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1. IDENTIFICATION

Document Category: Specific Safety Guide

Working ID: DS536

Proposed Title: Safety Assessment and Verification for Nuclear Power Plants

Proposed Action: Development of a new IAEA Safety Guide (NUSSC 52 Meeting Minutes, item 4.1)

Review Committee(s) or Group: NUSSC, EPReSC, NSGC

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2. BACKGROUND

Recommendations related to the safety assessment process for nuclear power plants (NPPs) were previously provided in IAEA Safety Standards Series No. NS-G-1.2, Safety Assessment and Verification for Nuclear Power Plants, published in 2001.

The recommendations in NS-G-1.2 mainly focused on the deterministic safety analysis, the probabilistic safety assessment and the engineering aspects of items important to safety for nuclear power plants. This Safety Guide was superseded by IAEA Safety Standards Series Nos GSR Part 4, Safety Assessment for Facilities and Activities, and SSG-2, Deterministic Safety Analysis for Nuclear Power Plants. These are now GSR Part 4 (Rev. 1) and SSG-2 (Rev. 1).

In addition, IAEA Safety Standards Series Nos SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, and SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, were developed to provide recommendations for the development and application of Level 1 and Level 2 PSA for NPPs.

As a result of these developments, IAEA Safety Standards provide recommendations for conducting probabilistic safety assessments and the deterministic safety analyses but not for the safety assessment and verification of engineering aspects of items important to safety of NPPs from a design perspective.

3. JUSTIFICATION FOR THE PRODUCTION OF THE DOCUMENT

As stated in para. 4.16 of GSR Part 4 (Rev. 1), the main elements for the performance of the safety assessment and verification for facilities and activities include the following:

- (a) Preparation for the safety assessment, in terms of assembling the expertise, tools and information required to carry out the work;
- (b) Identification of the possible radiation risks resulting from normal operation, anticipated operational occurrences or accident conditions;

- (c) Identification and assessment of a comprehensive set of safety functions;
- (d) Assessment of the site characteristics that relate to the possible radiation risks;
- (e) Assessment of the provisions for radiation protection;
- (f) Assessment of engineering aspects to determine whether the safety requirements for design relevant to the facility or activity have been met;
- (g) Assessment of human factor related aspects of the design and operation of the facility or the planning and conduct of the activity;
- (h) Assessment of safety in the longer term, which is of particular concern when ageing effects might develop and might affect safety margins, decommissioning and dismantling of facilities, and closure of disposal facilities for radioactive waste.

Currently, there are IAEA Safety Guides for conducting Deterministic Safety Analysis (SSG-2 (Rev. 1)), Level 1 and of Level 2 Probabilistic Safety Assessment for NPPs (SSG-3 and SSG-4, for which revisions are being prepared) and for the Periodic Safety Review for NPPs (SSG-25). However, there are no recommendations for conducting the assessment and verification of the design and the substantiation of key engineering aspects of items important to safety of NPPs (item (f) above). Full details are provided in the attached gap analysis.

In addition, during the 52nd NUSC Meeting, under item 4.1 as part of its medium-term work plan, NUSC members agreed on the development of the DPP for a new Safety Guide on safety assessment and verification for NPPs. Therefore, the development of this DPP aims at answering the NUSC members request.

The proposed new Safety Guide aims at providing recommendations to meet the requirements established in GSR Part 4 (Rev. 1) for conducting a comprehensive and cross-cutting safety assessment and verification process with regard to engineering aspects of items important to safety of NPPs.

Additionally, the Safety Guide will provide the basis for conducting the Technical Safety Review (TSR) service related to the verification of the design safety of NPPs, which is currently based on the guidelines presented in IAEA Services Series No. 41.

4. OBJECTIVE

The objective of this Safety Guide is to provide recommendations for meeting the requirements of GSR Part 4 (Rev. 1) regarding the comprehensive assessment and verification of key aspects considered in the design of engineering aspects of items important to safety of NPPs. The recommendations will target the phases of review for authorization (licensing) of the construction, modification and operation of new NPPs, and the re-evaluation of safety of existing NPPs during periodic safety reviews (e.g. for incorporating operating experience feedback, research and technical results and developments, improvements related to ageing effects as well as to in-service inspection, testing, monitoring and maintenance activities, site environmental changes, and new regulatory requirements).

In addition, the recommendations provided in this Safety Guide will focus on the assessment and verification of compliance with the requirements for the design and operation of items important to safety of NPPs established in SSR-2/1 (Rev. 1) and SSR-2/2 (Rev. 1), the requirements for radiation protection established in GSR Part 3, as well as requirements established in SSR-1 for the protection of items important to safety against hazards from site characteristics. These recommendations will

complement the recommendations in the IAEA Safety Guides on deterministic safety analysis as well as on probabilistic safety assessment.

It is expected that the Safety Guide will promote consistency in conducting the safety assessment and the verification of the design of engineering aspects of items important to safety of NPPs. In addition, this Safety Guide will provide a standard framework to facilitate a regulatory reviews and independent or peer reviews (e.g. TSR) of the safety assessment and its applications.

The Safety Guide is intended for use by designers, operating organizations, technical support organizations and regulatory bodies conducting the safety assessment and verification of the design of engineering aspects of items important to safety of NPPs during the phases of review for authorization (licensing) of the construction, modification and operation of new NPPs and the safety re-evaluation of existing NPPs during periodic safety reviews.

5. SCOPE

This Safety Guide will provide recommendations on safety assessment and verification of the design of engineering aspects of existing NPPs and new NPPs.

The consideration of hazards arising from malicious acts is out of the scope of this Safety Guide.

6. PLACE IN THE OVERALL STRUCTURE OF THE RELEVANT SERIES AND INTERFACES WITH EXISTING AND/OR PLANNED PUBLICATIONS

This Safety Guide will interface with at least the following IAEA Safety Standards Series and other publications (the list is not intended to be final or exhaustive):

- 1) GSR Part 4 (Rev. 1) – Safety Assessment for Facilities and Activities (2016);
- 2) SSR-2/1 (Rev. 1) – Safety of Nuclear Power Plants: Design (2016);
- 3) SSR-2/2 (Rev. 1) – Safety of Nuclear Power Plants: Commissioning and Operation (2016);
- 4) SSR-1 – Site Evaluation for Nuclear Installations (2019);
- 5) GSR Part 3 – Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards (2014);
- 6) SSG-2 (Rev. 1) – Deterministic Safety Analysis for Nuclear Power Plants (2019);
- 7) SSG-3 – Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants (under revision, DS523);
- 8) NS-G-2.13 – Evaluation of Seismic Safety for Existing Nuclear Installations (under revision, DS522);
- 9) SSG-4 – Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants (under revision, DS528);
- 10) SSG-25 – Periodic Safety Review for Nuclear Power Plants (2013);
- 11) SSG-28 – Commissioning for Nuclear Power Plants (2014);
- 12) SSG-30 – Safety Classification of Structures, Systems and Components in Nuclear Power Plants (2016);
- 13) SSG-39 – Design of Instrumentation and Control Systems for Nuclear Power Plants (2013);
- 14) SSG-51 – Human Factors Engineering in Nuclear Power Plants (2019);

- 15) SSG-52 – Design of the Reactor Core for Nuclear Power Plants (2019);
- 16) SSG-53 – Design of the Reactor Containment and Associated Systems for Nuclear Power Plants (2019);
- 17) SSG-54 – Accident Management Programmes for Nuclear Power Plants (2019);
- 18) SSG-56 – Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants (2020);
- 19) SSG-61 – Format and Content of the Safety Analysis Report for Nuclear Power Plants (2021);
- 20) SSG-62 – Design of Auxiliary and Supporting Systems for Nuclear Power Plants (2020);
- 21) SSG-69 – Equipment Qualification for Nuclear Installations (2021);
- 22) DS490 – Seismic Design of Nuclear Installations (revision of NS-G-1.6);
- 23) DS494 – Protection against Internal Hazards in the Design of Nuclear Power Plants (revision and combination of NS-G-1.7 and NS-G-1.11);
- 24) DS497a – Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants;
- 25) DS497b – Modifications to Nuclear Power Plants;
- 26) DS497e – Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants;
- 27) DS497g – Conduct of Operations at Nuclear Power Plants;
- 28) DS498 – Design of Nuclear Installations Against External Events Excluding Earthquakes (revision of NS-G-1.5);
- 29) DS503 – Protection against Internal and External Hazards in the Operation of Nuclear Power Plants (revision of NS-G-2.1);
- 30) DS508 – Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants;
- 31) INSAG-25 – A Framework for an Integrated Risk Informed Decision Making Process (2011);
- 32) IAEA Services Series No. 41 – Technical Safety Review (TSR) Service Guidelines (2019).

7. OVERVIEW

The planned table of contents is as follows:

1. INTRODUCTION

- 1.1. Background
- 1.2. Objective
- 1.3. Scope
- 1.4. Structure

2. GENERAL CONSIDERATIONS RELATED TO THE PERFORMANCE AND USE OF SAFETY ASSESSMENT AND VERIFICATION FOR NUCLAR POWER PLANTS

- 2.1. Project Management and Organization for Conducting the Safety Assessment
- 2.2. Consideration of Applicable Design and Regulatory Requirements

- 2.3. Familiarization with the Site Characteristics, Plant Design and Operation, Emergency Operating Procedures and Severe Accident Management Guidelines
- 2.4. Required Information to Conduct the Safety Assessment
- 2.5. Uses of the Safety Assessment
3. SAFETY ASSESSMENT AND VERIFICATION OF NUCLEAR POWER PLANT DESIGN AND MODIFICATIONS
 - 3.1. Safety Functions and Postulated Initiating Events
 - 3.2. Implementation of the Defence in Depth Concept
 - 3.3. Protection Against Internal Hazards and External Hazards
 - 3.4. Associated Functional Requirements and Criteria
 - 3.5. Safety Classification
 - 3.6. Design Basis, Margins, Loads and Load Combinations
 - 3.7. Human Engineering Factors
 - 3.8. Provisions for Ensuring Radiation Protection
 - 3.9. Auxiliary and Support Systems
 - 3.10. Operation and Technical Specifications
4. INDEPENDENT VERIFICATION OF THE SAFETY ASSESSMENT FOR NUCLEAR POWER PLANTS
 - 4.1. Purpose of the Independent Verification
 - 4.2. Scope of the Independent Verification
 - 4.3. Use of the Results of the Independent Verification
5. REFERENCES
6. CONTRIBUTORS TO DRAFTING AND REVIEW

8. PRODUCTION SCHEDULE

Provisional schedule for preparation of the document, outlining realistic expected dates for each step

| | |
|---|---------------|
| STEP 1: Preparing a DPP | DONE |
| STEP 2: Approval of DPP by the Coordination Committee | March 2022 |
| STEP 3: Approval of DPP by the relevant review Committees | June 2022 |
| STEP 4: Approval of DPP by the CSS | November 2022 |
| STEP 5: Preparing the draft | 4Q 2022 |
| STEP 6: Approval of draft by the Coordination Committee | 2Q 2023 |
| STEP 7: Approval by the relevant review Committees for submission to Member States for comments | 4Q 2023 |
| STEP 8: Soliciting comments by Member States | 1Q 2024 |
| STEP 9: Addressing comments by Member States | 2Q 2024 |
| STEP 10: Approval of the revised draft by the Coordination Committee Review in NSOC-SGDS (Technical Editorial review) | 4Q 2024 |
| STEP 11: Approval by the relevant review Committees | 2Q 2025 |

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|--|---------|
| STEP 12: - Submission to the CSS - Submission in parallel and approval by the Publications Committee - MTCD Editing - Endorsement of the edited version by the CSS | 4Q 2025 |
| STEP 13: Establishment by the Publications Committee and/or Board of Governors (for SF and SR only)) | |
| STEP 14: Target publication date | 3Q 2026 |

9. RESOURCES

It is estimated that the production of the Safety Guide would involve approximately 40 weeks of effort. This is based upon assuming 4 one-week consultant's meetings, involving no more than 5 experts, and an average of one week of work per expert between meetings.

Agency resources involved are estimated at 15 weeks of effort by the Technical Officer.

The safety assessment process for nuclear power plants, the IAEA Safety Guide NS-G-1.2, Safety Assessment and Verification for Nuclear Power Plants (2001), provided recommendations:

“(…) for carrying out a safety assessment during the initial design process and design modifications, as well as to the operating organization in carrying out independent verification of the safety assessment of new nuclear power plants with a new or already existing design.”

The topics covered by the recommendations in this Safety Guide were mainly focused on the deterministic safety analysis, the probabilistic safety assessment, and the engineering aspects important to safety for nuclear power plants. This Safety Guide was later declared superseded by the General Safety Requirements GSR Part 4, Safety Assessment for Facilities and Activities (2009), which were adopted to establish general requirements applicable to both facilities and activities. Later, based on the general requirements, the Specific Safety Guides SSG-2, Deterministic Safety Analysis for Nuclear Power Plants (2010), SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants (2010), and SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants (2010), were developed (see Figure 2).

These IAEA safety standards provide recommendations for conducting the probabilistic safety assessments and the deterministic safety analyses but not for the safety assessment and verification of engineering aspects important to safety for nuclear power plants from the design perspective.

Gaps finding

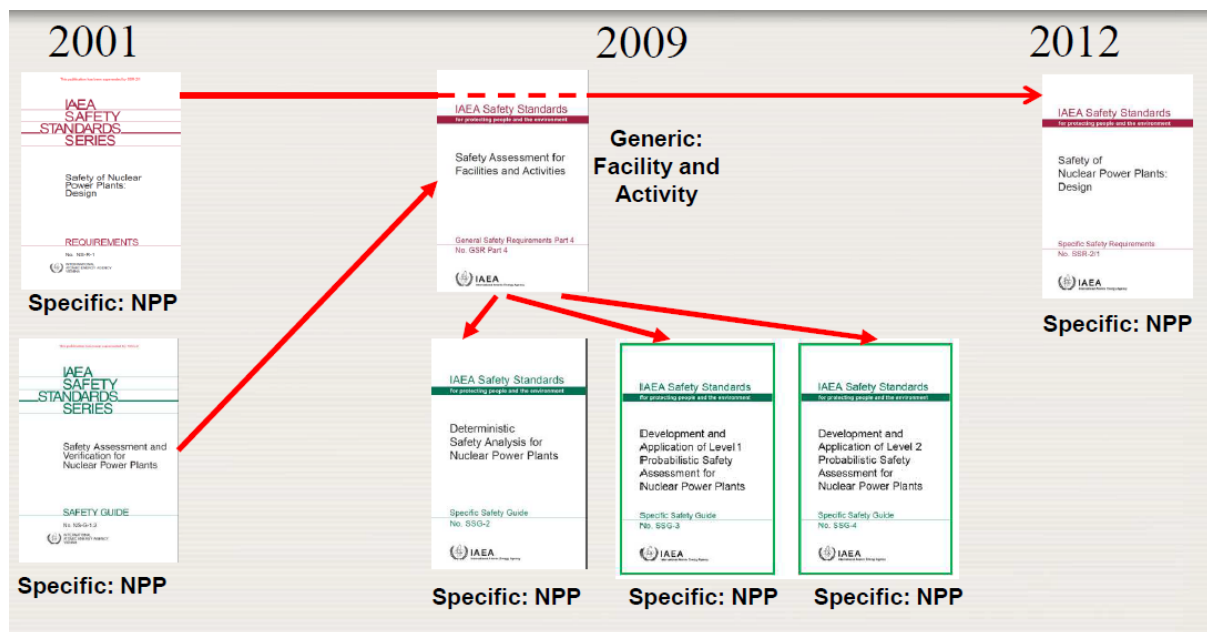


Figure 1. NS-G-1.2 superseded by GSR Part 4

Only aspects of potential interest for the proposed new Safety Guide on Safety Assessment and Verification for Nuclear Power Plants have been investigated.

Here, after a comparison of NS-G-1.2, GSR Part 4 (Rev. 1), SSG-2 (Rev. 1), SSG-3 and SSG-4, a gap analysis is presented.

| Section NS-G-1.2 | Comments |
|------------------|--------------|
| 1. INTRODUCTION | No commented |

| Section NS-G-1.2 | Comments |
|---|---|
| Background Objective Scope Structure | |
| 2. SAFETY ASSESSMENT, SAFETY ANALYSIS AND INDEPENDENT VERIFICATION | |
| Safety assessment and safety analysis | Covered in all GSR Part 4 (Rev. 1), SSG-2 (Rev. 1), SSG-3 |
| Independent verification | Covered in GSR Part 4 (Rev. 1) Req. 24, in SSG-2 (Rev. 1) but only related to deterministic safety analysis, and in SSG-3 and SSG-4 but only related to probabilistic safety assessment |
| Relationship between the design, safety assessment and independent | Covered but not explicitly presented in GSR Part 4 (Rev. 1) |
| 3. ENGINEERING ASPECTS IMPORTANT TO SAFETY | This section is not covered for NPPs and just general requirements are presented in GSR Part 4 (Rev. 1). There is a gap to be covered |
| General | |
| Proven engineering practices and operational experience | There is a gap to be covered |
| Innovative design features | |
| Implementation of defence in depth | |
| Radiation protection | |
| Safety classification of structures, systems and components | |
| Protection against external events | |
| Protection against internal hazards | |
| Conformity with applicable codes, standards and guides | |
| Load and load combination | |
| Selection of materials | |
| Single failure assessment and redundancy/independence | |
| Diversity | |

| Section NS-G-1.2 | Comments |
|---|--|
| In-service testing, maintenance, repair, inspections and monitoring of items important to safety | |
| Equipment qualification | |
| Ageing and wear-out mechanisms | |
| Human-machine interface and the application of human factor engineering | |
| System interactions | |
| Use of computational aids in the design process | |
| 4. SAFETY ANALYSIS | |
| General guidance | Covered in SSG-2 (Rev. 1) and SSG-3 |
| Postulated initiating events | Covered in SSG-2 (Rev. 1) and SSG-3 |
| Deterministic safety analysis | Covered and expanded in SSG-2 (Rev. 1) |
| Probabilistic safety analysis | Covered and expanded in SSG-3 and SSG-4 |
| Sensitivity studies and uncertainty analysis | Covered in both SSG-2 (Rev. 1) and SSG-3 |
| Assessment of the computer codes used | Covered in SSG-2 (Rev. 1), SSG-3 and SSG-4 |
| 5. INDEPENDENT VERIFICATION | Covered in general by Req. 24 of GSR Part 4 (Rev. 1), and particularly for deterministic safety analyses in SSG-2 (Rev. 1), and for probabilistic safety assessment in SSG-3 and SSG-4 |

Here, after a comparison of GSR Part 4 (Rev. 1), SSG-2 (Rev. 1) and SSG-3, a gap analysis is presented.

| GSR Part 4 (Rev. 1) | Comments |
|---|---------------|
| INTRODUCTION Background Objective Scope Structure | Not commented |
| BASIS FOR REQUIRING A SAFETY ASSESSMENT | Not commented |
| GRADED APPROACH TO SAFETY ASSESSMENT | |

| GSR Part 4 (Rev. 1) | Comments |
|---|---|
| | |
| <u>Requirement 1: Graded approach to safety assessment</u> | Covered by SSG-3, not by SSG-2 (Rev. 1) |
| SAFETY ASSESSMENT | |
| Overall requirements for safety assessment | |
| <u>Requirement 2: Scope of the safety assessment</u> | It is clear for NPPs and considered in SSG-2 (Rev. 1) and SSG-3 |
| <u>Requirement 3: Responsibility for the safety assessment</u> | It is clear for NPPs and considered in SSG-2 (Rev. 1) and SSG-3 |
| <u>Requirement 4: Purpose of the safety assessment</u> | Covered in both SSG-2 (Rev. 1) and SSG-3 |
| Specific requirements for safety assessment | |
| <u>Requirement 5: Preparation for the safety assessment</u> | In SSG-2 (Rev. 1) but specific for conducting deterministic safety analysis and in SSG-3 but specific for conducting probabilistic safety assessments |
| <u>Requirement 6: Assessment of the possible radiation risks</u> | Covered by both SSG-2 (Rev. 1) and SSG-3 |
| <u>Requirement 7: Assessment of safety functions</u> | Not covered |
| <u>Requirement 8: Assessment of site characteristics</u> | Not covered |
| <u>Requirement 9: Assessment of the provisions for radiation protection</u> | Covered by SSG-2 (Rev. 1) but limited |
| <u>Requirement 10: Assessment of engineering aspects</u> | Not covered |
| <u>Requirement 11: Assessment of human factors</u> | Not covered |
| <u>Requirement 12: Assessment of safety over the lifetime of a facility or activity</u> | Not covered |
| Defence in depth and safety margins | |
| Requirement 13: Assessment of defence in depth | Not covered |
| Safety analysis | |
| <u>Requirement 14: Scope of the safety analysis</u> | <u>Considered in SSG-2 (Rev. 1) and in SSG-3</u> |

| GSR Part 4 (Rev. 1) | Comments |
|---|--|
| <u>Requirement 15: Deterministic and probabilistic approaches</u> | Considered in SSG-2 (Rev. 1) and in SSG-3 |
| <u>Requirement 16: Criteria for judging safety</u> | Considered in SSG-2 (Rev. 1) but not explicitly mentioned, SSG-3 (partially with regard to probabilistic safety goals) |
| <u>Requirement 17: Uncertainty and sensitivity analysis</u> | Covered in SSG-2 (Rev. 1) and in SSG-3 |
| <u>Requirement 18: Use of computer codes</u> | Covered in SSG-2 (Rev. 1) and in SSG-3 |
| <u>Requirement 19: Use of operating experience data</u> | Covered in SSG-2 (Rev. 1) and in SSG-3 even though there is no specific reference |
| Documentation | |
| <u>Requirement 20: Documentation of the safety assessment</u> | Covered in SSG-2 (Rev. 1) and in SSG-3 |
| Independent verification | |
| <u>Requirement 21: Independent verification</u> | Explicitly covered by SSG-2 (Rev. 1) and SSG-3 |
| MANAGEMENT, USE AND MAINTENANCE OF THE SAFETY ASSESSMENT | |
| <u>Requirement 22: Management of the safety assessment</u> | Covered by SSG-2 (Rev. 1) and SSG-3 |
| <u>Requirement 23: Use of the safety assessment</u> | Explicitly covered by SSG-2 (Rev. 1) and SSG-3 |
| <u>Requirement 24: Maintenance of the safety assessment</u> | Explicitly covered by SSG-2 (Rev. 1) and SSG-3 |

Therefore, recommendations relating to safety assessment and verification should be provided on the following topics:

- Completeness and adequacy of the set of safety functions and the associated functional requirements assigned to engineering aspects important to safety across SSCs to different systems, and for all plant states with regard to the fundamental safety functions and the postulated initiated events and external events for which the SSCs are required;
- Adequate implementation of the defence in depth concept for NPPs, with consideration of independence among those SSCs required at different levels of defence;
- Adequate and effective implementation of the safety related classifications for SSCs and the associated relevant activities comprehensively across different systems to ensure their

required functions and the associated functional requirements in all plant states while coping with the expected internal and external hazards;

- Completeness and adequacy of the set of expected internal and external hazards resulting from the site evaluation to be considered in the design of SSCs and the adequate and effective implementation of design provisions for their protection against internal and external hazards and their combinations across different systems to ensure their required functions in all plant states;
- Completeness and adequacy of the set of expected loads and loads combinations (including those induced by internal and external hazards) to be considered for the design of SSCs important to safety and the adequate and sufficient implementation of design assumptions to enable SSCs capacity and the margins to withstand the identified loads and loads combinations across different systems to ensure their required functions in all plant states while preventing cliff-edge effects;
- Adequate and sufficient implementation of design safety principles, of human engineering factors and of provisions for ensuring radiation protection, across different systems, to enable SSCs to ensure their required functions in all plant states;
- Adequacy and representative consideration of proven engineering practices as well as of operating experience feedback in the design of SSCs for ensuring the performance of the required safety functions in all plant states.

Therefore, it is considered necessary to develop a new Safety Guide that provides recommendations to meet the requirements for conducting a comprehensive and cross-cutting safety assessment and verification process (as presented in GSR Part 4 (Rev. 1)) with regard to engineering aspects important to safety of nuclear power plants.