SAFETY GUIDES

ON IMPLEMENTATION OF THE LEGAL REQUIREMENTS

SAFETY GUIDE

Deterministic Safety Assessment

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1. GENERAL PROVISIONS

LEGAL BACKGROUND

1.1. Article 16, i. 3 of the Act on the Safe Use of the Nuclear Energy [1] stipulates that the licensees using nuclear energy or sources of ionising radiation or dealing with radioactive waste management and spent fuel management are required to perform assessment of nuclear safety and radiation protection at the nuclear facilities and sites with sources of ionising radiation and undertake actions and implement measures for the improvement of nuclear safety and radiation protection, taking into account national and international operating experience and scientific achievements in this field.

1.2. The Regulation on the procedure for issuing licenses and permits for safe use of nuclear energy [2] defines the cases, for which the applicants are required to perform and submit to NRA safety assessment of the activities associated with the use of nuclear energy. The specific requirements to the safety assessment of the facilities and activities are included in the Regulations under article 26, para 2 of [1] – Regulation on ensuring the safety of nuclear power plants [3], Regulation for safe management of spent nuclear fuel [4], Regulation for safe management of radioactive waste [5], Regulation on ensuring the safety of research nuclear installations [6], Regulation on emergency planning and emergency preparedness in case of nuclear and radiological emergencies [7], etc.

1.3. On the basis of §6 of the transitional and final provisions of [3], this publication gives guidance on application of the legal requirements with respect to the use of deterministic approach in the safety assessment. Some of the recommendations in this publication derive from the requirements and good practices as described in reference documents [8] – [16].

OBJECTIVE

1.4. This publication is intended to give guidance for safety assessment performed in the process of development and verification of the initial design of nuclear facilities, in the cases of design modifications as specified in article 15, para 4, i. 5 of the [1], as well as for performing of an independents verification of the safety assessment by the operating organisations[1] of a facilities with new or existing design.

1.5. The guidance for the deterministic safety assessment could be applied during (periodic) safety review of a nuclear facility in operation.

SCOPE

1.6. The Regulatory guide covers deterministic safety assessment performed in the process of design development, verification and review, where a conservative analysis approach is required as rule. The assessments made by the use of best-estimate analysis approach are not covered, such as the accident analyses performed for the development of emergency operating procedures, probabilistic safety analysis, operational event analysis, and for analytical validation of accident management guidelines, emergency plans and simulators.

1.7. The guidance and recommendations for safety assessment are directly applicable to nuclear power plants (NPPs), however they might be used also in the assessment of other nuclear facilities

[1] An organisation, which is a licensee or license holder according to the [1]

by applying of graded approach, where a judgment according to established criteria is made for the extent of applicability of the specific requirement to the particular facility. This approach should be used in determining the scope, extent, level of detail and effort that needs to be devoted to the safety assessment carried out for any particular facility or activity. The main factor taken into consideration in the application of a graded approach to the safety assessment should the magnitude of the potential radiation risks arising from the facility or activity. This needs to take into account any releases of radioactive material in normal operation, the potential consequences of anticipated operational occurrences and accidents, and the possibility of occurrence of very low probability events with potentially high consequences. Other relevant factors are the use of proven practices and procedures, proven designs and the complexity of the facility.[2]

1.8. The guidance for performing deterministic analysis as a method of the safety assessment relate mainly to NPPs and are focused on the 'internal' events and transients originating in the reactor or in its associated process systems. It excludes analytical methods used for events affecting broad areas of the plant, such as fires, floods, earthquakes, and aircraft crashes.

STRUCTURE

1.9. In separate sections of the Regulatory guide are presented the guidance to the engineering aspects of the safety assessment and the deterministic analysis, including the selection and categorization of the initiating events, the acceptance criteria, the analysis methods and rules, the analysis tools and the representation and assessment of the analysis results.

SAFETY ASSESSMENT

1.10. The safety assessment is a systematic process performed to verify and confirm the compliance of a proposed or existing design of a NPP with the established safety requirements.

1.11. The safety assessment should be based on the data received from the results of the safety analysis, previous operational experience, the results of supporting research activities and proven engineering practice.

1.12. The safety analysis should be implemented using deterministic and probabilistic methods, whereas the design safety level should be demonstrated by deterministic methods. The probabilistic analyses should be used as a complementary to the deterministic ones, as far as the probabilistic approach provides insights into plant performance, defence in depth and risk.

[2] More detailed information on the application of the graded approach is presented in Section 3 of the IAEA Safety Requirements GSR Part 4 Safety Assessment for Facilities and Activities.

2. ENGINEERING ASPECTS

IMPLEMENTATION OF THE DEFENCE IN DEPTH CONCEPT

2.1. The assessment of the implementation of defence in depth should be achieved through the demonstration of compliance with the safety requirements supported by a complete safety analysis. This assessment should confirm that possible initiating events are adequately dealt with on the respective defence in depth level by ensuring that the fundamental safety functions are performed as defined in the Regulations on implementation of [1].

2.2. The assessment of defence in depth should determine whether adequate provisions have been made at each of the levels of defence in order to:

- Address deviations from normal operation;
- Detect and intercept safety related deviations from normal operation;
- Control accidents within the limits established for the design;
- Identify measures to mitigate the consequences of accidents that exceed design limits;
- Mitigate the radiation risks of possible radioactive releases.

2.3. The safety assessment should identify the necessary layers of protection including physical barriers to confine radioactive material at specific locations and the need for supporting administrative controls to maintain them. This should include the identification of:

- Safety functions that must be fulfilled;
- Potential challenges to these safety functions;
- Mechanisms giving rise to these challenges and the responses to them;
- Provisions made to prevent these mechanisms from occurring; and
- Provisions to mitigate the consequences if the safety function fails.

2.4. Special attention has to be paid to the independence and effectiveness of the barriers, as well as to the internal and external hazards that have the potential to adversely affect more than one barrier at once or to cause simultaneous failures of safety systems.

2.5. The safety assessment should address the design provisions to detect the failure or bypass of each level of defence. The requested levels of defence should be specified for each operational mode (for example, an open containment may be allowed in certain shutdown modes, and the specified levels of defence should be available at all times when in that mode).

ASSESSMENT OF SAFETY FUNCTIONS

2.6. A key aspect of safety assessment should be the identification and assessment of the safety functions associated with a facility. This should include the safety functions associated with the engineered structures, systems and components (SSC), any physical or natural barriers and inherent safety features, and any human actions necessary to ensure the safety of the facility. An assessment should be undertaken to determine whether the safety functions can be achieved for all normal operational modes (including startup and shutdown where appropriate), all anticipated operational occurrences and the accident conditions that need to be taken into account.

2.7. The assessment of the safety functions should determine whether they will be carried out with an adequate level of reliability, whether vulnerabilities that could lead to a single failure or to a common cause failure for engineered equipment are present, and whether the SSC provided to carry out a safety function and the barriers have adequate levels of redundancy, diversity, separation, segregation, equipment qualification, etc.

APPLICATION OF PROVEN TECHNOLOGIES AND OPERATIONAL EXPERIENCE

2.8. Available operating experience should be taken into account in the safety assessment with the aim of ensuring that all relevant lessons in the area of safety have been adequately considered in the design. Operating experience feedback should be a fundamental source of information to improve the design of the plant.

2.9. For reactors of an evolutionary type, the design should use SSC with previous successful applications in operating plants. Where this is possible, the safety should be demonstrated taking into account the results of supporting research programs or relevant operational experience which has been gained at other plants.

2.10. Based on lessons learned from operating experience, safety analysis and safety research, the need for and value of design improvements beyond established practice should be considered. Where an innovative or non-proven design or design feature is introduced, compliance with the safety requirements should be demonstrated by an appropriate supporting demonstration programme and the features should be adequately tested before being put into service.

ASSESSMENT OF RADIOLOGICAL PROTECTION PROVISIONS

2.11. The safety assessment of the radiological protection provisions should address all operational states of the facility, and accident conditions. Two design objectives should be considered for normal operation and anticipated operational occurrences:

- keep the radiation doses of workers and public below the prescribed limits;
- keep the radiation doses as low as reasonably achievable.

2.12. The compliance with the first objective should be demonstrated by comparing the conservatively calculated equivalent dose with the limits specified in the national legislation. The relevant design calculations should be assessed by the designer to ensure the correctness of the input data and the validity of the methodology used.

2.13. The second design objective (meeting the ALARA principle) implies that all doses should be kept as low as reasonably achievable, taking economic and social factors into account. The process of optimization of radiation protection should involve some degree of balancing detriments (costs) and benefits (safety gains). In this optimization process, orientation values for radiation exposures and related design measures could be derived from similar existing plants with good operating records. The safety assessment should take into account the operational experience and consider additional design provisions or improvements to further reduce the radiation exposure to workers and members of the public. Such measures could be either direct (improved shielding) or indirect (reduction of equipment maintenance time).

2.14. The adequacy of design provisions for protection against accident conditions should be assessed by comparing the releases and/or the doses calculated in the safety analysis with the limits specified. It should be demonstrated for all design basis accidents that the intervention levels as specified in the Regulation on emergency planning and preparedness [7] have not been reached. In the safety assessment of beyond design basis accidents the designer should ensure that the relevant parameters for accident management and emergency planning have been adequately incorporated into the plant design. For new plant designs the protection measures should be limited in time and area in compliance with the criteria defined in section 4 (4.28) of the Regulatory Guide. The justification of the emergency preparedness provisions, including the co-ordination between the different organizations, should be documented in the Safety Analysis Report.

PROTECTION AGAINST EXTERNAL HAZARDS

2.15. The set of events, which should be addressed in the safety assessment, depends on the site chosen for the plant. The site specific hazards to be included in the design basis should be identified based on historical and physical data, and expressed by a set of values selected on the general probability distribution of each event according to specified thresholds. When such a probabilistic evaluation is not possible because of lack of confidence in the quality of data, deterministic approaches should be applied, relying upon enveloping criteria and engineering judgement.

2.16. The SSCs which are required to perform the fundamental safety functions should be designed to withstand the loads induced by the design basis events and able to perform their functions during and after such events. This should be achieved through adequate structural design, redundancy and separation.

2.17. The safety assessment should demonstrate a balanced design, where the radiological risk associated with external events does not exceed the range of radiological risk associated with the accident of internal origin. It should be verified that external events that are slightly more severe than those included in the design basis do not lead to a disproportionate increase in consequences.

2.18. The set of events addressed in the safety assessment would typically include the natural and man made events specified in Article 13 of [3]. A design basis event should be defined for each of the extreme weather conditions, including:

- Extreme atmospheric temperatures,
- Extreme cooling water temperatures and icing,
- Extreme wind loading,
- Extremes of rainfall and snowfall,
- Extreme amounts of sea vegetation.

2.19. The design basis should take into account the combinations of extreme weather conditions that could reasonably be assumed to occur at the same time.

2.20. It should be demonstrated by tests, experiments or engineering analyses that:

- structures in the nuclear power plant will withstand the loading imposed by the external events without inducing any failure of systems and components necessary to bring the plant back to and maintain it in a state where all fundamental safety functions can be guaranteed in the long term;
- safety systems can perform their safety functions in the range of conditions specified in the design basis (e.g. atmospheric temperatures, river water temperatures and levels).

PROTECTION AGAINST INTERNAL HAZARDS

2.21. The design should take into consideration specific loads and environmental conditions (temperature, pressure, humidity, radiation) imposed on structures and components by internal events specified in Article 12, para 4 of [3].

2.22. The safety assessment should demonstrate that the effects of pipe failures such as jet impingement forces, pipe whip, reaction forces, pressure wave forces, pressure buildup, humidity, temperature and radiation on components, building structures, electrical and instrumentation and control (I&C) equipment are sufficiently taken into account. Specifically, it should be shown that:

- Reaction forces have been taken into account in the design of safety classified equipment, supports for this equipment, and associated building structures;
- Components important to safety and their internals have been designed against credible pressure wave forces and flow forces;
- Pressure buildup has been considered for buildings important to safety such as the containment.

2.23. An internal flooding analysis for the relevant buildings of the plant should be performed and the following potential initiators of flooding should be considered: leaks and breaks in pressure retaining components, flooding by water from neighbouring buildings, spurious actuation of the fire fighting system, overfilling of tanks, and failures of isolating devices.

LOADS AND LOAD COMBINATIONS

2.24. Relevant safety classified structures and components should be designed to withstand all relevant loading resulting from operational states and design basis accidents including those resulting from internal and external hazards. A significant part of the safety assessment is therefore:

- To identify for each safety classified structure or component the relevant loading and loading combinations;
- To identify for each loading and loading combination the expected frequency of occurrence;
- To evaluate the stresses and strains in the safety classified structures and components for the identified loading and loading combinations;
- To evaluate the individual and cumulative damage in the structure or component taking account of all relevant deteriorations (e.g. creep, fatigue, ageing) and their potential interactions.

2.25. The set of loading and loading combinations should be complete and consistent with the assumptions of the safety analysis. The expected frequency of occurrence, together with the total number of anticipated transients during plant life, should be assessed based on historical records, operating experience, utility requirements or site characteristics.

OTHER ENGINEERING ASPECTS IMPORTANT TO SAFETY

2.26. The safety assessment should determine whether a suitable safety classification scheme has been formulated and applied to the SSC as required in Article 15 0f [3] and whether the following engineering aspects have been considered in the design:

- the materials used are suitable for their purpose with regard to the standards specified in the design and for the operational conditions that arise during normal operation and following anticipated operational occurrences or accidents that have been taken into account in the design;
- the specified equipment has been qualified to a sufficiently high level so that it will be able to perform its safety function in the conditions that it would experience in normal operation and following the anticipated operational occurrences and accidents that have been taken into account in the design;
- preference has been given to a fail-safe design where practicable, or alternatively means for detecting the failures that have occurred has been incorporated;

- any time related aspects such as ageing, wear-out or life limiting factors, such as cumulative fatigue, embrittlement, corrosion, chemical decomposition, etc. have been adequately addressed;
- procedures and measures are provided for all normal operational activities, in particular those necessary for implementation of the operational limits and conditions and those required in response to anticipated operational occurrences and to accidents, and ensure an adequate level of safety;
- the requirements for human factors, including those related to the ergonomic design, human–machine interfaces where operator actions are carried out, and future decommissioning and closure activities;
- the minimum staffing levels for maintaining safety as well as the aspects of safety culture for the existing NPPs.

3. SAFETY ANALYSIS

3.1. The safety analysis is a method of assessment used to demonstrate how safety requirements, such as ensuring the integrity of barriers against releases of radioactive material, are met for all initiating events that could occur over the range of operational states and accident conditions.

3.2. The aim of the safety analysis should be to establish and confirm, by the means of appropriate analytical tools, the design basis for the SSC important to safety, and to ensure meeting the prescribed limits for radiation doses and for each plant condition category as specified in Article 12, para 1 of [3]. The assessment of the engineering aspects important to safety (2.26 of this Guide) and the safety analysis should be carried out in parallel.

3.3. The plant design models and data, which are essential foundations for the safety analysis, should be kept up to date during the design phase and throughout the lifetime of the plant, including decommissioning. The updating process should incorporate new information as it becomes available, use more sophisticated tools and methods as they become accessible, and assess the performance of modifications to the design and operating procedures.

OBJECTIVE AND USE OF THE SAFETY ANALYSIS

3.4. The safety analysis should assess the performance of the plant against a broad range of operating conditions, postulated initiating events (PIEs) and other circumstances, in order to obtain a complete understanding of how the plant is expected to perform in these situations and to demonstrate that the plant can be kept within the safe operating regimes established in the design.

3.5. The safety analysis should identify potential weaknesses in the design, evaluate proposed design improvements and provide a demonstration that safety requirements and criteria are met. The safety analysis should assist in revealing initiating events and plant conditions that were not adequately considered in the early stages of design.[3]

3.6. The safety analysis should support safe operation of the plant by serving as an important tool in developing and confirming plant operational limits and conditions, including the margin to the safety system set points. It should also be used to establish and validate the normal operating procedures, maintenance and inspection requirements, and emergency procedures and guidelines.

[3] The safety analysis could also reveal the lack of necessity to consider particular initiating event, acceptance criterion or other aspect that does not impair safety due to its extremely low probability of occurrence, insignificant conditional probability or minimum impact on the potential consequences.

3.7. The safety analysis should also support the plant management decision making processes as new issues and questions arise during the life of the plant. The plant's initial safety analysis and the ability to re-perform all or part of that analysis to resolve new technical issues should be maintained over the life of the plant.

3.8. The safety analysis should assess whether:
- Sufficient defence in depth has been provided and the levels of defence are preserved;
- The SSC can withstand the physical and environmental conditions it would experience;
- Human factors and human performance issues have been adequately addressed.
- Long term ageing mechanisms that could detract from the SSC reliability over the plant life are identified, monitored and managed so that safety is not affected.

3.9. The safety analysis process should be with sufficient scope, quality, completeness and accuracy to engender the confidence of the designer, the operating organization, the regulator and the public in the safety of a plant's design.

DETERMINISTIC AND PROBABILISTIC ANALYSIS

3.10. According to the requirements set in [3], the safety analysis should incorporate both deterministic and probabilistic approaches. These approaches have been shown to complement each other.

3.11. The aim of the deterministic approach should be to address plant behaviour under specific predetermined operational states and accident conditions and to apply a specific set of rules in judging design adequacy. The probabilistic safety analysis (PSA) should set out to determine all significant contributors to risk from the plant and should evaluate the extent to which the design of the overall system configuration is well balanced, there are no risk outliers and the design meets basic probabilistic targets.

3.12. The insights gained from the deterministic analysis and the PSA should both be used in the decision making process. There are situations where the insights gained from the deterministic analysis and the PSA are not consistent. These should be considered on a case-by-case basis.

ESSENTIAL INFORMATION

3.13. The safety analysis process should be based on plant design information that is complete and accurate and should cover all plant SSC and site-specific characteristics. The plant design should be documented and the documentation should be kept up to date.

3.14. For an operating plant, the safety analysis used, for example, for design modifications should use plant specific operational data. This includes information on the radiological doses to operators during normal operation and routine discharges of radioactive material from the site. For plant systems, data collected should include normal operating temperatures, pressures, fluid levels and flow rates, and the transient response characteristics and timing for any operational occurrences. The operational data should also include information on component and system performance, initiating event frequencies, component failure rate data, modes of failure, system unavailability during maintenance or testing, and component and system repair times.

3.15. For a plant in the design phase, the data used should be derived from generic data from operating plants of similar design, or from research or test results. After the plant commissioning, some aspects of this generic database can be enhanced over time with plant specific data from the plant's own historical operating and maintenance data and experience and inspection results.

3.16. The safety analysis should cover all the sources of radioactive material in the plant. In addition to the reactor core, this includes irradiated fuel and stored radioactive waste.

4. DETERMINISTIC SAFETY ANALYSIS

4.1. Deterministic safety analyses predict the response of a NPP to postulated initiating events applying a specific set of rules and acceptance criteria. Typically, they focus on neutronic, thermohydraulic, radiological, thermo-mechanical and structural aspects, which are often analysed with different computational tools. The computations are usually carried out for predetermined operational states and accident conditions and include steady and transient states during normal operation, anticipated operational occurrences, postulated accidents, selected beyond design basis accidents (BDBAs) without core degradation as part of the design extension, and severe accidents. The results of computations are spatial-time dependencies of physical variables[4] or, the dose to workers or the public in the case of an assessment of radiological consequences.

4.2. Deterministic safety analyses for design or design modification purposes should be characterized by their conservative and bounding nature. This is achieved by an iterative process during the design phase where the limiting case or cases in terms of the minimum margin to the acceptance criteria are determined for each group of initiating events and sequences. In order to determine the limiting case, the consequential failures that are caused by the initiating event (internal or external) should be taken into account, coincident independent single failure[5], as well as an adequate set of conservative assumptions for the initial and boundary conditions. The analyses should cover all plant states and power levels.

4.3. The plant states to be reached for ending the safety analyses should be defined. The following end states are typically used in the analyses:

- Controlled state, defined as an intermediate state, reached only with automatic actions, when the fast transient is finished and the plant is stabilized,
- Safe shutdown state – following a DBA,
- Final state for BDBA analysis.

4.4. The end states should be defined for the analyses at all power levels and at shutdown conditions.

SELECTION OF POSTULATED INITIATING EVENTS AND PLANT STATES FOR THE SAFETY ANALYSIS

4.5. The term initiating event refers to an event, including operating errors or equipment failures or internal or external hazards, which, directly or indirectly challenges one or more safety functions. The set of PIEs developed for the safety analysis should be comprehensive and should be defined in such a way that they cover all credible failures of plant systems and components and human errors which could occur during any of the operating regimes including steady and transient states.

[4] such as neutron flux, thermal power of the reactor, pressure, temperature, flow rate and velocity of the primary coolant, stresses in structure materials, physical and chemical compositions, concentrations of radionuclides, etc.
[5] in correspondence with article 19, papa 2, t.4 of [3].

4.6. The set of plant specific PIEs should be identified in a systematic way. This should include adopting a structured approach to the identification of the PIEs.[6], where the following approach is recommended:

- Identification of all barrier failure mechanisms;
- Identification of all physical processes that could cause the failure mechanisms to start;
- Grouping of those processes by phenomenology;
- Identify scenarios for each of the above groups;
- Postulate initiating events that result in the above scenarios;
- Determine original cause – internal events, external events.

4.7. All the PIEs should be defined quantitatively in terms of their frequency of occurrence. Events of very low frequency or consequences should be included the beginning of the screening process, where it may be possible to eliminate some PIEs. Nevertheless, the elimination of any PIEs should be fully justified and the reasons well documented. The set of PIEs should be reviewed as the design and safety assessments proceed and should involve an iterative process between these two activities.

4.8. In accordance with the provisions set in article 12, para 1 of [3], the initiating events should be grouped into 4 categories that correspond to plant states, according to their annual probability of occurrence. They include steady and transient states during normal operation, anticipated operational occurrences and postulated accidents. Typical sets of PIEs and categorization of plant states, that should be considered in the safety analysis for pressurized and boiling water reactors, are provided in Attachment to 12 (1) of [3].

4.9. To limit the number of computational analyses, the initiating events might be further grouped using different approach based on the principal effect that could result in the degradation of safety functions. In this case the event groups typically include the following:

- Increase or decrease in heat removal from the reactor coolant system;
- Increase or decrease in reactor coolant system flow rate;
- Reactivity and power distribution anomalies;
- Increase or decrease in the reactor coolant inventory;
- Release of radioactive material from a subsystem or component.

4.10. A reasonable number of limiting (bounding) scenarios should be selected from each group of events, so that they present the greatest challenge to the relevant acceptance criteria and are limiting for the performance parameters of safety related equipment. There may be additional sequences to be analysed, such as an anticipated transient without scram (ATWS).

4.11. A different grouping might be more useful also when calculating the potential release of radioactive material to the environment. It is important to identify the accidents where major barriers, such as the containment, may be ineffective and ensure that analyses are performed for these. Examples include steam generator tube ruptures as initiating or consequential events, loss of coolant accidents outside the containment and faults at shutdown conditions when the containment is open.

[6] Such approach could include the following:
- Use of analytical methods such as failure mode, effect analysis (FMEA), hazard and operability analysis (HAZOP), and master logic diagrams;
- Comparison with the list of PIEs developed for safety analysis of similar plants (although this method should not be exclusively used since prior mistakes could be propagated);
- Analysis of operating experience data for similar plants.

4.12. Credible combinations of individual events, including internal and external hazards, that could lead to anticipated operational occurrences or design basis accident conditions, should be considered in the design. Engineering judgement and probabilistic methods can be used for the selection of the event combinations.

4.13. Complex sequences resulting from multiple failure conditions, as well as severe accidents have to be considered in the safety demonstration in addition to the analysis of PIEs. The significant event sequences that could lead to multiple failures conditions and severe accidents should be identified using a combination of probabilistic and deterministic methods and sound engineering judgment. These sequences should be selected by adding additional failures or incorrect operator responses to the PIE sequences (to include safety system failure) and to the dominant accident sequences from the PSA.

4.14. For new plant designs the accident sequences that could lead to multiple failure conditions, should be considered in the design basis and therefore their selection should be made at the design stage using the results of the preliminary PSA.

4.15. For existing plants the multiple failure conditions are usually considered as part of the design extension and might be categorised as beyond design basis accidents without severe core degradation in accordance with article 14 of [3]. In this case, the PSA results will have to be used to check the sufficiency of the foreseen additional safety improvement measures.

4.16. The multiple failure conditions typically include.
- anticipated transients without scram, if not considered as DBA;
- station blackout (Loss Of Off-site Power, plus turbine-generator trip, plus failure of the main on-site emergency AC electrical supply)[7];
- for PWRs, Main Steam-Line Break plus consequential Steam-Generator Tube Ruptures (SGTR);
- containment system bypass accidents, including multiple SGTR;
- loss of core cooling in residual heat removal mode;
- total loss of the spent fuel pool cooling system, etc.

4.17. Certain unlikely event sequences involving significant core damage (severe accidents) that have a potential to lead to significant releases into te environment, should be addressed in the design and safety assessment. The most rigorous way of identifying severe accident sequences is to use the results of the Level 1 PSA. However, it might also be possible to identify representative or bounding sequences from an understanding of the physical phenomena involved in severe accident sequences, the margin existing in the design, and the amount of system redundancy remaining in the DBAs..

4.18. For new plant designs the first priority is to prevent by a robust design ("practically eliminate") the severe accidents that could lead to large or early releases to the environment, including high pressure core melt scenarios, if they cannot be excluded as physically impossible. Each representative or limiting accident sequence should be assessed for the purpose of practical elimination. The prevention of a particular sequence should be demonstrated primarily by deterministic arguments complemented with probabilistic, where appropriate, taking into account the uncertainties resulting from the limited knowledge about particular physical phenomena. Important fact is that the practical elimination of accident sequences should not be demonstrated solely by very low frequency of occurrence, below an established threshold value.

[7] In the analysis of SBO sequences as DEC, proper credit could be given to diversified on-site power sources.

4.19. For the other severe accident scenarios, including low pressure core melt, the consequences should be evaluated in order to demonstrate the compliance with the acceptance criteria set in this Guide and with the requirements established by the Regulations under [1].

4.20. For existing plants it is needed to be shown that the sequences leading to severe accident are of very low frequency. The safety improvement measures resulting from the safety analyses should be assessed and implemented as far as they are reasonably practicable.

ACCEPTANCE CRITERIA

4.21. The main acceptance criteria for the results of the deterministic analyses represent a quantitative limit of specified calculated parameters, or qualitatively expressed requirements, conditions or limits established to be applied to the analysis result. The main acceptance criteria for the results of the safety analyses of existing NPPs in operational states and accident conditions are specified in the Regulations on implementation of [1].

4.22. For demonstration of safety of a NPP the following major acceptance criteria should be met:
- The annual individual effective doses from internal and external exposure of the population in all operational states and in accident conditions should be kept below the established limits and as low as reasonably achievable by implementation of measures for mitigation the radiological consequences;
- Maintaining the integrity of the barriers for the ionizing radiation (fuel pellet, fuel cladding, reactor pressure boundary and containment structure) depending on the category of the plant state for all accidents, for which their integrity is required;
- Ensuring the implementation of the safety functions by the designated SSC and of the operator actions in accidents, where this is needed;
- Practical elimination of large[8] or early[9] releases into the environment by the means of design provisions for new NPPs.

4.23. In addition to the main acceptance criteria, specific technical acceptance criteria (or surrogate criteria) should be defined in order to ensure the defence in depth and to confirm the adequacy of the margin to the main criteria. The surrogate criteria for the physical process should be set in such a way that, if not exceeded, will ensure that the barrier integrity limit, is not reached. Examples of surrogate variables are: peak clad temperature, departure from nuclear boiling ratio, fuel pellet enthalpy rise, etc.

4.24. Radiological and technical acceptance criteria should be assigned to each plant state such that frequent initiating events shall have only minor or no radiological consequences and that events that may result in severe consequences shall be of very low probability.

4.25. In defining the technical acceptance criteria for the DBA analyses of categories 1-4 the following aspects should be taken into account:
- For transients of categories 1 and 2, the integrity of the fuel cladding has to be maintained. This implies to define a limit for the departure from nucleate boiling ratio and, possibly, a criterion concerning pellet-cladding interaction.

[8] Accident conditions, for which the intervention level have been reached but the protection measures cannot be limited in area and time

[9] Accident conditions, for which the intervention levels have been reached, but the time for implementation of the protection measures is insufficient

- The acceptance criteria for the consequences of reactivity induced accidents of category 4, including the maximum specific energy released in the fuel, should be justified on the basis of detailed analytical and experimental investigations taking into account the precise characteristics of the fuel and the associated burn-up.
- In the analysis of loss of coolant accidents of categories 3 and 4 the acceptance criteria specified in article 20, i.2 of [3] should be applied It is also necessary to be shown the capability for long term core cooling and prevention of conditions which could lead to extended fuel damage.

Other technical criteria have to be proposed and justified, to deal with :

- the maximum numbers of fuel rods which could experience departure from nucleate boiling in plant conditions of categories 3 and 4,
- the maximum peak cladding temperature for fast transients to avoid cladding embrittlement in plant conditions of categories 3 and 4,
- the subcriticality requirements related to the shutdown states with regard to the accident conditions, which might occur during these states.

4.26. For demonstration of the barrier integrity in multiple failure conditions (beyond design basis accidents without severe core degradation) analysed as a design extension conditions, the radiological acceptance criteria set for category 4 plant states (DBA) should be applied.

4.27. For anticipated transients without scram when they are not considered as category 4 events (DBA), the maximum pressure of the primary and secondary circuits should be deemed acceptable if it does not exceed 1.3 times its design pressure for any core configuration.

4.28. For new plant designs the radiological consequences following multiple failure conditions including containment bypass should be judged acceptable if there is no need of implementation of protection actions for the public (sheltering and evacuation) in the vicinity of the plant site.,

4.29. As mentioned in section 4.18 of the Guide, large or early releases into the environment, including those resulting from a high pressure core melt scenario, should be prevented by incorporating measures for their practical elimination in the design of new NPPs,. Low pressure core melt sequences have to be dealt with so that the associated maximum conceivable releases would necessitate only very limited protective measures for the public. This would be expressed by no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant and no long term restrictions in consumption of food. The measures including sheltering and protection of the skin and the inhalation organs should be limited in time.

4.30. For the existing plants, the above mentioned criteria should be considered as reference basis for identification of reasonably practicable safety upgrading measures. The analyses of BDBA, including of severe accidents with account taken of the accident management measures, should demonstrate that the containment function is effective and the following conditions are met:

- possibility for containment isolation, including in case of severe accident occurring at reactor shutdown state with open containment[10]. If the accident is associated with containment bypass, mitigation measures should be considered;
- possibility for management of the pressure, temperature an concentration of combustible gases in the containment during severe accident;

[10] For these states the possibility for containment recovery within a justified period of time should be shown, or compensatory measures with the same effect should be required to be applied.

- possibility to keep the containment leak tightness for a justified period of time following severe accident;
- prevention of the high pressure core melt scenarios;
- prevention or mitigation of the containment degradation from direct containment heating.

ANALYSIS RULES AND ASSUMPTIONS

4.31. The safety analyses should be performed according the specific methodologies that identify the initial and boundary conditions, the applicable acceptance criteria, the availability assumptions with respect to the systems and components and the use of operator actions.

4.32. The analyses of plant states 1-4 should be performed using a conservative approach that implies use of unfavourable values of the input parameters with respect to their impact on the specific acceptance criteria. The analytical approach applied to the BDBA might be with relaxed conservatism with regard to the analysis assumptions, however the acceptance criteria for the corresponding plant state should be met.

4.33. The traditional conservative analyses use unfavourable values to characterize the plant state (initial and boundary conditions, availability assumptions) and the physical models of the computer codes. The analytical approach that combines the use of best-estimate computer code with conservative input data and availability assumptions could also be considered as conservative. However, for both options it should be demonstrated that the calculated results are conservative for each application.

4.34. The safety analysis incorporates predictions for the prevailing plant behaviour and this will always be associated with uncertainties that will depend mainly on the data accuracy. Therefore, the uncertainties should be characterized with respect to their source, nature and degree, and taken into account when considering the analysis results. Uncertainties that may have implications for the outcome of the safety analysis and for decisions made on that basis are to be addressed in uncertainty and sensitivity analyses. Uncertainty analysis refers mainly to the statistical combination and propagation of uncertainties in data, whereas sensitivity analysis refers to the sensitivity of results to major assumptions about parameters, scenarios or modelling.

Thermal hydraulic analyses of DBA and BDBA

4.35. The initial conditions are the assumed values of plant parameters at the start of the transient to be analysed. Examples of these parameters are reactor power level, power distribution, pressure, temperature and flow in the primary circuit.

4.36. The boundary conditions are the assumed values of parameters throughout the transient. Examples of boundary conditions are the actuation of safety systems, leading to changes in flow rates, external sources and sinks for mass and energy, and other parameters during the course of the transient.

4.37. For the purpose of conservative calculations, the initial and boundary conditions should be set to values that will lead to conservative results for those safety parameters that are to be compared with the acceptance criteria. One set of conservative values for initial and boundary conditions does not necessarily lead to conservative results for every safety parameter. Therefore, the appropriate conservatism should be selected for each initial and boundary condition, depending on the specific transient and the associated acceptance criterion.

4.38. In conservative analyses, the single failure criterion should be applied when determining the availability of systems and components.. A failure should be assumed in the system or component that would have the largest negative effect on the calculated safety parameter. Single failure might not be applied to a passive component in the cases where the extremely low probability of such failure has been justified and the component operability is not affected by the initiating event.

4.39. All the common cause and consequential failures associated with the postulated initiating event should also be included in the analysis in addition to the single failure. The system availability assumptions should be applied in a way that has the most adverse effect on the propagation of the initiation event. Furthermore, component unavailability due to on-line maintenance should be considered if demands have been made to the design.

4.40. In addition to the postulated initiating event, a loss of off-site power (LOOP) should be considered when analysing accidents of categories 3 and 4, if the assumption gives the most negative effect on the margin to the acceptance criterion. The initiating event should be assumed to precede the LOOP.

4.41. Likewise, equipment that is not qualified for specific accident conditions should be assumed to fail unless its continued operation results in more unfavourable conditions. The malfunction of control systems and delays in the actuation of protection systems and safety systems should be considered in the analysis (the issue of whether their continued functioning leads to more unfavourable conditions than their non-availability should be addressed).

4.42. For design purposes, manual operator action from the main control room to limit the evolution of a design basis accident can be assumed to take place at the earliest 30 min after the first significant information is given to the operator. For a local manual action, outside the main control room, the earliest time to be taken into account is 1 h. Exceptionally, the design may take credit for earlier operator action but, in these cases, the actuation times should be conservative and should be fully justified.

4.43. Although conservative and bounding analyses should be used for design purposes, it is practical to use more realistic analyses to evaluate the evolution and consequences of beyond design basis accidents, including severe accidents. In these cases it is appropriate to use best estimate methods and codes with an evaluation of the uncertainties.

4.44. For the assessment of multiple failures conditions, all systems can be deemed available, except those which are assumed to have failed in the multiple failures combination. No additional failure and no unavailability due to maintenance have to be deterministically postulated in the systems needed to reach the final state.

4.45. Severe accident conditions have to be assessed in the safety demonstration of nuclear power plants by imposing additional equipment failures leading to significant core degradation. Examples of such conditions are:

- loss of offsite power with unavailability of all the diesel generators, i.e. station blackout as combined with the unavailability of the diverse power supply source,
- total loss of feedwater combined with a failure of the primary feed and bleed,
- small break loss of coolant accident with the complete failure of the safety injection system,
- loss of coolant accident up to the surge line break with the complete failure of the safety injection system.

4.46. The uncertainties related to some of the phenomena which could occur during severe accidents sequences call for the consideration of various scenarios and the performance of sensitivity studies.

4.47. Concerning the loads resulting from hydrogen combustion, with the prevention of hydrogen detonation concept applied, local dynamic effects due to such phenomena as fast deflagration or deflagration to detonation transition sequences are only expected on internal structures of the containment building. For the internal wall of the containment, it has also to be demonstrated that, taking into account the mitigation means and whatever the selected scenario, the pressure load resulting from an adiabatic, isochoric and complete hydrogen combustion does not exceed the containment design pressure at any time.

Analysis of radiological consequences

4.48. In order to evaluate the source term from an NPP during operational states and under accident conditions, it is necessary to know the sources of radioactivity, evaluate the inventories of radionuclides that may occur on the plant and know the mechanisms by which radioactivity can be transmitted through the plant and released to the environment. The exposure pathways should be considered in the analysis of the radiological consequences for the public.

4.49. The evaluation of the source terms for normal operational states should include all the radionuclides that may make a significant contribution to the dose due to either liquid or gaseous discharges, such as corrosion and fission products.

4.50. For accident conditions, the evaluation of the source term should involve determining the behaviour of the radioactive species along this route, their retention in the containment, their release to the secondary containment, if one is provided, and the subsequent release to the atmosphere.

4.51. Separate analyses of the source term in the containment atmosphere should be carried out for each type accident sequebce, where the phenomena that would affect the source term would be different. For example:

- Reactivity faults where the rapid increase in reactivity would result in an enhancement of the release of fission products from the fuel matrix to the fuel-cladding gap and the failure of some of the fuel cladding;
- Large LOCA where the severe transient would also lead to an enhancement of the release of fission products from the fuel matrix to the fuel-cladding gap and failure of some of the fuel cladding;
- Small LOCA where the transient would be less severe and the release to the fuel-cladding gap would not be significantly enhanced but some of the fuel cladding may fail;
- Very small LOCA where the loss of coolant is less than the make-up flow, no fuel failures would occur and the release into the containment is limited to the radioactivity that is in the primary coolant.

4.52. A similar range of different types of faults should be considered in the evaluation of the source terms that would result from BDBA, including severe accidents that involve significant core degradation. In this case, the very small LOCA would not apply.

4.53. The evaluation of source terms should also include a comprehensive analysis of accidents in which the release of radioactivity occurs outside the containment. Accidents in which the release can bypass the containment[11] are a very important category because bypass accidents with a relatively small release from the fuel may have the same radiological consequences as an accident with a large release into the intact containment. Examples of bypass accidents in pressurized water reactors include:

- Leaks or pipe breaks in the secondary circuit accompanied by a steam generator tube rupture;
- Leaks or pipe breaks in systems that are connected directly to the primary circuit if these systems are outside the containment.

In addition, the source term should be evaluated also for other different types of accidents that would result in a release of radioactivity outside the containment, such as:

- Loss of cooling to the fuel in the spent storage pool;
- Fuel handling accidents;
- Leak or pipe break in any of the other auxiliary systems that carry radioactive liquid or gas;
- Fires or other вътрешни hazards;
- External hazards.

4.54. A comprehensive identification of all accident sequences with potential for releases will result in a large number and therefore it is practicable to group them and, for each group, to choose a bounding case for which the source term will be evaluated and will be considered to be representative of the other accidents in the same group. In each frequency band, the source term should be evaluated for each type of accident that will result in the greatest radiological consequence.

4.55. For a first approach at the design stage, the following assumptions can be made for the large break loss of coolant accident inside the containment, if they do not result in violation of the radiological acceptance criteria:

- fuel cladding failure rate: 1% for category 3 accidents and 10% for category 4 provided that these values have been justified, taking into account the fuel composition and burn up,
- primary containment structure leak rate: 1% per day of the free volume of the internal containment (with no direct leak to the outside),
- annulus filters efficiency: 1000 for molecular iodine and for aerosols, 100 for organic iodine.

4.56. If actions affecting the isolation of a leak or the dispersion of radioactive substances are automatic and designed to cope with a single failure, these actions can be assumed to be effective.

4.57. Radiological consequences must notably be calculated for accident situations during shutdown states, including a guillotine break of the residual heat removal system outside the containment building as well as for accident situations with long term circulation of contaminated fluids outside the containment building.

4.58. The methodology and assumptions to be applied to the determination of potential radiological consequences of multiple failure conditions are similar to those applied to design basis accidents of categories 1-4.

4.59. It is underlined that the calculations of radiological consequences for all conditions, including core melt situations, should deal with short term and long term consequences, considering the different ways of transferring radioactive materials to the environment (air, surface water, underground water) and to men (irradiation -cloudshine and groundshine-, radionuclides intake by

[11] Severe accidents with containment by-pass should be practically eliminated in the design of new plants.

ingestion or inhalation). Therefore it is necessary to be determined the atmospheric dispersion and deposition on vegetation, soil and other surfaces as well other exposure conditions.

4.60. The final results of the assessment of the selected design basis accidents with radiological significance should include individual effective doses of members of the public as a result of internal and external exposure at the boundary of the radiation protection area and beyond it for the first year following the accident. The calculated results should be compared with the legally prescribed limits. Additionally, it should be demonstrated for DBA that there is no need for intervention and implementation of any urgent protection action for the population.

4.61. The calculation results for the radiological consequences resulted from multiple failure conditions, including containment by-pass, should comply with the acceptance criteria specified in this Guide (no need for urgent protection actions – sheltering and evacuation – in the immediate vicinity if the plant site).

Use of computer codes and models

4.62. Any calculational methods and computer codes used in the safety analysis have to undergo verification and validation to a sufficient degree as required in article 18, para 2 of [3].

4.63. Model verification is the process of determining that a computational model correctly implements the intended conceptual model or mathematical model; that is, whether the controlling physical equations and data have been correctly translated into the computer code. System code verification is the review of source coding in relation to its description in the system code documentation.

4.64. Model validation is the process of determining whether a mathematical model is an adequate representation of the real system being modelled, by comparing the predictions of the model with observations of the real system or with experimental data. System code validation is the assessment of the accuracy of values predicted by the system code against relevant experimental data for the important phenomena expected to occur. In the validation process should be identified and specified the uncertainties, approximations made in the models, any shortcomings in the models and the underlying basis of data, and how these are to be taken into account in the safety analysis.

4.65. In addition, it has to be ensured that users of the code have sufficient experience in the application of the code to the type of facility to be analysed. In order to limit the user effects, the procedures, code documentation and user guidelines should be carefully followed. In some cases, the results produced by conservative analysis are sensitive to decisions that are made by the user, such as the number and structure of nodes that are used.

4.66. For each of the computer codes used to justify the design, it should be specified its experimental validation and qualification and how the remaining uncertainties are taken into account (e.g. sensitivity studies). This applies to computer codes used for neutronic and thermalhydraulic, and notably to computer codes of the new generation (3 D neutronic and thermalhydraulic coupled computer codes), in order to demonstrate that the envelope values determined by the results are actually conservative for the whole set of plant states studies.

4.67. For each of the computer codes used for severe accident analysis, it should be specified its experimental validation and qualification and how the remaining uncertainties are taken into account (e.g. sensitivity studies).

Presentation and interpretation of the results

4.68. The results of the safety analysis should be documented in a Safety Analysis Report. They should to be structured and presented in an appropriate format in such a way as to provide a good understanding and interpretation of the course of the accident and to permit easy checking of each individual acceptance criterion. A standardized format is suggested for similar analyses to facilitate interpretation and intercomparison of results. Each analysed case needs to be clearly characterized by a description of its conditions, including:

- Definition of initiating events,
- Initial conditions of the system,
- Control system conditions and logic,
- Availability of systems and components,
- Method of analysis,
- Acceptance criteria.

The summary report of the accident analysis results needs to contain the following information:

- A chronology (timing) of the main events as calculated,
- A description and evaluation of the accident on the basis of the parameters selected,
- Figures showing plots of the main parameters calculated,
- A statement in relation to the fulfilment of the acceptance criteria,
- An evaluation of alternative scenarios (alternative conditions and sensitivity studies),
- References.

4.69. The results of the analysis and particularly the key parameters defining the status of the safety functions during the development of the process, need to be presented and described in detail.

4.70. The presentation of the results needs to include a set of the important parameters in the course of a transient or accident as a function of time. This set needs to include all the parameters necessary to evaluate the status of the safety functions and the fulfilment of the acceptance criteria. It also needs to give information concerning the overall plant behaviour. Some of the parameters to be included in the lists are:

(1) Neutron power, decay heat and reactivity;
(2) Thermal power and heat fluxes in the core;
(3) Minimal departure from nucleate boiling ratio;
(4) Primary coolant conditions — temperatures, void fractions, flow and pressure;
(5) Maximal fuel temperatures;
(6) Maximal cladding temperatures;
(7) Reactor coolant inventory — total inventory and levels at key locations;
(8) Secondary system parameters showing heat flows;
(9) Containment pressure, temperature and the mass flow rate to the containment, if applicable;
(10) Activity of the release to containment and to the environment, if applicable;
(11) Hydrogen generation and distribution within containment;
(12) Level of core degradation, if applicable;
(13) Long term pressure buildup in the containment, if applicable;
(14) Parameters defining the performance of safety systems.

4.71. If radioactivity transport is involved in the analysis, the list of reported parameters needs to further include:

- Concentrations of radionuclides in the fuel, the coolant and their deposits on the reactor coolant system structures;

- Concentrations of radionuclides in the containment atmosphere and in deposits on the containment structures;
- Quantity, composition and time period for the discharge of radioactive materials to the environment;
- Annual individual effective dose of the public as a result of internal and external exposure at a various distances from the plat, suitable for demonstration of the implementation of the specified limits.

4.72. The format and structure of the results needs to be chosen in such a manner as to show:
- The sequence of events and system operation in the course of the accident (from initial state to the final safe stable state);
- Core and system performance;
- Physical barrier performance;
- Radiological consequences,;
- Assessment of the results and conclusions.

5. GLOSSARY

Acceptance criteria are quantitative limitations of selected parameters or qualitative requirements set up for the results of safety analysis. They comprise specified bounds on the value of a functional or conditional indicator used to assess the ability of a system, structure or component to perform its design functions [11].

Accident conditions deviations from the normal operation that are more severe than the anticipated operational occurrences and include DBA and BDBA [3].

Anticipated operational occurrence is an operational process deviating from normal operation which is expected to occur at least once during the plant operating lifetime but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions [3].

Conservative assumptions for availability of systems are the assumptions about availability of equipment chosen to give a pessimistic result in a safety analysis.

Conservative input data are initial, boundary conditions and plant parameters leading to pessimistic results in a safety analysis.

Operational states include normal operation as well as anticipated operational occurrences [3].

Realistic assumptions for availability of systems are the equipment availability assumptions chosen to give a realistic (also ‘as built’, ‘as operated’) result in a safety analysis.

Realistic input data. Realistic input data are initial, boundary conditions and plant parameters chosen to give a realistic (also ‘nominal’, ‘as operated’) result in a safety analysis.

Safety assessment a review of all aspects of the design and operation of a nuclear facility or another source of ionising radiation which are relevant to its safety and to the protection of persons, including an analysis of the provisions for nuclear safety and radiation protection and of the risks associated with normal operation and with accidents [1].

Severe accidents are plant states with significant core degradation [3].

Source term is the amount and isotopic composition of material released or postulated to be released from a facility to the atmosphere or surface waters during operational states or to the atmosphere following an accident. It includes the physical and chemical form of the radioactive material and the timing, height, energy, mode.

6. REFERENCES

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[4] Regulation for safe management of spent nuclear fuel, promulgated in the State Gazette No. 71 of 13 August 2004.

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