~~03~~ 27 ~~April~~ May 2025

# IAEA SAFETY STANDARDS
**for protecting people and the environment**

**STATUS: Step 7**

**First review of the draft publication by the review Committee(s)**

# Periodic Safety Review of Nuclear Power Plants

## DRAFT SPECIFIC SAFETY GUIDE

## DS535 (Revision of SSG-25)

## CONTENTS

# 1. INTRODUCTION

BACKGROUND

1.1. Routine reviews of nuclear power plant operation (including reviews of modifications to hardware and procedures, significant events, operating experience, plant management and personnel competence) and special reviews following major events of safety significance are the primary means of ensuring safety. In addition, some States have initiated systematic safety reassessments, termed periodic safety review (PSR), to assess the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects, and use PSR in support of justifications of long term operation. A PSR includes an assessment of plant design and operation against applicable current safety standards and operating practices and has the objective of ensuring a high level of safety throughout the plant's operating lifetime. It is complementary to the routine and special safety reviews conducted at nuclear power plants and does not replace them.

1.2. This Safety Guide supports IAEA Safety Standards Series Nos: SF-1, Fundamental Safety Principles [1]; SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [2]; and GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [3]. The terms used in this Safety Guide are as defined in the IAEA Safety Glossary [4].

1.3. This Safety Guide supersedes IAEA Safety Standards Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants issued in 2013[1].

OBJECTIVE

1.4. The objective of this Safety Guide is to provide recommendations on the conduct of a PSR for an existing nuclear power plant. This Safety Guide is intended for use by operating organizations, regulatory bodies and their technical support organizations, consultants and advisory bodies.

SCOPE

1.5. This Safety Guide addresses PSR for operating nuclear power plants. PSR is a comprehensive safety review of all important aspects of safety, conducted at regular intervals, typically every ten years. In addition, a PSR may be used in support of the decision making process for licence renewal or long term operation, or for the restart of a nuclear power plant following a prolonged shutdown.

1.6. The PSR process considered in this Safety Guide is valid for nuclear power plants of any age and may have a wider applicability, for example to research reactors, nuclear fuel cycle facilities and radioactive waste management facilities, through the application of a graded approach[2].

---

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-25, IAEA, Vienna (2013).
[2] An example of such application can be found in INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review for Research Reactors, Safety Reports Series No. 99, IAEA, Vienna (2020) [5], or

1.7. This Safety Guide also provides recommendations on specific considerations for the PSR that will cover the transition to decommissioning and for the PSR conducted during the decommissioning of nuclear power plants.

STRUCTURE

1.8. The rationale for and the objectives of PSR for operating nuclear power plants and recommendation on the general approach to PSR are provided in Section 2. Recommendations on the roles and responsibilities of the operating organization, the regulatory body and external experts when conducting a PSR are provided in Section 3. Section 4 presents a recommended review process. Section 5 provides recommendations on the post-review activities. Recommendations on the general review methodology and strategic considerations relating to the conduct of PSR are provided in Section 6. Section 7 provides recommendations on the review of safety factors, i.e. the important aspects of safety of an operating nuclear power plant that are addressed in a PSR. Section 8 provides recommendations on global assessment. Section 9 provides recommendations on long term operation aspects. Section 10 provides recommendations on the PSR process for nuclear power plants in permanent shutdown or in decommissioning.

1.9. Appendix I describes the interfaces between the various safety factors and Appendix II provides recommendations on the content of the various documents and reports relating to PSR. Annex I provides information on typical inputs and outputs in the review of safety factors and lists relevant IAEA and other publications. ~~Annex II provides information on the implementation of the PSR safety improvements plan.~~

## 2. RATIONALE, OBJECTIVES AND APPROACH TO PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS

2.1. Since operation of the first generation of commercial nuclear power plants there have been substantial developments in safety standards, operating practices and in technology. Lessons have been learned from operating experience, including the response of nuclear power plants to external events exceeding their design basis values. Additionally, better analytical methods have been developed. All of these developments should be considered by operating organizations and regulatory bodies in the interests of continuous safety improvement.

2.2. Requirement 12 of SSR-2/2 (Rev. 1) [2] states:

> "**Systematic safety assessments of the plant, in accordance with the regulatory requirements, shall be performed by the operating organization throughout the plant's operational lifetime, with due account taken of operating experience and significant new safety related information from all relevant sources.**"

Although operating nuclear power plants are subject to routine and special safety reviews, such reviews are generally not sufficiently comprehensive to meet this requirement. Thus, it is

---

in INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review for Nuclear Fuel Cycle Facilities, Safety Reports Series No. 124, IAEA, Vienna (2024) [6].

common international practice for operating organizations to undertake proactive, strategic, detailed and comprehensive PSRs.

Additionally, Requirement 12 of GSR Part 4 (Rev. 1) [3] states:

> **"The safety assessment shall cover all the stages in the lifetime of a facility or activity in which there are possible radiation risks."**

2.3. The scope and content of the PSR, the manner of its implementation and the regulatory activities relevant to the PSR will vary depending on regulatory requirements. PSR should provide a means for regulating the safety of plant operation in the long term and for addressing requests by licensees for authorization to continue plant operation beyond an established licensed term or for a further period established by a safety evaluation. A recent PSR should provide reassurance that there continues to be a valid licensing basis taking account of, for example, plant ageing and current safety standards and operating practices.

2.4. PSR should be used to provide an overall view of actual plant safety and the quality of the safety documentation, and to determine corrective actions to ensure safety or reasonably practicable safety improvements to enhance safety to an appropriate high level at least until the next PSR period. To do this, the PSR should identify any lifetime limiting features at the plant in order to plan future modifications and to determine the timing of future reviews.

2.5. A PSR should be performed about ten years after the start of plant operation, and then at ten year intervals until the end of operation. Ten years is considered to be an appropriate interval for such reviews in view of the likelihood, within this period, of the following:

(a) Changes in safety standards, operating practices, technology, underlying scientific knowledge or analytical techniques;
(b) The potential for the cumulative effects of plant modifications to adversely affect safety or the accessibility and usability of the safety documentation;
(c) New information, experience and lessons from the occurrence of major~~external~~ events that affected the safety of another nuclear installation or an industrial facility;
(d) Changes of hazards over time for which new information and assessments have become available;
(e) Identification of significant ageing effects or trends;
(f) Accumulation of operating experience;
(g) Changes in how the plant is, or will be, operated;
(h) Changes in the natural, industrial or demographic environment in the vicinity of the plant;
(i) Changes in staffing levels or in the competence or experience of staff;
(j) Changes in the management structures and procedures of the plant's operating organization.

Extension of the period between PSRs beyond about ten years could delay the identification of important safety issues and could lead to a loss of the direct knowledge and experience gained in previous reviews and to a loss of continuity.

2.6. The period between PSRs should not be defined based on the lengths of refuelling cycles or other fuel or core considerations. For example, in cases of significantly longer refuelling cycles, periodic replacements of reactor cores, or even periodic replacements of whole power modules[3], the design, operational, and ageing aspects of such structures, systems and components (SSCs) should be subject to PSR at appropriate periods, taking into account the factors listed in para. 2.5.

2.7. In case of plants with multiple identical modules[4] ~~plants~~, to achieve consistent PSR results across individual modules, it should be preferred to conduct a PSR considering all the operational modules on site, even if these modules have been commissioned over time. This approach might result in some modules undergoing the first PSR in a shorter timeframe. However, for plants with different module types or sites with multiple independent facilities, a PSR should be conducted separately, as appropriate.

2.8. The length of the review process will depend on the availability and retrievability of relevant information and the organizational structure of the operating organization. To provide a timely input, the entire PSR process should be completed within three years, and normally less for subsequent PSRs.

2.9. It is recognized that some States may prefer alternative arrangements to a PSR. For example, some States apply routine comprehensive safety assessment programmes that deal with specific safety issues, significant events and changes in safety standards and operating practices as they arise. Such programmes can, if applied with appropriate scope, frequency, depth and rigour, achieve the same outcomes as the process recommended in this Safety Guide. They allow safety to be improved on a continuous basis and avoid the need to implement concurrently a large programme of corrective actions, or safety improvements. This Safety Guide is not intended to discourage such alternative arrangements. However, when an alternative approach is followed, it should meet the objectives of PSR (as set out in para. 2.10), together with other relevant objectives and requirements of licensing, regulation and operating processes.

2.10. The objective of PSR is to determine by means of a comprehensive and proportionate assessment:

(a) The adequacy and effectiveness of the arrangements and of the structures, systems and components (SSCs) that are in place to ensure plant safety until the next PSR or, where appropriate, until the end of planned operation (that is, if the nuclear power plant will cease operation before the next PSR is due);
(b) The extent to which the plant conforms to current safety standards and operating practices;
(c) Any necessary corrective actions and reasonably practicable safety improvements and timescales for their implementation;
(d) The extent to which the safety documentation, including the licensing basis, remains valid.

2.11. A PSR can be used for various purposes:

(a) As a systematic safety assessment performed at regular intervals (see Requirement 12 of

[3] Reactor module (sometimes abbreviated as 'module') is a nuclear reactor with its associated structures, systems and components.
[4] Reactor module (sometimes abbreviated as 'module') is a nuclear reactor with its associated structures, systems and components. Multi-module unit/plant is a unit/plant having the possibility of including more than one reactor module.

SSR-2/2 (Rev. 1) [2]);

(b) In support of the decision making process for licence renewal;
(c) In support of the decision making process for long term operation;
(d) In support of the transition phase to decommissioning;
(e) In support of the decommissioning.

## GENERAL APPROACH TO PERIODIC SAFETY REVIEW

2.12. The operating organization has the prime responsibility for ensuring that an adequate PSR is performed (see Requirement 12 of SSR-2/2 (Rev. 1) [2]).

2.13. A PSR should provide a comprehensive and proportionate assessment of the safety of the nuclear power plant. The PSR process is complex and can be aided by appropriate subdivision of tasks; consequently, this Safety Guide sets out these tasks in accordance with 14 safety factors that are intended to cover all aspects important to the safety of an operating nuclear power plant. This subdivision is not the only possible approach: in cases where the number of safety factors and/or their grouping is different (e.g. to meet the specific needs of the operating organization or regulatory body or owing to particular aspects of the plant under review), the comprehensiveness of the PSR should be ensured by other means.

2.14. The 14 safety factors recommended in this Safety Guide are listed below and described in detail in Section 7. The grouping, order and numbering of the safety factors listed above is not intended to imply any order of importance.~~:~~

*Safety factors relating to the plant*

(1) Plant design;
(2) Actual condition of SSCs;
(3) Equipment qualification;
(4) Ageing.

*Safety factors relating to safety analysis*

(5) Deterministic safety analysis;
(6) Probabilistic safety assessment;
(7) Hazard analysis.

*Safety factors relating to performance and feedback of experience*

(8) Safety performance;
(9) Feedback of operating experience.

*Safety factors relating to management*

(10) Organization, the management system and safety culture;
(11) Operational limits and conditions and operating procedures;
(12) Human factors;
(13) Emergency planning.

*Safety factors relating to the environment*

(14) Radiological impact on the environment.

~~The grouping, order and numbering of the safety factors listed above is not intended to imply any order of importance.~~

2.15. The effectiveness of nuclear security arrangements to prevent unauthorized actions that could jeopardize nuclear safety should be reviewed periodically~~ by the appropriate national authorities~~. Guidance on nuclear security measures is provided in the IAEA Nuclear Security Series. Some operating organizations may decide to review physical security as a separate safety factor within the PSR. Aspects related to the interfaces of safety, nuclear security and safeguards are expected to be addressed within the PSR as a means of ensuring compliance with Requirement 8 of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [7]~~[7]~~~~[5]~~, and Requirement 17 of SSR-2/2 (Rev. 1) [2].

2.16. The review of safety factors should identify findings of the following types:

(a) Positive findings (strengths): Where current operating practices are equivalent to good practices as established in current codes and standards.
(b) Negative findings (deviations): Where current operating practices do not meet current codes, standards or industry practices, or do not meet the current licensing basis, or are inconsistent with plant documentation or operating procedures.

2.17. The PSR should address the period until the next PSR or, where appropriate, until the end of planned operation, and should consider whether there are any foreseeable circumstances that could affect the safe operation of the nuclear power plant. If such circumstances are identified, the operating organization should take appropriate action to ensure that the licensing basis remains valid.

2.18. The operating organization should perform a global assessment of safety at the plant to integrate the results of the reviews of individual safety factors. The operating organization should ensure that this global assessment considers all findings and proposed improvements from the individual safety factor reviews as well as the interfaces between different safety factors (see Appendix I).

2.19. The PSR should be conducted in four phases, which may overlap or be further subdivided as appropriate:

(a) Preparation of the PSR project: This should include an agreement with the regulatory body with regard to the scope and timing of the review and the codes and standards that will be used.
(b) Conduct of the PSR: In this phase, the operating organization should conduct the review in accordance with an agreed ‘basis document’ for the PSR (see para. 6.6). The review should identify findings (which may be positive (strengths) or negative (deviations)) and should lead to proposals for corrective actions (in cases of non-compliance with the current licensing basis) or safety improvements and an integrated implementation plan.
(c) Regulatory review: The regulatory body should proportionately review the PSR report prepared by the operating organization and the proposed corrective actions and safety improvements, should identify any issues it wishes to raise (for example, whether further safety improvements need to be considered), should review the proposed integrated

implementation plan and should determine whether the licensing basis for the nuclear power plant remains valid.

(d) Finalization of the integrated implementation plan: The integrated implementation plan, comprising corrective actions and reasonably practicable safety improvements to be implemented in accordance with a time schedule agreed with the regulatory body, should be finalized in this phase.

2.20. The corrective actions and safety improvements resulting from the PSR may partially overlap with actions or activities from recent safety assessments. The operating organization may choose not to repeat activities from previous assessments if no changes have occurred. However, any actions identified in those assessments should be incorporated into the integrated implementation plan of the PSR. In cases where potential non-compliance with the current licensing basis is identified, the operating organization should take immediate action to ensure the licensing basis remains valid, even if the PSR has not yet been finalized.

~~2.20.~~2.21. The phase following PSR in which the safety improvements are implemented is not considered an activity of PSR and so is not addressed in detail in this Safety Guide. Further recommendations on the phases of the PSR are provided in Section 4.

# 3. ROLES AND RESPONSIBILITIES FOR PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS

3.1 The responsibility for conducting the PSR and reporting its findings lies solely with the operating organization of the plant. The operating organization is required to report all safety significant findings from the PSR to the regulatory body, subject to regulatory requirements (see para. 4.45 of SSR-2/2 (Rev. 1) [2]).

3.2 Depending on national legal and regulatory ~~requirements~~framework, the regulatory body has the responsibility for:

(a) Specifying or approving the requirements to perform the PSR;
(b) Approving the documentation to be provided by the operating organization prior to the PSR (i.e. the PSR basis document including the project plan);
(c) Reviewing the actual scope, conduct and findings of the PSR and the resulting corrective actions and safety improvements;
(d) Seeking assurance on the safe operation of the plant until the next PSR;
(e) Taking appropriate licensing actions;
(f) Informing the government and the public about the results of the PSR and resulting safety improvements.

3.3 Both the operating organization and the regulatory body should have sufficient technical competence to discharge their responsibilities as set out in paras 3.1 and 3.2. This should include competence to manage effectively any contracted work (e.g. from external consultants or technical support organizations) and to assess the outputs produced.

3.4 Certain aspects of the PSR can be conducted more effectively by external consultants. For example, the review of the safety factor relating to organization, the management system and safety culture, and the safety factor relating to human factors could benefit from reviews conducted by specialists completely independent from the operating organization of the nuclear power plant. The operating organization should seek to identify aspects of the PSR where external consultants might be better than internal staff at conducting an impartial, independent and objective review, noting that the engagement of external organizations does not diminish the responsibility of the operating organization for carrying out an adequate PSR.

## 4. THE REVIEW PROCESS OF PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS

4.1 The overall process for undertaking the PSR of a nuclear power plant is shown in Fig. 1. The process consists of parallel activities independently performed by the operating organization (shown in Figs 2–4) and the regulatory body (shown in Fig. 5). Major interactions between the operating organization and the regulatory body occur throughout the PSR process, in particular during the assessment of PSR reports (see para. 4.30).

4.2 The activities of the operating organization can be divided into three steps:

(1) Preparation for the PSR project;
(2) Conduct of the reviews of safety factors;
(3) Analysis of the findings (including the global assessment), implementation of corrective actions, and preparation of a programme of safety improvements.

The regulatory body's activities are performed throughout the PSR project.

4.3 The starting point of a PSR should be an agreement between the operating organization and the regulatory body on the general scope, regulatory requirements and standards to be used for the review, and the objectives and outputs for the PSR, as described in the basis document. As part of this agreement, the operating organization and the regulatory body should determine an appropriate cut-off date for the set of documents to be reviewed and the status of the safety performance of the plant to be taken as the basis for the PSR, so as to ensure consistency across all parts of the PSR.

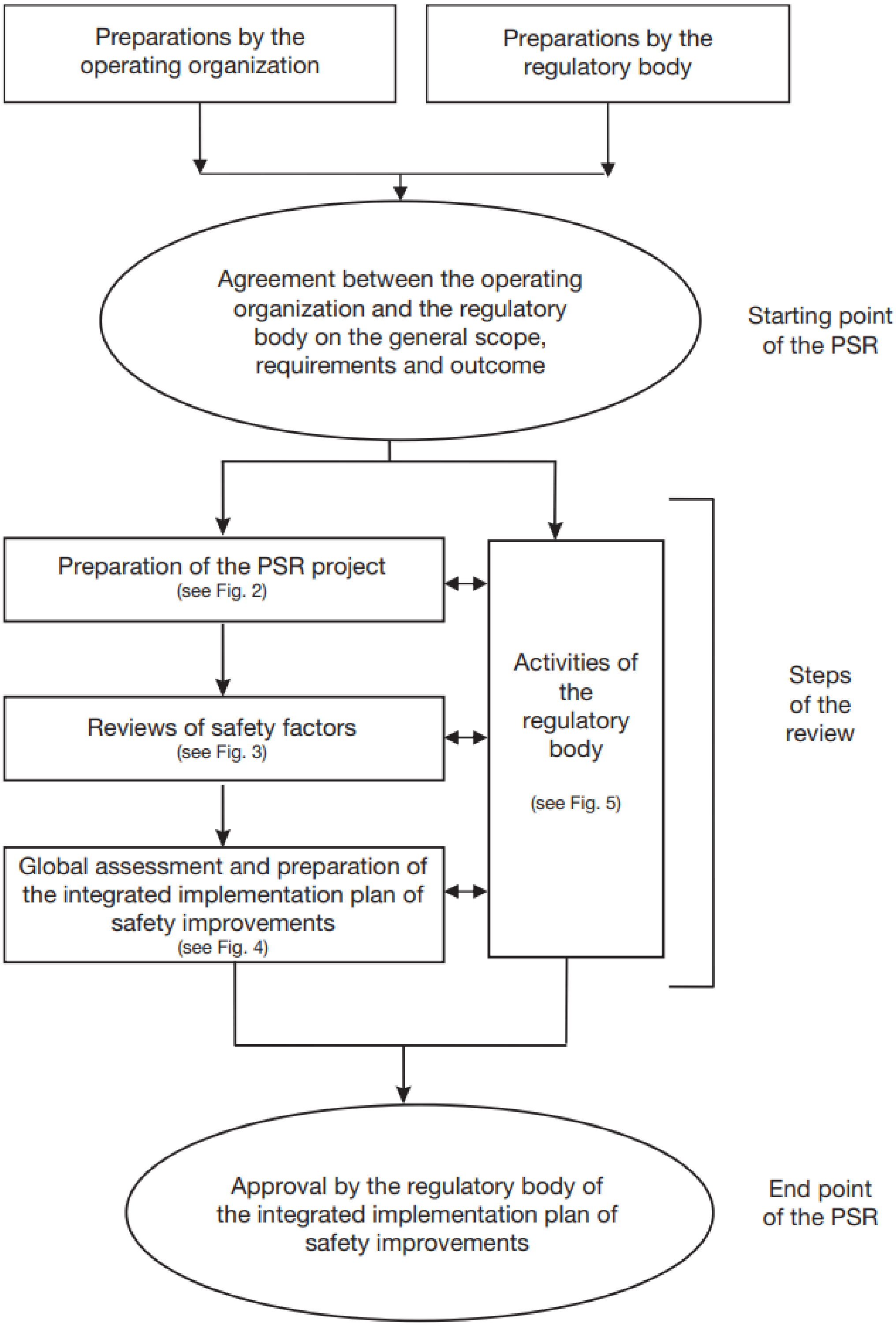


*FIG. 1. Overall process for a PSR of a nuclear power plant.*

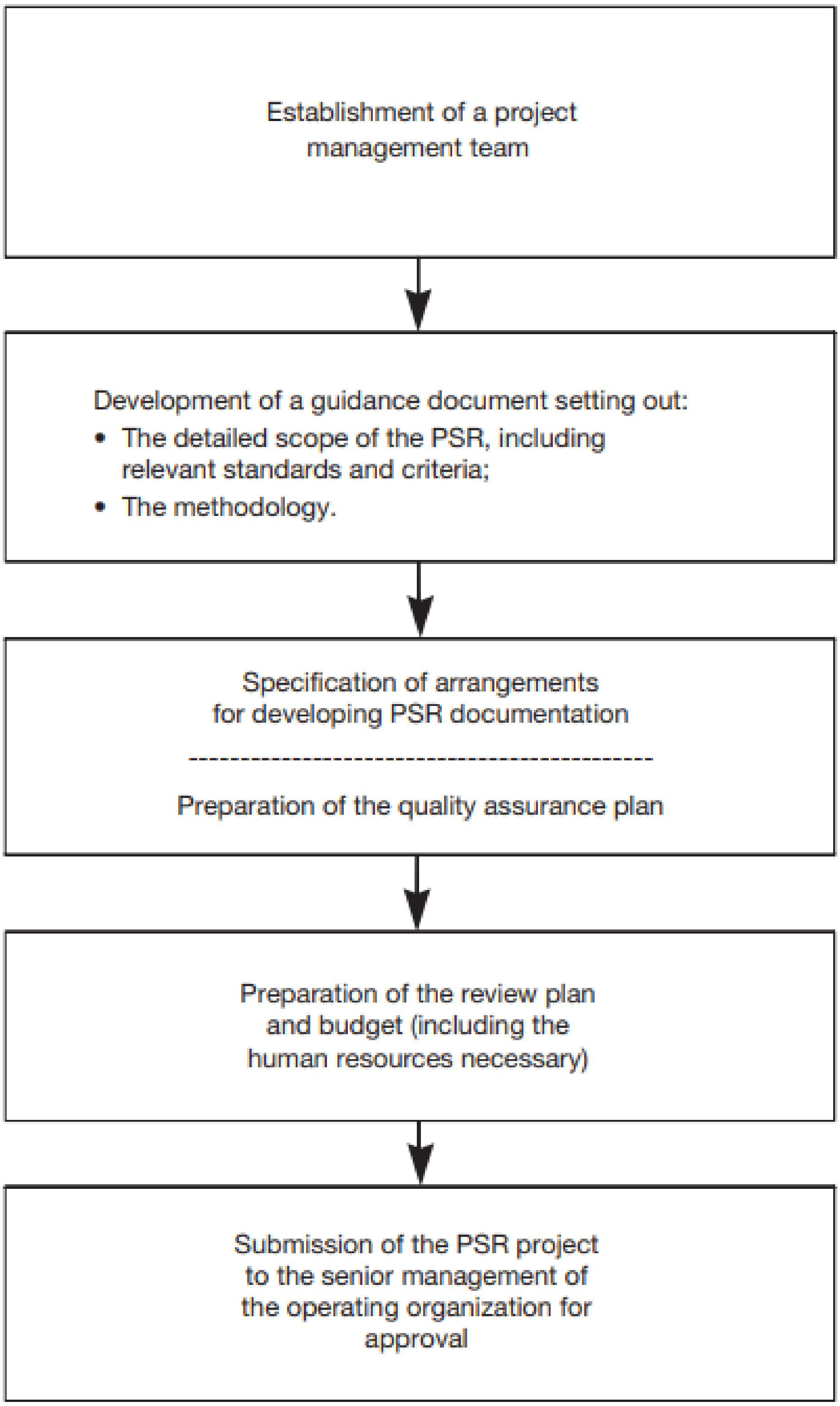


*FIG. 2. Process for the preparation of the project for the PSR for a nuclear power plant.*

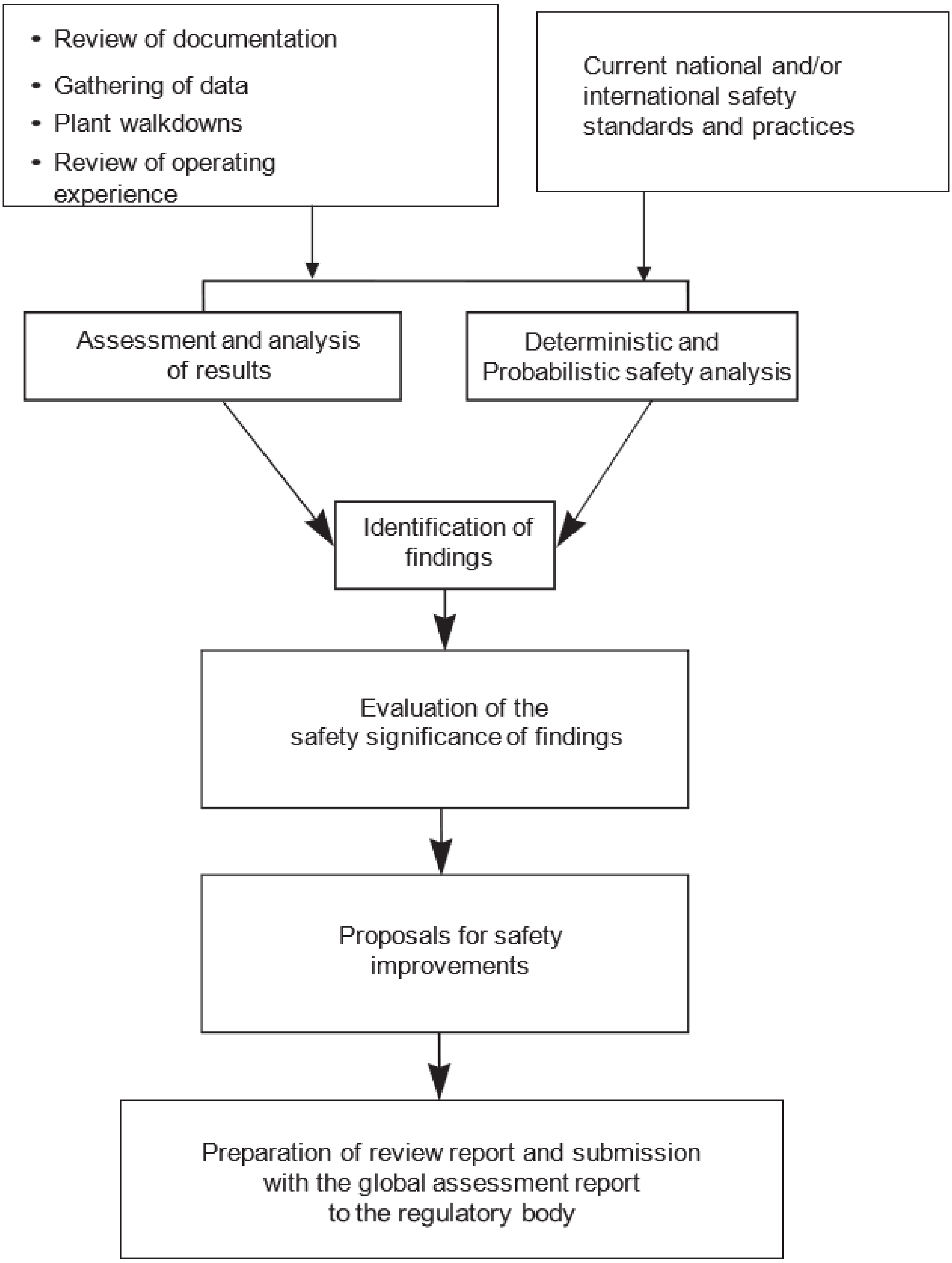


*FIG. 3. Process for the review of each safety factor in the PSR for a nuclear power plant.*

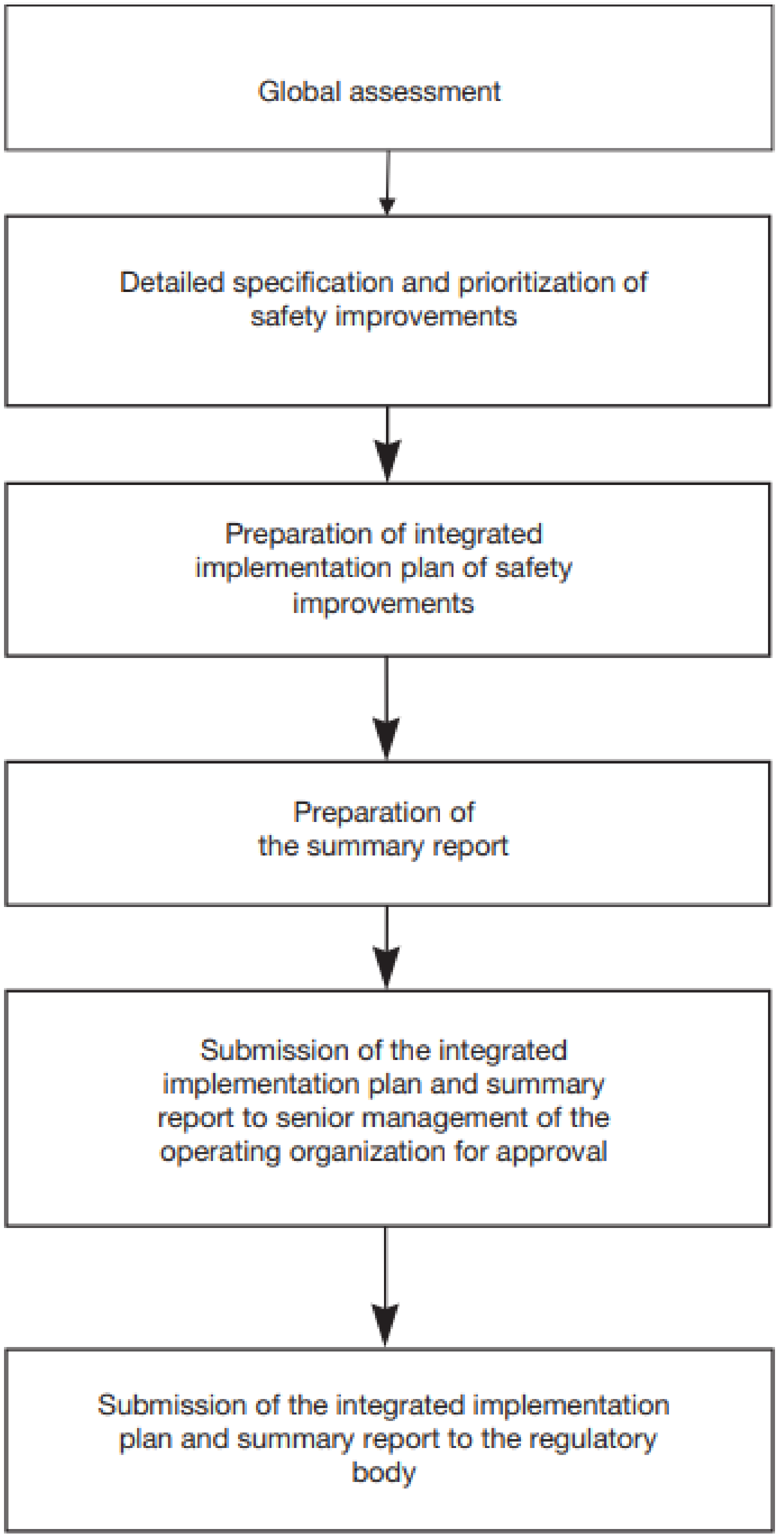


*FIG. 4. Process for global assessment and preparation of the integrated implementation plan of safety improvements arising from a PSR for a nuclear power plant.*

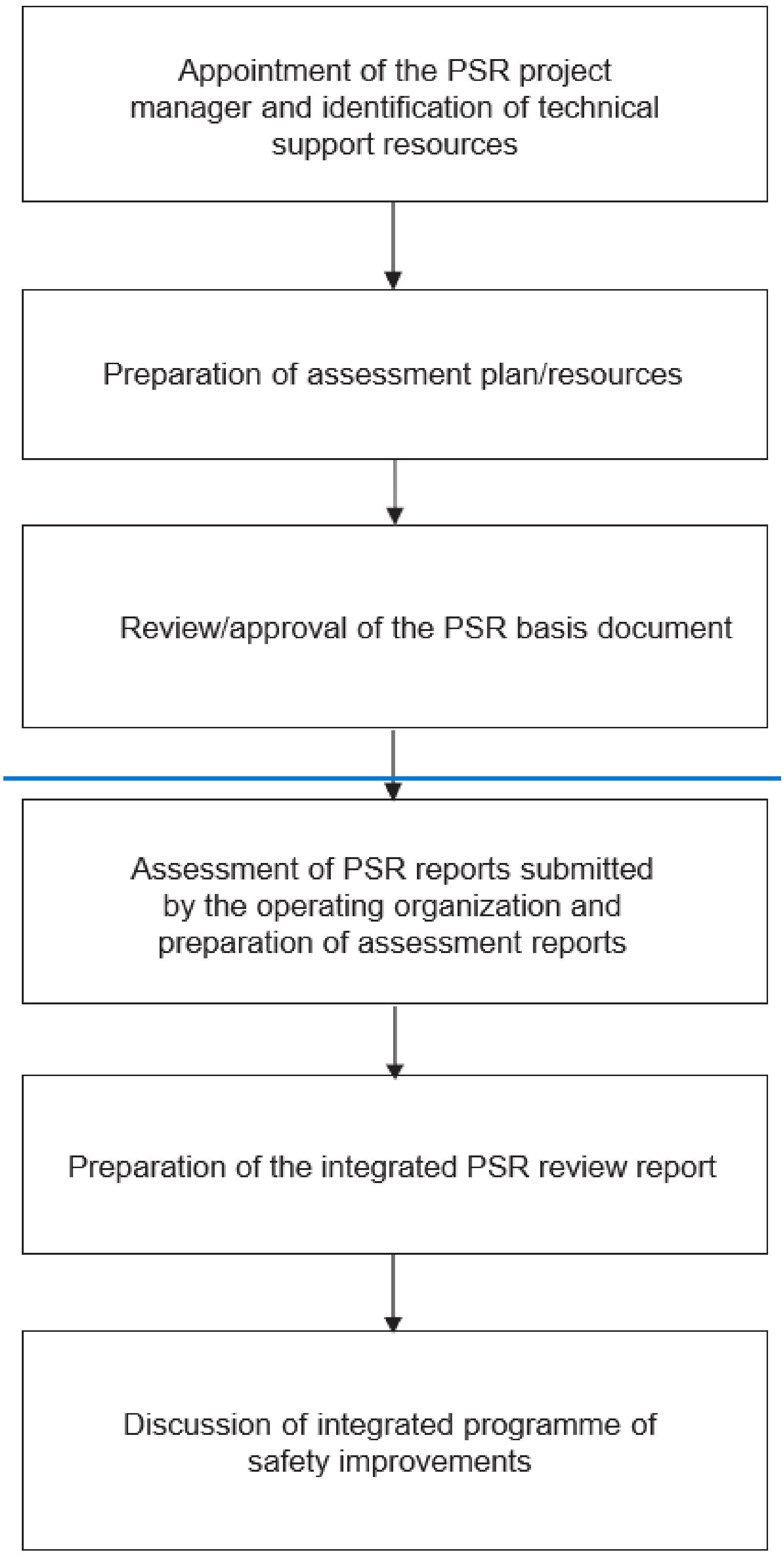
Appointment of the PSR project manager and identification of technical support resources
Preparation of assessment plan/resources
Review/approval of the PSR basis document
Assessment of PSR reports submitted by the operating organization and preparation of assessment reports
Preparation of the integrated PSR review report
Discussion of integrated programme of safety improvements


*FIG. 5. Activities of the regulatory body in relation to the PSR for a nuclear power plant.*

ACTIVITIES OF THE OPERATING ORGANIZATION IN RELATION TO PSR

**Preparation of the PSR project**

4.4 The operating organization should establish an appropriate project management team and develop a reasonable time schedule at the outset of the project, to ensure that the PSR is completed within the agreed timescale, quality, and budget. In developing the schedule, the operating organization should take into account the iterative nature of the review of safety factors, including interfaces between the various safety factors, and should allow time for these interfaces to be dealt with.

4.5 The operating organization should allocate an overall budget for the PSR, in which the scope of the review, organizational aspects, the need to employ external organizations and the schedule for the PSR are all taken into account. Specific consideration should be given to those review activities that might require intensive resources.

4.6 A PSR is typically performed by a number of review teams that work in parallel. A document should therefore be prepared by the operating organization to provide guidance to the review teams on how to review the different safety factors so as to ensure a comprehensive, consistent and systematic approach. This guidance document should elaborate on the agreed scope of the PSR. It should also identify applicable safety standards, methods and practices, which, in most cases, should be based on regulatory requirements and current safety standards and operating practices and should reflect current knowledge. Relevant international standards should be used as appropriate, specifically in the case of unavailability of adequate national standards. These standards, methods and practices should also be referenced in the PSR basis document.

4.7 To ensure the appropriate quality and format of the PSR documents, the operating organization should prepare a quality assurance plan that defines the requirements for the preparation and verification of the PSR documentation. The quality assurance plan should also ensure that all reviewers use consistent input data for the review of individual safety factors — and consider interfaces between these factors — to maintain consistency across all areas of the review.

4.8 Appropriate training and briefings should be conducted to facilitate the effective and efficient completion of the PSR.

**Review of safety factors**

4.9 To improve overall efficiency and consistency, an updated and coherent set of databases may be developed for use within the ~~should be used for~~ safety factor reviews. These databases should include the necessary input data for the safety factor reviews, for example, relevant design information and safety analyses, including information on design modifications, operational history data, operating events data, data from on-site monitoring networks, non-conformance data, and maintenance and testing data.

4.10 Each safety factor should be reviewed (see Section 7) for all relevant operational states and accident conditions, and an assessment in terms of compliance with current safety standards and operating practices should be made.

4.11 Areas where either the licensing basis or current safety standards and operating practices are not achieved should be identified. The safety significance of all findings should be evaluated using deterministic and probabilistic methods as appropriate (see Section 7). A list of proposed corrective actions and safety improvements should be prepared for each negative finding. If no reasonably practicable safety improvement are identified, an adequate justification should be provided.

4.12 If the operating organization identifies a finding that poses an immediate and significant risk to workers, the public or the environment, prompt corrective actions should be taken and should be reported to the regulatory body.

4.13 A safety factor report should be prepared to summarize the results of the review of each safety factor (see Appendix II). Areas where current safety standards and operating practices are exceeded (that is, plant strengths) should be identified and stated in the safety factor report.

4.14 A global assessment should then be performed, and a report of the global assessment should be prepared (see Section 8 and Appendix II).

4.15 A final PSR report should be prepared by the operating organization and should include the following:

(a) A summary of the outcomes from the safety factor reviews, including a list of findings indicating areas where current safety standards and operating practices are not achieved, and a list of areas where current safety standards and operating practices are exceeded (plant strengths);
(b) A summary of the outcomes from the global assessment;
(c) An integrated implementation plan of proposed safety improvements, including their safety significance and prioritization.

**Preparation of the integrated implementation plan of safety improvements**

4.16 The safety improvements and the integrated implementation plan proposed in the final PSR report should be updated to address outcomes of the regulatory review. The updated report should include the outcomes with respect to the scope and adequacy of the proposals for safety improvements and any proposed changes to their ranking, prioritization and timing.

4.17 The integrated implementation plan should take into account interactions between individual safety improvements, with consideration given to appropriate configuration management. The plan should also specify the schedules for implementation of safety improvements and the necessary resources. The implementation of safety improvements will have different time frames; however, the majority of the safety improvements should be completed far in advance of the next PSR.

4.18 If safety improvements are proposed to be implemented in stages (e.g. for PSRs performed for multiple units of the same design, or for complex safety improvements) this should be justified by the operating organization.

4.19 The integrated implementation plan should be subject to approval by the operating organization, to secure the necessary human and financial resources to implement the proposed safety improvements in accordance with a reasonably practicable schedule. The approved plan should then be submitted to the regulatory body for review and, if required, for approval, in accordance with regulatory requirements.

4.20 A summary report should be prepared to present the highlights of the PSR review process. This report should be made available to the public, in accordance with regulatory requirements.

## ACTIVITIES OF THE REGULATORY BODY IN RELATION TO PSR

4.21 The requirements for the PSR should be established by the regulatory body.

4.22 Milestones and time frames provided by the operating organization should be reviewed by the regulatory body.

4.23 The regulatory body should appoint a project manager for assessment of the PSR. The responsibilities of the project manager should include:

(a) Coordination of all PSR related activities within the regulatory body (and any external sources of assistance);
(b) Acting as a focal point for communication with the operating organization.

4.24 The regulatory body should assess the PSR basis document provided by the operating organization and should reach an agreement with the operating organization on the general scope, requirements, objectives and outputs of the PSR and on the format and content of the PSR reports (see also para. 6.~~6~~5).

4.25 A plan should be prepared by the regulatory body for performing the regulatory review and assessment of the PSR reports. The plan should state the review criteria to be used, and should identify the source and availability of the technical experts who will conduct the regulatory review.

4.26 Appropriate training and briefings should be provided for persons who will undertake the regulatory review and assessment to ensure that criteria are applied consistently and to facilitate the effective and efficient completion of the regulatory review.

4.27 The regulatory body should review the PSR reports and should assess the PSR findings and proposals for safety improvements submitted by the operating organization. To do this, the regulatory body may use its own analysis methods and verification and validation calculations, for example, using alternative computer codes.

4.28 During the review process, the regulatory body and/or its technical support staff should communicate with the operating organization to clarify issues, including discussion of any additional issues identified, and to acquire any necessary additional information. The results of these interactions should be documented for future reference.

4.29 The regulatory body should ensure that reports of the PSR regulatory review and assessment are prepared that clearly identify all significant safety issues that need to be resolved. Such reports could also give an initial indication of the acceptability of safety improvements proposed by the operating organization.

4.30 In the event that the PSR identifies a finding that poses an immediate and significant risk to workers, the public or the environment, the regulatory body should verify that the operating organization takes prompt corrective action.

4.31 Using the reports from the review and assessment of individual safety factors, the regulatory body (usually ~~the PSR~~a project manager for assessment of the PSR) should prepare an integrated PSR review report. The integrated PSR review report should present, in a concise way, the following:

(a) The regulatory body's view of the adequacy of the PSR as documented in the reports submitted, including the safety improvements already implemented by the operating organization;
(b) The regulatory body's view of the adequacy of corrective actions and safety improvements identified by the operating organization but not yet implemented;
(c) An evaluation of the time schedule for the integrated implementation plan proposed by the operating organization.

4.32 The regulatory body should discuss the integrated PSR review report with the operating organization. This may involve several meetings but should lead to an agreement from both parties on an updated integrated implementation plan of safety improvements. The regulatory body should then take appropriate licensing or other actions consistent with regulatory requirements. Any subsequent changes in integrated implementation plan of safety improvements should be discussed and agreed with the regulatory body.

# 5. ACTIVITIES FOLLOWING PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS

5.1 The operating organization and the regulatory body should maintain adequate arrangements for project management after the completion of the PSR. These arrangements should ensure that the regulatory body is notified when safety improvements are implemented and is notified of any significant delays in completing the improvements later than the agreed schedule.

5.2 All PSR documentation should be stored by the operating organization using a suitable system to allow easy retrieval and examination by both the operating organization and the regulatory body. The documentation should contain the final versions of the PSR reports and information on lessons learned from the PSR.

5.3 The outcomes of the PSR and the resulting safety improvements will often necessitate changes to plant documentation. Therefore, the operating organization should update all plant documentation including, for example, the safety analysis report[5], operating and maintenance procedures and training materials, to reflect the outcomes of the PSR.

5.4 The outcomes of the PSR and the resulting safety improvements will often necessitate changes to design, operation and licensing documentation to reflect the actual configuration of

[5] Recommendations on the safety analysis report are provided in IAEA Safety Standards Series No. SSG-61, Format and Content of the Safety Analysis Report for Nuclear Power Plants [8]~~[6]~~.

the nuclear power plant. The operating organization should modify all affected documentation (e.g. manuals relating to the operating organization, the emergency plan, training plans) as necessary.

5.5 Where a final safety analysis report is part of the documentation of the nuclear power plant, this should be updated after completion of the PSR to reflect the findings. The final safety analysis report (or other equivalent safety documents) should be updated by the operating organization to incorporate all the design changes and the results of safety analyses obtained in support of the safety improvements.

5.6 The operating organization and/or the regulatory body should report the outcomes of the PSR to the government in accordance with regulatory requirements and national practice. A summary of the results of the PSR should be made available to the public, as recommended in para. 4.20.

# 6. SCOPE, STRATEGY AND GENERAL METHODOLOGY FOR PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS

6.1 The scope of the PSR should include all safety aspects of a nuclear power plant and should be agreed with the regulatory body. The review should cover the operation of all facilities and SSCs on the site covered by the operating licence (including, if applicable, waste management facilities and on-site simulators), including a review of the operating organization and its staff. In addition, any accepted exemptions from code requirements in the licensing basis at the time of the code cut-off date should be documented and re-assessed or re-validated.

6.2 When performing PSR of a nuclear power plant with multiple units, aspects such as radiation protection, emergency planning and radiological impact on the environment could be covered in reviews that are common to all units. Other aspects (e.g. the actual condition of SSCs important to safety, ageing and safety performance) should be covered in reviews that are specific to each unit.

6.3 The conduct of a generic PSR of multiple units of the same design and operation, whether or not located on the same site, can decrease the resources and effort needed. However, a generic PSR should only be conducted for safety factors, or parts of a safety factor, that are similar. If the units are located at different sites or differ in other respects (e.g. different design features, site conditions, organizational and human factors) site specific or unit specific aspects should be reviewed separately.

6.4 The precise approach to PSR of the safety factors identified should be adapted to the legal and regulatory framework and processes within the State. In particular, the list of safety factors (see para. 2.13) may be extended (e.g. by considering radiation protection or other issues as separate safety factors) or reduced (e.g. by combining or grouping the safety factors differently).

6.5 Before the review work is started, a number of prerequisites should be satisfied. The main prerequisite is an agreement between the operating organization and the regulatory body as to the scope and objectives of the PSR, including current national and international standards and codes to be used (see also para. 4.24).

6.6 The PSR basis document prepared by the operating organization should identify the scope, major milestones, including cut-off dates (beyond which changes to codes and standards and new information will not be considered), and methodology of the PSR, the safety factors to be reviewed, the structure of the documentation and the applicable national and international standards, codes and practices. The process for categorizing, prioritizing and resolving findings should also be agreed upon and set out in the PSR basis document.

6.7 The PSR should apply all relevant regulations and standards within the State. Other requirements such as international safety standards and operating practices, and national or international guidance should be met to the fullest extent practicable. The selection and hierarchy of safety standards and operating practices considered should be clearly stated in the PSR basis document. Special consideration should be given to safety standards issued by the State of origin of the technology.

6.8 If there are no adequate national standards, reference should be made to international codes and standards (such as those of the IAEA, the International Organization for Standardization and the International Electrotechnical Commission) or, where appropriate, to codes and standards of a recognized organization of a particular State. Information on good practices collected by industry associations and owners' groups could also be relevant and should be taken into account.

6.9 The PSR basis document should outline or reference the management processes to be followed in conducting the PSR and producing the various documents relating to the review (see Appendix II) so as to ensure a complete, comprehensive, consistent and systematic approach. Requirements for the management system are established in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [9]~~[9][7]~~.

6.10 The PSR basis document should provide or reference a project plan that identifies all the activities to be performed during the review, together with associated timelines and responsibilities. This should present a realistic and reasonable schedule for the conduct of the PSR, including sufficient time for completion of the review and assessment process by the regulatory body. A typical content of a PSR basis document is shown in Appendix II.

6.11 The PSR schedule should take into account that the review of safety factors is an iterative process and that the interface between safety factors also needs to be considered. The teams reviewing different safety factors should communicate with each other throughout the review process, starting in the preparation phase. Some of the findings identified in the review of a particular safety factor may need to be considered in the review of other safety factors. In addition, the outputs from the review of some safety factors may be relevant as inputs to the review of other safety factors. Typical lists of input and output information for each safety factor are provided in Appendix I with the IAEA safety requirements to be considered in this context.

6.12 Unless otherwise stated in regulatory requirements, the starting point for the PSR should be taken to be the time of the agreement between the operating organization and the regulatory body in the preparation phase (see para. 2.18); the end point of the PSR should be the finalization of the integrated implementation plan.

6.13 International experience suggests that a first PSR at an older nuclear power plant may reveal discrepancies between the design documentation and the actual configuration, or that information on the design basis of SSCs important to safety is incomplete. Where this is the case, the design documentation should be updated and a proper safety justification should be provided (for example, renewal of the obsolete or incomplete final safety analysis report).

6.14 The effort necessary to conduct a second (or subsequent) PSR of a nuclear power plant will often be considerably reduced compared with that for the first PSR. In general, subsequent PSRs should focus on changes in requirements, plant conditions, operating experience and new information, rather than repeating the activities of previous reviews. However, a subsequent PSR should consider explicitly whether the earlier PSR continues to remain valid. This is particularly important in management systems, where a change in one component cannot be evaluated in isolation from the rest. Any changes made to the management system since the initial or previous PSR should be reassessed in a comprehensive manner.

6.15 The PSR should take account of existing ongoing operational programmes, such as configuration management and ageing management; the results of trend analyses from these programmes should be reviewed to evaluate their effectiveness.

6.16 Experience has shown that licensees with good configuration management programmes find it easier to perform a PSR. The PSR should consider how effective the plant's configuration management programme has been in keeping the safety documentation (e.g. the final safety analysis report) up to date in light of subsequent modifications, refurbishment and changes to operating, testing, maintenance and other practices.

6.17 Some safety factors (or parts of a safety factor) might be assessed more efficiently and effectively in other contexts or through different means than by PSR (for example by continuous review through other programmes). In such cases, the PSR should focus on the assessment methodology applied at the nuclear power plant and should review relevant trends.

6.18 As part of the review of each safety factor, all the documents listed in the PSR basis document should be checked for completeness. If there is no overall technical database for the plant, a common set of databases should be established early in the review process for the review of the 14 safety factors and the global assessment.

6.19 Findings from the reviews of safety factors should be evaluated and the scheduling of any proposed safety improvements should be determined. The proposed schedule should take into account the need to implement reasonably practicable safety improvements in a timely manner in accordance with the global assessment of safety at the plant. If there is an immediate and significant risk to workers, the public or the environment, this should be addressed urgently by the operating organization and should not await completion of the PSR process. Instead, the operating organization should determine prompt corrective actions and, where relevant, submit these without delay to the regulatory body for agreement or approval.

6.20 The level of plant safety should be determined by a global assessment reflecting, among other things, the combined effects of all safety factors. It is possible that a negative finding (deviation) in one safety factor can be compensated for by a positive finding (strength) in another safety factor. Section 8 provides further recommendations on the global assessment of safety at the plant.

6.21 If the design basis for the nuclear power plant is not currently documented, the operating organization should re-establish the design basis before the start of the PSR, or early in the PSR process. Otherwise, the PSR should review the design basis documentation using the final safety analysis report where this is part of the safety and/or licensing documentation.

6.22 The results of relevant studies and of routine and special safety reviews, as well as activities relating to licensing, compliance or operations, should be used, as appropriate, as inputs into the PSR to minimize any duplication of effort. The origins of all information used should be referenced appropriately and an explanation should be provided of how each reference has been used.

6.23 Safety improvements should be implemented in accordance with the integrated implementation plan submitted to the regulatory body for agreement or approval. For a PSR of nuclear power plants with multiple units, or in the case of a PSR for multiple plants of identical design, safety improvements may be implemented in a lead unit and lessons learned may then be used for the implementation of safety improvements in the other units. This process should be described in the integrated implementation plan.

6.24 The global assessment should take into account all the positive and negative findings from the PSR, and the corrective actions and/or safety improvements proposed, and should assess the overall level of safety that will be achieved at the nuclear power plant following the PSR. The risks associated with any unresolved negative findings should be assessed and an appropriate justification for continued operation should be provided.

6.25 The results of the PSR should be documented by the operating organization and the documentation should be submitted to the regulatory body either during the PSR or during a structured continuous improvement programme, as required. The documentation should include:

(a) Reports on the review of each safety factor;
(b) A report documenting the results of the global assessment;
(c) The final PSR report, including information on the proposed safety improvements and integrated implementation plan and a summary of the reports on safety factors and the global assessment.

The contents of these documents are described in Appendix II

## 7. SAFETY FACTORS IN A PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS

7.1 The important aspects of safety of an operating nuclear power plant that are addressed in a PSR are termed ‘safety factors’. Fourteen safety factors are identified in this Safety Guide (see para. 2.13), which may be used to subdivide the PSR. This section provide recommendations on these safety factors, their individual objectives, scope and tasks and also the specific methodology for their review. Information on interfaces between safety factors is provided in Appendix I and information on typical inputs, outputs and references for each safety factor is given in Annex I. The content of a typical report on the review of each safety factor is set out in Appendix II.

7.2 Radiation protection is not regarded as a separate safety factor in this Safety Guide since it is related to most of the other safety factors. The arrangements for radiation protection and their effectiveness should generally be reviewed as specific aspects of the safety factors relating to: plant design; actual condition of SSCs important to safety; safety performance; and operational limits and conditions and operating procedures. Alternatively, the operating organization may decide to review radiation protection as a separate safety factor.

7.3 Findings from the review of individual safety factors may indicate that plant safety is acceptable; however, a global assessment of safety at the plant should be performed to review interactions, overlaps and gaps between safety factors and to form an overall view.

7.4 The review should determine the status of each safety factor at the time of the PSR and should assess future safety at the nuclear power plant at least until the next PSR and, where appropriate, up to the end of planned operation. This should include a review of the capability of the operating organization to identify potential failures and either prevent them or mitigate their consequences before they could lead to a radiological incident. Ageing related degradation mechanisms that could lead to failures of SSCs important to safety that could potentially limit the plant's operating lifetime should be identified to the extent possible. The assessment should also include a potential of external hazards and site characteristics to change over time until the next PSR.

7.5 The outputs from the review of safety factor 8 on safety performance, and of safety factor 9 relating to the use of feedback of operating experience can be used as early inputs to the reviews of other safety factors. Therefore, the majority of the tasks in the review of these safety factors should be addressed at an early stage in the PSR.

~~7.5~~7.6 The level of detail of the review could vary from safety factor to safety factor. For some safety factors, a high level or programmatic review could be performed. Where such an approach is adopted, this should be set out and justified in the PSR basis document.

~~7.6~~7.7 The review of safety factors should assess all relevant documents identified in the PSR basis document. If further documents are identified as being relevant during the PSR process, these should also be reviewed. The level of effort necessary to review a safety factor will depend on the quality, availability and retrievability of relevant information.

~~7.7 The outputs from the review of safety factor 8 on safety performance, and of safety factor 9 relating to the use of feedback of operating experience can be used as early inputs to the reviews of other safety factors. Therefore, the majority of the tasks in the review of these safety factors should be addressed at an early stage in the PSR.~~

7.8 Prior to commencing the review of individual safety factors, methods to assess, categorize, rank and prioritize findings should be established and these methods should be documented (e.g. in the review reports).

7.9 The review of safety factors should be used to identify positive and negative findings (see para. 2.15), which should be documented in the safety factor review report. If there are no changes in relevant safety standards or to the plant, a statement to this effect should be made in the report.

7.10 Negative findings should be divided into:

(a) Deviations for which no reasonably practicable safety improvements can be identified;

(b) Deviations for which identified safety improvements are not considered necessary;
(c) Deviations for which safety improvements are considered necessary.

7.11 The approach taken to negative findings should be justified by the operating organization and agreement by the regulatory body should be sought, in accordance with regulatory requirements.

7.12 In the case of negative findings for which no reasonably practicable safety improvements can be identified, the reason(s) should be documented and the issue revisited after an appropriate period of time to determine whether a practicable solution is available.

7.13 For negative findings for which safety improvements are not considered necessary, the reason(s) should be documented and the action considered completed.

7.14 Negative findings for which safety improvements are necessary, including updating or extending plant documentation or operating procedures, should be categorized and prioritized in accordance with their safety significance. The categorization and prioritization of safety improvements should be based on the results of safety assessment. Safety improvements from the safety factor reviews, together with safety improvements resulting from the global assessment, should be included in the operating organization's integrated implementation plan.

7.15 Findings that have an interface with other safety factors should be discussed immediately with the relevant review team(s).

## SAFETY FACTORS RELATING TO THE PLANT

### Safety factor 1: Plant design

7.16 In accordance with the requirements established in SSR-2/1 (Rev. 1) [7]~~[7][5]~~, the nuclear power plant is required to be designed to ensure that safety functions can be performed with the necessary reliability, that radiation doses to workers at the plant and to members of the public do not exceed the dose limits and are kept as low as reasonably achievable, and that plant states that could lead to high radiation doses or to a large radioactive release have been 'practically eliminated'.

7.17 The design basis of SSCs important to safety should be made available to provide for the safe operation and maintenance of the plant throughout its operating lifetime and to facilitate plant modifications.

*Objective*

7.18 The objective of the review of plant design is to determine the adequacy of the design and its documentation by assessment against the current licensing basis and national and international standards, requirements and practices.

*Scope*

7.19 The review of plant design should include the following aspects:

(a) Review of the adequacy of the design basis documentation;

(b) Identification of differences between codes and standards met by the design (e.g. the standards and criteria in force when it was built) and the current nuclear safety and design standards;
(c) Review of compliance with plant design specifications;
(d) Review of the completeness and adequacy of the list of SSCs important to safety;
(e) Review of SSCs important to safety to ensure that they have appropriate design characteristics, that adequate engineering design rules and proven engineering practices were considered in their design to ensure that fundamental safety functions are fulfilled for all operational states and accident conditions.
(f) Review of the protection against internal and external hazards;
(g) Review of the strategy for spent fuel storage and an assessment of the condition of the storage facilities, the records management and the inspection regimes being used.
(h) Review of the safety analysis report or licensing basis documents following plant modifications, taking into account their cumulative effects and updates to the site characterization.
(i) Review of the implementation of defence in depth and engineered barriers for preventing the dispersion of radioactive material (integrity of fuel, reactor coolant system and containment) to verify that design and other characteristics are appropriate to meet the requirements for plant safety and performance for all plant states and the applicable period of operation.

The scope of this review will depend on the extent of changes in standards and/or the licensing basis since the previous PSR or the start of operation.

7.20 The review of plant design should consider safety requirements for the design as established in SSR-2/1 (Rev. 1) [7]~~[7][5]~~, and recommendations relating to the safety analysis report as provided in SSG-61 [8]~~[8][6]~~.

*Methodology*

7.21 The review should be performed systematically by means of a clause-by-clause review of national and international requirements and standards listed in the PSR basis document and other requirements and standards identified as relevant during the course of the review. Where this would assist the review, the evolution of these requirements and standards from the versions used for the original design should be evaluated to assess the impact of changes on the plant design. The impact of climate change to design basis should be considered.

7.22 In some cases, comparison with requirements and standards may be best performed by means of a high level or programmatic review. If this approach is to be adopted, the PSR basis document should clearly indicate this intention and, where appropriate, this should be agreed with the regulatory body.

7.23 In the review, consideration should be given to subdivision into topics in accordance with plant systems, such as the reactor core, reactor coolant system, containment system, instrumentation and control systems, electrical power systems and auxiliary systems.

7.24 The review of this safety factor should be conducted for all SSCs important to safety. The review should seek to identify deviations between the plant design and current safety standards and regulatory requirements (including relevant design codes) and to determine their safety significance. If a suitable list of SSCs is not available, one should be developed by the operating organization as part of the PSR.

7.25 The review should consider the adequacy of defence in depth in the plant design. This should include an examination of:

(a) The degree of independence of the levels of defence in depth;
(b) The adequacy of implementation of preventive and mitigatory safety measures;
(c) Redundancy, separation and diversity of SSCs important to safety;
(d) Defence in depth in the design of structures, systems, and components;
(e) The consideration of defence in depth in human factors engineering.

7.26 The review should confirm that adequate margins are available to avoid cliff edge effects and early radioactive releases or large radioactive releases.

7.27 Where the plant has undergone a significant number of modifications over its lifetime or in the period since the last PSR, the cumulative effects of all modifications on the design should be examined (e.g. review of the loading on electrical supplies or post-trip cooling demands on water supplies). Adequate means should be available to allow for effective tracking of these effects.

7.28 The PSR should verify that significant documentation relating to the original and/or reconstituted design basis has been obtained, securely stored and updated to reflect all the modifications made to the plant since its commissioning. Recommendations on meeting the requirements established in GSR Part 2 [9]~~[9]~~~~[7]~~ for document control are provided in IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [10]~~[10]~~~~[8]~~.

7.29 A design re-evaluation should be undertaken if the design information is inadequate or there is significant uncertainty over the adequacy of an SSC important to safety to fulfil its safety function (e.g. because of its actual condition: see safety factor 2).

7.30 The review should aim to verify that the design ensures that the generation of radioactive waste and discharges are kept to the minimum practicable in terms of both activity and volume, by means of appropriate design measures and operational and decommissioning practices.

7.31 The review should be performed to verify the existence of adequate safety margins to take due account of changes in site-specific design parameters.

7.32 The potential impact on safety due to design changes related to safety measures, nuclear security measures, and arrangements for the State system of accounting for and control of nuclear material should be reviewed within the PSR. These changes might include the following aspects: security procedures that might potentially delay or prevent the implementation of safety measures; security and safeguards-related equipment that might trigger new safety hazards; and cyber-security-related arrangements that might affect the instrumentation and control systems. Recommendations on managing the interfaces can be found in Ref. [11]~~[11]~~~~[9]~~[20]~~[20]~~~~[17]~~.

**Safety factor 2: Actual condition of SSCs**

7.33 The actual condition of SSCs important to safety within the nuclear power plant is an important factor in any review of the safety of the plant. Hence, it is important to document thoroughly the condition of each SSC important to safety. Additionally, knowledge of any existing or anticipated obsolescence of plant systems and equipment should be considered part of this safety factor.

*Objective*

7.34 The objective of the review of this safety factor is:

(a) To determine the actual condition of SSCs important to safety;
(b) To consider whether SSCs important to safety are capable of meeting design requirements, at least until the next PSR;
(c) To verify that the condition of SSCs important to safety is properly documented;
(d) To verify that procedures for maintenance, testing, surveillance and inspection of SSCs important to safety are in place, reviewed and updated as necessary;
(e) To review the ongoing maintenance, surveillance, testing and in-service inspection programmes of SSCs important to safety, as applicable.

*Scope*

7.35 The review of the actual condition of SSCs important to safety, including spent fuel storage facilities, should include examination of the following aspects for each SSC:

(a) Actual ageing effects and processes against anticipated and predicted ones;
(b) Obsolescence;
(c) Design requirements and standards including design basis assumptions;
(d) Plant programmes, including the chemistry programme;
(e) Results of inspections and/or walkdowns of SSCs;
(f) Maintenance records and operating history of SSCs;
(g) Dependence on essential services and/or supplies external to the plant;
(h) Availability of spare parts and their adequate storage.

7.36 The review of safety factor 2 should consider at a minimum Requirements 15, 28, 29, and 31 of SSR-2/2 (Rev. 1) [2]. Recommendations on chemistry programme for water cooled nuclear power plants are provided in IAEA Safety Standards Series No. SSG-13 (Rev. 1), Chemistry Programme for Water Cooled Nuclear Power Plants [12].

*Methodology*

7.37 The actual condition of SSCs important to safety should be reviewed using knowledge of:

(a) Any existing or anticipated ageing processes;
(b) Knowledge of any existing or anticipated obsolescence of plant systems and equipment, especially those for which no direct substitute is available;
(c) Modification history;
(d) Operating history.

7.38 For practical purposes, the review may group SSCs important to safety in accordance with functional systems or types.

7.39 The frequency of maintenance, testing, surveillance and inspection is required to be determined considering the importance to safety of the component, supported by probabilistic assessments; its reliability and availability for operation; its potential for degradation due to ageing; operating experience, or recommendations from vendors (see para. 8.5 of SSR-2/2 (Rev. 1) [2]).

7.40 Paragraph 8.14A of SSR-2/2 (Rev. 1) [2] states that “The operating organization shall establish maintenance programmes for non-permanent equipment to be used for accidents more severe than design basis accidents”.

7.41 The implications of changes to design standards on the actual condition of SSCs since the plant was designed, or since the last PSR should be examined.

7.42 Inputs to the review of this safety factor should be made available from the ageing management and obsolescence programmes of the operating organization. However, if these programmes do not provide adequate information, the necessary inputs should be derived at an early stage of the PSR. Further recommendations are provided in IAEA Safety Standards Series No. SSG-48, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants [12]~~[12]~~~~[10]~~.

7.43 Where needed, inputs to the review of this safety factor with respect to maintenance, surveillance and inspection programmes should be provided by the safety assessment.

7.44 Where data on actual condition of SSCs are lacking, they should be generated or derived by performing special tests, plant walkdowns and inspections as necessary.

7.45 Requirement 28 of SSR-2/2 (Rev. 1) [2] states that “**The operating organization shall develop and implement programmes to maintain high standards of material conditions, housekeeping and cleanliness in all working areas.**”

7.46 The validity of existing records should be checked to ensure that they accurately represent the actual condition of SSCs important to safety, including any significant findings from ongoing maintenance, tests and inspections.

7.47 It might not always be possible to determine the actual condition of SSCs important to safety in some areas of the plant, for example due to limited accessibility. Such instances should be highlighted and the safety significance of the resulting uncertainty in the actual condition of those SSCs should be determined.

7.48 Uncertainties in relation to the actual condition of SSCs may be reduced by considering evidence from similar components from other plants or facilities that are subject to similar conditions and/or knowledge of the relevant ageing processes and operating conditions.

7.49 The actual condition of spent fuel storage facilities should be assessed to verify their adequacy and their conformance with the spent fuel storage strategy for the nuclear power plant.

7.50 After determining the actual condition of SSCs important to safety, each SSC should be assessed against the current design basis (or updated design basis: see safety factor 1) to confirm that design basis assumptions have not been significantly challenged and will remain valid until the next PSR.

7.51 Where consistency with the design basis has been significantly challenged, the PSR should make proposals for corrective action (for example, additional inspections or tests, further safety analysis or the replacement of components). These proposals should then be considered further in the global assessment.

7.52 The methodology and results of the ageing management review described in SSG-48 [12]~~[12][10]~~ should also be used to provide information on the current performance and condition of the structure or component, including assessment of any indication of significant ageing effects.

**Safety factor 3: Equipment qualification**

~~7.52~~7.53 Requirement 13 of SSR-2/2 (Rev. 1) [2] states:

> "**The operating organization shall ensure that a systematic assessment is carried out to provide reliable confirmation that safety related items are capable of the required performance for all operational states and for accident conditions.**"

Equipment qualification should take into account the prevailing environmental conditions, throughout the design life, with due account taken of plant conditions during maintenance and testing.

*Objective*

~~7.53~~7.54 The objective of the review of equipment qualification is to demonstrate for the period until at least the next PSR, that:

(a) The equipment is capable of performing its intended safety function(s) under the range of service conditions it is subject to, specified for the nuclear power plant in operational states and in accident conditions, as well as under the effects caused by specified service conditions during plant states and during external events (e.g. seismic events, electromagnetic phenomena such as arcing, lightning).

(b) The status of each item of qualified equipment is preserved and properly documented throughout the lifetime of the plant.

~~7.54~~7.55 The review of safety factor 3 should consider at a minimum Requirement 30 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~, and Requirement 13 of SSR-2/2 (Rev. 1) [2].

*Scope*

~~7.55~~7.56 The review of equipment qualification should address all aspects affecting the suitability of each qualified system or component with respect to its intended functions, which includes:

(a) Equipment qualification programme;
(b) Suitability and correctness of equipment functions and performance during their service life until at least the next PSR;
(c) Environmental qualification, including possible changes in environmental conditions under all plant states since the programme was devised;
(d) Qualification for the effects of internal hazards and external hazards;
(e) Electromagnetic qualification;
(f) Equipment qualification documentation.

*Methodology*

~~7.56~~7.57 The assessment of the effectiveness of the equipment qualification programme should include the evaluation of activities performed by:

(a) The operating organization;
(b) Vendors and manufacturers of qualified equipment;
(c) Third party providers of equipment qualification services;
(d) Equipment qualification testing facilities.

~~7.57~~7.58 The review of this safety factor should confirm whether the equipment qualification is an active and ongoing process. The review should verify whether:

(a) A list of equipment subject to qualification is available and up to date.
(b) The methods and criteria used in the equipment qualification programme reflect licensing conditions and the design basis.
(c) The original assumptions regarding the safety, operability and performance of equipment were reasonable and remain valid.
(d) The equipment qualification documentation is available in an auditable and traceable form, provides evidence of qualification for each item of equipment in the equipment qualification list, and includes a system for locating supporting documentation.
(e) The supporting documentation is traceable and includes the following:
    (i) Test and analysis documentation;
    (ii) Evaluation of operating experience and information from feedback programmes;
    (iii) Procurement documents;
    (iv) Quality assurance data from the manufacturing of qualified equipment;
    (v) Criteria for the storage, transport and installation of qualified equipment;
    (vi) Criteria for the surveillance and maintenance of qualified equipment.
(f) There is sufficient evidence of the following:
    (i) The technical basis and assumptions used in the modelling of qualified life remain valid;
    (ii) The installed equipment matches the qualified equipment;
    (iii) The equipment is installed correctly;
    (iv) The equipment and any protective barriers are appropriately maintained;
    ~~(iv)~~(v) Obsolescence of the equipment or of spare parts is adequately considered;
    ~~(v)~~(vi) Corrective actions are identified and performed in a timely manner;
    ~~(vi)~~(vii) Personnel are capable of identifying the characteristics of ageing degradation effects.
(g) The measures necessary to preserve the status of qualified equipment during the service life of equipment are documented in appropriate procedures or instructions (e.g. for storage and handling of qualified spare parts, and for installation, surveillance,

maintenance and component replacement) and are implemented.

(h) The relevant personnel have appropriate qualifications and training to establish and preserve equipment qualification.

(i) Maintenance and testing of qualified equipment, surveillance and inspection of equipment conditions, and monitoring of environmental conditions have been established to ensure that the ageing degradation and functional capability of qualified equipment remain acceptable.

(j) A programme is in place to analyse premature degradation or failures of qualified equipment, and to implement appropriate corrective actions, including revisions of conclusions on the status of qualified equipment.

(k) An operating experience programme is in place to collect and review information relevant to the status of qualified equipment.

(l) The equipment qualification programme reflects the as-built design of the nuclear power plant, including any recent modifications.

(m) There is adequate evidence that controls implemented within the equipment qualification programme (e.g. corrective actions, configuration management) are effective.

~~7.58~~7.59 The equipment qualification programme should have clearly defined interfaces with other programmes and processes and activities to ensure the status of qualified equipment is preserved.

~~7.59~~7.60 The process for making modifications to the plant should also ensure that equipment qualification documentation is updated to reflect any design changes.

~~7.60~~7.61 Further recommendations are provided in IAEA Safety Standards Series No. SSG-69, Equipment Qualification for Nuclear Installations [14]~~[14]~~~~[11]~~.

**Safety factor 4: Ageing**

~~7.61~~7.62 All SSCs important to the safety — and other SSCs whose failure might prevent SSCs important to safety from fulfilling their intended functions or are credited in the safety analyses (deterministic and probabilistic) as performing the function of coping with certain types of event — are subject to ageing, which could eventually impair their fulfilment of safety functions and their service lives.

*Objective*

~~7.62~~7.63 The objective of the review is to determine whether SSCs subject to ageing are being effectively managed and whether an effective ageing management programme is in place so that all required safety functions will be fulfilled over the design lifetime of the plant and, if applicable, for long term operation.

*Scope*

~~7.63~~7.64 The review of ageing should include review of the ageing management programme established at the nuclear power plant. The review should evaluate both programmatic and technical aspects. The following aspects of the ageing management programme should be evaluated:

(a) The timely detection and mitigation of ageing mechanisms and/or ageing effects;

(b) The comprehensiveness of the programme, i.e. does it address all SSCs subject to ageing?
(c) The effectiveness of operating and maintenance policies and/or procedures for managing the ageing of replaceable components;
(d) Evaluation and documentation of potential ageing degradation that might affect the safety functions of SSCs important to safety;
(e) Management of the effects of ageing on those parts of the nuclear power plant that will be needed for safety purposes when the reactor has ceased operation, for example the spent fuel storage facilities;
(f) Performance indicators;
(g) Record keeping.

~~7.64~~7.65 The review should evaluate the following technical aspects:

(a) The ageing management methodology (see SSG-48 [12]~~[12][10]~~);
(b) The operating organization's understanding of dominant ageing mechanisms and phenomena, including knowledge of actual safety margins;
(c) Availability of data for assessing ageing degradation, including baseline data and operating and maintenance histories;
(d) Acceptance criteria and safety margins for SSCs important to safety;
(e) Operating guidelines aimed at controlling and/or moderating the rate of ageing degradation;
(f) Methods for monitoring ageing and for mitigation of ageing effects;
(g) Awareness of the physical condition of SSCs important to safety and any features that could limit service life;
(h) Understanding and control of ageing of all materials (including consumables, such as lubricants) and SSCs that could impair their safety functions;
(i) Obsolescence of technology used in the nuclear power plant.

~~7.65~~7.66 The review of safety factor 4 should consider at a minimum Requirement 31 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~, and Requirement 14 of SSR-2/2 (Rev. 1) [2].

*Methodology*

~~7.66~~7.67 The ageing management programme should be reviewed to confirm that it provides for the timely detection and prediction of ageing degradation that might affect the safety functions and service lives of SSCs subject to ageing, and that it identifies appropriate measures for the maintenance of these functions. Programme descriptions, evaluation of programmes and technical bases for programmes, plans for the reliability and availability of SSCs, the detection and mitigation of ageing effects, and the actual physical condition of structures and components should be examined. The review should focus on the integrated performance of the systems important to safety and on the results of periodic inspection and testing programmes and trends in important safety parameters.

~~7.67~~7.68 The review should examine whether effective control of ageing degradation is achieved by means of a systematic ageing management process, in accordance with Requirement 31 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~, and Requirement 14 of SSR-2/2 (Rev. 1) [2] and the recommendations provided in SSG-48 [12]~~[12][10]~~. Such a process consists of the following ageing management tasks, which should be performed on the basis of a proper understanding of the ageing of the SSCs important to safety:

(a) Operation of SSCs within operating guidelines with the aim of minimizing the rate of ageing degradation;
(b) Inspection and monitoring with the aim of timely detection and characterization of any ageing degradation;
(c) Assessment of observed ageing degradation in accordance with appropriate guidelines in order to assess the integrity and functional capability of the structure or component;
(d) Repair or replacement of parts to prevent or remedy unacceptable ageing degradation.

~~7.68~~7.69 The review should assess whether:

(a) A systematic, effective and comprehensive ageing management programme is in place for all SSCs important to safety;
(b) Any non-safety-classified SSCs whose failure might prevent SSCs important to safety from fulfilling their intended functions are addressed to an adequate extent;
(c) Any non-safety-classified SSCs that are credited in the safety analyses (deterministic and probabilistic) as performing the function of coping with certain types of event are addressed to an adequate extent;
(d) All relevant ageing degradation mechanisms are identified, and the models used to predict the evolution and advancement of ageing degradation are properly supported in accordance with current operating practices pertaining to ageing degradation;
(e) Adequate measures are taken to monitor and control ageing processes;
(f) The ageing management programme will ensure continued safe operation for at least the period until the next PSR.

~~7.69~~7.70 Structures and components that are periodically replaced or refurbished in accordance with predefined rules can be excluded from the scope of ageing management. However, the adequacy of the predefined rules should be assessed.

SAFETY FACTORS RELATING TO SAFETY ANALYSIS

**Safety factor 5: Deterministic safety analysis**

~~7.70~~7.71 Deterministic safety analysis is required to be conducted for each nuclear power plant, in order to confirm the design basis for SSCs important to safety and to demonstrate~~d~~ that the nuclear power plant as designed is capable of complying with authorized limits on discharges with regard to radioactive releases and with the dose limits in all operational states, and is capable of meeting acceptable limits for accident conditions (see para. 5.71 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~.

*Objective*

~~7.71~~7.72 The objective of the review of this safety factor is to determine to what extent the existing deterministic safety analysis is complete and remains valid when the following aspects have been taken into account:

(a) The actual plant design, including all modifications of SSCs since the last update of the safety analysis report or the last PSR;
(b) Current operating modes and fuel management;
(c) The actual condition of SSCs important to safety and their predicted state at the end of the period covered by the PSR;

(d) The use of modern, validated computer codes;
(e) Current deterministic methods;
(f) Current safety standards and knowledge (including research and development outcomes);
(g) The existence and adequacy of safety margins;
(h) Relevant changes of site characteristics.

*Scope*

~~7.72~~7.73 The review of the deterministic safety analysis should include the following tasks:

(a) Review of the application of analytical methods, guidelines and computer codes used in the existing deterministic safety analysis and comparison with current safety standards and regulatory requirements;
(b) Review of the current state of the deterministic safety analysis (original analysis and updated analysis) for the completeness of the set of postulated initiating events forming the design basis, with consideration given to feedback of operating experience from plants of a similar design, in the State or in other States;
(c) Review of the adequacy of the safety assessment in terms of addressing the planned decommissioning actions and potential incidents related to decommissioning, including radiological hazards and personnel exposure;
(d) Evaluation of whether the assumptions made in performing the deterministic safety analysis remain valid given the actual condition of the plant;
(e) Evaluation of whether the actual operational conditions of the plant meet the acceptance criteria for the design basis;
(f) Evaluation of whether the assumptions used in the deterministic safety analysis are in accordance with current safety standards and regulatory requirements;
(g) Review of the application of the concept of defence in depth;
(h) Evaluation of whether appropriate deterministic methods have been used for development and validation of emergency operating procedures and the accident management programme at the plant, including the determination of sufficient grace time periods for operator diagnostics and manual actions;
(i) Evaluation of whether calculated radiation doses and releases of radioactive material in operational states and in accident conditions meet regulatory requirements and expectations and that the plant event sequences that could lead to high radiation doses or to a large radioactive release have been 'practically eliminated';
(j) Analysis of: the functional adequacy, reliability, redundancy, diversity, separation, segregation, independence and equipment qualification of systems and components; the impact on safety of internal and external events, equipment failures and human errors; and the adequacy and effectiveness of engineering and administrative measures to prevent and mitigate accidents.

~~7.73~~7.74 The review of safety factor 5 should consider at a minimum Requirement ~~15~~ 14-18 of GSR Part 4 (Rev. 1) [3], Requirements 5, 10, 13, 16, 19, 20 and 42 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~, and Requirement 3 of IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [15]~~[15][12]~~. Recommendations on deterministic safety analysis are provided in IAEA Safety Standards Series No SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [16]~~[16][13]~~.

*Methodology*

~~7.74~~7.75 The review of deterministic safety analysis should provide a systematic re-examination of how operating experience feedback, new knowledge (e.g. of physical phenomena), changes in analysis and modelling techniques affect safety at the nuclear power plant.

~~7.75~~7.76 The existing deterministic safety analysis should be reviewed against regulatory requirements and current safety standards and operating practices to verify that the design basis for SSCs important to safety is correct and that plant behaviour for postulated initiating events is properly addressed.

~~7.76~~7.77 The review should seek to identify (or confirm) any major ~~weaknesses~~ deviations as well as strengths of the plant design in relation to the application of defence in depth and should evaluate the importance of systems and measures for preventing and managing accidents.

~~7.77~~7.78 The adequacy of the shared resources (whether human resources or material resources) in accident conditions should be reviewed in order to demonstrate that the required safety functions can be fulfilled at each facility in accident conditions.

~~7.78~~7.79 The capabilities of the plant in its current state (where relevant, with account taken of planned safety improvements), should be demonstrated to meet regulatory requirements and expectations in operational states and accident conditions.

~~7.79~~7.80 If it is necessary to repeat the analysis, consideration should be given to using current analytical methods, particularly with regard to computer codes for transient analyses. If the earlier approach is still used, its continuing validity should be verified explicitly in the review, including the assumptions used, the degree of conservatism applied, consideration of uncertainties and the availability of adequate margins ~~and the inherent uncertainties~~ in the analysis to avoid cliff edge effects.

~~7.80~~7.81 The review should include an evaluation of the supporting analyses for design extension conditions. This should determine whether the arrangements aimed at preventing or mitigating severe core damage continue to be sufficient and whether any reasonably practicable safety improvements are available.

~~7.81~~7.82 With regard to design extension conditions, the extent of the inclusion and evaluation of combinations of events and their consequential effects, which could lead to anticipated operational occurrences or to accident conditions, should be reviewed. Depending on their likelihood of occurrence, events which might be consequence of other events, such as flooding after an earthquake, should be considered as to be part of the original postulated initiating event.

~~7.82~~7.83 The review should confirm the adequacy of the documentation of deterministic safety analyses, to ensure that it facilitates independent verification and that this verification is being performed.

**Safety factor 6: Probabilistic safety assessment**

~~7.83~~7.84 A review of the probabilistic safety assessment (PSA) should be conducted to identify potential weaknesses in the design and operation of the nuclear power plant and, as part of the global assessment, to evaluate and compare proposed safety improvements. The review should also confirm that all important contributors to risk (and the associated uncertainties) are properly captured in the plant's overall risk profile and that the implications for decision making are well understood.

*Objective*

~~7.84~~7.85 The objectives of the review of the PSA are to determine:

(a) The extent to which the existing PSA study remains valid as a representative model of the nuclear power plant;
(b) Whether the results of the PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operating modes;
(c) Whether the scope (which should include all operating modes and identified internal and external hazards), methodologies and extent (i.e. Level 1, 2 or 3) of the PSA are in accordance with current safety standards and operating practices;
(d) Whether the existing scope and application of PSA are sufficient.
(e) Whether multiple unit aspects, if relevant, are addressed in the risk profile.
(f) Whether the results of the PSA have been used to support the demonstration that plant states that could lead to high radiation doses or to a large radioactive release have been 'practically eliminated', and that there would be no, or only minor, potential radiological consequences for plant states with a significant likelihood of occurrence.

*Scope and tasks*

~~7.85~~7.86 The review of the PSA should include the following aspects:

(a) The existing PSA, including the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case;
(b) Whether accident management programmes for accident conditions (design basis accidents and design extension conditions) are consistent with PSA models and results;
(c) Whether the scope and applications of the PSA are sufficient;
(d) The status and validation of analytical methods and computer codes used in the PSA;
(e) Whether the results of PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operating modes, and meet relevant probabilistic safety criteria;
(f) Whether uncertainty and sensitivity analysis have been performed and taken into account in the results of the PSA and the conclusions drawn.
(g) Whether the existing scope and application of the PSA are sufficient to assist in the PSR global assessment, for example, to compare proposed improvement options.

~~7.86~~7.87 The review of safety factor 6 should consider at a minimum Requirement 15 of GSR Part 4 (Rev. 1) [3], Requirements 5, 10, 13, 16, 19, 20 and 42 of SSR-2/1 (Rev. 1) [7]~~[7]~~~~[5]~~ and Requirement 3 of GSR Part 6 [15]~~[15]~~~~[12]~~.

~~7.87~~7.88 Recommendations on PSA are provided in IAEA Safety Standards Series Nos SSG-3 (Rev. 1), Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [17]~~[17]~~~~[14]~~, and SSG-4 (Rev. 1), Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [18]~~[18]~~~~[15]~~.

*Methodology*

~~7.88~~7.89 The PSA should be reviewed to confirm that the modelling reflects the current design and operating features, takes account of all relevant operating experience, includes all modes of operation and, where relevant, has a scope agreed with the regulatory body.

~~7.89~~7.90 The PSA should be reviewed for completeness against an appropriate set of postulated initiating events, considering all foreseeable failures of SSCs of the plant, as well as operating errors and possible failures arising from internal and external hazards, whether in full power, low power or shutdown states.

~~7.90~~7.91 The extent to which hazards are represented in the PSA should be reviewed to verify that omissions are based on site specific justifications and that these omissions do not weaken the overall risk assessment for the plant.

~~7.91~~7.92 It should be verified that probabilistic hazard curves representing external events as an input to the probabilistic safety assessment for external hazards have been developed with reference to the specific site.

~~7.92~~7.93 The analytical methods and computer codes used in the PSA should be reviewed to verify that the methods used, and validation standards adopted continue to be appropriate.

~~7.93~~7.94 If it is necessary to repeat parts of the PSA, consideration should be given to using current PSA methodology (analytical methods and computer codes). If the earlier approach is still used, its continuing validity should be verified explicitly in the review, including the assumptions used, the degree of conservatism applied and inherent uncertainties in the analysis.

~~7.94~~7.95 The extent to which the potential for unidentified cross-links and the effects of common cause events are taken into account in the model should be reviewed, as these are often not adequately considered in plants of earlier design.

~~7.95~~7.96 The human reliability analysis performed in the PSA should be reviewed to ensure that the actions are modelled on a plant specific and scenario dependent basis, and that current methods are applied.

~~7.96~~7.97 It should be confirmed that an assessment has been made of the potential for human errors to worsen an event sequence through erroneous operation of equipment or incorrect diagnosis of the necessary recovery process.

~~7.97~~7.98 For sites with multiple units that would share resources (whether human resources or material resources) in accident conditions, it should be verified that the PSA for each unit has taken into account the availability of resources if hazards were to affect several units on the site.

~~7.98~~7.99 For sites with multiple units, it should be confirmed that all risk significant multiple unit initiating events[6] and hazards, as well as all plant operating modes are addressed, and that relevant risk metrics for multiple unit PSA are defined to capture different combinations between the reactor cores and spent fuel pools on the site, to facilitate the use of the results of the multiple unit PSA for decision making.

~~7.99~~7.100 The results of the PSA should be compared with relevant probabilistic safety criteria (e.g. for system reliability, core damage and releases of radioactive material) defined for the plant or set by the regulatory body.

~~7.100~~7.101 The history of updates to the PSA to reflect changes in plant status should be reviewed. Ideally, a living PSA should be maintained; where this is not practicable, the PSA should be kept sufficiently up to date throughout the lifetime of the plant to make it useful for safety related decision making.

~~7.101~~7.102 The results of the PSA should be reviewed to confirm that they have been used to determine appropriate safety related improvements to the design and operation of the facility and to determine priorities for modifications.

[6] A multiple unit initiating event is an initiating event that challenges normal operation of two or more units (or a degraded condition that eventually leads to a trip or challenge to normal operation) and that necessitates successful mitigation to prevent core damage of affected units, or that can directly lead to core and/or fuel damage.

**Safety factor 7: Hazard analysis**

~~7.102~~7.103 To ensure the fulfilment of required safety functions and operator actions, SSCs important to safety, including the control room and the emergency control centre, are required to be adequately protected against relevant internal and external hazards (see para. 5.15A of SSR-2/1 (Rev. 1) [7]~~[7]~~~~[5]~~.

*Objective*

~~7.103~~7.104 The objective of the review of hazard analysis is to determine the adequacy of protection of the nuclear power plant against internal and external hazards, including their potential to change over time, with account taken of the plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of the period covered by the PSR, current analytical methods, safety standards and knowledge.

*Scope*

~~7.104~~7.105 For each internal or external hazard identified, the review should evaluate the adequacy of the protection, with account taken of the following:

(a) The severity and associated frequency of occurrence of the hazard and potential hazard combinations, including considerations of their potential to change over time, as appropriate;
(b) Current safety standards and operating practices;
(c) Current understanding of environmental effects;
(d) The capability of the plant to withstand the hazard as claimed in the safety case, based on its current condition and with allowance given to predicted ageing degradation and the potential for external natural hazards to change over time;
(e) The appropriateness of procedures to cover operator actions claimed to prevent or mitigate the hazard.

~~7.105~~7.106 If it has not been previously done, a list of relevant internal and external hazards that might affect safety over the lifetime of the plant should be established. Where such a list has already been established, this should be reviewed for completeness.

~~7.106~~7.107 The following representative internal hazards that might affect plant safety should be reviewed (additional site specific internal hazards should be included under this safety factor if appropriate):

(a) Fire (including measures for prevention, detection and suppression of fire);
(b) Flooding;
(c) Pipe whip;
(d) Missiles and drops of heavy loads;
(e) Steam release;
(f) Hot gas release;
(g) Cold gas release;
(h) Deluge and spray;
(i) Explosion;
(j) Electromagnetic or radio frequency interference;
(k) Toxic and/or corrosive liquids and gases;

(l) Vibration;
(m) Subsidence;
(n) High humidity;
(o) Structural collapse;
(p) Loss of internal and external services (e.g. cooling water, electricity);
(q) High voltage transients;
(r) Loss or low capacity of air conditioning (i.e. which may lead to high temperatures).

~~7.107~~7.108 The following representative external hazards that might affect plant safety should be reviewed (additional site specific internal hazards should be included under this safety factor if appropriate):

(a) Floods, including tsunamis and storm surge;
(b) High winds, including tornadoes and tropical cyclones;
(c) Fire;
(d) Meteorological hazards (extreme temperatures, extreme weather conditions, high humidity, drought, snow, buildup of ice);
(e) Solar storm;
(f) Sand and dust storms;
(g) Toxic and/or corrosive liquids and gases, other contamination in the air intake (for example, industrial contaminants, volcanic ash);
(h) Hydrogeological and hydrological hazards (extreme groundwater levels, seiches);
~~(h)~~(i) Hazards from floating objects and hazardous liquid on water intakes and components of the ultimate heat sink;
~~(i)~~(j) Seismic hazards;
~~(j)~~(k) Volcanic hazards;
~~(k)~~(l) Aircraft crashes, external missiles;
~~(l)~~(m) Explosion;
~~(m)~~(n) Biological fouling;
~~(n)~~(o) Lightning strike;
~~(o)~~(p) Electromagnetic or radio frequency interference;
~~(p)~~(q) Vibration;
~~(q)~~(r) Traffic;
~~(r)~~(s) Loss of internal and external services (cooling water, electricity, etc.).

~~7.108~~7.109 The review of safety factor 7 should consider at a minimum Requirements 8, 10, and 13 of GSR Part 4 (Rev. 1) [3], all Requirements of IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [19]~~[19]~~~~[16]~~, and Requirement 17 of SSR-2/1 (Rev. 1) [7]~~[7]~~~~[5]~~.

*Methodology*

~~7.109~~7.110 For each relevant hazard, and their credible combinations, the review should verify, by means of current analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate.

~~7.110~~7.111 The analytical methods, safety standards and information used for the hazard analysis should be up to date and valid. Both aleatory uncertainties and epistemic uncertainties are required to have been considered in the ~~analysis~~ establishment of site specific design parameters (see para. 4.21 of SSR-1 [19]~~[19]~~~~[16]~~). If this is not the case, the analysis should be repeated or revised as necessary. The analysis and/or methods should take account of the plant design, site characteristics, the condition of SSCs important to safety (both at present and predicted for the end of the period covered by the PSR) and relevant international practice. Amongst other things, changes in plant design, the prevailing climate, the potential for floods and earthquakes, and transport and/or industrial activities near the site should be considered.

~~7.111~~7.112 In considering the risk of a particular hazard, consideration should be given to experience of hazards and operating practices at nuclear power plants and at other facilities, both in the State and in other States.

~~7.112~~7.113 Knowledge gained from actual events, in particular those that have occurred at nuclear power plants, should be identified. Any experience from managing such events (for example, external floods, seismic events and tornadoes) should be used to improve existing procedures at the plant.

~~7.113~~7.114 Climate change should be considered for all relevant external natural hazards, in particular when re-assessing the severity and associated frequency of occurrence of such hazards.

~~7.114~~7.115 External natural hazards should be grouped in accordance with their sensitivity to climate change and the related state of knowledge. The grouping of these hazards should be re-assessed and confirmed in the preparation phase of the PSR based on the feedback from scientific monitoring.

~~7.115~~7.116 The effects of climate change (at a minimum, up to the next PSR) should be factored into the studies of the external natural hazards classified as sensitive to climate change. Application, where relevant, of specific margins, typically derived from statistical extrapolation methods that take into account the current trend of climate change and predict it over a relevant period of time (typically 20–30 years) should be considered. Where appropriate, international data, in particular from the Intergovernmental Panel for Climate Change (IPCC), should be used as relevant input.

~~7.116~~7.117 The adequacy of the procedures used to prevent a hazard or to mitigate its consequences should be reviewed, including the extent to which these are tested and rehearsed. The adequacy of the preventive and mitigatory measures can be evaluated by deterministic safety analysis (safety factor 5) or PSA (safety factor 6).

~~7.117~~7.118 Recommendations on evaluation of and protection against internal and external hazards in the design and operation of nuclear power plants are provided in Refs [20]~~[20]~~~~[17]~~-[28]~~[28]~~~~[25]~~.

## SAFETY FACTORS RELATING TO PERFORMANCE AND FEEDBACK OF EXPERIENCE

### Safety factor 8: Safety performance

~~7.118~~7.119 Safety performance is determined from assessment of continuous monitoring of the safety of the plant, assessment of operating experience, including safety related events, and records of the unavailability of safety systems, occupational radiation doses and the generation and management of radioactive waste.

*Objective*

~~7.119~~7.120 The objective of the review of safety performance is to determine whether the plant's safety performance, including radiation doses and the generation and management of radioactive waste indicate any need for safety improvements.

*Scope*

~~7.120~~7.121 The review of safety performance should evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including:

(a) Analysis and control of safety related activities including radiation protection;
(b) Continuous monitoring and periodic review of the safety of the plant and of the performance of the operating organization;
(c) Unavailability of safety systems;
(d) Safety related incidents, low level events and near misses;
(e) Safety related operational data;
(f) Maintenance, inspection and testing;
(g) Reactivity management, including core management and fuel handling;
(h) Radiation monitoring, including assessment of occupational exposure and workers' health surveillance;
(i) Generation and management of radioactive waste, including characterization and classification and processing of radioactive waste;
(j) Storage of radioactive waste, including arrangements for subsequent disposal;
(k) Monitoring for verification of compliance with regulatory requirements.

~~7.121~~7.122 The review of safety performance is closely linked to the review of feedback of operating experience (safety factor 9), but the review of safety performance should be restricted to events at the plant under review.

~~7.122~~7.123 The review should consider the adequacy and effectiveness of safety performance indicators, applying trend analysis and comparing performance levels with those for other plants in the State or in other States, where practicable.

~~7.123~~7.124 Records of radiation doses should be reviewed to determine whether these are within prescribed limits, as low as reasonably achievable and adequately managed. Although radiation risks will need to be considered in all safety factors, the review of this safety factor should specifically examine data on radiation doses and the effectiveness of the radiation protection measures in place. The review should take into account the types of activity being undertaken at the plant, which might not be directly comparable with those at other nuclear power plants in the State or in other States.

~~7.124~~7.125 Data on the generation of radioactive waste should be reviewed to determine whether operation of the plant is being optimized to minimize the quantities and radioactivity levels of waste being generated and accumulated, taking into account the national policy on radioactive discharges and radwaste management and international treaties, standards and criteria.

~~7.125~~7.126 The review of the safety factor 8 should consider at a minimum Requirements 8, 9, 15, 20, 21, 29, 30 and 32 of SSR-2/2 (Rev. 1) [2], Requirements 11, 12, 14, 20, 24, 25 and 28 of IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [29]~~[29][26]~~, and Requirements 6 and 8–11 of IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [30]~~[30][27]~~.

*Methodology*

~~7.126~~7.127 Where available, the review should utilize a set of safety performance indicators, which should cover in a systematic manner all aspects of operation important to safety. These indicators should provide information on both positive and negative aspects of safety performance. The sets of safety performance indicators developed by the IAEA, by certain Member States and by WANO could be used for this purpose. Reference [31]~~[31][28]~~[31]~~[31][28]~~ provides practical guidance on the use of safety indicators. Recommendations on the use of such indicators for verifying compliance with the requirements for safe plant operation established in SSR-2/2 (Rev. 1) [2] are provided in IAEA Safety Standards Series Nos SSG-74, Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants [32]~~[32][29]~~, and SSG-70, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants [33]~~[33][30]~~.

~~7.127~~7.128 The review should also examine any other records of operating experience from the review period that are relevant to safety but have not been considered on the basis of the plant's safety performance indicators.

~~7.128~~7.129 The review of safety performance should evaluate the adequacy of the plant's safety performance methodologies and processes with regard to:

(a) Safety related incidents, low level events and near misses;
(b) Methods for the selection and recording of safety related operating data, including data on maintenance, testing, surveillance and inspection;
(c) Methods and results of reactivity management, including core management and fuel handling;
(d) Records of on-site radiation monitoring, including radiation doses and contamination levels;
(e) Records of radioactive waste;
(f) Management of radioactive waste , including generation, characterization and classification, processing and ~~accumulation~~storage;
(g) Compliance with regulatory requirements.

~~7.129~~7.130 The analysis of trends over the operating lifetime of the plant (or since the last PSR) should be reviewed to identify potential future safety concerns (for example, precursors to accidents) or deteriorating safety performance. Where relevant, the results of the previous PSR should be examined to detect any long term trends in deteriorating safety performance.

~~7.130~~7.131 Consideration should be given to the effects of any changes in operation at the plant (e.g. the use of a new design of fuel) on safety performance. In particular, the review should evaluate the continuing relevance of the current safety indicators and other safety performance methods in the context of current and future operations, and ensure that only relevant data and records are used.

~~7.131~~7.132 Requirement 20 and paras 5.10–5.16 of SSR-2/1 (Rev. 1) [2] establish the requirements for a radiation protection programme in a nuclear power plant, including requirements on the assessment of occupational exposure. Requirement 21 and paras 5.17–5.20 of SSR-2/1 (Rev. 1) [2] establish requirements for the management of radioactive waste ~~and effluents~~ arising from the operation of a nuclear power plant. SSG-70 [33]~~[33][30]~~ provides relevant recommendations ~~and further guidance with regard to small modular reactors is provided in Ref. [31]~~for operational limits and conditions. These publications should be considered when reviewing records relating to radiation doses and the generation of radioactive waste.

~~7.132~~7.133 The use of safety performance indicators also enables comparisons to be made with other nuclear power plants and provides an opportunity for operating organizations to benefit from each other’s experience. The extent to which this is being undertaken should be examined in the review.

~~7.133~~7.134 Where the review indicates a weak performance or trend, the possible root causes (e.g. deficiencies in procedures, training or safety culture) should be identified.

~~7.134~~7.135 For the purpose of providing data for other safety factors and for consideration in the global assessment, the results of the routine evaluations should be summarized (e.g. using indicators or trends) to provide an overall assessment of the safety performance for each year of the plant’s operation over the review period. Trends should be reported and, where necessary, further analysis should be undertaken to highlight any potential safety issues.

~~7.135~~7.136 Recommendations on the structure of the operating organization for nuclear power plants, the conduct of operations, core management and fuel handling, occupational radiation protection and predisposal management of radioactive waste are provided in Refs [34]~~[34][32]~~–[38]~~[38][35]~~, respectively.

**Safety factor 9: Feedback of operating experience**

~~7.136~~7.137 Experience from other nuclear power plants, and sometimes from non-nuclear facilities, together with research findings, can support identification of previously unknown safety issues or contribute to solving existing ones. Operating organization should obtain and evaluate information on operating experience at other facilities, including those for which the operating organization is also responsible, and should derive lessons for its own operations. Wider experience in the State and in other States, including relevant information from non-nuclear facilities, are required to be included in the feedback of operating experience (see Requirement 24 of SSR-2/2 (Rev. 1) [2]).

*Objective*

~~7.137~~7.138 The objective of the review of this safety factor is to determine whether adequate processes are in place to establish, implement, assess and continuously improve the operating experience programme at the plant to prevent or minimize the risk of future events by learning from research findings and from events that have already occurred at the plant or elsewhere~~.~~

*Scope*

~~7.138~~7.139 The review should include examination of the following aspects of the operating experience programme;

(a) Identification and reporting of internal operating experience;
(b) Collection of operating experience from other facilities;
(c) Screening of operating experience, including immediate review of events of specific interest;
(d) Investigation and in-depth analysis of relevant operating experience;
(e) Identification and analysis of trends for timely recognition of developing issues;
(f) Management of corrective actions resulting from investigation and analysis of operating experience, including approval, implementation, tracking and evaluation of their effectiveness;
(g) Usage, dissemination and exchange of operating experience, including through national and international reporting systems;
(h) Monitoring, assessment, and use of the results of safety research and technical development;
(i) The effectiveness of the operating experience programme;
(j) Maintenance of a storage, retrieval and documentation system for operating experience.

~~7.139~~7.140 The review of the safety factor 9 should consider at a minimum Requirement 24 of SSR-2/2 (Rev. 1) [2], Requirement 16 of GSR Part 3 [29]~~[29][26]~~, and Requirement 19 of GSR Part 4 [3].

*Methodology*

~~7.140~~7.141 The review of the operating experience programme should verify whether:

(a) operating experience is reported in a timely manner to reduce the potential for recurring events in-house and in the industry;
(b) Internal and external sources of operational experience feedback are considered in the programme to improve safety and reliability from lessons learned;
(c) Operating experience information is appropriately screened to select and prioritize those items that need further investigation;
(d) Analysis is performed for appropriate events, depending on their severity or frequency, to ensure root causes and corrective actions are identified;
(e) Corrective actions are defined, prioritized, scheduled and followed up to ensure effective implementation and effective improvement of safety and reliability;
(f) operating experience information is analysed for trends, and results are used throughout the plant to effectively improve safety and reliability;
(g) Assessments and indicators are effectively used to review and monitor plant's performance and the effectiveness of the programme;
(h) Opportunities for improvements in technical and organisational safety, identified from operating experience, safety research and technical developments are assessed and implemented to an appropriate extent;
(i) A system for the storage, retrieval and searching of operating experience is established

and maintained.

~~7.141~~7.142 Arrangements have been established for the dissemination of operating experience at nuclear power plants by the IAEA (e.g. International Reporting System for Operating Experience), and by organizations, such as the Nuclear Energy Agency of the OECD (OECD/NEA), WANO, and various plant owners' groups. The operating organization should have a process for receiving, analysing and acting upon such operating experience. The PSR should provide a summary of the findings from this process and should evaluate the effectiveness of the process.

~~7.142~~7.143 The review of an operating experience programme is a cross functional process. Therefore, any input from the review of other safety factors is beneficial to support the review of the programme.

~~7.143~~7.144 Arrangements for the dissemination of research findings might not be as well established as those from operating experience. The PSR should therefore pay particular attention to the adequacy of these arrangements and the timely implementation of research findings.

~~7.144~~7.145 Recommendations on operating experience feedback are provided in IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [39]~~[39]~~~~[36]~~.

SAFETY FACTORS RELATING TO MANAGEMENT

**Safety factor 10: Organization, the management system and safety culture**

~~7.145~~7.146 The operating organization is required to have in place a management system that ensures that the plant is operated in a safe, efficient and effective manner (see para. 3.4 of SSR-2/2 (Rev. 1) [2]). Similarly, the organization is required to promote a strong culture for safety (see para. 4.1 of SSR-2/2 (Rev. 1) [2]), and should ensure that individuals correctly perform duties that are important to safety, with alertness, due thought, full knowledge, sound judgement and a proper sense of accountability.

~~7.146~~7.147 The review of safety factor 10 should provide results of the review ~~of the~~ of the leadership and management for safety, and culture for safety at operating organization, in accordance with the requirements of GSR Part 2 [9]~~[9]~~~~[7]~~.

*Objective*

~~7.147~~7.148 The objective of the review of this safety factor is to determine whether the organization, leadership and management system and culture for safety are adequate and effective for ensuring the safe operation of the nuclear power plant.

*Scope*

~~7.148~~7.149 The review of this safety factor should include a review of the following aspects:

(a) Leadership for safety.
(b) Management system including:

(i) Responsibilities, policies, goals and objectives;
(ii) The involvement of interested parties;
(iii) Organizational structures, processes and interfaces;
(iv) The application of a graded approach;
(v) Documentation and records;
(vi) Resources, competencies, and staffing including contractors;
(vii) Management of radiation protection;
(viii) Management of radioactive waste;
~~(viii)~~
(ix) Management of safety assessment;
~~(ix)~~(x) Management of modifications and control of plant configuration;
(xi) Management for preparedness and response to a nuclear or radiological emergency~~.~~;
~~(x)~~(xii) Preliminary decommissioning plan.

(c) Culture for safety.

~~7.149~~7.150 The review of safety factor 10 should consider at a minimum Requirements 1–14 of GSR Part 2 [9]~~[9][7]~~ Requirements 1–5, 7–~~9~~11, 15,18, 21, 23 and 24 of SSR-2/2 (Rev. 1) [2] , Requirements 3 and 5 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~, Requirements 3, 11, 22 and 24 of GSR Part 4 (Rev. 1) [3], Requirement 4 of GSR Part 5 [30]~~[30][27]~~, and Requirement 26 of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [40]~~[40][37]~~.

*Methodology*

~~7.150~~7.151 The review of the leadership and management for safety should verify whether:

(a) Managers demonstrate leadership for safety and commitment to safety, and are trained, coached and assessed to improve leadership skills.
(b) Managers at all levels are involved in field activities, and they assess and discuss conduct of work and compliance with management expectations and objectives.
(c) Managers encourage an open reporting culture and a readiness to challenge acts or conditions that are adverse to safety.
(d) Managers lead by example and demonstrate a motivation to improve plant safety performance and achieve the established safety goals and objectives.
(e) Managers and supervisors are held accountable in relation to safety and for the achievement of assigned objectives.
(f) All elements of management, including safety, health, radiation and environmental protection~~ environmental~~, including preliminary decommissioning plan, quality, social and economic elements, are integrated in the management system and it is ensured that safety is not compromised.
(g) Processes and activities are ~~developed~~defined, effectively managed, documented and kept up-to-date to ensure that requirements are met without compromising safety.
(h) Managers demonstrate commitment to and responsibility for the establishment, implementation, assessment and continuous improvement of the management system.
(i) The risk assessment is integrated in the management system and changes that could have significant implications for safety are appropriately analysed.

(j) The effectiveness of the management system is monitored and measured to confirm expected results are attained and of areas for improvement are identified.
(k) The prime responsibility for safety is assigned and this responsibility for safety is discharged.
(l) The goals, strategies, plans and objectives of the organization are consistent with organization's safety policy and are periodically reviewed.
(m) Interested parties are identified and a strategy for interaction with them is established.
(n) Liaison with the regulatory body is established.
(o) The organizational structure, responsibilities and accountabilities are defined and documented in the management system.
(p) Proposed changes to the management system, including organizational changes are assessed for their impact to safety, including cumulative impact of changes.
(q) The scope of staff services provided from outside the operating organization are defined in the management system. A clear division of responsibilities and authority between all parts of the operating organization and relevant outside organizations are defined.
(r) A graded approach is applied in the management system.
(s) The process for issuance, validation, approval, dissemination, review and periodic updating of documentation, records and reports is established.
(t) Provisions for adequate resources and funding, including for the long term management and disposal of radioactive waste, as well as for decommissioning of plant and during an emergency response are in place.
(u) The necessary knowledge, skills, attitudes and safety expertise are sustained at the plant, and long term objectives for staffing including succession planning are developed and met.
(v) The management system includes arrangements for the supply of items, products and services.
(w) The management system includes arrangements for qualification, selection, evaluation, procurement, and oversight of the supply chain.
(x) The management system includes arrangements for radiation protection and the management of radioactive waste, as well as it enables the planning of decommissioning throughout the lifetime of the plant.
(y) Arrangements are in place to manage plant design modifications to ensure that all modifications are properly identified, specified, screened, designed, evaluated, authorized, implemented and recorded.
~~(x)~~(z) Arrangements are in place for maintaining the configuration of the nuclear power plant and operations are justified by the safety analysis of the plant.
~~(y)~~(aa) Arrangements for safety assessment are in place.~~;~~
~~(z)~~(bb) Arrangements for preparedness and response for a nuclear or radiological emergency are in place.

~~7.151~~7.152 The review of safety culture is an assessment of commitment to safety and should include the following:

(a) A review of the safety policy to verify that it states that safety takes precedence over production and to confirm that this policy is effectively implemented;
(b) Verification that all individuals in the organization contribute to fostering and sustaining a strong safety culture, including verification that managers demonstrate and support attitudes and behaviours that result in a sustainably strong safety culture;

(c) ~~A review~~ Verification of how the organization ensures that all individuals accept their personal accountability for their attitudes and conduct with regard to safety;

(d) Verification that the organization ensures that its managers and workforce understand and appropriately discharge their responsibility for safety and a review how a common understanding of safety and safety culture, including awareness of radiation risks and hazards related to work, is being developed;

(e) Verification that the organization promotes and assures procedure adherence and a review of how it is assured that individuals understand the standards that they are expected to apply in completing their tasks and follow relevant processes, procedures and work instructions;

(f) Verifications that the organization promotes and assures a culture that supports trust and collaboration as well as reporting of problems relating to technical, human and organizational factors;

(g) ~~A review~~Verification of how plant staff are encouraged to acknowledge errors and seek help when needed and to challenge potentially unsafe practices and identify and report deficiencies and correct unsafe behaviours, wherever and whenever they encounter them;

(h) Verification that the organization promotes and assures a culture that supports questioning and learning attitude and review what the main mechanisms, tools or resources assuring continuous learning are;

(i) Verification that the organization promotes and assures conservative and safety oriented decision making;

(j) A review how safety culture is assessed and how ~~are~~ assessments are analysed, communicated to staff and acted upon.

~~7.152~~7.153 An assessment of safety culture could include interviews of personnel at all levels of the operating organization and personnel providing support services.

~~7.153~~7.154 Recommendations on the management system and the operating organization for nuclear power plants are provided in GS-G-3.1 [10]~~[10][8]~~, SSG-72 [34]~~[34][32] and~~, IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [41]~~[41][38]~~ and IAEA Safety Standards Series No. SSG-71, Modifications to Nuclear Power Plants .

### Safety factor 11: Operational limits and conditions and operating procedures

~~7.154~~7.155 Procedures important to the safety of the nuclear power plant should be comprehensive, validated, formally approved, appropriately distributed and subject to rigorous management control. In addition, the procedures should be unambiguous and relevant to the actual plant (with modifications taken into account); they should reflect current operating practices, and due consideration should be given to human factor aspects (for example, whether they are user friendly).

*Objective*

~~7.155~~7.156 The objective of the review of operational limits and conditions (OLCs) and operating procedures is to determine whether the operating organization's activities for managing, implementing and adhering to operating procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.

*Scope*

~~7.156~~7.157 The review should examine the following documentation:

(a) Operational limits and conditions;
(b) Operating procedures and guidelines for operational states and accident conditions covering all modes of operation, including low power and shutdown, and all fuel locations on the site;
(c) Maintenance, testing, surveillance and inspection procedures;
(d) Procedures for issuing work permits;
(e) Procedures for fuel handling and storage, including reactivity management;
(f) Procedures for radiation protection, radioactive waste management, and for the monitoring and control of discharges of radioactive effluents.

~~7.157~~7.158 The extent of the review should consider, at a minimum Requirements 6, 19, 20, 21, 26, 30 and 31 of SSR-2/2 (Rev. 1) [3], Requirement 28 of SSR-2/1 (Rev. 1) [2], Requirements 15 and 24 of GSR Part 3 [29]~~[29][26]~~, Requirements 23 and 24 of GSR Part 4 [3], and Requirements 4 and 12 of GSR Part 5 [30]~~[30][27]~~.

*Methodology*

~~7.158~~7.159 The review of this safety factor should verify that:

(a) The operational limits and conditions include safety limits, limiting settings for safety systems, limits and conditions for normal operation, requirements for surveillance and testing, specified operational configurations and action statements for deviations from normal operation;
(b) Procedures and reference materials are clearly identified and are readily accessible in the control room and other operating locations;
(c) Strict adherence to written operating procedures is an essential part of safety policy;
(d) Procedures are developed for normal operation to ensure the plant is operated within the operational limits and conditions;
(e) Procedures are developed, verified and validated for use in the event of anticipated operational occurrences, and design basis accidents;
(f) Guidelines or procedures are developed, verified and validated for the management of design extension conditions;
(g) Instruction for the utilization of available equipment exist in the accident management programme and adverse working conditions are taken into account;
(h) The analysis used in support of development of operating procedures and guidelines is justified and documented;
(i) Operating procedures and supporting documentation are issued under controlled conditions and are subject to approval and periodic review and revision;
(j) Procedures are updated in a timely manner in the light of operating experience and the plant configuration.

~~7.159~~7.160 The review of this safety factor should focus on those procedures that have the highest safety significance and need not necessarily include a full review of every procedure. The safety significance of procedures can be determined from deterministic safety analysis and/or PSA. For procedures assigned lower safety significance, a sampling approach could be followed to review the overall adequacy of procedures (and the management processes used to develop and control them).

~~7.160~~7.161 Recommendations on operational limits and conditions, operating procedures and accident management guidelines are provided in SSG-70 [33]~~[33][30]~~, SSG-72 [34]~~[34][32]~~, SSG-74 [32]~~[32][29]~~, and IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [43]~~[43][39]~~.

### Safety factor 12: Human factors

~~7.161~~7.162 Human factors influence all aspects of the safety of a nuclear power plant. The review should examine the human factors at the plant and within the operating organization to determine whether these correspond to accepted good practices and to verify that they do not present an unacceptable contribution to risk. In particular, the review should determine whether operator actions claimed to be in support of safety are feasible and properly supported.

*Objective*

~~7.162~~7.163 The objective of the review of this safety factor is to evaluate the various human factors that might affect the safe operation of the nuclear power plant and to seek to identify reasonably practicable safety improvements.

*Scope*

~~7.163~~7.164 The review of human factors should consider the procedures and processes and safety measures in place at the nuclear power plant to ensure the following:

(a) Adequate staffing levels exist for operating the plant, with due recognition given to absences, shift working and restrictions on overtime;
(b) Qualified staff are available on duty at all times;
(c) Adequate programmes are in place for initial training, refresher training and upgrading training, including the use of simulators;
(d) Operator actions needed for safe operation have been assessed to confirm that assumptions and claims made in safety analyses (for example, PSA, deterministic safety analysis and hazard analysis) are valid to allow for effective and safe execution of these actions (e.g. considering accessibility, environmental conditions, drills);
(e) Human factors in maintenance are assessed to promote error-free execution of work;
(f) Adequate competence requirements exist for operating, maintenance, technical and managerial staff;
(g) Staff selection methods (for example, testing for aptitudes, knowledge and skills) are systematic and validated;
(h) Appropriate fitness for duty guidelines exist relating to working hours, types and patterns of work, good health and substance abuse;
(i) Policies exist for maintaining the knowledge of staff and for ensuring adequate succession management in accordance with good practices;
(j) Adequate facilities and programmes are available for staff training.

~~7.164~~7.165 The following aspects of the human–machine interface should also be verified in the review:

(a) Sound ergonomic principles are followed in the design of equipment, in particular in the design of the control room and other workstations relevant to safety;
(b) Operating procedures take into account human information requirements and workloads, and the clarity and achievability of procedures.

~~7.165~~7.166 The extent of the review should consider at a minimum Requirement 11 of GSR Part 4 (Rev. 1) [3], Requirement 32 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~, Requirements 4, 7, 8 and 27 of SSR-2/2 (Rev. 1) [2], and Requirement 9 of GSR Part 2 [9]~~[9][7]~~.

*Methodology*

~~7.166~~7.167 The review should be conducted with the assistance of properly qualified specialists. Because of the difficulties associated with carrying out an objective review of what is essentially the performance of its own staff, the operating organization may decide that specific elements of the review should be conducted by external consultants.

~~7.167~~7.168 The review of the human–machine interface should examine the actual condition of the plant using, for example, plant walkdowns.

~~7.168~~7.169 If deficiencies in the procedures and processes or in the design of the human–machine interface represent a potentially significant contribution to risk, the PSR should make proposals for corrective actions to be considered in the global assessment. These may include improvements in procedures, enhanced training or redesign of human–machine interfaces.

~~7.169~~7.170 Recommendations on human factors are provided in IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [44]~~[44][40]~~ and further guidance can be found in Ref. [45]~~[45][41]~~.

**Safety factor 13: Emergency planning**

~~7.170~~7.171 The design and operation of a nuclear power plant is expected to prevent or otherwise minimize releases of radioactive substances that could give rise to risks to workers, the public or the environment. Emergency planning for the possibility of such releases is a prudent and necessary action, not only for the operating organization but also for local and national authorities.

*Objective*

~~7.171~~7.172 The objective of the review of emergency planning is to determine:

(a) Whether the operating organization has in place adequate plans, staff, facilities and equipment for dealing with emergencies;

(b) Whether the operating organization's arrangements have been adequately coordinated with the arrangements of local and national authorities and are regularly exercised.

*Scope*

~~7.172~~7.173 The PSR should include an overall review to check that emergency planning at the plant continues to be satisfactory and to check that emergency plans are maintained in accordance with current safety analyses, accident mitigation studies and good practices.

~~7.173~~7.174 The PSR should verify that the operating organization has given adequate consideration to: significant changes at the site of the nuclear power plant and in its use; organizational changes at the plant; changes in the maintenance and storage of emergency equipment; and developments around the site that could influence emergency planning.

~~7.174~~7.175 The review of emergency planning should:

(a) Evaluate roles and responsibilities in the emergency preparedness and response;
(b) Evaluate interfaces of emergency planning and response with security and integration with security plans;
(c) Evaluate the adequacy of the hazard assessment and the protection strategy;
~~(c)~~(d) Evaluate arrangements are in place to confirm that operations in response to a nuclear or radiological emergency can be effectively managed.
(e) Evaluate the capability of taking mitigatory actions and of managing radioactive waste;
~~(d)~~(f) Evaluate the capability of taking urgent protective actions and other response actions;
~~(e)~~(g) Evaluate the adequacy of on-site equipment and facilities for emergencies;
~~(f)~~(h) Evaluate the adequacy of on-site technical and operational support centres;
~~(g)~~(i) Evaluate the efficiency of communications in the event of an emergency, in particular the interaction with organizations outside the plant;
~~(h)~~(j) Evaluate the content and efficiency of emergency training and exercises and check records of experience from such exercises;
~~(i)~~(k) Evaluate arrangements for the regular review and updating of emergency plans and procedures;
~~(j)~~(l) Examine changes in the maintenance and storage of emergency equipment, including non-permanent equipment for accident management;
~~(k)~~(m) Evaluate arrangements in place to cope with beyond design basis external hazards, or pandemics;
~~(l)~~(n) For sites with multiple units, evaluate arrangements in place to cope with concurrent accidents affecting all units on the site;
~~(m)~~(o) Evaluate the effects of any recent residential and industrial developments around the site.

~~7.175~~7.176 The extent of the review should consider, at a minimum Requirements 2, 4–15, and 18–26 of GSR Part 7 [40]~~[40]~~~~[37]~~, Requirement 67 of SSR 2/1 (Rev. 1) [7]~~[7]~~~~[5]~~, and Requirements 17–19 of SSR-2/2 (Rev. 1) [2].

~~7.176~~7.177 Relevant recommendations on emergency preparedness for and response to a nuclear or radiological emergency are provided in IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [46]~~[46]~~~~[42]~~, SSG-76 [35]~~[35]~~~~[33]~~ and SSG-54 [43]~~[43]~~~~[39]~~, and further information is provided in Refs [47]~~[47]~~~~[43]~~, [48]~~[48]~~~~[44]~~.

*Methodology*

~~7.177~~7.178 The roles and responsibilities for preparedness and response to a nuclear emergency should be reviewed to confirm that they are clearly allocated among the operating organization, regulatory body and response organizations.

~~7.178~~7.179 Emergency plans should be reviewed to evaluate their interfaces with security and ~~that an adequate level of~~ integration with security plans ~~is achieved~~.

~~7.179~~7.180 The hazard assessment and protection strategy should be reviewed to evaluate the adequacy of the basis for emergency preparedness and response and that protective actions and other response actions can be taken effectively.

~~7.180~~7.181 The capability for taking mitigatory actions should be reviewed to evaluate whether the operating organization has adequate arrangements for implementing functions that are essential for an effective emergency response.

~~7.181~~7.182 Records of emergency exercises should be reviewed to evaluate the effectiveness and competence of the staff of the operating organization and of off-site (emergency) organizations, the functional capability of equipment (including communications equipment) and the adequacy of emergency planning.

~~7.182~~7.183 The operating organization's arrangements for interaction with relevant off-site organizations such as the police, fire departments, hospitals, ambulance services, regulatory bodies, local authorities, government, public welfare authorities and the news media should be evaluated.

~~7.183~~7.184 The review of the adequacy of on-site equipment and facilities for emergency response and off-site emergency response facilities or locations should include walkdowns of relevant areas on and off the site.

~~7.184~~7.185 The review of functional capabilities of emergency response should include the arrangements to safety and effectively manage radioactive waste generated during an emergency.

~~7.185~~7.186 The content and effectiveness of emergency training and exercises should be evaluated by reviewing the records of these exercises with respect to their frequency and results, and the actions taken in case of deficiencies. These can be compared with national and international guidelines and current safety standards and operating practices.

~~7.186~~7.187 For sites with multiple units, the review should include an evaluation of arrangements to cope with simultaneous accidents affecting all the units on site, with a focus on the availability of adequate resources, accident management means, staffing, and training programmes.

~~7.187~~7.188 Arrangements for regular reviews of emergency plans and procedures and their periodic updating can be evaluated as part of the review of the operating organization's management processes (safety factor 11).

## SAFETY FACTOR RELATING TO THE ENVIRONMENT

### Safety factor 14: Radiological impact on the environment

The operating organization is required to have a monitoring programme that provides data on the radiological impact of the nuclear power plant on its surroundings during normal operation and ensures verification of compliance with the requirements for radiation protection and safety to protect members of the public against exposure (see para. 5.20 of SSR-2/2 (Rev. 1) [2].

*Objective*

~~7.188~~7.189 The objective of the review of this safety factor is to determine whether the operating organization has appropriate monitoring equipment, an adequate and effective programme and methods for monitoring the radiological impact of the plant on the environment and for assessment of the environmental impact and public exposure during normal operation, which ensures that ~~effluents~~ discharges of effluents are properly controlled and that protection and safety in relation to public exposure is optimized.

*Scope*

~~7.189~~7.190 The arrangements for monitoring the radiological impact on the environment outside the site area in normal operation is the subject of the review.

7.191 The extent of the review should consider, at a minimum Requirements 12, 14, 30, 31, and 32 of GSR Part 3 [29]~~[29][26]~~, Requirements 5, 78 and 79 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~, and Requirement 21 of SSR-2/2 (Rev. 1) [2].

~~7.190~~7.192 Relevant recommendations on environmental impact assessment are provided in IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and the Environment [49], GSG-9, Regulatory Control of Radioactive Discharges to the Environment [50] and GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [51].

*Methodology*

~~7.191~~7.193 The review should establish whether the monitoring programme is appropriate and sufficiently comprehensive to ensure verification of compliance with the requirements for protection of the environment and the public. In particular, the review should verify that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

~~7.192~~7.194 In some States, monitoring programmes are also undertaken by public organizations. This can facilitate the independent validation of data provided by the operating organization. Examples of data collected by other organizations include data on the concentrations of radionuclides in air, water (including river water, sea water and groundwater), soil, agricultural and marine products and wild flora and fauna.

~~7.193~~7.195 As part of the review, it should be verified that:

(a) Discharges of effluents are being monitored and the results analysed for trends, and that appropriate actions are taken to keep such discharges as low as reasonably achievable within authorized limits;
(b) On-site monitoring is undertaken at locations and using methods that have a high probability of the prompt detection of a release of radioactive material to the environment;
(c) Off-site concentrations of radionuclides in air, water (including river water, sea water and groundwater, as appropriate), soil, agricultural and marine products and animals are being monitored and the results are analysed for trends, and that appropriate corrective actions are taken in the event that action levels are exceeded;
(d) Suitable monitoring equipment is provided and procedures for verification are implemented;

(e) Equipment used for monitoring is properly maintained, tested and calibrated at appropriate intervals with reference to standards traceable to national or international standards;

(f) The dispersion pathways in air and water by which radioactive releases from the nuclear power plant could potentially affect the public and the environment have been identified and evaluated considering specific regional and site characteristics, including the population distribution in the region with special attention paid to the transport and accumulation of radionuclides in the biosphere;

(g) Calculation of doses to the public and assessment of radiological environmental impacts based on effluents monitoring, and considering pathways of releases and related uncertainties, ~~during the lifetime of the plant~~ meet regulatory requirements and reflect international good practice during the lifetime of the plant;

(h) Assumptions and models used in assessment of public exposure and the assessment of radiological environmental impacts are adequate;

(i) Exposures to members of the public are restricted, so that neither authorized limits set by regulatory body nor the effective dose or the equivalent dose to tissues or organs specified in Schedule III of GSR Part 3 [29]~~[29][26]~~ are exceeded;

(j) Specific restrictions and procedures are followed to ensure that dose limits are not exceeded owing to possible combinations of doses from exposures due to different authorized practices at the plant's site;

(k) Records of the results of monitoring, estimated doses to members of the public and verification of compliance, are maintained as required by the regulatory body, including records of the tests and calibrations of equipment;

(l) The results of monitoring and verification of compliance are shared at approved intervals with the regulatory body and appropriate data have been published on the environmental impact of the plant;

(m) Any levels exceeding the operational limits and conditions relating to public exposure and radiological environmental impacts, including authorized limits on discharges or any significant increase in dose rate or concentrations of radionuclides in the environment are promptly reported to the regulatory body in accordance with reporting criteria;

(n) Possible changes in any conditions that could affect exposure of members of the public or radiological environmental impacts, such as plant modifications and the actual conditions of SSCs important to safety, identification of new sources of radiological impact, changes in environmental dispersion conditions, changes in exposure pathways and population distribution, or changes in values of parameters used for the determination of the representative or the most exposed person ~~has~~ have been considered in monitoring programmes, in the calculation of doses to the public and in the assessment of radiological environmental impacts.

~~7.194~~7.196 Radiological monitoring data should be compared with the values measured before the nuclear power plant was put into operation and/or historical values examined in the last PSR. In the event of significant deviations, an explanation should be provided by the operating organization, with account taken of relevant factors external to the nuclear power plant. The review should also verify that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

~~7.195~~7.197 Where environmental data have not been provided since the start of operation of the plant or since the last PSR, these data should be submitted to the regulatory body for information.

# 8. GLOBAL ASSESSMENT IN PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS

## METHODOLOGY FOR GLOBAL ASSESSMENT

8.1. The objective of the PSR global assessment is to arrive at a judgment of the nuclear power plant’s suitability for continued operation based on a balanced view of all findings from the reviews of the separate safety factors. This judgment should take into account positive and negative findings, results of the global assessment, and proposed corrective measures and safety improvements.

8.2. The global assessment should evaluate the impact on safety on the basis of the findings from all the separate safety factors and should be performed after completing all the individual safety factor reviews. The total effect of the negative findings, safety improvements, and positive findings (strengths) identified in the PSR should be examined to ensure that the overall level of plant safety is adequate.

8.3. The global assessment should also consider interface issues, overlaps, and omissions between safety factors to determine additional corrective measures and safety improvements arising from more than one safety factor review. Reviewing the implementation of the concept of ‘practical elimination’ of plant event sequences that could lead to an early radioactive release or a large radioactive release is an example of identifying a possible interface issue.

8.4. The global assessment should examine supporting information such as documents on the scope and methodology of the PSR, regulatory requirements, feedback from the regulatory body on previously submitted PSR documents, particular issues raised by the regulatory body, and additional reference material as appropriate.

8.5. The global assessment should be performed by an interdisciplinary team with appropriate expertise in operation, design, and safety at the plant, including an appropriate number of participants from the safety factor reviews. The team should also include members who are independent of the safety factor review teams.

8.6. In performing the global assessment, the findings from other relevant safety reviews of the plant should be incorporated as appropriate, for example, findings from long term operation studies if the PSR is performed to support long term operation.

### Assessment of ~~P~~practical elimination of plant event sequences that could lead to an early radioactive release or a large radioactive release

8.7. Paragraph 4.3 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~ states (footnote omitted):

> “The design shall be such as to ensure that plant states that could lead to high radiation doses or to a large radioactive release have been ‘practically eliminated’, and that there would be no, or only minor, potential radiological consequences for plant states with a significant likelihood of occurrence.”

8.8. Furthermore, para. 5.31 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~ in relation to design extension conditions states that (footnote omitted): “The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is ‘practically eliminated’.”

8.9. Evaluation of various aspects related to the practical elimination of plant event sequences that could lead to an early radioactive release or a large radioactive release is expected to be performed in different PSR safety factors (typically safety factors 1 and 5–7). Within the global assessment, the proof of implementation of this concept in practice should be checked by integrating the review results of relevant safety factors.

8.10. Relevant recommendations are provided in IAEA Safety Standards Series No.SSG-88, Design Extension Conditions and the Concept of Practical Elimination in the Design of Nuclear Power Plants [52]~~[52]~~~~[45]~~.

**Assessment of high level requirements**

8.11. An analysis of the various safety factor review results within the global assessment should be conducted using appropriate general, high level requirements consistent with the Fundamental Safety Principles [1] or with regulatory requirements.

8.12. The objective of the assessment is to determine whether those high level requirements are met by analysing and integrating the review results from individual safety factors.

8.13. An ultimate example of this analysis is an assessment of the implementation of defence in depth in line with Principle 8 of SF-1 [1] and Requirement 7 of SSR-2/1 (Rev. 1) [7]~~[7]~~~~[5]~~.

**Assessment of defence in depth**

8.14. Paragraph 3.31 of SF-1 [1] states:

> “The primary means of preventing and mitigating the consequences of accidents is ‘defence in depth’. Defence in depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. When properly implemented, defence in depth ensures that no single technical, human or organizational failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability. The independent effectiveness of the different levels of defence is a necessary element of defence in depth.”

8.15. The concept of defence in depth for the design of nuclear power plants is described in paras 2.12–2.14 of SSR-2/1 (Rev. 1) [7]~~[7]~~~~[5]~~. Paragraph 2.14 of SSR-2/1 (Rev. 1) [7]~~[7]~~~~[5]~~ states:

> “A relevant aspect of the implementation of defence in depth for a nuclear power plant is the provision in the design of a series of physical barriers, as well as a combination of active, passive and inherent safety features that contribute to the effectiveness of the physical barriers in confining radioactive material at specified locations. The number of barriers that will be necessary will depend upon the initial source term in terms of the amount and isotopic composition of radionuclides, the effectiveness of the

individual barriers, the possible internal and external hazards, and the potential consequences of failures."

8.16. Requirement 7 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~ states that "**The design of a nuclear power plant shall incorporate defence in depth. The levels of defence in depth shall be independent as far as is practicable.**" Paragraphs 4.9–4.13A of SSR-2/1 (Rev. 1) [7]~~[7][5]~~ develop this overarching requirement.

8.17. Paragraph 4.12 of GSR Part 4 (Rev. 1) [3] states:

> "It shall be determined in the safety assessment whether adequate defence in depth has been provided, as appropriate, through a combination of several layers of protection (i.e. physical barriers, systems to protect the barriers, and administrative procedures) that would have to fail or to be bypassed before there could be any consequences for people or the environment."

8.18. The PSR global assessment should review the extent to which the above safety requirements related to the implementation of defence in depth are fulfilled.

8.19. The assessment of the implementation of defence in depth should determine whether the necessary layers of protection, including physical barriers to confine radioactive material at specific locations, are in place, and whether supporting administrative controls for achieving defence in depth are implemented.

8.20. Paragraph 4.46 of GSR Part 4 (Rev. 1) [3] in relation to the identification of necessary layers of protection in the safety assessment of defence in depth states:

> "This shall include identification of:
> (a) Safety functions that have to be fulfilled;
> (b) Potential challenges to these safety functions;
> (c) Mechanisms that give rise to these challenges, and the necessary responses to them;
> (d) Provisions made to prevent these mechanisms from occurring;
> (e) Provisions made to identify or monitor deterioration caused by these mechanisms, if practicable;
> (f) Provisions for mitigating the consequences if the safety functions fail."

8.21. Reference [53]~~[53][46]~~ provides a comprehensive approach to the assessment of the implementation of defence in depth, covering all aspects including siting, design, manufacturing and construction, commissioning, operation, accident management, and emergency preparedness.

8.22. Other methods based on engineering judgment, deterministic or probabilistic methods, or their combinations as appropriate, could also be used for specific assessments of the application of defence in depth in plant design.

8.23. In order to obtain a complete picture of the plant's defence in depth, all identified gaps should be included in the PSR global assessment, including any findings related to long term operation.

**Assessment of combined effects**

8.24. Although negative findings may be individually acceptable, their combined effects should also be reviewed for acceptability. Global issues from the combined effects of individual findings should be identified in the global assessment and should be considered in the preparation of the integrated implementation plan. These global issues could indicate system deficiencies that have not been identified as a result of the review of individual safety factors; however, their occurrence emerges from a set of interrelated deviations.

8.25. It is possible that a weakness in one safety factor can be compensated for by a strength in another. For example, it may be acceptable (on a temporary or permanent basis) to use a strength in human factors (such as operator action supported by adequate procedures) to compensate for a weakness in design or equipment (such as a lack of automatic protection against a postulated slow type of reactor fault of very low probability).

## PRINCIPLES FOR RANKING RESULTS FROM PERIODIC SAFETY REVIEW, INCLUDING CATEGORIZATION OF DEVIATIONS

8.26. A method for determining the safety significance of negative findings (deviations), their ranking, and the prioritization of corrective measures and safety improvements should be established prior to performing the global assessment and, where required, agreed with the regulatory body.

8.27. The safety significance of all deviations identified in the safety factors review should be evaluated using engineering judgment, as well as deterministic and probabilistic methods, as appropriate, covering all relevant requirements. The safety significance should represent the level of risk associated with each deviation, thus providing a sound basis for an integrated, risk-informed approach to making decisions on safety matters.

8.28. While probabilistic methods can provide useful insights into relative contributions to risk, help judge priorities, and compare options, a decision making process that is solely based on numerical assessment of risk should not be adopted.

8.29. For deviations affecting fundamental safety functions, the safety significance determination should be based on the evaluation of the frequency and radiological consequences of event sequences where the deviation is the cause of the event sequence, or where it negatively affects the event sequence in question. Results of existing deterministic and probabilistic safety analyses should be used as sources of information for such evaluation.

8.30. For deviations with no straightforward link to fundamental safety functions, alternative approaches should be developed to determine the safety significance using general criteria for the deterioration of processes and activities and the resulting risk for safety. Such an alternative approach should provide balanced results with the approach based on the evaluation of safety functions.

8.31. The results of the assessment of defence in depth could be integrated into the safety significance determination using criteria for the robustness and independence of individual layers of defence in depth.

8.32. The safety significance of findings should be used as a basis for prioritizing corrective measures and safety improvements and should be considered in the justification for deviations for which no reasonably practicable safety improvements could be identified.

## IDENTIFICATION, DEVELOPMENT AND JUSTIFICATION OF SAFETY IMPROVEMENTS FOR THE NEXT PSR PERIOD

8.33. Paragraph 4.47 of SSR-2/2 (Rev. 1) [2] states:

> "On the basis of the results of the systematic safety assessment, the operating organization shall implement any necessary corrective actions and reasonably practicable modifications for compliance with applicable standards with the aim of enhancing the safety of the plant by further reducing the likelihood and the potential consequences of accidents."

8.34. The method for prioritizing corrective measures and safety improvements should be based on the safety significance of each deviation and the effectiveness of eliminating the deviation's effects.

8.35. In identifying corrective measures and safety improvements, the following aspects should be considered as appropriate in addition to the safety significance of deviations:

(a) Findings from the review of practical elimination of large or early releases;
(b) The results of the assessment of high level requirements;
(c) Any deterioration of defence in depth detected by the global assessment;
(d) The results of combined effects of findings;
(e) The ability of corrective measures to improve safety beyond the elimination of deviations;
(f) The time necessary for implementing corrective actions and/or safety improvements;
(g) Root and/or direct causes of deviations (if they have been determined);
(h) Cost-benefit analysis.

8.36. Cost-benefit analysis and/or risk analysis, or a combination thereof, could be used for the prioritization of corrective measures and proposed safety improvements. Consideration should be given to the actual benefit to safety that will be achieved and the duration of the benefit, considering the remaining planned lifetime of the plant.

8.37. For corrective actions and/or safety improvements that need a long time for their implementation, adequate interim measures should be implemented.

8.38. If a modification is necessary on the grounds of unacceptable risk, then relevant operations should be halted until the modification or adequate interim measures have been implemented and, where required, approved by the regulatory body.

8.39. Corrective measures and proposed safety improvements should be included in the integrated implementation plan with the implementation timing reflecting the results of the global assessment and, where required, agreed with the regulatory body.

8.40. Overall conclusions of the global assessment, including appropriate justification for continued operation and the integrated implementation plan, should be included in the final

PSR report (see Appendix II). Corrective measures and proposed safety improvements should be implemented in accordance with the agreed integrated implementation plan.

8.41. Justification for continued operation should address operations both in the short term prior to the implementation of identified corrective measures and safety improvements, and in the long term if the global assessment concludes that addressing some of the negative findings is not reasonably practicable.

## 9. PERIODIC SAFETY REVIEW FOR AND DURING LONG TERM OPERATION OF NUCLEAR POWER PLANTS

9.1 Continuation of operation of a nuclear power plant beyond the time frame originally anticipated for its operation has become a priority for many operating organizations. Long term operation of a nuclear power plant may be defined as operation beyond an established time frame defined, for example, by the licence term, the plant design, relevant standards, or regulatory requirements. Long term operation should be justified by safety assessment, with consideration given to the life limiting processes and features of SSCs important to safety (see SSG-48 ~~[5]~~[12]~~[12]~~~~[10]~~]).

9.2 Member States have different regulatory strategies for long term operation of nuclear power plants, resulting in differences in the regulatory approval framework and process. However, there are key technical aspects and subjects of the decision for long term operation and programmes that should be considered, irrespective of the pursued regulatory strategy, approval framework and process applied.

9.3 A PSR conducted during the originally anticipated timeframe for its operation should provide an overall view of actual plant safety and determine reasonably practicable safety improvements. These improvements should ensure that a high level of safety is maintained throughout the plant's anticipated operational lifetime. This often calls for assessing life-limiting features of the plant, considering the current knowledge and operating experience, including ageing-related issues. The goal should be to identify, and possibly anticipate, any need to modify, refurbish, or replace certain SSCs to ensure safety during the entire originally anticipated operating lifetime of the nuclear power plant.

9.4 In general, PSR can be used to support the justification of long term operation, provided that specific aspects of long term operation are considered. Additionally, if long term operation is approved by the regulatory body, periodic safety reviews are generally still necessary and should continue to be performed throughout the entire period of extended operation. These reviews should be conducted on a ten-year cycle or at a frequency specified by the regulatory body.

9.5 It is recognized that some States employ alternative arrangements to PSR, which may be equally adequate for justifying extension of the lifetime of a nuclear power plant.

9.6 When the PSR is used to support long term operation decisions or to assess the safety status of plants during the long term operation period, the review should include specific aspects directly related to long term operation, such as ageing analysis and qualification extension analysis. The PSR should cover both the technical conditions for long term operation and conditions related to the current safe operation of the plant. These aspects and assessments performed as part of the periodic safety review should be clearly defined in the PSR basis document and in the PSR assessment reports produced as part of the PSR.

9.7 When the PSR is used to support long term operation decisions or to assess the safety of plants during the period of long term operation, the differences between a typical periodic safety review and a periodic safety review for or during long term operation should be taken into account. The main differences relate to:

(a) The scope of the periodic safety review, which should be set considering Requirement 16 of SSR-2/2 (Rev. 1) [2], specifically:
   (i) The review of the preconditions for long term operation, covering programmes and documents relevant to ageing management;
   (ii) The review of ageing management and its alignment with SSG-48 [12]~~[12][10]~~, with particular attention to scope setting in accordance with para. 5.16 of SSG-48 [12]~~[12][10]~~.
(b) Ageing effects, degradation mechanisms and ageing management programmes, which should be given increased importance.
(c) The PSR objectives related to preserving and extending equipment qualification and addressing technological obsolescence issues, which should be carefully defined.
(d) The time span considered in the review of the safety factors, which should be extended to cover the entire intended period of long term operation, depending on regulatory requirements and the operating organization's intentions.
(e) The results of the assessment of a periodic safety review for long term operation, which should confirm the assumptions of the feasibility study for long term operation.
(f) The scheduling of the PSR and the implementation plan, which should consider the need for extended and/or specific long term operation assessments and improvements and/or modifications, and the timeline for implementing all the PSR and long term operation safety improvements.

9.8 The safety factors evaluated in the PSR that provide justification for long term operation or are performed during the period of long term operation are usually the same as those for a standard PSR. However, some of these safety factors might become increasingly challenged over time due to extended operation and factors such as ageing mechanisms, time-related physical degradation and wear, evolution of knowledge and technology, new operating experience, and new developments in requirements, codes and standards for long term operation. Therefore, the safety factors that are most relevant for long term operation may need a specific analysis in a PSR for or during long term operation compared to a standard PSR.

9.9 The 14 safety factors presented in Section 2 and detailed in Section 7 should be considered relevant for long term operation. The related objectives and scope of these safety factors should be adapted to include aspects that have the potential to challenge or question the safe long term operation. These should be clearly outlined in the PSR basis document and agreed with the regulatory body, and included in the PSR assessment reports. Table 9.1 shows examples long term operation considerations for the most relevant safety factors (i.e. 1–4) that should be used to support the justification of long term operation.

TABLE 9.1. PURPOSE OF REVIEW OF SAFETY FACTORS 1–4: ASPECTS SPECIFIC TO LONG TERM OPERATION

| Safety factor | Name | Purpose of the review with regard to long term operation |
|---|---|---|
| 1 | Plant design | The review of this safety factor should be conducted to identify additional safety improvements necessary to ensure that the licensing basis remains valid during the period of long term operation. Such improvements might include refurbishment, provision of additional SSCs, safety features, and/or additional safety analysis and engineering justifications. |
| 2 | Actual condition of SSCs | The review of this safety factor should be conducted to determine the actual conditions of SSCs. Specific focus for long term operation should be given to those SSCs that cannot be replaced or repaired, or are difficult to replace or repair, for example the reactor pressure vessel, the containment building, and those that are within the scope of safety factor 4. |
| 3 | Equipment qualification | The review of this safety factor should be conducted to confirm that plant equipment subject to qualification has been properly identified and qualified for the relevant intended conditions. Specific focus for long term operation should be given to those SSCs for which the existing qualification justification does not cover the next PSR period, or the long term operation period. Particular attention should be given to those SSCs that are difficult to replace, and those SSCs for which ageing might affect the existing qualification. |
| 4 | Ageing | The review of this safety factor should be conducted to determine whether the ageing aspects affecting in-scope SSCs are identified and effectively managed. An ageing management programme should be in place and reviewed to ensure that it remains up-to-date and effective in delivering all required safety functions for long term operation. The review should also assess the effectiveness of the plant obsolescence management programme with respect to long term operation. |

9.10 For the other safety factors, the review should be done in accordance with the recommendations provided in Section 7. However, some parts of their review scope should be adapted for long term operation, as appropriate. Table 9.2 lists the relevant areas of the different safety factors to be adapted for the justification of long term operation and provides some examples of the purpose of the review with regard to long term operation.

TABLE 9.2. PURPOSE OF REVIEW OF SAFETY FACTORS 5–14: ASPECTS SPECIFIC TO LONG TERM OPERATION

| Safety Factor | Name | Purpose of the review with regard to long term operation |
|---|---|---|
| 5 | Deterministic safety analysis | The review of this safety factor should determine to what extent the existing deterministic safety analysis remains valid, taking into account the actual plant design, the current condition of SSCs important to safety, and their predicted state at the end of the PSR period. When the periodic safety review is used to support long term operation, this review should consider the entire intended period of long term operation, particularly regarding the predicted state of SSCs important to safety. |
| 6 | Probabilistic safety assessment | The review of this safety factor should determine the extent to which the existing PSA study remains valid. It should assess whether the study reflects the latest plant configuration, identifies weaknesses in the design and operation of the plant, and evaluates and compares proposed safety improvements in the global assessment. |
| 7 | Hazard analysis | The review of this safety factor should determine to what extent the existing protection against internal and external hazards remains adequate, taking into account the plant design, site characteristics, the current condition of the in-scope SSCs important to safety, their predicted state at the end of the PSR period, and the potential for hazards to change over time. When the periodic safety review is used to support long term operation, this review should cover the entire intended period of long term operation, particularly regarding the predicted state of SSCs important to safety and the impact of climate change, considering notably knowledge evolution and available information on future climate conditions, as appropriate. Further recommendations on the assessment of meteorological and hydrological hazards are provided in Ref. [23]. |
| 8 | Safety performance | The review of this safety factor should determine whether the plant's safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, are effective or indicate a need for safety improvements. It should also consider whether the extrapolation of safety performance trends has been addressed for the PSR period. When the periodic safety review is used to support long term operation, this review should consider the entire intended period of long term |

| Safety Factor | Name | Purpose of the review with regard to long term operation |
|---|---|---|
| | | operation, particularly with regard to the extrapolation of safety performance trends and the production of solid radioactive waste, considering the on-site storage capacity and the off-site disposal facilities. |
| 9 | Feedback of operating experience | The review of this safety factor should determine whether there is adequate feedback of operating experience from other NPPs and research findings, and whether this feedback is used to introduce reasonably practicable safety improvements at the plant or operating organization. Consideration of the latest international experience and research findings related to long term operation should be of special focus for the review of this safety factor. |
| 10 | Organization, the management system and safety culture | The review of this safety factor should determine whether the organization, the management system, and the safety culture are adequate and effective to ensure the safe operation of the NPP. When the periodic safety review is used to support long term operation, the review should consider whether an adequate policy regarding long term operation is present, and whether the dedicated organizational structures will remain effective and whether sufficient resources are available for long term operation. |
| 11 | Operational limits and conditions and operating ~~P~~procedures | The review of this safety factor should determine whether the operating organization's processes for managing, implementing, and adhering to operating procedures, and for maintaining compliance with operational limits and conditions, and regulatory requirements, are adequate and effective to ensure plant safety. When the PSR is used to support long term operation, this review should cover the entire intended period of long term operation. |
| 12 | Human factors | The review of this safety factor should evaluate the various human factors that might affect the safe operation of the nuclear power plant and seek reasonably practicable safety improvements. When the periodic safety review is used to support long term operation, this review should cover the entire intended period of long term operation. Issues related to the availability of sufficiently qualified staff, including effective knowledge and competence management necessary for the long term operation period, should be addressed by the review of this safety factor. |
| 13 | Emergency planning | The review of this safety factor should determine whether the operating organization has adequate plans, |

| Safety Factor | Name | Purpose of the review with regard to long term operation |
|---|---|---|
| | | staff, facilities, and equipment for dealing with emergencies, and whether the operating organization's arrangements have been adequately coordinated with the arrangements of local and national authorities and are regularly exercised. Due consideration of changes at the plant site, its surroundings, and the status of equipment and facilities used for emergency preparedness and response should be provided by the review of this safety factor to confirm their adequacy. When the periodic safety review is used to support long term operation, this review should cover the entire intended period of long term operation. |
| SF14 | Radiological impact on the environment | The review of this safety factor should determine if the operating organization has an adequate and effective programme for monitoring the radiological impact of the plant on the environment, ensuring that discharges are properly controlled and are as low as reasonably achievable. When the periodic safety review is used to support long term operation, this review should include the impact on the environment of the activities performed during the long term operation period of the plant. |

9.11 When PSR is used to support decisions on long term operation, the results of the assessment should be used to confirm the assumptions of the feasibility study for such long term operation. They might need to be integrated with other findings from analyses, in accordance with regulatory requirements.

9.12 When PSR is used to support decisions on long term operation or is conducted during the long term operation period:

(a) The global assessment should be performed as recommended in Section 8, with due consideration of any specific long term operation assessments not performed as part of the PSR, for example, in accordance with regulatory requirements and any reasonably practicable safety improvements identified.
(b) The PSR documentation should follow the format recommended in Appendix II and any additional long term operation documentation specified in regulatory requirements should be considered and added, as relevant.

# 10.PERIODIC SAFETY REVIEW OF NUCLEAR POWER PLANTS IN PERMANENT SHUTDOWN OR IN DECOMISSIONNING

10.1. Decommissioning refers to the administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility [4]. Decommissioning is the final stage in the lifetime of an authorized facility. The decommissioning stage follows the operational stage of a facility. These two ~~phases~~ stages have common characteristics and interdependencies, as well as significant differences, such as the nature of the key associated activities, the hazards to the public and the environment, and the quantities of radioactive waste that are generated and disposed of or discharged.

10.2. It is usual for the decommissioning stage to be divided into a number of phases depending on the selected decommissioning strategy, with a systematic transition phase between operation and decommissioning, where defueling activities and preparations for decommissioning typically take place. In some cases, the transition phase might be considered as the last phase of the operational stage. This transition phase should normally be covered in the last operational PSR, if any or in the safety documentation supporting the regulatory approval~~authorization~~ for final shutdown and the launch of the decommissioning ~~programme~~. During the decommissioning planning, as early as possible, the operating organization should clarify how the transition phase would be addressed. This section provides recommendations for the PSR that addresses the transition phase, referred to as the 'last operational PSR', and also for the PSR conducted during the decommissioning phase for a nuclear power plant, should this approach is followed.

~~10.3. The review process, key principles, and recommendations in this section apply to nuclear power plants. They may also be applicable to research reactors, radioactive waste management facilities, or nuclear fuel cycle facilities under decommissioning, using a graded approach that considers different levels of residual radiological risks for each decommissioning phase of these facilities.~~

10.3. A graded approach should be systematically applied for PSR of facilities under permanent shutdown, in the transition phase or undergoing active decommissioning. In particular, the scope and depth of the PSR for these facilities should be tailored depending on the decommissioning activities in the facility and be commensurate to the current hazards and risk profile or anticipated throughout the PSR period. Cognizance should also be taken of the planned decommissioning programme, particularly where this is extensive.

10.4. The PSR scope and objectives should be determined considering the expected duration of the decommissioning, SSCs important to safety, the remaining radioactive source term and the risk profile for the entire review period. A graded approach should be applied to ensure adequate scoping and objectives setting. A graded approach should be used to focus the PSR on ensuring safety improvements are directed towards relevant safety or environmental protection issues for a plant under decommissioning with due consideration of relevant uncertainties. Where available, the safety case for decommissioning should be used as the baseline ~~for~~ when setting the scope of the PSR and defining its objectives.

~~10.4.~~10.5. It is recognized that some States may prefer alternative arrangements to safety assessment during decommissioning than PSR. Such arrangements can, if applied with appropriate scope, frequency, depth and rigour, achieve the same outcomes as the process recommended in this Safety Guide. They allow safety to be appropriately managed. This Safety Guide is not intended to discourage such alternative arrangements or set unnecessary burden on operators or regulators.

~~10.5.~~10.6. Defuelling, spent fuel management, and preparatory decommissioning activities should be included in the last operational PSR, as they ~~are~~ might be essential for starting decommissioning. Regulatory processes for final shutdown and decommissioning

~~plans~~ may overlap with ~~or replace~~ this PSR In cases of overlaps, the PSR might be ~~, making it potentially redundant or part~~replaced by the process of ~~the~~ license application for authorization.

~~10.6.~~10.7. Any other facility on the decommissioning site that is in the operational stage (e.g. interim storage facilities for intermediate level radioactive waste or spent fuel interim storage facilities) should be reviewed against relevant standards, guides, and good practices consistent with the operational status of the facility. The recommendations provided in this section, therefore, do not generally apply to such other operating facilities, unless justified in appropriate circumstances.

~~10.7.~~10.8. The approach to safety assessment in decommissioning differs to that for operational facilities because of differences in a number of key aspects, for example, risk profile, staff experience and hazard analysis. A facility under permanent shutdown or undergoing decommissioning has significantly reduced nuclear and radiological hazards, when compared to facility in operation. ~~However~~Moreover, conventional and chemical hazards ~~are generally~~may be more significant during decommissioning activities and depend on the decommissioning phase of the facility. Conventional and chemical hazards are, however not in the scope of current Safety Guide, unless they impact nuclear safety. It is also recognized that the risk profile of the facility under decommissioning progressively decreases throughout the decommissioning process, while for a facility in operation, the risk profile does not significantly change over the entire operating lifetime.

~~10.8. A graded approach should be systematically applied for PSR of facilities under permanent shutdown, in the transition phase or undergoing active decommissioning. In particular, the scope and depth of the PSR for these facilities should be tailored depending on the decommissioning phase of the facility and be commensurate to the current hazards and risk profile or anticipated throughout the PSR period.~~

10.9. The PSR for an operating plant should be structured around the 14 safety factors listed in para 2.1~~5~~4. A similar approach could be adopted for plants in permanent shutdown, in the transition phase or under active decommissioning. In general, only a subset of these safety factors is expected to be relevant for consideration for such facilities and this subset is highly likely to vary throughout the whole decommissioning period**.** Table 10.1 shows examples of subset of the safety factors, their relevance for the last operational PSR and for PSRs in decommissioning, as well as the key principles to be considered for their review.~~These safety factors, their relevance for the last operational PSR and for PSRs in decommissioning, as well as the key principles to be considered for their review are presented in Table 10.1.~~ For each safety factor, the reviews should address the associated risks and hazards and their expected evolution over the PSR period. A graded approach should be applied, considering the expected reduction of hazards and risks that will occur during the PSR period, due to the transition from operation to final shutdown or by the progress of the decommissioning activities.

10.10. As the level of detail of the review is expected to vary between individual safety factors and decommissioning phases, for some safety factors, a high level or a programmatic review could be performed. Where such an approach is adopted, this should be set out and justified in the PSR basis document.

10.11. It is recommended that an updated safety case for permanent shutdown and decommissioning reflecting the current site configuration or decommissioning progress is used to realign PSR timelines with any major changes in the facility and corresponding hazards, where relevant.

10.12. For nuclear power plants that are transferring the site license ownership when the PSR is due, relevant arrangements to ensure organizational capabilities, management system and safety culture, should be considered in the PSR.

10.13. Findings from the reviews of individual safety factors should form an input to a global assessment that should be performed to provide a judgement on safety for continued decommissioning activities, taking into account proposed corrective measures and safety improvements.

10.14. The review process, key principles, and recommendations in this section apply to nuclear power plants. They may also be applicable to research reactors, radioactive waste management facilities, or nuclear fuel cycle facilities under decommissioning, using a graded approach that considers different levels of residual radiological hazards for each decommissioning phase of these facilities.

TABLE 10.1: INDIVIDUAL SAFETY FACTORS, THEIR RELEVANCE FOR DECOMMISSIONING, AND KEY CONSIDERATIONS FOR THEIR REVIEW

| Safety factor | Name | Purpose of the PSR and key considerations |
|---|---|---|
| 1 | Plant design | The review should focus on SSCs important to safety, considering the operations planned for the PSR period and the implications of:<br>• The final shutdown and the resulting operations performed during the transition phase; or<br>• The activities during decommissioning.<br><br>The recommendations on safety factor 1 in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with the activities planned on site for the upcoming PSR period. The review should also cover site infrastructure and its configuration~~, including cooling systems and buildings~~. |
| 2 | Actual condition of SSCs important to safety | The review should focus on SSCs important to safety, considering the operations planned for the PSR period and the implications of:<br>• The final shutdown and the resulting operations performed during the transition phase; or<br>• The activities during decommissioning.<br><br>The recommendations on safety factor 2 in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with the |

| | | |
|---|---|---|
| | | activities planned on site for the upcoming PSR period. The review should also cover site infrastructure and its configuration, including cooling systems and buildings. |
| 3 | Equipment qualification | The review should focus on SSCs important to safety, considering the operations planned for the PSR period and the implications of:<br>• The final shutdown and the resulting operations performed during the transition phase; or<br>• The activities during decommissioning.<br><br>The recommendations on safety factor 3 in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with the activities planned on site for the upcoming PSR period. During decommissioning, qualification and environmental conditions are less severe compared to the operational phase, for example, ambient temperature. At some point during decommissioning, this safety factor may no longer be relevant. |
| 4 | Ageing | The review should focus on SSCs important to safety, considering the operations planned for the PSR period and the implications of:<br>• The final shutdown and the resulting operations performed during the transition phase; or<br>• The activities during decommissioning.<br><br>The recommendations on safety factor 4 in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with the activities planned on site for the upcoming PSR period. In case of deferred dismantling, this safety factor will be of high importance with respect to equipment reliability and maintenance. At some point during decommissioning, this safety factor may no longer be relevant.- |
| 5 | Deterministic safety analysis | The review should focus on SSCs important to safety, considering the operations planned for the PSR period and the implications of:<br>• The final shutdown and the resulting operations performed during the transition phase; or |

| | | |
|---|---|---|
| | | • The activities during decommissioning and potential incidents related to de-commissioning, including radiological hazards and personnel exposure.<br><br>The recommendations on safety factor 5 in Section 7 are applicable and the review should consider Requirement 3 of IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [12]. When these recommendations are taken into account, a graded approach should be applied, commensurate with the activities planned on site for the upcoming PSR period. Th~~is~~e scope of the review of this safety factor will decrease as the decommissioning progresses. At some point during decommissioning, this safety factor might not be relevant anymore. |
| 6 | Probabilistic safety assessment | This safety factor will not be relevant after the completion of transition phase once all fuel is removed from the site. |
| 7 | Hazard analysis | The review should focus on SSCs important to safety, considering the operations planned for the PSR period and the implications of:<br><br>• The final shutdown and the resulting operations performed during the transition phase; or<br>• The activities during decommissioning.<br><br>The recommendations on safety factor 7 in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with the activities planned on site for the upcoming PSR period. The list of relevant hazards may vary over the course of decommissioning, considering:<br><br>• The ~~reduction~~ change in the probability of occurrence, and hence in the hazard level;<br>• The use rate, which for some components might be higher during the decommissioning phase than in the operational phase (e.g. cranes), inducing a higher hazard rate.<br><br>For these hazards, a graded approach should be applied throughout decommissioning, as they will vary, sometimes significantly. The scope of this safety factor will decrease as decommissioning progresses, though some hazards may occasionally be of higher significance. At some point during decommissioning, this safety factor might no longer be relevant. |
| 8 | Safety performance | This safety factor should be considered and incorporated into the PSR. The recommendations on safety factor 8 provided in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with |

| | | |
|---|---|---|
| | | the activities planned on site for the upcoming PSR period. The review of this safety factor should cover:<br><br>• Review of feedback from operating experience;<br>• Inventory of the remaining radioactive waste;<br>• Review of site programmes, for example radiation protection~~.~~;<br>• Analysis of trends. |
| 9 | Feedback of operating experience | The recommendations on safety factor 9 provided in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with the activities planned on site for the upcoming PSR period. ~~Benchmarking~~ Taking into account lessons learnt against defuelling and decommissioning activities from other nuclear power plants should be considered where relevant and possible. However, using operating experience from other plants may be challenging due to differences in plant design, plant history, decommissioning programmes and techniques used. |
| 10 | Organization, the management system and culture for safety | Considering potential changes in staffing and organization, as well as multiple and diverse operations and activities, it might not be practical to only perform a review of this safety factor in the PSR. Instead, this safety factor might be addressed outside of PSR, as the 10 year frequency might not be appropriate.<br><br>If this safety factor is selected for inclusion in the PSR, the scope of the review should be clearly defined by the operating organization, taking into account Requirements 6 and 7 of GSR Part 6 [15]~~[15][12]~~. |
| 11 | Operational limits and conditions and operating procedures | Same as for safety factor 10 |
| 12 | Human factors | This safety factor is of high importance~~t~~ for decommissioning, including the transition phase. The recommendations for safety factor 12 provided in Section 7 are applicable. When these recommendations are taken into account, a graded approach should be applied, commensurate with the activities planned on site for the upcoming PSR period.-<br><br>- |
| 13 | Emergency planning | This safety factor should be reviewed based on:<br><br>• The radioactive source term, which will be removed progressively as decommissioning progresses. |

| | | |
|---|---|---|
| | | • The hazards and the potential for a significant release of radioactivity, which will reduce significantly as decommissioning progresses.<br><br>Most of the recommendations provided for safety factor 13 in Section 7 are applicable, for as long as there is a sufficient source term to justify the need for emergency planning. |
| 14 | Radiological impact on the environment | This safety factor is relevant to all decommissioning ~~stages~~ phases for as long as there is a sufficient source term that can be released to the environment. The recommendations made for safety factor 14 in Section 7 are applicable, but a graded approach should be applied and be commensurate with the remaining source term. |

# APPENDIX I. INTERFACES BETWEEN SAFETY FACTORS

I.1. The teams reviewing each safety factor should communicate with each other during the review process, starting from the preparation phase of the PSR. Communication between review teams should be well organized, because findings (or outputs) identified in the review of one safety factor could be an important input to the review of other safety factors. All findings that are related to other safety factors should be provided immediately to the reviewers of the relevant safety factors. Potential likely correlations between the different safety factors are shown in Table 1 and paras I.2–I.15 list the relevant safety requirements. The safety factors listed on the upper horizontal axis may provide input to the safety factors listed on the vertical axis on the left.

TABLE 1. MATRIX OF INTERFACES BETWEEN SAFETY FACTORS

| | | Safety factors providing input | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | **SF1** | **SF2** | **SF3** | **SF4** | **SF5** | **SF6** | **SF7** | **SF8** | **SF9** | **SF10** | **SF11** | **SF12** | **SF13** | **SF14** |
| **Safety factors receiving input** | **SF 1** | | X | X | X | X | X | X | X | X | | X | X | X | X |
| | **SF 2** | X | | X | X | X | | | X | X | X | X | X | | |
| | **SF 3** | X | X | | X | X | X | X | X | X | X | X | X | X | |
| | **SF 4** | X | X | X | | X | X | X | X | X | X | X | X | | |
| | **SF 5** | X | X | X | X | | X | X | X | X | | X | X | X | |
| | **SF 6** | X | X | X | X | X | | X | X | X | | X | X | X | |
| | **SF 7** | X | X | X | | X | X | | X | X | | X | X | X | X |
| | **SF 8** | X | X | | | | | | | X | X | | X | x | X |
| | **SF 9** | X | X | | | | | | X | | X | | X | X | X |
| | **SF 10** | X | X | | | | | | X | X | | | X | X | X |
| | **SF 11** | X | X | X | X | X | X | X | X | X | X | | X | X | X |
| | **SF 12** | X | X | X | X | X | X | X | X | X | X | X | | X | |
| | **SF 13** | X | X | | | X | X | X | X | X | X | X | | | |
| | **SF 14** | X | X | | | X | | X | X | X | X | X | X | | |

SF 1: Plant design.
SF 2: Actual condition of SSCs.
SF 3: Equipment qualification.
SF 4: Ageing.
SF 5: Deterministic safety analysis.
SF 6: Probabilistic safety assessment.
SF 7: Hazard analysis.
SF 8: Safety performance.
SF 9: Feedback of operating experience.
SF 10: Organization, the management system and safety culture.
SF 11: Operational limits and conditions and operating procedures.
SF 12: Human factors.
SF 13: Emergency planning.
SF 14: Radiological impact on the environment.

I.2. The review of the safety factor 1 should consider relevant interfaces, as appropriate, with safety factors 2–9 and 11–14. The relevant safety requirements are:

- Requirement 7 of SSR-1 (Rev. 1) [19]~~[19][16]~~
- Requirements 6, 10, 11, 13, 14, 15, 18–22, 24, 26–32 of SSR-2/2 (Rev. 1) [2]
- Requirements 13 and 15 of GSR Part 3 [29]~~[29][26]~~
- Requirements 6–11, 13–19 of GSR Part 4 (Rev. 1) [3]
- Requirements 8–12 of GSR Part 5 [30]~~[30][27]~~
- Requirements 1, 1, 12 and 14 of GSR Part 6 [15]~~[15][12]~~
- Requirement 8, 15 and 16 of GSR Part 7 [40]~~[40][37]~~.

I.3. The review of the safety factor 2 should consider relevant interfaces, as appropriate, with safety factors 1, 3, 4, 5, 8 – 12. The relevant safety requirements are:

- Requirements 29–31 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
- Requirements 28, 29 and 31 of SSR-2/2 (Rev. 1) [2]
- Requirements 10, 23 and 24 of GSR Part 4 (Rev. 1) [3]
- Requirement 11 of GSR Part 5 [30]~~[30][27]~~.
-

I.4. The review of the safety factor 3 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 4–13. The relevant safety requirements are:

- Requirements 6, 10, 11, 14, 15, 18–22, 24, 26–32 of SSR-2/2 (Rev. 1) [2]
- Requirements 10 and 11 of GSR Part 4 (Rev. 1) [3].

I.5. The review of the safety factor 4 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 3, 5 – 12. The relevant safety requirements are:

- Requirements 29–31 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
- Requirements 28, 29 and 31 of SSR-2/2 (Rev. 1) [2]
- Requirements 10, 23 and 24 of GSR Part 4 (Rev. 1) [3]
- Requirement 11 of GSR Part 5 [30]~~[30][27]~~.

I.6. The review of the safety factor 5 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 3, 4, 6, 7, 8, 9, 11, 12 and 13. The relevant safety requirements are:

- Requirement 7 of SSR-1 (Rev. 1) [19]~~[19][16]~~
- Requirement 17 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
- Requirements 6, 10, 11, 13, 14, 15, 24, 26, 28–31 of SSR-2/2 (Rev. 1) [2]
- Requirement 11 of GSR Part 4 (Rev. 1) [3]
- Requirement 8 of GSR Part 7 [40]~~[40][37]~~.

I.7. The review of the safety factor 6 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 3, 4, 5, 7, 8, 9, 11, 12 and 13. The relevant safety requirements are:

- Requirement 7 of SSR-1 (Rev. 1) [19]~~[19][16]~~
- Requirement 17 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
- Requirements 6, 10, 11, 13, 14, 15, 24, 26, 28 – 31 of SSR-2/2 (Rev. 1) [2]

— Requirement 11 of GSR Part 4 (Rev. 1) [3]
— Requirement 8 of GSR Part 7 [40]~~[40][37]~~.

I.8. The review of the safety factor 7 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 3, 5, 6, 8, 9, 11–14. The relevant safety requirements are:

— Requirements 2, 3, 4, 6, 10, 14 and 29 of SSR-1 (Rev. 1) [19]~~[19][16]~~
— Requirement 17 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
— Requirements 13, 14, 15, 18, 19, and 24 of SSR-2/2 (Rev. 1) [2]
— Requirement 7 of GSR Part 4 (Rev. 1) [3]
— Requirement 4, 5, 6, 8 and 9 of GSR Part 7 [40]~~[40][37]~~.

I.9. The review of the safety factor 8 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 9, 10, 12, 13 and 14. The relevant safety requirements are:

— Requirement 10 of GSR Part 2 [9]~~[9][7]~~
— Requirements 12, 14, 30 and 31 of GSR Part 3 [29]~~[29][26]~~
— Requirement 19 of GSR Part 4 (Rev. 1) [3]
— Requirements 4, 6, 8 and 12 of GSR Part 5 [30]~~[30][27]~~
— Requirements 1, 10–12 and 14 of GSR Part 6 [15]~~[15][12]~~
— Requirement 11 of GSR Part 7 [40]~~[40][37]~~.

I.10. The review of the safety factor 9 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 8, 10, 12, 13 and 14. The relevant safety requirements are:

— Requirement 10 of GSR Part 2 [9]~~[9][7]~~
— Requirements 12, 14, 30 and 31 of GSR Part 3 [29]~~[29][26]~~
— Requirement 19 of GSR Part 4 (Rev. 1) [3]
— Requirements 4, 6, 8 and 12 of GSR Part 5 [30]~~[30][27]~~
— Requirement 11 of GSR Part 7 [40]~~[40][37]~~.

I.11. The review of the safety factor 10 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 8, 9, 12, 13 and 14. The relevant safety requirements are:

— Requirement 10 of GSR Part 2 [9]~~[9][7]~~
— Requirements 12, 14, 30 and 31 of GSR Part 3 [29]~~[29][26]~~
— Requirement 19 of GSR Part 4 (Rev. 1) [3]
— Requirements 4, 6, 8 and 12 of GSR Part 5 [30]~~[30][27]~~
— Requirements 1, 10–12 and 14 of GSR Part 6 [15]~~[15][12]~~
— Requirement 11 of GSR Part 7 [40]~~[40][37]~~.

I.12. The review of the safety factor 11 should consider relevant interfaces, as appropriate, with all safety factors. The relevant safety requirements are:

— Requirements 6, 19, 20, 21, 26, 30 and 31 of SSR-2/2 (Rev. 1) [2]
— Requirement 28 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
— Requirements 9, 15, 24 of GSR Part 3 [29]~~[29][26]~~
— Requirements 23 and 24 of GSR Part 4 (Rev. 1) [3]

— Requirements 4 and 12 of GSR Part 5 [28]~~[28][25]~~.

I.13. The review of the safety factor 12 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 13. The relevant safety requirements are:

— Requirements 6, 16, and 32 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
— Requirements 4, 6, 7, 8, 11, 18, 19, 22, and 30 of SSR-2/2 (Rev. 1) [2]
— Requirements 9 and 11 of GSR Part 2 [9]~~[9][7]~~
— Requirement 26 of GSR Part 3 [29]~~[29][26]~~
— Requirements 11, 23, and 24 of GSR Part 4 (Rev. 1) [3].

I.14. The review of the safety factor 13 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 5, 6, 7, 8, 9, 10, and 11. The relevant safety requirements are:

— Requirements 1, 13, 26, and 31 of SSR-2/2 (Rev. 1) [2]
— Requirements 59 and 66 of SSR-2/1 (Rev. 1) [2]
— Requirements 9, 15, 24 of GSR Part 3 [29]~~[29][26]~~
— Requirement 24 of GSR Part 4 (Rev. 1) [3].

I.15. The review of the safety factor 14 should consider relevant interfaces, as appropriate, with safety factors 1, 2, 5, 7, 8, 9, 10, 11, and 12. The relevant safety requirements are:

— Requirements 2, 4, 10, 14, 26, and 29 of SSR-1 (Rev. 1) [19]~~[19][16]~~
— Requirements 81 and 82 of SSR-2/1 (Rev. 1) [7]~~[7][5]~~
— Requirements 20 and 21 of SSR-2/2 (Rev. 1) [2]
— Requirements 3, 12, 14, 30, 31, and 32 of GSR Part 3 [29]~~[29][26]~~
— Requirement 9 of GSR Part 4 (Rev. 1) [3]
— Requirements 8 and 12 of GSR Part 5 [30]~~[30][27]~~.

# APPENDIX II. DOCUMENTATION OF THE PSR

II.1. The following documents should be produced during the conduct of the PSR to provide the information needed in the different stages of the process described in this Safety Guide:

(a) The basis document for the PSR;
(b) Safety factor report(s);
(c) The global assessment report;
(d) The final PSR report, including the integrated implementation plan.

RECOMMENDED CONTENTS OF THE PSR BASIS DOCUMENT

II.2. The PSR basis document should include three main parts:

(1) *General*

(a) The scope and objectives of the PSR and the future operating period that will be considered by the review;
(b) The cut-off date(s) to be used, that is, the dates beyond which updates to standards and codes and new information (e.g. more recent plant operating experience) will not be considered during this PSR;
(c) The plant licensing basis at the time of initiating the PSR;
(d) Relevant regulatory requirements;
(e) The list of safety factors to be reviewed within the PSR and interfaces between them;
(f) A description of the systematic review approach to be used to ensure a complete and comprehensive review;
(g) Processes for identifying, categorizing, prioritizing and resolving negative findings;
(h) The process for ensuring that any immediate and significant risks to workers, the public or the environment identified during the PSR will be addressed without delay;
(i) The methodology to be used for the global assessment and the planned document structure of the global assessment report;
(j) Guidance for preparation of the integrated implementation plan of safety improvements;
(k) The systematic method to be used for recording outputs from the PSR, including the proposed formats of:
    (i) The safety factor reports;
    (ii) The global assessment report;
    (iii) The final PSR report, including the integrated implementation plan of safety improvements.

(2) *Safety factors*

The following information should be provided for each safety factor:

(a) Objectives and scope of the review;
(b) The applicable regulatory requirements, national, international and industry safety standards, codes and methods, and operating practices selected as the basis for the safety factor review and, where relevant, their hierarchy;
(c) The input documents and processes to be reviewed;
(d) The specific methodologies to be used for the review and a justification for the approach to be followed;

(e) Expected outputs.

(3) *Project plan for the PSR*

(a) Organization of the project, including roles and responsibilities;
(b) Time schedule including any major milestones and cut-off date(s);
(c) Project and quality management processes;
(d) Processes for ensuring consistency between separate safety factor reviews, for example, for establishing a common set of technical databases (see para. 4.9);
(e) Training;
(f) Internal communications;
(g) The plan for communicating and interfacing with and gaining relevant approvals and agreements from, the regulatory body.

## RECOMMENDED CONTENTS OF EACH SAFETY FACTOR REPORT

II.3. The safety factor report should include the results from the review of each safety factor following the approach detailed in the PSR basis document. The findings specific to each safety factor should be documented and ranked in accordance with their safety significance. In some States, the findings on all safety factors are included in a single report; however, multiple reports can be developed. If multiple reports are to be developed, a general template or structure should be provided to maintain consistency and to ensure that all the items to be reviewed are covered by the different teams performing the PSR.

II.4. The following is an example of the structure of a typical safety factor report:

(a) Title (name of the safety factor);
(b) Introduction;
(c) Scope of the review, including a list of the documents and aspects of safety reviewed (e.g. organizational capability);
(d) Review criteria (e.g. reference standards, operating practices, safety assessment criteria);
(e) Review methodologies applied;
(f) Review of performance since the previous PSR;
(g) Comparison with review criteria and discussion of the results;
(h) Evaluation of the safety significance of negative findings, together with proposed safety improvements and their prioritization;
(i) Review of future safety for the period addressed in the PSR;
(j) Conclusions;
(k) References;
(l) Appendices.

## RECOMMENDED CONTENTS OF THE GLOBAL ASSESSMENT REPORT

II.5. The PSR results for all safety factors should be evaluated through a global assessment, and the following items should be documented:

(a) Significant PSR outcomes, including positive and negative findings (strengths and deviations);
(b) Analysis of interfaces, overlaps and omissions between safety factors and between individual negative findings;

(c) An overall analysis of the combined effects of the positive and negative findings;
(d) The category, ranking and priority of safety improvements proposed to address negative findings;
(e) An assessment of defence in depth;
(f) An assessment of the overall risk;
(g) Justification for proposed continued operation in both the short term and long term.

RECOMMENDED CONTENTS OF THE FINAL PSR REPORT

II.6. The final PSR report should provide an overview of the PSR and should include the following topics:

(a) Summary of the outcomes of the safety factor reports;
(b) Summary of the outcomes of the global assessment report, including:
    (i) Identification of negative findings arising from deviations between the present state of the plant and current safety standards and operating practices;
    (ii) An evaluation of the safety significance of these negative findings;
    (iii) An overall judgement on the acceptability of continued plant operation;
(c) The integrated implementation plan, including proposals for resolving negative findings by reasonably practicable safety improvements or corrective actions, and their safety significance and priority;
(d) An assessment of the safety of future plant operation over the period addressed in the PSR.

# REFERENCES


[1] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006), https://doi.org/10.61092/iaea.hmxn-vw0a

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2 (Rev.1), IAEA, Vienna (2016).

[3] INTERNATIONAL ATOMIC ENERGY, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

[4] INTERNATIONAL ATOMIC ENERGY, IAEA Nuclear Safety and Security Glossary: Terminology Used in Nuclear Safety, Nuclear Security, Radiation Protection and Emergency Preparedness and Response, 2022 (Interim) Edition, IAEA, Vienna (2022), https://doi.org/10.61092/iaea.rrxi-t56z

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review for Research Reactors, Safety Report Series No. 99, IAEA, Vienna (2020).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review for Nuclear Fuel Cycle Facilities, Safety Report Series No. 124, IAEA, Vienna (2024).

~~[4]~~[7] INTERNATIONAL ATOMIC ENERGY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

~~[5]~~[8] INTERNATIONAL ATOMIC ENERGY, Format and Content of the Safety Analysis Report for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-61, IAEA, Vienna (2021).

~~[6]~~[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016), https://doi.org/10.61092/iaea.cq1k-j5z3

~~[7]~~[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

~~[8]~~[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of the Interfaces between Nuclear and Radiation Safety and Nuclear Security, IAEA Safety Standards Series (DS533) / IAEA Nuclear Security Series (NST067) (in preparation).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Chemistry Programme for Water Cooled Nuclear Power Plants, IAEA Safety Standards Series No. SSG-13 (Rev. 1), IAEA, Vienna (2024).

~~[9]~~[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-48, IAEA, Vienna (2018).

~~[10]~~[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Equipment Qualification for Nuclear Installations, IAEA Safety Standards Series No. SSG-69, IAEA, Vienna (2021).

~~[11]~~[15] INTERNATIONAL ATOMIC ENERGY, Decommissioning of Facilities, IAEA Safety Standards Series No. GSR Part 6, IAEA, Vienna (2014).

~~[12]~~[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

~~[13]~~[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3 (Rev. 1), IAEA, Vienna (2024), https://doi.org/10.61092/iaea.3ezv-lp49

~~[14]~~[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4 (Rev. 1), IAEA, Vienna (in publication).

~~[15]~~[19] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSR-1, IAEA, Vienna (2019).

~~[16]~~[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection Against Internal and External Hazards in the Operation of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-77, IAEA, Vienna (2022).

~~[17]~~[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 (Rev. 1), IAEA, Vienna (2022).

~~[18]~~[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-79, IAEA, Vienna (2002).

~~[19]~~[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Meteorological and Hydrological Hazards in Site Evaluations for Nuclear Installations, IAEA Safety Standards Series No. SSG-18, IAEA, Vienna (2011) (currently under revision as DS541).

~~[20]~~[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, IAEA, Vienna (2004).

~~[21]~~[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, IAEA, Vienna (2021).

~~[22]~~[26] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Nuclear Installations Against External Events Excluding Earthquakes, IAEA Safety Standards Series No. SSG-68, IAEA, Vienna (2021).

~~[23]~~[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Safety for Nuclear Installations, IAEA Safety Standards Series No. SSG-89, IAEA, Vienna (2024), https://doi.org/10.61092/iaea.jtrh-4y3z

~~[24]~~[28] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-67, IAEA, Vienna (2021).

~~[25]~~[29] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION

OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, Standards Series No. GSR Part 3, IAEA, Vienna (2014), https://doi.org/10.61092/iaea.u2pu-60vm

~~[26]~~[30] INTERNATIONAL ATOMIC ENERGY, Predisposal waste management of Radioactive Waste, General Safety Requirements, International Safety Standards Series No. GSR Part 5, IAEA, Vienna (2009).

~~[27]~~[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Safety Performance Indicators for Nuclear Power Plants, IAEA-TECDOC-1141, IAEA, Vienna (2000).

~~[28]~~[32] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-74, IAEA, Vienna (2002).

~~[29]~~[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-70, IAEA, Vienna (2022).

~~[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety, Security and Safeguards by Design in Small Modular Reactors, TecDoc (in preparation)~~

~~[31]~~[34] INTERNATIONAL ATOMIC ENERGY AGENCY, The Operating Organization for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-72, IAEA, Vienna (2022).

[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Conduct of Operations at Nuclear Power Plants, IAEA Safety Standards Series No. SSG-76, IAEA, Vienna (2022).

~~[32]~~[36] INTERNATIONAL ATOMIC ENERGY AGENCY, Core Management and Fuel Handling for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-73, IAEA, Vienna (2022).

~~[33]~~[37] INTERNATIONAL ATOMIC ENERGY AGENCY, Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, IAEA, Vienna (2018).

~~[34]~~[38] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40, IAEA, Vienna (2016).

~~[35]~~[39] INTERNATIONAL ATOMIC ENERGY AGENCY, Operating Experience Feedback for Nuclear Installations, IAEA Safety Standards Series No. SSG-50, IAEA, Vienna (2018).

~~[36]~~[40] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS OFFICE FOR THE CO-ORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7, IAEA, Vienna (2015), https://doi.org/10.61092/iaea.3dbe-055p

[41] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5, IAEA, Vienna (2009).

[42] INTERNATIONAL ATOMIC ENERGY AGENCY, Modifications to Nuclear Power Plants, IAEA Safety Standards Series No. SSG-71, IAEA, Vienna (2022).

~~[37]~~[43] INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-54, IAEA, Vienna (2019).

~~[38]~~[44] INTERNATIONAL ATOMIC ENERGY AGENCY, Human Factors Engineering in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-51, IAEA, Vienna (2019).

~~[39]~~[45] INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Safe Management of the Operating Lifetimes of Nuclear Power Plants, INSAG-14, IAEA, Vienna (1999).

~~[40]~~[46] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, Arrangements for Preparedness for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007). (a revision of this publication is in preparation.)

~~[41]~~[47] INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Maintaining Knowledge, Training and Infrastructure for Research and Development in Nuclear Safety, INSAG-16, IAEA, Vienna (2003).

[48] INTERNATIONAL ATOMIC ENERGY AGENCY, Human Reliability Analysis for Nuclear Installations, IAEA Safety Reports Series No 127 [IAEA Preprint] (2024).

[49] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Radiation Protection of the Public and the Environment, IAEA Safety Standards Series No. GSG-8, IAEA, Vienna (2018).

[50] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Regulatory Control of Radioactive Discharges to the Environment, IAEA Safety Standards Series No. GSG-9, IAEA, Vienna (2018).

[51] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Prospective Radiological Environmental Impact Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSG-10, IAEA, Vienna (2018).

~~[42]~~[52] INTERNATIONAL ATOMIC ENERGY AGENCY, Design Extension Conditions and the Concept of Practical Elimination in the Design of Nuclear Power Plants, SSG-88, IAEA, Vienna (2024), https://doi.org/10.61092/iaea.la1m-dy8m

[53] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of Defence in Depth for Nuclear Power Plants, Safety Report Series No. 46 (Rev. 1), IAEA, Vienna (2024).

# Annex I

# TYPICAL INPUTS, OUTPUTS AND RELEVANT PUBLICATIONS FOR THE REVIEW OF SAFETY FACTORS FOR A NUCLEAR POWER PLANT

## SAFETY FACTOR 1: PLANT DESIGN

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and national and international requirements and codes on design and site evaluation;<br>— Current national and international operating practices in design and site evaluation.<br><br>Plant specific documents:<br><br>— Relevant chapters of the final safety analysis report;<br>— The site evaluation (from the final safety analysis report or similar safety document);<br>— The list of SSCs important to safety and their safety classification (from the final safety analysis report or similar safety document);<br>— The documented design basis (original or reconstituted and updated) including the list of postulated initiating events;<br>— Detailed description of the plant design, supported by drawings of the layout, systems and equipment (from the final safety analysis report or similar safety document);<br>— Technical specifications (as set out in the final safety analysis report);<br>— Results of tests in the commissioning phase;<br>— Review compliance with plant design specifications.<br><br>Operating experience:<br><br>— Operating experience from similar plants in the State and in other States;<br>— Actual physical condition of the plant. | The review of plant design may lead to findings in one or more of the following areas:<br><br>— Compliance with current safety standards and design requirements;<br>— Defence in depth in the prevention and mitigation of events (faults and hazards) that could jeopardize safety;<br>— Dependability requirements for SSCs important to safety;<br>— Records of the design basis, modifications to the plant and test results;<br>— The final safety analysis report;<br>— Recommended plant modifications;<br>— New operational limits and conditions.<br><br>On the basis of the results of the review, reassessment of safety margins against current safety standards and regulatory requirements may be necessary.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I), for example in the following areas:<br><br>— New safety margins;<br>— Plant design modifications. |

| | |
|---|---|
| The review of this safety factor may require input from other safety factors (see Appendix I), for example in the following areas:<br><br>— New results of reviews of tests, inspections and maintenance and ageing margins;<br>— Negative findings from equipment qualification;<br>— Results from the evaluation of hazards;<br>— Results of root cause analyses;<br><br>New postulated initiating events and new technical solutions. | |

RELEVANT IAEA PUBLICATIONS


Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

Design of Electrical Power Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-34, IAEA, Vienna (2016).

Design of Fuel Handling and Storage Systems in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-63, IAEA, Vienna (2020).

Design of the Reactor Containment and Associated Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-53, IAEA, Vienna (2019).

Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-56, IAEA, Vienna (2020).

Design of the Reactor Core for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-52, IAEA, Vienna (2005).

Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

Investigation of Site Characteristics and Evaluation of Radiation Risks to the Public and the Environment in Site Evaluation for Nuclear Installations, DS529, IAEA, Vienna (in preparation).

Evaluation of Seismic Safety for Nuclear Installations, IAEA Safety Standards Series No. SSG-89, IAEA, Vienna (2024).

Design of Nuclear Installations Against External Events Excluding Earthquakes, IAEA Safety Standards Series No. SSG-68, IAEA, Vienna (2021).

Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-79, IAEA, Vienna (2023).

Format and Content of the Safety Analysis Report for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-61, IAEA, Vienna (2021).

Geotechnical Aspects in Siting and Design of Nuclear Installations, IAEA Safety Standards Series No. DS531, IAEA, Vienna (in preparation).

Design of Instrumentation and Control Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-39, IAEA, Vienna (2016).

Protection against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, IAEA, Vienna (2021).

Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40, IAEA, Vienna (2016).

Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, IAEA, Vienna (2018).

Radiation Protection Aspects of Design for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-90, IAEA, Vienna (2024).

Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-30, IAEA, Vienna (2014).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

Seismic Design for Nuclear Installations, IAEA Safety Standards Series No. SSG-67, IAEA, Vienna (2021).

Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 (Rev. 1), IAEA, Vienna (2022).

Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSR-1, IAEA, Vienna (2019).

Design of Instrumentation and Control Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-39, IAEA, Vienna (2016).

Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

Design Extension Conditions and the Concept of Practical Elimination in the Design of Nuclear Power Plants, SSG-88, IAEA, Vienna (2024).

## SAFETY FACTOR 2: ACTUAL CONDITION OF STRUCTURES, SYSTEMS AND COMPONENTS

<table>
<tr><th>Inputs</th><th>Outputs</th></tr>
<tr><td>
Standards and requirements:
<ul>
<li>Current safety standards and national and international requirements and codes on design;</li>
<li>Appropriate standards on assessment;</li>
<li>Operating experience from plants, in the State and in other States, containing similar SSCs.</li>
</ul>
Plant specific documents:
<ul>
<li>The list of SSCs important to safety and their safety classification;</li>
<li>Information about the integrity and functional capability of SSCs important to safety, including material case histories;</li>
<li>Descriptions of the actual condition of SSCs important to safety;</li>
<li>The assessment methods applied by the operator;</li>
<li>Technical specification of the SSCs important to safety;</li>
<li>Equipment qualification results;</li>
<li>Description of the support facilities available to the plant both on and off the site, including maintenance and repair shops;</li>
<li>Reports of walkdowns;</li>
<li>Maintenance records;</li>
<li>Inspection results;</li>
<li>Findings of tests that demonstrate the functional capability of SSCs important to safety;</li>
<li>Operational data history and trends;</li>
<li>Outstanding maintenance and modifications;</li>
<li>Maintenance data, including data on repeated maintenance and corrective maintenance and reports of obsolescence;</li>
<li>Records of modifications.</li>
</ul>
The review of this safety factor may require input from other safety factors (see Appendix I), for example in the following areas:
<ul>
<li>Negative findings from equipment qualification;</li>
<li>Predictions of ageing, effectiveness of the ageing management programme;</li>
<li>New postulated initiating events;</li>
<li>New internal and external hazards;</li>
<li>Operating history;</li>
<li>Configuration management.</li>
<li>Records of modifications.</li>
</ul>
</td><td>
Examples of findings from the review of the actual condition of the plant’s SSCs are the following:
<ul>
<li>Confirmation that the design basis assumptions have not been significantly challenged, with account taken of the actual condition of the plant, and will remain unchallenged until the next PSR;</li>
<li>The actual condition of the SSCs important to safety of the nuclear power plant is such that the design basis assumptions are not significantly challenged and will not be challenged before the next PSR;</li>
<li>Additional surveillance measures are necessary to ensure the timely detection of ageing effects;</li>
<li>Maintenance and testing needs to be improved;</li>
<li>Processes do not maintain adequate records of the actual state of the plant, ageing processes and obsolescence of components;</li>
<li>Validity of existing records is sufficient or has to be improved.</li>
</ul>
Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I).
</td></tr>
</table>

## SAFETY FACTOR 3: EQUIPMENT QUALIFICATION

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and national and international requirements and codes on equipment qualification;<br>— Current national and international operating practices in equipment qualification.<br><br>Plant specific documents:<br><br>— The site evaluation (from the final safety analysis report or similar safety document);<br>— The list of SSCs important to safety and their safety classification;<br>— The documented design basis (original and updated) including the list of postulated initiating events and specific environmental parameters;<br>— The list of equipment covered by the equipment qualification programme and the procedure for control of this list;<br>— Equipment qualification report and other supporting documents (e.g. equipment qualification specifications and qualification plan);<br>— Records of all qualification measures taken during the installed service life of the equipment.<br><br>Operating experience:<br><br>— Operating experience from similar plants in the State and in other States<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | The review of equipment qualification may lead to findings in one or more of the following areas:<br><br>— The equipment qualification programme, its procedures (including for design extension conditions) and records;<br>— The final safety analysis report;<br>— Environmental conditions;<br>— Maintenance and ageing management programmes.<br><br>Findings in the review of equipment qualification may result in one or more of the following:<br><br>— Equipment qualification is adequate or justification is necessary;<br>— Additional qualification or protection is needed for particular components;<br>— Proposal for replacement of particular SSCs;<br>— Improvements to the maintenance programme;<br>— Improvements to the ageing management programme.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

## RELEVANT IAEA PUBLICATIONS


Ageing Management for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-48, IAEA, Vienna (2018).

Equipment Qualification for Nuclear Installations, IAEA Safety Standards Series No. SSG-69, IAEA, Vienna (2021).

Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA Safety Standards Series, IAEA Safety Standards Series No. SSG-30, IAEA, Vienna (2014).

Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), IAEA, Vienna (2016).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

Seismic Design for Nuclear Installations, IAEA Safety Standards Series No. SSG-67, IAEA, Vienna (2021).

## SAFETY FACTOR 4: AGEING

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards on ageing management;<br>— Relevant guidance on the management of plant ageing and record keeping.<br><br>Plant specific documents:<br><br>— Manuals on ageing management used by the operating organization;<br>— Documentation on the method and criteria for identifying SSCs important to safety covered by the ageing management programme;<br>— The list of SSCs important to safety covered by the ageing management programme and records that provide information in support of the management of ageing;<br>— Data for assessing ageing degradation, including baseline data and operating and maintenance histories.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I), for example in the area of operating history. | The review of ageing may lead to findings in one or more of the following areas:<br><br>— The rapidity of the ageing process;<br>— Plant design review.<br><br>Examples of outputs are:<br><br>— Proposals for replacement of particular SSCs important to safety;<br>— Improvements to the maintenance programme;<br>— Improvements to the ageing management programme.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

## RELEVANT IAEA PUBLICATIONS


Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-48, IAEA, Vienna (2018).

Periodic Safety Review of Nuclear Power Plants: Experience of Member States, IAEA-TECDOC-1643, IAEA, Vienna (2010).

Ageing Management and Long Term Operation of Nuclear Power Plants: Data Management, Scope Setting, Plant Programmes and Documentation, Safety Reports Series No. 106, IAEA, Vienna (2022).

Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), Safety Reports Series No. 82 (Rev. 2), IAEA, Vienna (2024).

Regulatory Oversight of Ageing Management and Long Term Operation Programme of Nuclear Power Plants, Safety Reports Series No. 109, IAEA, Vienna (2022).

Use of Periodic Safety Review for Long Term Operation of Nuclear Power Plants, Safety Reports Series No. 121, IAEA, Vienna (2024).

Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), IAEA, Vienna (2016).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

SAFETY FACTOR 5: DETERMINISTIC SAFETY ANALYSIS

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current national and international guidelines for deterministic safety analysis, including guidelines for application of the single failure criterion and for redundancy, diversity and separation of SSCs important to safety.<br><br>Plant specific documents:<br><br>— The final safety analysis report, if available;<br>— Compilation of the existing deterministic safety analysis and the assumptions used;<br>— Operational limits and conditions and permitted operational states of the plant;<br>— Anticipated operational occurrences, including the list of all postulated initiating events that could affect the safety of the plant;<br>— Analytical methods and computer codes used in deterministic safety analysis and comparable current methods (e. g. those for use for a modern nuclear power plant), including their validation;<br>— Calculated radiation doses and limits on releases of radioactive material for design basis accidents.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | Examples of outputs are:<br><br>— New postulated initiating events;<br>— Revised operational limits and conditions;<br>— Correctness of the assumptions used in the analysis;<br>— Assessment of the capability of the design to provide for defence in depth;<br>— Proposed improvements to the deterministic analysis methodologies and/or modelling.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

RELEVANT IAEA PUBLICATIONS

Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-54, IAEA, Vienna (2019).

Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Vienna (2016).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

Design Extension Conditions and the Concept of Practical Elimination in the Design of Nuclear Power Plants, SSG-88, IAEA, Vienna (2024).

Format and Content of the Safety Analysis Report for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-61, IAEA, Vienna (2021).

## SAFETY FACTOR 6: PROBABILISTIC SAFETY ASSESSMENT

| Inputs | Outputs |
|---|---|
| Standards and requirements<br><br>— Current national and international guidelines and codes for PSA, in particular those addressing operator actions, common cause events, cross-link effects and redundancy and diversity of SSCs important to safety.<br><br>Plant specific documents:<br><br>— Existing PSA documentation and models, including those used in risk informed applications of the PSA;<br>— Postulated initiating events (those used for the existing PSA and a comparable list for a modern nuclear power plant);<br>— Reports of external peer reviews and/or independent reviews;<br>— A compilation or selection of guidelines, assessment principles, standards, regulatory requirements, etc. that represent what is considered the ‘current standard’ in performance of the PSA and the best practices known, available and applicable (all these are used to derive criteria for the review of PSA);<br>— The accident management programme for design extension conditions together with results of the PSA.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | Examples of outputs are:<br><br>— Revised operational limits and conditions;<br>— Analysis of the correctness of the assumptions used in the analysis;<br>— Assessment of the capability of the design to provide for defence in depth;<br>— Proposed improvements to the deterministic analysis methodologies and/or modelling;<br>— Assessment of the adequacy of the accident management programme;<br>— Identification of operational activities that are significant to safety;<br>— Improvements to the PSA reliability database.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

## RELEVANT IAEA AND OECD/NEA PUBLICATIONS


Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3 (Rev. 1), IAEA, Vienna (2024).

Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4 (Rev. 1), IAEA, Vienna (in publication).

Human Reliability Analysis for Nuclear Installations, IAEA Safety Reports Series No 127 [IAEA Preprint] (2024).

Accident Management Programmes in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-54, IAEA, Vienna (2019).

OECD NUCLEAR ENERGY AGENCY, Level 2 PSA Methodology and Severe Accident Management, Rep. OECD/GD (97)198, OECD, Paris (1997).

Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

## SAFETY FACTOR 7: HAZARD ANALYSIS

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and national and international design codes, and safety assessment codes and standards;<br>— Regulatory requirements;<br>— Control procedures, safety assessment processes and procedures of the operating organization.<br><br>Plant specific (and site specific) documents:<br><br>— Results of previous hazards analyses;<br>— Flood risk assessments;<br>— Climate change assessments;<br>— Seismic assessments and records;<br>— Fire protection plans;<br>— PSA assumptions (where used);<br>— Emergency plans;<br>— Data from on-site monitoring systems (e.g. meteorology tower and seismic monitoring network)<br>— Local patterns or trends of aircraft movement and records of overflight incidents;<br>— Recent planning applications (future changes in industrial or transport activity near the plant);<br>— Records of wind speeds and direction;<br>— Records of volcanic activity and hazards;<br>— Records of ambient and sea and river temperature;<br>— Records of river and sea levels;<br>— Records of meteorological hazards;<br>— Records of hydrological hazards.<br><br>Operating experience<br><br>— Operating experience from similar plants or sites, both in the State and in other States;<br>— Records of hazard incidents affecting the plant.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | Findings from the review of the hazard analysis may include one or more of the following:<br><br>— The design basis assumptions will not be significantly challenged by internal or external hazards until at least the next PSR;<br>— The need for reassessment of safety margins;<br>— Procedures for mitigating the consequences of hazards need to be improved;<br>— Equipment qualification needs to be reassessed;<br>— Modifications are necessary to detect hazards or to improve mitigation of the consequences of hazards, for example, flood barriers need to be raised;<br>— Additional monitoring and improved record keeping is necessary;<br>— Updates to the final safety analysis report are needed;<br>— Plant modification processes or maintenance procedures do not take adequate account of requirements for hazards qualification.<br><br>Results from the review of this safety factor may provide inputs to other safety factors (see Appendix I). |

## RELEVANT IAEA PUBLICATIONS

Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSR-1, IAEA, Vienna (2019).

Evaluation of Seismic Safety for Nuclear Installations, IAEA Safety Standards Series No. SSG-89, IAEA, Vienna (2024).

Design of Nuclear Installations Against External Events Excluding Earthquakes, IAEA Safety Standards Series No. SSG-68, IAEA, Vienna (2021).

Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-79, IAEA, Vienna (2023).

Meteorological and Hydrological Hazards in Site Evaluations for Nuclear Installations, IAEA Safety Standards Series No. SSG-18, IAEA, Vienna (2011) (currently under revision as DS541).

Protection against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, IAEA, Vienna (2021).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 (Rev. 1), IAEA, Vienna (2022).

Volcanic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-21, IAEA, Vienna (2012).

Investigation of Site Characteristics and Evaluation of Radiation Risks to the Public and the Environment in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. DS529 (in preparation).

Geotechnical Aspects in Siting and Design of Nuclear Installations, IAEA Safety Standards Series No. DS531, IAEA, Vienna (in preparation).

## SAFETY FACTOR 8: SAFETY PERFORMANCE

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and operating practices.<br>— Regulatory requirements.<br><br>Operating experience:<br><br>— Best international practice in the use of safety performance indicators developed by the IAEA and WANO.<br><br>Plant specific documents:<br><br>— Records of operating experience relevant to safety, including the following:<br>• Frequency of unplanned trips while the reactor is critical;<br>• Frequency of unplanned operator actions in the interests of safety and their success rate;<br>• Selected actuations of and/or demands on safety systems;<br>• Failures of safety systems;<br>• Unavailability of safety systems;<br>• Trends in causes of failures (for example, operator errors, hardware faults);<br>• The backlog of outstanding maintenance and configuration management;<br>• The extent of repeat maintenance;<br>• The extent of corrective (breakdown) maintenance;<br>• The integrity of physical barriers for the containment of radioactive material;<br>• Radiation doses to persons on the site (including collective doses);<br>• ~~Data from off-site radiation monitoring;~~<br>• The annual rate of generation of radioactive waste and the quantity of waste stored on the site;<br>~~• Quantities of radioactive effluents produced;~~<br>• Reports on the routine analysis of safety performance indicators;<br>• Procedures, documentation and outputs from the plant's routine processes for the review of operating experience~~.~~;<br>• Incidents and events with consequences relevant to the future transition to decommissioning and decommissioning, including site cleanup;<br>• Any dismantling, decontamination and radioactive waste management activities already performed for on-site structures and buildings;<br>• Any potential presence of and contamination of underground structures (e.g. pipes and tanks), | The review of safety performance may lead to findings in one or more of the following areas:<br><br>— Training relating to safety performance;<br>— Plant processes and procedures, for example, operating procedures, maintenance procedures;<br>— Safety culture;<br>— The final safety analysis report;<br>— Strengths and weaknesses demonstrated by performance indicators;<br>— Input data for the PSA.<br><br>Results from the review of this safety factor may provide inputs to other safety factors (see Appendix I). |

| | |
|---|---|
| any groundwater contamination and surface contamination as well as any non-radiological contaminants that might require cleanup during transition to decommissioning or decommissioning, including site cleanup.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | |

## RELEVANT IAEA PUBLICATIONS

Operating Experience Feedback for Nuclear Installations, IAEA Safety Standards Series No. SSG-50, IAEA, Vienna (2018).

Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-74, IAEA, Vienna (2022).

Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-70, IAEA, Vienna (2022).

Operational Safety Performance Indicators for Nuclear Power Plants, IAEA-TECDOC-1141, IAEA, Vienna (2000).

Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40, IAEA, Vienna (2016).

Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, IAEA, Vienna (2018).

Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), IAEA, Vienna (2016).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

## SAFETY FACTOR 9: FEEDBACK OF OPERATING EXPERIENCE

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and regulatory requirements;<br>— Relevant guidelines from the OECD/NEA, WANO and INPO.<br><br>Operating experience:<br><br>— International databases collecting operating experience, such as the IAEA's International Reporting System for Operating Experience (IRS) database and databases of WANO, INPO and owners' groups;<br>— Highlight reports and topical studies of the IRS and 'Significant Event Reports' and 'Significant Operating Experience Reports' issued by WANO;<br>— Operating experience from similar plants in the State and in other States.<br><br>Plant specific documents:<br><br>The review of the use of experience from other plants and research finding includes the following plant specific inputs:<br><br>— Reports from the operating organization's routine assessment of operating experience at other plants;<br>— Procedures and documentation governing the operating organization's process for the review of operating experience at other plants;<br>— Assessments from the operating organization's review of emerging research findings;<br>— Procedures and documentation governing the operating organization's routine process for the assessment of research findings;<br>— Independent internal or external audits and self-assessments regarding operating experience and research findings.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations, provides examples of typical outcomes for this safety factor. Additional outcomes could include:<br><br>— Proposals for improving arrangements for receipt of operating experience feedback from other plants;<br>— Proposals for improved dissemination of operating experience feedback within the operating organization;<br>— Arrangements for the receipt of findings from relevant research programmes (including international programmes).<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). This safety factor needs to be reviewed early in the PSR programme. |

## RELEVANT IAEA PUBLICATION

Operating Experience Feedback for Nuclear Installations, IAEA Safety Standards Series No. SSG-50, IAEA, Vienna (2018).

Improving the International System for Operating Experience Feedback, INSAG-23, IAEA, Vienna (2008).

SAFETY FACTOR 10: ORGANIZATION, THE MANAGEMENT SYSTEM AND SAFETY CULTURE

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and regulatory requirements;<br>— Current national and international operating practices.<br><br>Plant specific documents:<br><br>— The operating organization's safety policy and related documentation;<br>— Procedures and documentation of the management system (for example, on quality management, configuration management and ageing management);<br>— Outputs from application of management system procedures, including quality plans;<br>— Records (for example, on training, commissioning, maintenance, testing);<br>— Documentation describing the organizational structure and safety related roles and responsibilities of individuals and groups;<br><br>Corrective action programme and processes<br><br>— For reporting of events;<br>— Surveys of safety culture.<br><br>Operating experience:<br><br>— Operating experience with respect to the organization and administration at plants in the State and in other States;<br>— Internal audit and surveillance reports;<br>— External audits (for example, reports from IAEA Operational Safety Review Team (OSART) missions);<br>— Self-assessments;<br>— Safety performance assessments;<br>— Safety culture assessments.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | The review of organization, the management system and safety culture may lead to findings in one or more of the following areas:<br><br>— Clarity of policy statements;<br>— Adequacy of the documentation of the management system;<br>— Structure of the operating organization;<br>— Work processes (how work is specified, prepared, reviewed, performed, recorded, assessed and improved);<br>— Control of documents, products and records;<br>— The purchasing process;<br>— Communication;<br>— Organizational change management;<br>— Commitment to safety;<br>— Compliance with procedures;<br>— The existence of a questioning attitude among personnel;<br>— Whether the operating organization has a 'learning culture';<br>— Prioritization of safety issues;<br><br>Clarity of roles and responsibilities;<br><br>— Training on safety culture;<br>— Regular safety culture assessments.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

RELEVANT IAEA/INSAG PUBLICATIONS

Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

Format and Content of the Safety Analysis Report for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-61, IAEA, Vienna (2021).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Key Practical Issues in Strengthening Safety Culture, INSAG-15, IAEA, Vienna (2002).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Maintaining Knowledge, Training and Infrastructure for Research and Development in Nuclear Safety, INSAG-16, Vienna, IAEA (2003).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Management of Operational Safety in Nuclear Power Plants, INSAG-13, IAEA, Vienna (1999).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Safe Management of the Operating Lifetimes of Nuclear Power Plants, INSAG-14, IAEA, Vienna (1999).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Safety Culture, INSAG-4, IAEA, Vienna (1991).

Recruitment, Qualification and Training of Personnel for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-75, IAEA, Vienna (2022).

Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), IAEA, Vienna (2016).

Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5, IAEA, Vienna (2009).

The Operating Organization for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-72, IAEA, Vienna (2022).

SAFETY FACTOR 11: OPERATIONAL LIMITS AND CONDITIONS AND OPERATING PROCEDURES

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and regulatory requirements for procedures;<br>— Current national and international operating practices in procedures.<br><br>Plant specific documents:<br><br>— Plant operating procedures for normal operation, fault conditions and symptom-based emergency operating procedures for restoring critical safety functions;<br>— Procedures supporting plant operating procedures (e.g. for their development, validation, acceptance, modification withdrawal);<br>— Audits and self-assessments that question adherence to plant procedures.<br><br>Operating experience:<br><br>— Operating experience involving procedural issues at plants in the State and in other States;<br>— Safety significant events involving procedural issues.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | The review of procedures may lead to findings in one or more of the following areas:<br><br>— The process for development, elaboration, validation, acceptance, modification, and withdrawal of procedures;<br>— Clarity of procedures;<br>— Compliance with procedures;<br>— Effectiveness and adequacy of procedures;<br>— Safety culture.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

RELEVANT IAEA/INSAG PUBLICATIONS


Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

Conduct of Operations at Nuclear Power Plants, IAEA Safety Standards Series No. SSG-76, IAEA, Vienna (2022).

Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2022).

Format and Content of the Safety Analysis Report for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-61, IAEA, Vienna (2021).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Maintaining Knowledge, Training and Infrastructure for Research and Development in Nuclear Safety, INSAG-16, IAEA, Vienna (2003).

Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-74, IAEA, Vienna (2022).

Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-70, IAEA, Vienna (2022).

Protection against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, IAEA, Vienna (2021).

Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), IAEA, Vienna (2016).

Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-54, IAEA, Vienna (2019).

Leadership and Management for Safety, IAEA Safety Standards Series No GSR Part 2, IAEA, Vienna (2016).

The Operating Organization for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-72, IAEA, Vienna (2022).

## SAFETY FACTOR 12: HUMAN FACTORS

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and regulatory requirements;<br>— Current national and international operating practices for ensuring that human factors do not affect the safe operation of the nuclear power plant.<br><br>Plant specific documents:<br><br>— Policy to maintain the knowledge of the plant staff;<br>— Training records, also for training in safety culture, particularly for staff in management positions;<br>— Staffing records;<br>— Fitness for duty requirements;<br>— Programmes for the feedback of operating experience for failures and/or errors in human performance that have contributed to safety significant events and their causes, and consequent corrective actions and/or safety improvements;<br>— Audits and self-assessments of hours of work and time records.<br><br>Operating experience:<br><br>— Operating experience involving human factors at plants in the State and in other States;<br>— Safety significant events involving human factors.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I). | The review of human factors may lead to findings in one or more of the following areas:<br><br>— Staffing levels;<br>— Training programmes;<br>— Operating, maintenance and engineering practices;<br>— Competency management;<br>— Staff selection and recruitment and succession management;<br>— Knowledge management;<br>— Use of external technical resources;<br>— The human–machine interface;<br>— Communications.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

## RELEVANT IAEA/INSAG PUBLICATIONS


Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

Human Reliability Analysis for Nuclear Installations, IAEA Safety Reports Series No 127 [IAEA Preprint] (2024).

INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Safety Culture, INSAG-4, IAEA, Vienna (1991).

Recruitment, Qualification and Training of Personnel for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-75, IAEA, Vienna (2022).

Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5, IAEA, Vienna (2009).

The Operating Organization for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-72, IAEA, Vienna (2022).

Performing Safety Culture Self-Assessments, IAEA Safety Reports Series No. 83, IAEA, Vienna (2016).

## SAFETY FACTOR 13: EMERGENCY PLANNING

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Current safety standards and regulatory requirements on emergency planning.<br><br>Plant specific documents:<br><br>— The emergency plan of the operating organization;<br>— Strategy, procedures and organization for emergencies;<br>— Studies of the mitigation of consequences of accidents;<br>— Procedures for the management of design extension conditions and accident management guidelines.<br><br>Operating experience:<br><br>— Records of emergency exercises held and lessons learned;<br>— Lessons learned from exercises held in the State and in other States and from international exercises.<br><br>The review of this safety factor may require input from other safety factors (see Appendix I), particularly input from the review of PSA if appropriate analyses are available (Level 3 PSA or at least Level 2 PSA). | The review of emergency planning may lead to findings in one or more of the following areas:<br><br>— Status of the emergency preparedness of the plant;<br>— Confirmation that an effective emergency planning process is in place;<br>— Technical and/or administrative improvements for communication with external bodies are necessary;<br>— Emergency planning and response training with other organizations needs to be improved;<br>— Emergency plans need to be updated in accordance with the results of current safety analyses, accident mitigation studies and good practices.<br><br>Results from the review of this safety factor may provide inputs for other safety factors (see Appendix I). |

## RELEVANT IAEA PUBLICATIONS


Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7, IAEA, Vienna (2015).

Arrangements for Preparedness for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007).

Conduct of Operations at Nuclear Power Plants, Safety Standards Series No. SSG-76, IAEA, Vienna (2022).

Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident, IAEA-TECDOC-955, IAEA, Vienna (1997).

Preparation, Conduct and Evaluation of Exercises to Test Preparedness for a Nuclear or Radiological Emergency, IAEA-EPR-EXERCISE, IAEA, Vienna (2006).

Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-2, IAEA, Vienna (2011).

Arrangements for the Termination of a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-11, IAEA, Vienna (2020).

Arrangements for Public Communication in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-14, IAEA, Vienna (2020).

Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency, IAEA-EPR-METHOD, IAEA, Vienna (2003).

## SAFETY FACTOR 14: RADIOLOGICAL IMPACT ON THE ENVIRONMENT

| Inputs | Outputs |
|---|---|
| Standards and requirements:<br><br>— Relevant national standards;<br>— IAEA Safety Requirements publications and Safety Guides, including SSR-2/1 (Rev. 1), SSG-90 and DS529;<br>— Relevant guidelines from the OECD/NEA, WANO and INPO.<br><br>Plant specific documents:<br><br>— Potential sources of radiological impact;<br>— Authorized limits for discharges;<br>— Off-site monitoring for contamination levels and radiation levels;<br>— Availability of alarm systems to respond to unplanned releases of effluents from on-site facilities;<br>— Recent and future changes in the use of areas around the site;<br>— Records and quantities of discharges of effluents;<br>— Records and data from off-site environmental monitoring;<br>— Published environmental data.<br><br>The review of this safety factor may require input from other safety factors<br><br>(see Appendix I), particularly from the reviews of plant design and of safety performance | The review of radiological impact on the environment may lead to findings in one or more of the following areas:<br><br>— Monitoring equipment, programmes and methods for monitoring the radiological impact of the plant on the environment;<br>— Assessment of the environmental impact and public exposure during normal operation;<br>— Control of ~~effluents~~ discharges of effluents;<br>— Optimization of protection and safety in relation to public exposure.<br><br>Results from the review of this safety factor may provide inputs to the reviews of all the other safety factors (see Appendix I). |

## RELEVANT IAEA PUBLICATIONS


Operating Experience Feedback for Nuclear Installations, IAEA Safety Standards Series No. SSG-50, IAEA, Vienna (2018).

Investigation of Site Characteristics and Evaluation of Radiation Risks to the Public and the Environment in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. DS529, IAEA, Vienna (in preparation).

Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40, Vienna (2016).

Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, Vienna (2018).

Radiation Protection Aspects of Design for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-90, IAEA, Vienna (2024).

Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

# CONTRIBUTORS TO DRAFTING AND REVIEW

| | |
|---|---|
| Alm-Lytz, K. | International Atomic Energy Agency |
| Barquin Duena, A. | Consejo de Seguridad Nuclear, Spain |
| Bertocchi, F. | Nuclear Research and Consultancy Group, Netherlands |
| Bouhrizi, S. | Électricité de France, France |
| Bznuni, S. | International Atomic Energy Agency |
| Duchac, A. | International Atomic Energy Agency |
| Gajdos, M. | International Atomic Energy Agency |
| Lipar, M. | Consultant, Slovakia |
| Machacek, J. | CEZ a.s., Jaderna Elektrarna Temelin, Czech Republic |
| Murakami, K. | University of Tokyo, Japan |
| Nieto Mora, J. | Centrales Nucleares Almaraz-Trillo, Spain |
| Salmanov, R. | Framatome, Germany |
| Shaw, P. | International Atomic Energy Agency |
| Sun, K. | International Atomic Energy Agency |
| Walz, D. | Framatome, Germany |
| Zaidi, I. | International Atomic Energy Agency |