissioning phase.

Success criteria in PRA to passive systems

According to the answers the success criteria in PRA is defined quite similar way. Operability domain of the passive system is defined and the system is assumed to work accordingly. Additional failures (failures, operator errors) are taken into account similar way as for active systems.

Degradation of passive systems

Consideration of gradual degradation of passive systems in analysis is not clearly defined in regulations in accordance with the responses. However, safety systems are mostly dimensioned so that minor degradation does not threaten performance of the safety systems.

Performance of passive system working in connection with other passive or active systems

Regulations in member states do not give clear guidance on the issue. However, it is seen as an important issue to be taken care of and justified to have not detrimental effect on plant safety. In SEC NRS response a couple of practical examples of considered scenarios is presented.

What kind of evidence from the commissioning tests you would expect to receive/have received that proofs the performance of the safety systems as used in the safety analysis?

SEC NRS Response

There is not only minor differences between active and passive safety systems in the regulatory framework concerning safety demonstration through the testing or analyses.

“Safety important systems and elements shall be capable of performing their functions according to NPP design taking into account external impacts (.....) and (or) under possible (....) impacts occurred as a result of accidents at which the function of the considered systems and elements is required”.

Para 3.1.8 NP-001-15

Safety important systems should be tested during commissioning and during operation to the extent feasible to assure their compliance with the design characteristics.

Direct and full checking are preferable but if performance of direct and/or full checking is not be possible indirect and/or partial checking shall be carried out.

Adequacy of indirect and/or partial checking shall be confirmed in SAR.

Para 3.1.14 NP-001-15

The operating organization shall ensure development and implementation of the commissioning program for the NPP unit according to the design reflected in SAR.

Equipment and devices for testing of safety important systems shall be prepared and verified before the physical start-up. Programs and procedures shall be prepared for:

- to confirm the performance of systems and components;
- to test the systems for compliance with design parameters and characteristics;
- to check the signal sequence and equipment actuation;
- to control of the metal (including welded joints) state of equipment and pipelines;
- to perform the metrological tests of instrumentation means and measuring channels for compliance with design requirements.

Commission tests, physical and power start-up, trial operation shall confirm that power unit as a whole and the safety important systems and elements are available and function in accordance with design, the revealed defects are eliminated.

Documents regulating the performance of the commission tests, physical and power start up, and trial commercial operation shall contain the list of nuclear hazardous activities and measures for protection of nuclear accidents.

When NPP units is commissioned, actual characteristics of safety important systems shall be defined and documented, specifications for the equipment and systems, control systems operation set points shall be given.

List of characteristics and parameters to be documented is defined in appropriate test programs.

Extractions from para 4.2 NP-001-15

HAEA Response

The FSAR must contain every information that created or modified during construction and commissioning. No specific criteria for such kind of evidence.

STUK Response

YVL A.5 sets the following requirements:

410. The commissioning plan shall be updated well in advance of the start of commissioning and shall be supplemented with the following information at the minimum:

- a list of test programmes to be drawn up for commissioning testing
 - a report of the use of the PRA in the drawing up of the testing programmes to assess the scope and balance of the programmes and to reduce the risks of testing (Guide <https://www.stuklex.fi/en/ohje/YVLA-7>, para 325)
- mutual performance order of the test programmes
- conditions for phase-to-phase progress and other hold and witness points for testing
- a testing schedule specifying the planned duration of different tests
- identification of items requiring special attention and summaries of the tests planned for them
- procedures for review and reporting of test results and a description of the procedure in case some test results do not comply with the acceptance criteria
- a plan for maintenance during commissioning
- familiarisation of the personnel involved in commissioning with their tasks.

414. For the purpose of testing, testing programmes shall be drawn up for suitable entities. Each system and phase, for example, can have their own testing programme. A testing programme may comprise several individual tests.

415. Test procedures for individual tests shall be drawn up in advance.

416. The testing programme shall include the following information at the minimum

- preface
- objective of the test programme
- tests included in the test programme
- description and objectives of each test
- organisations involved in the performance of the testing programme and delineation of their responsibilities
- references to detailed test procedures of the tests
- acceptance criteria for each test.

418. At the minimum, the test procedures shall present for each test

- prerequisites for conducting the test
- restrictions on plant operation and other conditions for the performance of the test
- test conditions
- initial state of the systems
- instruments to be employed and other testing equipment and systems required; also in so far as not part of the facility's fixed equipment
- a description of what provisions are made for malfunctions during the test performance
- specific regulations concerning occupational safety and component shielding
- personnel necessary for the test and special expertise possibly required
- the person responsible for the test and his/her deputy
- instructions for performing the test
- completion of the test
- recording of data to be monitored during the test
- the reporting method of the results.

419. A third party independent of the facility and systems design organisation shall review the test programmes of safety-significant plant and system performance tests.

420. Testing programmes of safety-classified systems and plant tests (such as low power tests and power tests) shall be submitted to the Radiation and Nuclear Safety Authority for approval and testing procedures for the testing programmes shall be submitted for information.

421. Testing programmes of Class EYT/STUK (classified non-nuclear) systems shall be submitted to the Radiation and Nuclear Safety Authority for information.

422. If a testing programme or procedure must be altered, the updated testing programme or procedure shall be submitted to STUK for review well in advance before the test in question is conducted.

423. If a testing programme is subject to STUK's approval, the test may take place only after receipt of the approval. Commencement of the test means the first measure taken to demonstrate the performance of the tested item. However, inspections and tuning of the I&C equipment, flushing of piping and other preparatory measures can be carried out without STUK's approval for the testing programme.

In summary, a comprehensive presentation of test is to be presented. Reference plant tests are noted, but if not documented in sufficient detail they may not be used to justify technical solutions (in accordance with MDEP FPOT common position).

Are the commissioning test results used to update analyses?

[SEC NRS Response](#)

The draft of Final Safety Analysis Report shall be developed before the nuclear fuel delivery to the NPP unit.

After completion of the "trial-commercial operation" stage, SAR shall be upgraded to consider the results obtained at the commission tests, stages of physical, power startups and the trial-commercial operation.

Para. 4.2 NP-001-15

In practice: if the results of the testing revealed significant differences from the design characteristics, they shall be considered in the analyses. If more conservative characteristics were used in safety analyzes, it is not necessary to update the analyzes. However test results should be presented in the SAR.

As an example: the mass flow characteristic of the emergency and planned core cooling system was refined after the commission tests for NV-2 NPP, however safety analyses presented in SAR were performed with more conservative mass flow, so a new safety analyzes were not required

[HAEA Response](#)

Generally all experiences from the commissioning must be used to update FSAR, so indirectly also the analyses too.

1.2.5.0600. In the operation license application: a) the modifications concerning the Final Safety Analysis Report which became necessary during the execution of the commissioning programme shall be summarised and substantiated.

1.2.5.0700. The following shall be attached to the license application, as having been reviewed based on the experience of the activities performed on the basis of the commissioning license:

- the updated Final Safety Analysis Report which shall verify – taking into consideration the results of commissioning tests – that
- the nuclear facility operates in compliance with the valid design basis,

[...]

[STUK Response](#)

If during commissioning activities it is found that assumptions in the analyses are incorrect it is required to either justify the situation or to update the analysis.

Best estimate calculations are not necessarily required to be updated, as commissioning test should show fulfillment of criteria. Commissioning test may cause need to update best estimate calculations made for the emergency operating procedures.

Is there performance tests (mockup/factory test/test facility) expected to be done to certain components/phenomena by the licensee or vendor? Does the regulator (or regulators TSO) make or plan to make such testing for code validation or safety system characterization?

[SEC NRS Response](#)

Technical and organizational decisions made to ensure NPP safety shall be proven through the accumulated experience, tests, research. Such approach shall be applied for both for

manufacturing and design equipment, for, NPP constructing and operating, reconstruction and modernization of systems and elements as well as for NPP decommissioning.

Para 1.2.7NP-001-15

In practice: there are several passive systems in the design of AES-2006: hydro accumulators of the first, second and third stages (for different versions of the design), passive heat removal system from SG, passive heat removal system from containment, passive recombiners). All mentioned passive systems were tested at full scale or scaled experimental facilities. Mostly it was performed by vendor.

According to requirement p. 1.2.9 NP-001-15 all computer codes applied for safety justification should be certified. Code certification is a procedure within which V&V report is examined. The requirements to the structure and content of V&V report are installed in RD-00-23. One of the requirements is to present the validation matrix where the important phenomena are listed together with experimental facility where these phenomena were reproduced.

[HAEA Response](#)

There are no requirements for such kind of tests. The regulatory body is not planning to make such kind of tests.

3a.2.1.2400. Systems, structures and components important to safety shall be designed by using design methods that have been previously tried and proven under similar conditions. In other cases, such technologies and products shall be used, the applicability of which has been examined and demonstrated. In the case of new design solutions, which differ from well-established solutions in engineering practices, the applicability shall be demonstrated from a safety point of view by appropriate research, tests and analysis of experience from other applications. New solutions shall be tested before commissioning. The operation of systems, structures and components shall be monitored during their operation for confirming the demonstration of suitability.

[STUK Response](#)

Demonstration of systems operability is expected.

STUK Regulation (STUK Y/1/2016) states: *Nuclear power plant safety and the technical solutions of its safety systems shall be assessed and substantiated analytically and, if necessary, experimentally.*

YVL B.3-407: *If reliable calculation methods are not available, the acceptability of the technical solution in question shall be justified by means of experiments.*

STUK is planning to make some mockup (certain components/phenomena) tests especially for passive systems. The proof that the systems work in accordance with expectations needs to come from the licensee/vendor.

How is success criteria set in PRA to passive systems?

[SEC NRS Response](#)

Success criteria can be defined as the minimal configuration system that ensures the performance of the safety function, the failures and operator errors are defined in the same way, as for active systems.

[HAEA Response](#)

The regulatory guide for PSA recommends the following: The success criteria for passive systems are determined by background analysis (thermal hydraulic calculations).

STUK Response

Thermal hydraulic boundary conditions are defined where the system is assumed to operate. Then it is checked whether these conditions are reached in the scenarios. Failures and human errors are considered the same way as for active components.

Is the possible performance degradation of passive systems assessed in analysis? Is the conservative approach assuming the worst possible performance is used, or is there some sort of performance degradation curve? Requirements concerning this issue?

SEC NRS Response

There are no special requirements concerning consideration of partly degradation of the passive systems.

Degradation of the elements of passive system usually considered in safety analyses. For PHRS SG it is considered in the design: three channels from four are enough to cope with BDBDA, for PHRS SG- degradation of the part of heat exchange surface.

As for PARs, usually failure of some PARs are considered, however there is no analytical or experimental basis to assess how many PARs are failure. There are not enough representative experimental data concerning the degradation under the harsh condition.

HAEA Response

There are no such specific requirements.

STUK Response

Operation of all systems (passive and active) should take into account uncertainties, including potential performance degradation. Even for best estimate calculations if degradation is expected it needs to be taken into account. Technical specifications give the initial state of the systems. For Technical specifications, it should be taken into account how much the system performance may be degraded, if the system is needed for longer time. Accident environmental conditions need to be taken into account in the plant design adequately.

Are there any requirements or practical approaches to prove the performance of passive system working in connection with other passive or active system?

SEC NRS Response

There are no special requirements concerning concurrent operation of the passive systems or passive and active systems.

Practical approach: if any negative effects (systems or trains disturbing each other) were identified due to concurrent operation, such effects should be analyzed and if necessary verified by analyses or by experiments.

Practical example is concurrent operation of passive heat removal system from SG and passive core flooding system (second stage) in case of LOCA accompanied by loss of active part of ECCS on AES-2006 (NV version) . It was identified that concurrent operation of these systems can be affected by non-condensable gases entered to heat exchanger surfaces from containment through leakage hole. Experimental justification based on specially constructed experimental facility was submitted by Licensee to Regulatory Body. The acceptable capacity of the mentioned above systems with consideration of non-condensable gas influence was confirmed.

One more examples: joint operation of the passive heat removal system from SG and the passive heat removal system from the containment in case of SBLOCA, and total loss of AC sources. Mentioned systems have a common water tank as a final heat sink. Additional analyzes were performed confirming the sufficient effectiveness of the systems operating together.

Joint work of PHRS SG together with relief valve (BRU-A). In this case the influence of the BRU-A on the PHRS SG were assessed as not significant.

[HAEA Response](#)

There are no such specific requirements.

[STUK Response](#)

Requirements do not give clear guidance how to justify. Thorough justification however is expected. Justification may include (but not limiting to) for example literature review, thermal hydraulic analysis, scaled tests, commissioning tests and tests in reference plants.

Appendix D

Task force 4 Approaches for regulatory review of safety analyses

Contents

1. STAFF AND COMPETENCE	75
1.1. How many safety analysts and safety assessment reviewers are available at the regulatory body for the PSAR?	75
1.2. How many experts for review does the regulatory body have for deterministic safety assessment, severe accident analysis, probabilistic safety assessments?	76
1.3. How does the regulatory body ensure that the knowledge of the reviewers are up to date and satisfying?.....	76
1.4. Does the regulatory body have special trainings/education for the reviewers of the safety analyses and what kind?.....	77
1.5. Does the regulatory body have special requirements for competence/qualification for the safety analysis reviewers?	78
1.6. How many experts from the TSOs of the regulatory body are available for safety analyses?	78
2. REVIEW PROCESS	79
2.1. Does the regulatory body have standard procedures for reviewing safety analysis? If the answer is yes, please refer to these procedures and give a brief description on their content!.....	79
2.2. Usually how many safety assessment analysts and reviewers participate in the review in the case of major projects, like the review of FSR, PSR, etc.?	81
2.3. Does the regulatory body develop Quality Assurance Plans for the major review projects?.....	82
2.4. Usually what is the depth of a review? (e.g.: review calculations on certain topics or all analyses, review of TRASS reports, review of preliminary and boundary conditions, etc.)	82
2.5. What is the basis on determining the depth of a review?	83
2.6. How does the regulatory body distribute the tasks between the reviewers (for example by chapters, specialization etc. or the same part is reviewed by more than 1 reviewer etc.)?.....	84
2.7. What are the main aspects (how to and what to inspect) for reviewing a safety analysis?	84
2.8. Does the regulatory body have checklists for reviewing safety analysis? Is it different for requirements and recommendations?	85
2.9. Does the regulatory body have evaluation forms (to record the observations/comments) for reviewing safety analysis? How does this evaluation form look like?	85
2.10. How does the reviewer document the observations?.....	86
2.11. What is the main outcome of the review process?	87
2.12. How does the changes of the safety analysis are reviewed?	87

3. INDEPENDENT ANALYSIS	88
3.1. Does the regulatory body have TSOs to perform independent analysis?	88
3.2. In what areas does the regulatory body use TSOs to perform independent analysis (for example severe accidents, PSA, deterministic safety analysis etc.)?	88
3.3. Does the regulatory body have procedures or guides for determining the scope of the independent assessments?	88
3.4. What is the scope of the independent assessments?	89

APPROACHES FOR REGULATORY REVIEW OF SAFETY ANALYSES (COMMONALITIES, DIFFERENCES AND POSSIBLE GAPS)

This paper summarizes the results of comparison of regulatory approaches of Finland, Hungary and Russia related to review of safety analyses. The paper was prepared in the framework of Technical expert subgroup on accidents and transients analysis of MDEP VVER Working Group. The objective of the analysis was to identify commonalities among the Regulatory Authorities and their TSOs within the review methods concerning accident and transient analysis in order to:

- promote understanding of each country's regulatory requirements to accident and transient analyses;
- identify areas where harmonization and convergence of review methods could be further developed;
- enhance communication among the members;

Regulatory framework concerning review of safety analyses

Staff and competence for review of safety analysis

In all participating member countries the number of employees at the regulatory bodies are in the range of hundreds. For reviewing the chapters related to safety analyses in the PSAR there are at least a few experts. All member countries consider the international and domestic information exchange and learning as a form to keep up to date. All participating countries mention also the use of their external services (TSOs, universities) as a way to keep updated. In Finland there are external training courses as special course and Hungary also planning to organize such kind of events for their inspectors. In the response the member countries indicate that there are no specific requirements for safety analysis reviewers, however there are general requirements for the employees. Hungary and Finland use a few people from TSOs for safety analysis review, in Russia the TSOs are established by the regulatory body and have special status on the level of federal law (for example only TSO can perform review of safety analysis for NPP and other nuclear installation subjected for permanent state supervision).

Review process

In Hungary there are no standard procedures for reviewing safety analyses, Russia have a standard plan for this, and Finland have written guidance for some cases. In major projects like PSAR in Hungary there are 6-8, in Finland 7-9 and in Russia 49 reviewers for safety analysis. In all member states the depth of the review covers all the important aspects and there are independent calculations for specific cases. The basis of the depth of the review are mainly the requirements for all member states. Finland is using a requirement management software and Hungary is also planning to use such kind of tools. Hungary also uses checklists based on regulatory guides. In the participating countries the distribution of the review work is based mainly on the expertise of the reviewers. All member states are using some kind of evaluation forms to record the results

of the reviews. The observations are recorded in the evaluation forms in Hungary, in Russia reports are needed, and in Finland requirement management software is used beside memorandums. The main outcome of the review process are reports in Finland and Russia, in Hungary group evaluation forms are the basis of the resolutions.

Independent analysis

All member states are using TSOs for independent analysis. In Finland and Russia the TSOs are used for independent analysis in the area of DSA, severe accident analysis, PSA. In Hungary for DSA, partly for severe accidents and in some cases for PSA. In Russia and Finland there are no formal procedures for the scope of independent analysis, expert judgment is used in Finland, Rostechnadzor Technical Assignment is used in Russia. In Hungary there is a regulatory guide to determine the scope for this.

6. STAFF AND COMPETENCE

6.1. How many safety analysts and safety assessment reviewers are available at the regulatory body for the PSAR?

[SEC NRS Response](#)

According to the Federal Law “On the use of nuclear energy” Russian regulatory authority (Rostechnadzor) is authorized to arrange the safety reviews. The review itself is carried out by the expert organization, which must have a license for safety review. Moreover, if a nuclear installation were a subject to the permanent state supervision, safety review shall be only carried out by a technical support organization authorized by Rostechnadzor. The arrangement of safety review requires having a competent staff for: developing a review task; assessing the results of the review; reflecting results of the review in license condition (about 20 experts for these tasks).

Headquarters of Rostechnadzor have three departments involved in arrangement of safety reviews for nuclear installations: department for Safety Regulation of Nuclear Power Plants and Research reactors; department for Safety Regulation of Nuclear Fuel Cycle Facilities, Nuclear Power Installations of Vessels and Radiation Hazardous Facilities; department of Special Security.

Departments have their own structure, for instance, department for Safety Regulation of Nuclear Power Plants and Research Reactors has: review and licensing of existing NPP section, review and licensing of new builds section, review and licensing of research reactors section, inspections of NPP section, inspections of research reactors section, supervision for construction section, supervision for equipment, supervision for I&C and electrical infrastructure. Each section has 3 to 6 specialists in its staff.

The maximum number of employees is set by the Government of the Russian Federation, so the maximum number of the headquarters of Rostechnadzor in 2016 was 660 units, branch offices - 7,085 units (of them number of employees performing function of atomic supervision, was - 526).

[HAEA Response](#)

The HAEA is planning to involve 4-6 reviewers for safety assessments in the case of PSAR.

[NSC Response](#)

[STUK Response](#)

STUK is planning to involve approximately 7-9 reviewers for safety assessments in the case of PSAR. Full time analysts are not available in house, but reviewers have varying amount of expertize in the field of safety analysis.

NDK Response

6.2. How many experts for review does the regulatory body have for deterministic safety assessment, severe accident analysis, probabilistic safety assessments?

SEC NRS Response

See response to 1.1 question.

HAEA Response

Most of the staff are not experts yet in the area of safety assessments, however there are 4-6 colleagues for deterministic safety assessments, 2-3 for severe accident analysis and 2 for probabilistic safety assessments.

NSC Response

STUK Response

The office responsible for deterministic safety assessments review has 9 experts for the review. From these 2 are concentrating on severe accident analysis.

PSAR PRA review takes about 2,5 man-years, the whole PRA review is about 5 man-years. For PRA approximately 10 persons are involved in review.

NDK Response

6.3. How does the regulatory body ensure that the knowledge of the reviewers are up to date and satisfying?

SEC NRS Response

In accordance with the requirements of the legislation on public service, civil servants of the headquarters and branch offices of Rostechnadzor must possess the necessary education, professional experience and experience of the state civil service. The replacement of open vacancies in Rostechnadzor is carried out on the basis of competitive procedures, regulated by the methodology for holding a competition for filling the position of the federal state civil service in Rostechnadzor, approved by order of Rostechnadzor No. 907 of November 20, 2008.

In 2016, 71 federal civil servants of the Rostechnadzor headquarters passed through appraisal. All branch offices of Rostechnadzor have established appraisal commissions. Experience exchange through international cooperation is also considered as part of knowledge management.

SEC NRS has formal procedure for periodical (each 3 year) assessment of qualification and working progress of experts. The expert should submit a report on his past works (review projects as well as R&D) and training courses to the Qualification Committee, who takes the decision on the expert qualification after the review the report and holding the interview with the expert. Qualification criteria and procedure are established in the internal SEC NRS document.

HAEA Response

The colleagues working in the area of safety assessments are sent to different workshops, trainings and meetings organized by IAEA and other organizations like OECD, VVER Forum etc. The HAEA experts also learn by funding researches and through consultations with TSOs and licensees.

NSC Response

STUK Response

Knowledge and knowhow management is described in STUK management system documentation YTV 6.b. Main idea is to firstly recruit skilled personnel. Personal development and training programs are created for each new recruit. Experts are taking part in external trainings and workshops if they are assessed to be valuable. However on the job learning is considered to be the most valuable mean of professional development.

Inspectors are also communicating with TSOs when the independent assessment tools (models) are created and used. STUK is also involved in experimental studies, ordered from TSOs or universities.

New recruits are put through introduction training program and their level of competence is evaluated through the program by their supervisor and senior colleagues. As the level of competence is adequate for independent work the formal qualification is granted based on shown performances and knowledge.

Experienced inspectors are evaluated annually based on their performances at the same time their level of competence and development are assessed and discussed (personal development discussions, annually). These evaluations are input for personal development plans, which are updated and executed.

NDK Response

6.4. Does the regulatory body have special trainings/education for the reviewers of the safety analyses and what kind?

SEC NRS Response

In order to assist new employees in acceleration of process of their professional and social adaptation in team Institute of mentoring is used.

In 2016, the organization of additional professional education of Federal civil servants of Rostechnadzor was held in within the framework of implementation of the state order for additional professional education for 2016.

Training with the use of distance learning technologies was carried out on 13 programs of additional professional education (659 people were trained), which allowed to reduce transport and business trips expenses.

SEC NRS has a special training course for the new employees, it covers all aspects of regulatory process, such as regulatory requirements and guidelines, interaction with the stakeholders, quality assurance of the deliverables and so on. The first part of the initial training based on online learning platform, the second part – lectures given by the management of SEC NRS (each division gives one lecture once a year). The most important findings of safety review and associated challenges are discussed on the meetings of Scientific Council (with representatives of Rostechnadzor and other interested parties), so the staff can be in line with relevant expert activities.

HAEA Response

The HAEA is planning to organize trainings by TSO experts to gain knowledge on safety assessments (mainly deterministic safety assessments). The training will include presentations on V&V, model developing, inspection aspects etc. supplemented with some basic on-the-job trainings. There are also trainings for inspectors at the nuclear power plant to have knowledge on safety systems.

NSC Response

STUK Response

Tutoring is used to train newer reviewers. Internal training in safety regulation and inspection practices related to safety analyses are conducted regularly. Inspectors are obligated to familiarize themselves with internal guidance and safety analyses by studying related documents.

Additionally for example thermal hydraulics issues are trained in external courses. External training courses are attended when needed.

NDK Response

6.5. Does the regulatory body have special requirements for competence/qualification for the safety analysis reviewers?

SEC NRS Response

According to the Provision on Expertise, during the review of safety analysis the expert organization is guided by the quality management system established in the expert organization. According to the SEC NRS Instructions, experts are selected from specialists who have:

- correspondence of basic education to the main subject of review;
- At least 2 years of work experience in the main field of review;
- The skills necessary to perform a qualified review and presentation of the results of the review;
- Availability (working load compared to other candidates);
- Not been involved in the development of the documents submitted for review;

HAEA Response

There are no formal requirements for competence/qualifications for safety analysis reviewers, however the beginners follow the experts guidance and they have to pass exams on the safety systems of the nuclear power plant e.g., primary circuit engineering exam. The primary circuit training last about 3 months theoretical + at least 2 weeks practical training. The training covers all primary systems in the NPP, operational aspects and other safety related aspects (safety culture, fire protection, radiation protection).

NSC Response

STUK Response

Professional competence of the reviewers of safety analysis are evaluated in a holistic way. Special requirements do not exist as such. However, the abilities of reviewers are observed throughout the development program. These observations pay attention to training results and shown progress in in daily work (learning by doing, tutoring, etc.).

The competence/qualification of the safety analysis reviewers is constantly evaluated against the requirements of the daily work of each individual.

NDK Response

6.6. How many experts from the TSOs of the regulatory body are available for safety analyses?

SEC NRS Response

Review of the safety analysis of nuclear installations for which a permanent state supervision regime has been established can only be performed by the Rostekhnadzor's TSO: SEC NRS and VO Safety. The maximum number of employees of SEC NRS is 350 people, of which 2/3 are employees of scientific and technical units that could arrange and carry out a review of safety analysis of nuclear installations.

[HAEA Response](#)

There are only 3 TSOs for PSA (around 10 experts all in all) and severe accident analysis (2--3 experts), and 2 for DSA (4-6 experts). In Hungary not only the licensees but the regulatory body can also use TSOs.

[NSC Response](#)

[STUK Response](#)

Independent safety analysis work may also be purchased from abroad, if necessary.

STUK mostly uses annually 2-4 person yrs from TSOs, more resources would be available if needed.

[NDK Response](#)

7. REVIEW PROCESS

7.1. Does the regulatory body have standard procedures for reviewing safety analysis? If the answer is yes, please refer to these procedures and give a brief description on their content!

[SEC NRS Response](#)

Organization and implementation of safety review is defined by the following regulatory and legal acts:

“Provision on Licensing Activities in the Field of Atomic Energy Use” (approved by the Government of the Russian Federation No. 280 dated of March 29, 2013);

“Provision on Assignment of a Legal Entity to Scientific and Technical Support Organization of the Authorized Body for State Regulation of Safety at Atomic Energy Use” (enacted by Decree of the Government of the Russian Federation No.387 dated of April 30, 2013);

“Administrative Regulations for the Federal Environmental, Industrial and Nuclear Supervision Service on Execution its State Function for Licensing the Activities in the Field of Atomic Energy Use” (approved by Order of the Federal Environmental, Industrial and Nuclear Supervision Service No. 453 dated of October 08, 2014.

“Provision on Safety Review (Safety Analysis Review) Procedure of Nuclear Facilities and (or) Types of Activities in the Field of Use of Atomic Energy” (approved by Order of the Federal Environmental, Industrial and Nuclear Supervision Service No. 160 dated of April 21).

Phases of the review process

- Phase 0. Preliminary phase
- Phase 1. Initiation
- Phase 2. Works planning & organization
- Phase 3. Implementation
- Phase 4. Monitoring & general management
- Phase 5. Completion

Phase 0 Preliminary phase

Actions of Rostekhnadzor:

Preparation of Technical Assignment (TA) for safety review

Provision of the expert organization with approved TA (topic, scope, deadlines) and applicant safety justifications for safety review

- Action of the expert organization:

Confirmation of TA acceptance for implementation

Phase 1 Initiation

- (1) Selection of specialists responsible for review safety review leader (chief expert) and coordinator (responsible for communication between team members)
- (2) Negotiation with the responsible specialists before their appointment
- (3) Appointment of the responsible specialists

Phase 2 Planning & arrangement

- (4) Selection of experts
- (5) Preparation of the work schedule
- (6) Estimation of man-power, costs, other recourses needed
- (7) Approval of the experts' list and works schedule
- (8) Preparation of individual TAs for experts
- (9) Discussion on review methods and tools to be used
- (10) Provision of experts with Applicant documents to be reviewed
- (11) Provision of experts with additional information

Phase 3 "Implementation of the review"

Individual stage of review

- (12) Individual analysis and review by experts
- (13) Development of individual review reports by expert on topics

Comparative stage of review

- (14) Discussion of the results among experts
- (15) Refinement of individual review reports (if necessary)
- (16) Signing of individual review reports by experts

Results assessment & summarization. Findings & conclusions.

- (17) Quality assessment and acceptance of individual review reports

Phase 4 "Monitoring & general management"

Routine monitoring over work progress
Management of the work schedule
Routine risks monitoring and management
Routine monitoring and control over changes
Control of changes
Management over the performance if any changes
Routine monitoring over quality
Reporting on work implementation
Information exchange
Cost management
Administration of contracts

Phase 5 "Completion"

Drawing up of review results

- (18) Draft of the safety review report
- (19) Assessment of the draft safety review report by experts

- (20) Submission of expert's questions (findings) to the Applicant
- (21) Discussion of expert's findings with the Applicant
- (22) Final draft of the safety review report
- (23) Signing of the safety review report by experts

Assessment of review quality

- (24) Assessment for compliance with the established procedure
- (25) Quality assessment for compliance with layout, wording, etc.
- (26) Validation of the safety review report (each section and in total)
- (27) Acceptability assessment and approval of the review report
- (28) Submission of the safety review report to Regulator

Distribution, archiving, payment

- (29) Duplication and binding the safety review report
- (30) Distribution and archiving the safety review report
- (31) Estimation of the man-power in fact
- (32) Drawing up of the payment documents

[HAEA Response](#)

At the moment the HAEA does not have standard procedures for reviewing safety analysis, however it is under development. Also for each project, there is a quality assurance plan which contains all the relevant information, regulations and aspects that must be taken into account during the review.

[NSC Response](#)

[STUK Response](#)

YTV 3.b.1 present the main requirements and gives some background information to the requirements that need to be fulfilled by the analysis of deterministic nature.

YTV 3.b.2 gives general view on the PRA inspection with 2 phases. First phase of review consists of general inspection, reviewing formal coverage and recognizes approaches, methods and reported results. Second phase consists of thorough review with qualitative and quantitative approaches. A more detailed inspection manual is referred to from this document.

Failure tolerance analysis (CCF, FMEA, internal/external hazards,...) review is covered by YTV 3.b.3

[NDK Response](#)

7.2. Usually how many safety assessment analysts and reviewers participate in the review in the case of major projects, like the review of FSR, PSR, etc.?

[SEC NRS Response](#)

Major projects require considerable resources. For instance, review of safety analysis report of unit 1 of Leningrad NPP-II (application for operating license) required 49 experts for review and assessment of safety analysis documentation and 16 specialists for independent calculations.

[HAEA Response](#)

Depending on the size and relevance of the project 2-8.

[NSC Response](#)

STUK Response

Graded approach is used. STUK is planning to involve approximately 7-9 reviewers for safety assessments in the case of PSAR.

NDK Response

7.3. Does the regulatory body develop Quality Assurance Plans for the major review projects?

SEC NRS Response

According to the Provision on Expertise, during the review of safety analysis the expert organization is guided by the quality management system established in the expert organization. Key criteria for assessment of safety review acceptability.

- Compliance with the procedure established for safety review
- Completeness of Ta performance
- Timeliness of review methods and tools applied by experts
- Adequacy of review methods and tools applied by experts
- Logic and argumentation of review results
- Compliance of safety review results with other known data

HAEA Response

For major reviews the HAEA develop Quality Assurance Plans to determine the framework and the most important aspects, scheduling, project structure, relevant regulations, evaluation form etc.

NSC Response

STUK Response

Quality Assurance Plan is not developed, but similar issues are described in project and subproject plans.

NDK Response

7.4. Usually what is the depth of a review? (e.g.: review calculations on certain topics or all analyses, review of TRASS reports, review of preliminary and boundary conditions, etc.)

SEC NRS Response

Reviewing documentation submitted to obtain license, the following is to be examined:

- design, engineering and process engineering solutions;
- completeness of technical and administration measures aimed at implementation of the licensed type of activity;
- compliance with the requirements for safety provision; meeting the requirements for safety ensuring in respect to the measures directed at protection of population and personnel of a nuclear facility in case of accident origination and the preparedness to implement such measures together with the system on quality assuring and the required engineering and technical assistance in respect to the licensed type of activity;
- capability of a license applicant to provide the conditions for safe implementation of the licensed type of activity, the safety of a nuclear facility and activities under realization along with the quality of activities carried out and the services rendered which meet the federal regulations and rules in the sphere of the use of atomic energy;

- availability of the appropriate means and their preparedness to eliminate the emergency situations in case of nuclear and radiation accident origination at a nuclear facility;
- capability of a license applicant to provide for safe termination of the licensed type of activity and decommissioning a nuclear facility together with availability of the appropriate design documentation.

Reviewing the AT analysis results experts can ask for additional details of AT analysis (ask for additional documents). Experts can also perform independent calculations.

[HAEA Response](#)

The HAEA usually inspect the documentation only and don't prepare independent calculations. In some cases (i.e. if the results, methods, assumptions are not convincing) the HAEA ask for detailed to documentation or independent analysis to make sure that the safety assessments are in compliance with the regulations.

[NSC Response](#)

[STUK Response](#)

Methodology reports are reviewed. In the analysis reports it is reviewed whether they follow given methodology and that references are correct. All analyses that have been sent are reviewed. Boundary conditions, initial state and failure assumptions are part of the review. Independent analyses for comparison use are done (mostly by TSO) from the most limiting or otherwise interesting cases. Then the results are compared. Also some sensitivity analyses may be ordered from TSO.

Analyses are reviewed against plant design documentation, so that it is evident that the analysis represent the plant design. In PSAR phase mostly the design solutions that are hard or impossible to change later are given the most effort.

[NDK Response](#)

7.5. What is the basis on determining the depth of a review?

[SEC NRS Response](#)

Depth of the review is based upon provision for arrangements of the review and upon regulatory requirements to safety of nuclear installations. There are also regulatory requirements on the content of SAR for different types of nuclear installations (for VVER its NP-006-16), which provide basis for determining the Rostekhnadzor's Technical Assignment for the review.

[HAEA Response](#)

There are no guiding documents to determine the depth of the review. For the PSAR we will use a requirement management software to make sure that every requirement is fulfilled, also we have a checklist for recommendations to understand how much the licensee followed the recommendations from the regulatory guides.

[NSC Response](#)

[STUK Response](#)

Depth of review may be adjusted if the reviewed material is reviewed before by the regulator and is submitted as updated material. STUK has requirement management software in use to track fulfillment of requirements.

For failure analysis review a clearly stated graded approach is given based on safety class.

[NDK Response](#)

7.6. How does the regulatory body distribute the tasks between the reviewers (for example by chapters, specialization etc. or the same part is reviewed by more than 1 reviewer etc.)?

[SEC NRS Response](#)

Each expert has individual review topic from the Rostechnadzor's Technical Assignment for the review. The level of detailization may differ, for instance review of chapter 15 of VVER-1200 SAR require around 60 topics, usually one topic is one type of accident or transient. Each topic can be assessed by one expert, sometimes more experts can be assigned (if the audit calculations are required or in case of experienced expert would like to train the less experienced one). Experts can ask for additional documents to support SAR, such requests first go to Rostechnadzor, which asks applicant to provide them. After additional documents arrived, Rostechnadzor make a decision if additional review topics (and experts) are required and send the additional documents as well as updated Technical Assignment to the expert organization.

[HAEA Response](#)

In the case of PSAR we would like every reviewer to review all chapters related to deterministic safety assessments and also the chapters related to safety systems, to understand the safety features, functions and philosophy at the same time. However, our practice is that the team leaders selects the scope for each reviewer.

[NSC Response](#)

[STUK Response](#)

Subproject manager coordinates the work with the help of line organization. Some overlapping in review work is justifiable, but unnecessary work duplication is to be avoided. Overlapping helps in communicating and avoiding omissions. In assigning work each reviewers expertise is taken into account. Therefore, a common rule cannot be given here.

Generic chapters are reviewed by many, as these help to understand specific issues in different chapters.

[NDK Response](#)

7.7. What are the main aspects (how to and what to inspect) for reviewing a safety analysis?

[SEC NRS Response](#)

Regulatory requirements and guidelines can be used as basis for what to inspect. How to inspect (for example, review and assessment or audit calculations) is up to the team leader of the review process, however an expert can initiate a proposal for expanding the review (additional documentation could be requested from the Applicant, topics for discussion and clarification could be proposed).

[HAEA Response](#)

Naturally the most important is to check whether the licensee is in compliance with requirements and followed the regulatory guide recommendations. Furthermore the HAEA checks if the documentation is fulfilling the formal requirements and whether the licensee used the latest and internationally accepted methods, are they valid and verified, are there any outlier, inappropriate results, is the nodalization, time steps and the timeframes (end of calculations) are proper, are the conditions and assumptions proper, are the results credible using engineering judgment, is there enough information to recreate the calculations etc.

NSC Response

STUK Response

Firstly it needs to be assessed whether the analysis are in line with the plant design. Methodology and potential screening (exclusions) need to be carefully assessed when assessing the coverage of the analyses. In the analyses themselves boundary conditions, failure assumptions and fulfillment of criteria is to be reviewed.

NDK Response

7.8. Does the regulatory body have checklists for reviewing safety analysis? Is it different for requirements and recommendations?

SEC NRS Response

No, there are no any checklists for the review, however SEC NRS has a template for expert findings and digital database with all regulations and guidelines is used as an essential tool for the review process.

HAEA Response

Yes, the HAEA developed a checklist based on the regulatory guide to ensure that there are not any recommendation that the licensee did not followed. Also the HAEA will use a requirement management system to ensure that every requirement is fulfilled.

NSC Response

STUK Response

Guidance for the review is given in YTV 3.b.1. Requirement management tool is used an it could be seen as a “check list”. In addition, inspection memorandum templates have been used in previous projects with success.

NDK Response

7.9. Does the regulatory body have evaluation forms (to record the observations/comments) for reviewing safety analysis? How does this evaluation form look like?

SEC NRS Response

Rostechnadzor established requirements for what should be analyzed and assessed during the review and what should be presented in the review results (requirements for the content). The form of review results are established by expert organization in its quality management documents. SEC NRS has following requirements to the form of review results:

The expert's opinion on a particular issue should have the following structure:

1. Background information.
2. Evaluation criteria used for review and assessment.
3. Brief description of the review topic.
4. Result of review.

The expert shall state the following aspects of the review:

- what are the requirements, criteria, safety principles (paragraphs or sections of existing regulations) met by the applicant in the justification;
- what approaches, methods, techniques, initial data, boundary conditions, etc. are used by the Applicant in the SAR;

- what are the values, results, conclusions, etc. obtained by the Applicant in the SAR.

Assessments, comments, recommendations the expert should state the following aspects:

- whether all requirements, criteria, safety principles (points or sections of the existing normative documents) are covered by the applicant at SAR;
- how the approaches, methods, techniques, initial data, boundary conditions, etc. used by the applicant meet the requirements of regulatory documents;
- as approaches, methods, techniques, initial data, boundary conditions, etc., used by the applicant, correspond to the state of the art science and technology, domestic or foreign practice.

The comments (non compliances), as well as the recommendations, must comply with the following rules:

- The formulation of comments (as well as assessments and recommendations) should be clearly understandable and adequately concise. Direct quotations from normative documents should be avoided, limited to references to the relevant paragraphs.
- In the preparation of comments (and recommendations) should be aware that they must be clearly understandable, subject to the exceptions of the context of expert opinion.

The findings and suggestions of the expert should reflect the conclusion that

The safety justification provided by the applicant meets the requirements of Federal regulations in the field of nuclear energy and other regulatory documents and is sufficient.

Or

The safety justification provided by the applicant does not meet the requirements of Federal regulations in the field of nuclear energy and other regulatory documents and is insufficient ("as evidenced by the observations № ... of this expert opinion).

[HAEA Response](#)

Yes, the evaluation form includes the following information: name of the reviewer, identifying number for comment/observation, comment/observation, safety relevance of comment/observation (on a scale from 1 to 4), related regulation (requirement or recommendation), related notes, proposed text for decision.

[NSC Response](#)

[STUK Response](#)

Inspection memorandum templates have been used in previous projects. This would include name of the reviewer, issue found with related regulation (requirement or recommendation), related notes, guidance on important issues and proposed text for decision.

[NDK Response](#)

7.10. How does the reviewer document the observations?

[SEC NRS Response](#)

Expert should develop the individual report on the reviewed topic.

[HAEA Response](#)

The reviewer fills the evaluation forms.

[NSC Response](#)

[STUK Response](#)

Fulfilment of requirements is traced in requirement management tool. Observations may be documented in the database or in separate inspection memorandums.

[NDK Response](#)

7.11. What is the main outcome of the review process?

[SEC NRS Response](#)

Based on safety review findings, expert organization prepares expert report, which shall be endorsed by an authorized person of the expert organization.

[HAEA Response](#)

The reviewer evaluation forms are summarized in a group evaluation form and the decision is developed based on the proposed texts for decision from the group evaluation forms.

[NSC Response](#)

[STUK Response](#)

YTV 3.b.1 gives guidance to reviewing the analyses

During the construction license phase STUK inspects the analysis and their adequacy when reviewing the construction license application. Based on this inspection an inspection report is issued, which is based on the observations documented during review. Inspection report summarizes the fulfillment of requirement. The inspection report is the basis for STUKs statement and assessment of safety that is given in the statement for construction license.

[NDK Response](#)

7.12. How does the changes of the safety analysis are reviewed?

[SEC NRS Response](#)

Safety review shall be carried out with regard:

- to issue (or not to issue) of a license;
- to change of the license conditions.

Requirements to periodic safety assessment of nuclear facilities (in each 10 years).

In case of ongoing review process, any changes in safety analysis submitted by the applicant to the Rostekhnadzor, which makes a decision if additional review topics (and experts) are required and send the modified documents as well as updated Technical Assignment to the expert organization.

[HAEA Response](#)

There is no formal process for this.

[NSC Response](#)

[STUK Response](#)

Depending on chance the graded approach is used. If the change is major, the same review as for first time issue is used.

NDK Response

8. INDEPENDENT ANALYSIS

8.1. Does the regulatory body have TSOs to perform independent analysis?

SEC NRS Response

Rostechnadzor have two TSOs – SEC NRS and VO Safety.

HAEA Response

The HAEA have TSO-s to perform independent analysis which the HAEA use to have a better understanding of special phenomenas and also to confirm and check the results of the licensee's analysis.

NSC Response

STUK Response

Yes, STUK uses TSOs to perform independent verification analysis to support decision making.

NDK Response

8.2. In what areas does the regulatory body use TSOs to perform independent analysis (for example severe accidents, PSA, deterministic safety analysis etc.)?

SEC NRS Response

As a TSO, SEC NRS must provide services for Rostechnadzor in all areas of the review of safety analysis, including independent calculation (third parties could also be involved in that activity by SEC NRS).

HAEA Response

Mostly in the area of DSA, but also for severe accidents case by case and for independent PSA event analysis (regularly).

NSC Response

STUK Response

All mentioned examples in the question are the ones in which TSOs are used.

NDK Response

8.3. Does the regulatory body have procedures or guides for determining the scope of the independent assessments?

SEC NRS Response

There is no formal procedure for that, but SEC NRS has to follow the Technical Assignment prepared by Rostechnadzor.

HAEA Response

The HAEA have a regulatory guide for independent analysis which appoints the scope of the independent analysis that the licensee should have.

NSC Response

STUK Response

Expert judgement is used to determine the scope. For novel features it is seen important to gain confidence by doing independent assessments.

Licensee needs to make a safety assessment of the plant. As part of this assessment independent assessments are somewhat mandatory, to support conclusions.

NDK Response

8.4. What is the scope of the independent assessments?

SEC NRS Response

Scope of independent assessment is defined by Rostechnadzor in Technical Assignment for TSO depending on case.

HAEA Response

The scope is based on the following defined by a special regulatory guide. The licensee should develop independent analysis for analysis that are containing the characteristics of the designs, which are decisive from a safety point of view. The independent analysis should cover the following cases:

1. From each initiating event group at least the enveloping events with examining (justifying) the enveloping nature of the events.
2. Analysis that justifies the adequacy of unusual, new, unique solutions. Mainly design solutions that was not analyzed by other independent organizations and real experiences, experimental results are limited (typically passive systems).
3. Analysis where the analysis made by the designer shows small safety margins related to the criteria (typically when the difference between the results and the criteria is comparable with the estimated or evaluated uncertainty).
4. Underlying or basic calculations (reactor physics, hot channel calculations etc.) and calculations related to examine the ultimate/final criteria (i.e. activity release).
5. Analysis where the approximation, simplification are unusual or not convincingly justified in the analysis made by the designer or where the assumptions have a large effect on fulfilling the criteria.
6. Analysis where they justify that one of the physical barriers can withstand a maximal load for any of the physical parameters (pressure, temperature etc.)
7. At least one scenario for each of the safety systems.

From the above mentioned if there are any other reason for not trusting the results the licensee should prepare independent analysis to clarify.

NSC Response

STUK Response

Not defined in regulations.

NDK Response
