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Document Preparation Profile (DPP)
Version 1~~0~~1 dated 25~~2~~ May~~rch~~ 2022

1. IDENTIFICATION

Document Category: **Specific Safety Guide**

Working ID: **DS536**

Proposed Title: **Safety Assessment and Independent Verification of Engineering Aspects Important to Safety 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, WASSC, RASSC**

Technical Officer: **Jorge LUIS HERNANDEZ (SAS/NSNI)**

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[1] 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 This Safety Guide was superseded by IAEA Safety Standards Series Nos GSR Part 4, Safety Assessment for Facilities and Activities, and SSG-2, Deterministic Safety Analysis for Nuclear Power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~~Currently, IAEA Safety Standards provide specific set of recommendations for conducting probabilistic safety assessments and the deterministic safety analyses but not for the safety assessment and independent 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:

[1] "Engineering aspects" is understood as all the topics to be covered in the safety assessment as required in Requirement 10 of GSR Part 4 (Rev. 1).

(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 NUSSC Meeting, under item 4.1 as part of its medium-term work plan, NUSSC members agreed on the ~~development~~ preparation 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 NUSSC 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 independent verification[2] 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 independent 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 modification and 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

[2] “Independent verification” is understood as the independent verification of the safety assessment as required in Requirement 21 of GSR Part 4 (Rev. 1).

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 independent 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. The recommendations in this guide will be appropriately graded with regard to the safety assessment of the safety significant of engineering aspects across all levels of defence in depth. These recommendations will complement the recommendations in the IAEA Safety Guides on deterministic safety analysis as well as on probabilistic safety assessment and assessment of the defence-in-depth and DEC conditions.

It is expected that the Safety Guide will promote consistency in conducting the safety assessment and the independent 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 independent 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 independent verification of the design of engineering aspects important to safety for nuclear power plants[3] with a new or already existing design. The safety guide will cover aspects related to safety assessment that are generally applicable to different type of nuclear power reactor technologies. However, specifics issues related to the innovative technologies will be covered in another safety guide. This safety guide will expand the key aspects to be considered under the safety assessment in relation to engineering aspects important to safety which are not covered by current safety guides while avoid repetition.

The recommendations for performing a safety assessment are suitable also as guidance for the safety review of an existing plant.~~of existing NPPs and new NPPs~~.

~~The consideration of hazards arising from malicious acts is out of the scope of this Safety Guide.~~ The interfaces between safety and security in the assessment of possible radiation risks will be identified on an overall level, while specific guidance on the assessment of hazards arising from malicious acts will not be included within 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):

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[3] The nuclear power plant includes the reactor unit with the spent fuel pool.

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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 (2014~~6~~);
13) SSG-39 – Design of Instrumentation and Control Systems for Nuclear Power Plants (2016~~3~~);
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~~ SSG-67 – Seismic Design for~~of~~ Nuclear Installations (~~revision of NS-G-1.6~~2021);
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~~ SSG-68 – Design of Nuclear Installations Against External Events Excluding Earthquakes ~~(revision of NS-G-1.5)~~ (2021);

29) ~~DS503~~ SSG-77 – Protection against Internal and External Hazards in the Operation of Nuclear Power Plants ~~(revision of NS-G-2.1)~~ (2022);

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);

~~31)~~32) DS524 - Radiation Protection Aspects of Design for Nuclear Power Plants (NS-G-1.13);

~~32)~~33) IAEA Services Series No. 41 – Technical Safety Review (TSR) Service Guidelines (2019).

7. OVERVIEW

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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 INDEPENDENT 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~~and Applications
3. SAFETY ASSESSMENT OF ENGINEERING ASPECTS IMPORTANT TO SAFETY FOR ~~AND VERIFICATION OF~~ NUCLEAR POWER PLANT DESIGN AND MODIFICATIONS
   - 3.1. ~~Safety~~ System Functions and Postulated Initiating Events for System Design
   - 3.2. Implementation of the Defence in Depth Concept
   - 3.3. Protection Against Internal Hazards and External Hazards including combinations and beyond design basis external hazards
   - 3.4. ~~Associated Functional~~ Safety Requirements and Functional Criteria for the System and its Supporting Systems including Non-permanent Equipment
   - 3.5. ~~Safety~~ Classification and Categorization of Structures, Systems and Components
   - 3.6. Safety Design Principles
   - ~~3.5.~~3.7. Associated Relevant Activities
   - ~~3.6.~~3.8. Design Basis, Margins, Loads and Load Combinations
   - ~~3.7.~~3.9. Human Factors Engineering ~~Factors~~

~~3.8.~~3.10. Provisions for Ensuring Radiation Protection

~~3.9.~~3.11. ~~Auxiliary and Support Systems~~Operational limits and Conditions for Safe Operation

~~3.10.~~3.12. ~~Operation and Technical Specifications~~Design Safety Provisions for the Decommissioning Stage

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~~ Process to Conduct 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
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 independent verification of engineering aspects important to safety for nuclear power plants from the design perspective.

Gaps finding

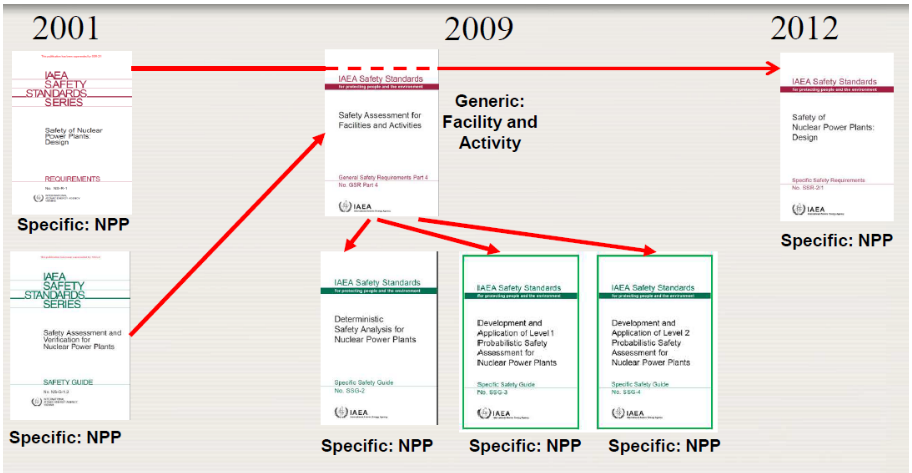

Figure 1~~2~~. NS-G-1.2 superseded by GSR Part 4

Only aspects of potential interest for the proposed new Safety Guide on Safety Assessment and Independent Verification of Engineering Aspects Important to Safety 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 Background Objective Scope Structure	No commented
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. 21 (§4.66-4.71)~~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 verification	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	

Section NS-G-1.2	Comments
Single failure assessment and redundancy/independence	
Diversity	
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~~ 21 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

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

GSR Part 4 (Rev. 1)	Comments
Requirement 14: Scope of the safety analysis	Considered in SSG-2 (Rev. 1) and in SSG-3
Requirement 15: Deterministic and probabilistic approaches	Considered in SSG-2 (Rev. 1) and in SSG-3
Requirement 16: Criteria for judging safety	Considered in SSG-2 (Rev. 1) but not explicitly mentioned, SSG-3 (partially with regard to probabilistic safety goals)
Requirement 17: Uncertainty and sensitivity analysis	Covered in SSG-2 (Rev. 1) and in SSG-3
Requirement 18: Use of computer codes	Covered in SSG-2 (Rev. 1) and in SSG-3
Requirement 19: Use of operating experience data	Covered in SSG-2 (Rev. 1) and in SSG-3 even though there is no specific reference
Documentation	
Requirement 20: Documentation of the safety assessment	Covered in SSG-2 (Rev. 1) and in SSG-3
Independent verification	
Requirement 21: Independent verification	Explicitly covered by SSG-2 (Rev. 1) and SSG-3
MANAGEMENT, USE AND MAINTENANCE OF THE SAFETY ASSESSMENT	
Requirement 22: Management of the safety assessment	Covered by SSG-2 (Rev. 1) and SSG-3
Requirement 23: Use of the safety assessment	Explicitly covered by SSG-2 (Rev. 1) and SSG-3
Requirement 24: Maintenance of the safety assessment	Explicitly covered by SSG-2 (Rev. 1) and SSG-3

Therefore, recommendations relating to safety assessment and independent 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 engineering 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 independent verification process (as presented in GSR Part 4 (Rev. 1)) with regard to engineering aspects important to safety of nuclear power plants.