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Safety Demonstration of Innovative Technology in Nuclear Power Plants

(DRAFT)

SPECIFIC SAFETY GUIDE

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. New nuclear power plants may employ new approaches and concepts at a component, system and facility level that are different from those current used at conventional nuclear power plants[1]. These new approaches and concepts are referred to in this Safety Guide as innovative technology, which is defined in detail in Section 2. Although nuclear power plants using innovative technology may also incorporate known engineering practices and utilize existing designs for their structures, systems and components (SSCs), nuclear power plants using innovative technology have not yet reached the same level of maturity as the current proven designs, for example with respect to regulatory scrutiny, operating experience and knowledge.

1.2. The IAEA has conducted a high level review of the applicability of its safety standards to various technologies, including small modular reactors and non-water-cooled reactors The results of this review were published as Ref. [1]. The review concluded that, in general, the IAEA safety standards relating to safety assessment (i.e. IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2] and supporting Safety Guides) are applicable, however there are areas that need further enhancement in relation to nuclear power plants using innovative technology, as described in Ref. [1].

1.3. There are specific challenges connected with the safety demonstration[2] for innovative technology in nuclear power plants. These challenges are in particular connected to such aspects as limited knowledge of phenomena relevant to innovative technology and associated uncertainties, lack of adequate simulation tools, limited (or no) operating experience, lack of applicable regulations, codes and standards, and issues with the application of traditional safety assessment approaches. These challenges are explained in more detail in para. 2.11 and are expected to be addressed for adequate safety demonstration.

OBJECTIVE

1.4. The objective of this Safety Guide is to provide recommendations on approaches to address challenges associated with innovative technology in safety demonstrations for nuclear power plants.

1.5. This Safety Guide also provides recommendations on the use of specific strategies for the safety demonstration of innovations such as new fuel concepts, new coolants, innovative safety features, innovative modes of operation, innovative materials, and advanced manufacturing techniques.

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[1] In the context of this Safety Guide the term ‘conventional nuclear power plants’ includes commercial nuclear power plants that have already been built and operated or are under construction and which could not be considered innovative technologies as it is specified in paragraph 2.4.

[2] For the purposes of this Safety Guide, a term ‘safety demonstration’ refers to a comprehensive process to validate and substantiate the safety claims made during the NPPs design, which considers findings of a safety assessment supported by a statement of confidence in these findings (see safety case), and implies acceptance of interested parties (e.g. designer, operating organisation, regulatory body).

SCOPE

1.6. This Safety Guide focuses on specific issues for safety demonstration related to the introduction of innovative technology in nuclear power plants. In particular, it covers innovations for commercial NPPs as well as innovative and first-of-a-kind designs, which are not as mature as current (proven) designs with respect to knowledge on relevant phenomena, regulatory scrutiny and operating experience.

1.7. The recommendations provided are applicable to the safety demonstration of a wide range of nuclear power plant designs in which innovative technology is used, from currently operating water cooled NPPs to small modular reactors and to non-water cooled reactors.

1.8. The Safety Guide is intended to complement existing IAEA safety standards (not to substitute), in particular focusing on areas in which the applicability of such standards might be unclear, or which might benefit from additional recommendations in relation to interpretation and the application of a graded approach. Such safety standards include GSR Part 4 (Rev. 1) [2], IAEA Safety Standards Series Nos SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3], SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [4], and supporting Safety Guides. The safety implications of innovative technologies used in nuclear power plants, or their operation can often be assessed against these standards with judgement. Importantly, the recommendations in this Guide should not be construed as allowing for any weakening of the safety requirements in GSR Part 4 (Rev. 1) [2], SSR-2/1(Rev. 1) [3] and SSR-2/2 (Rev. 1) [4].

1.9. The following are out of scope of this Safety Guide:

– Innovations not having safety implications;
– Non-innovative technology (i.e. that does not fall under the concept of ‘innovative’ as described in Section 2);
– Nuclear installations, other than nuclear power plants[3];
– Aspects related to the waste and decommissioning of NPPs with innovative technology. Except for the ‘by design’ aspects which have to be considered in the safety demonstration in the design stage, in accordance to the Requirement 12 in SSR-2/1 (Rev. 1) [3], which are addressed.

STRUCTURE

1.10. Section 2 defines innovative technology and briefly describes the issues related to the safety demonstration of specific aspects of innovative technology. Section 3 provides recommendations on approaches to the safety demonstration of nuclear power plants that employ innovative technology, focusing on the identification and management of relevant issues, and the application of a graded approach.

1.11. Sections 4 and 5 complement each other, with systematic cross-connections between the recommendations provided in the two sections. Section 4 provides specific recommendations on potential strategies to address the challenges for safety demonstration when there are issues with

[3] Recommendations in this guide might also be considered for the safety assessment of innovations in nuclear installations other than NPPs (e.g. research reactors), if these recommendations are adapted to the applicable safety requirements and the necessary judgement is exercised.

the practical implementation of existing approaches for safety assessment, limited knowledge of relevant phenomena, a lack of suitable computer codes and simulation tools, limited operating experience and a lack of applicable regulations, codes and technical standards. Using the strategies described in Section 4, Section 5 provides concrete recommendations for the safety demonstration of specific innovations such as innovative fuel concepts, non-water reactor coolants, innovative human–machine interfaces, innovative instrumentation and control systems, advanced manufacturing techniques, cogeneration applications, multi-modularity, and transportable nuclear power plants. Section 6 provides recommendations on the consideration of interfaces between safety measures, security measures and safeguards arrangements during the safety demonstration of innovative technology in nuclear power plants.

2. DEFINITION OF INNOVATIVE TECHNOLOGY IN NUCLEAR POWER PLANTS AND THE CHALLENGES IT POSES

2.1. The IAEA safety standards make reference to the use of proven technologies and practices. In particular Requirement 9 para. 4.14 of SSR-2/1 (Rev. 1) [3] states that

> "Items important to safety for a nuclear power plant shall preferably be of a design that has previously been proven in equivalent applications, and if not, shall be items of high quality and of a technology that has been qualified and tested."

2.2. At the same time, the design and operation of nuclear power plants continue to improve with advanced technology. In this context, Requirement 9 para. 4.16 of SSR-2/1 (Rev. 1) [3] states:

> "Where an unproven design or feature is introduced or where there is a departure from an established engineering practice, safety shall be demonstrated by means of appropriate supporting research programmes, performance tests with specific acceptance criteria or the examination of operating experience from other relevant applications. The new design or feature or new practice shall also be adequately tested to the extent practicable before being brought into service and shall be monitored in service to verify that the behaviour of the plant is as expected."

2.3. Similarly, in a number of IAEA safety standards, the introduction of new and innovative features or design solutions is taken into account through recommendations to apply the safety standards with judgement considering the specific context.

2.4. An innovation in the context of the safety demonstration of nuclear power plants is considered to be a new type for an SSC or a specific mode of operation relevant to safety that has not previously been used or is used in a new way, for which:

(a) Proven engineering practices for nuclear power plants are not fully defined; or
(b) Existing practices or safety standards need to be interpreted, and judgement used for their application.

The degree of innovation could vary from the evolutionary changes of existing solutions with some new characteristics to technologies with new characteristics or properties previously not used in nuclear power plants.

2.5. New nuclear power plant designs with innovations that includes conceptual changes compared to conventional NPPs is referred to as an innovative design (see also Ref. [5]). For operating reactors, the introduction of innovative technology has a more limited scope, but the definition applies as well to any modifications using innovations.

2.6. For the purposes of this Safety Guide, an innovative technology is any innovation in a nuclear power plant, including new modes of operation, that is relevant to safety. This guide does not define separate categories of innovative technology, instead referring to the degree of innovation or to innovative designs where appropriate.

2.7. Requirement 6 of SSR-2/1(Rev. 1) [3] states:

> "**The design for a nuclear power plant shall ensure that the plant and items important to safety have the appropriate characteristics to ensure that safety functions can be performed with the necessary reliability, that the plant can be operated safely within the operational limits and conditions for the full duration of its design life and can be safely decommissioned, and that impacts on the environment are minimized.**"

2.8. Requirement 24 of IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [6] states that "**The applicant shall be required to submit an adequate demonstration of safety in support of an application for the authorization of a facility or an activity.**"

2.9. While the safety impact of some innovations can be evaluated using established safety standards and proven engineering practices for nuclear power plants, some innovative technologies need additional consideration. In evaluating the safety performance of innovative technology, vendors, operating organizations and regulatory bodies might face the following challenges, among others:

(a) Lack of knowledge about performance and potential safety impacts;
(b) Increased uncertainties due to the innovation;
(c) Difficulties in performing a safety assessment or even defining what an acceptable safety assessment would include.

Such challenges should be handled in a cautionary way for the safety assessment to be convincing. Recommendations on achieving this are provided in further sections of this Safety Guide.

2.10. Innovative technologies include first-of-a-kind technological features or the use of an existing technology for different or novel purposes, in a different way, or in a different mode of operation. Innovative technology is expected to pose specific challenges to the safety demonstration for one or more of the following reasons:

(a) **Use of proven engineering practices in a different context**: while there might be sound engineering practices for the technology overall, there is not sufficient engineering practice for its specific use in nuclear power plants.
(b) **Technological solutions and working principles:** the design and its working principles are new for nuclear applications or for specific use in the nuclear context. The performance of the technology under operating conditions as well as accident conditions is therefore difficult to predict.
(c) **Surveillance, inspection and maintenance**: the technologies, processes or approaches for the surveillance, inspection or maintenance of SSCs are innovative or have not been used previously for nuclear power plants.
(d) **Phenomena:** the phenomena of working media, materials or nuclear fuel used in the technological solution are not well understood, difficult to predict or there is a lack of knowledge about relevant physical and chemical phenomena of working fluids, materials, or nuclear fuels under the specific 'new' conditions, for which they are being proposed;

(e) **Computer codes:** the computer codes needed to support the safety assessment lack models or methods relevant to the technology or its specific use, are not sufficiently predictive, or are not sufficiently validated for their intended use.

(f) **Operating experience**: there is a lack of operating experience for the innovative technology or its specific use, and existing operating experience for similar technologies is not transferrable to use in nuclear power plants.

(g) **Regulatory frameworks and codes and standards**: there are few or no established regulatory positions pertinent to the innovative technology or its specific use. Regulatory guides or available codes and standards are not sufficiently applicable without considerable levels of interpretation and judgement, or there are gaps in regulatory frameworks, codes and standards. Safety assessment approaches meeting regulatory expectations might be missing or difficult to achieve.

(h) **Materials**: the materials used have not previously been used in a nuclear context or in the same conditions as those proposed, so there is a lack of knowledge of relevant failure mechanisms, corrosion, wear and tear, and ageing mechanisms. Codes and standards do not always cover such materials or the specific conditions they are used in. The supply chain for such materials might be new, and there is a lack of a transferrable track record.

(i) **Nuclear fuels**: the performance of new types of nuclear fuel or fuel types used under different conditions, the behaviour of which during operation, accident conditions, and the fuel lifetime are not well understood. In addition, claims for inherent safety of the fuel might have implications for other levels of defence in depth.

(j) **Reactor coolant and working medium:** reactor coolants or other working media are used that have not previously been used in a nuclear context, or are used in conditions leading to new effects and phenomena (e.g. as a supercritical fluid). In these cases, there might be a lack of knowledge of some characteristics of such media relevant to safety.

(k) **Instrumentation and control**: the instrumentation and control systems are based on innovative technology or approaches not previously used for nuclear power plants.

(l) **Approaches to reactor operation:** there are new modes of operation (e.g. remote operation) or new expectations on operator interactions with the innovative technology compared to existing practices, and approaches not previously used for nuclear power plants.

(m) **Approaches to human–machine interfaces:** human–machine interfaces are based on principles and approaches not previously used for nuclear power plants.

(n) **Manufacturing and construction**: the processes used to manufacture or construct the technology or some of its parts are innovative for a nuclear power plant. Examples may include additive manufacturing or modular construction techniques. The impact on failure mechanisms, reliability and ageing of the technology are not fully known or there are relevant impacts of the manufacturing or construction processes on the safety of the plant.

(o) **Non-electrical applications**: a part of the nuclear power produced is used for non-electrical applications, and there are novel feedbacks and interactions between the non-nuclear energy conversion and its utilization and the nuclear reactor system, which are relevant to safety.

(p) **Multi-module designs**: the unit consists of several modules in close proximity, which might share structures or systems, in which case the safety assessments need to consider the impact that each module can have on the others.

(q) **Transportable concepts or siting concepts**: the transport of factory sealed cores or an entire transportable nuclear power plant poses additional challenges for the safety demonstration in the phases of manufacturing, commissioning, transport, operation, refuelling and decommissioning. In addition, siting at novel locations or in novel environments (e.g. on the sea floor) might have an impact on safety demonstration.

(r) **Possible radiation risks**: the use of innovative technologies may result in possible radiation risks that are different to conventional NPPs and might influence the application of graded approach in safety assessment (e.g. because power levels and radioactive inventories are very small as in a micro-reactor design).

2.11. The aspects listed in para. 2.10 might lead to various degrees of uncertainty and various unknowns. Their implications should be understood and assessed.

3. GENERAL APPROACHES TO SAFETY DEMONSTRATION FOR INNOVATIVE TECHNOLOGY

IDENTIFICATION OF ISSUES, KNOWLEDGE GAPS AND UNCERTAINTIES

3.1. Paragraph 3.15 of IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [7] states that:

> "Safety assessments cover the safety measures necessary to control the hazard, and the design and engineered safety features are assessed to demonstrate that they fulfil the safety functions required of them. Where control measures or operator actions are called on to maintain safety, an initial safety assessment has to be carried out to demonstrate that the arrangements made are robust and that they can be relied on. A facility may only be constructed and commissioned or an activity may only be commenced once it has been demonstrated to the satisfaction of the regulatory body that the proposed safety measures are adequate."

3.2. Requirement 10 of GSR Part 4 (Rev. 1) [2] further elaborates on the assessment of engineering aspects, stating that "It **shall be determined in the safety assessment whether a facility or activity uses, to the extent practicable, structures, systems and components of robust and proven design.**"

3.3. GSR Part 4 (Rev. 1) [2] also establishes requirements for the use of innovative technology. In particular, para 4.29 of GSR Part 4 (Rev. 1) [2] states:

> "Where innovative improvements beyond current practices have been incorporated into the design, it shall be determined in the safety assessment whether compliance with the safety requirements has been demonstrated by an appropriate programme of research, analysis and testing complemented by a subsequent programme of monitoring during operation."

3.4. Para. 2.5 of GSR Part 4 (Rev. 1) [2] states that "The concept of the graded approach applies to all aspects of safety assessment, including the scope and the level of detail of the safety assessment required" and also para 3.4 of GSR Part 4 (Rev. 1) [2] states that "Other relevant factors, such as the maturity or complexity of the facility or activity, shall also to be taken into account in a graded approach to safety assessment. ...". In the context of innovative technology in nuclear power plants, this should consider any additional uncertainties associated with the performance of the technology as well as the combined significance of all innovative technologies in the nuclear power plants.

3.5. If the requirements in SSR-2/1 (rev.1) [3] are applied to innovative technologies beyond "land based stationary nuclear power plants with water cooled reactors", these requirements may also be applied, with judgement, to other reactor types, to determine the requirements that have to be considered in developing the design" (IAEA SSR-2/1 (rev.1), para. 1.6) [3]. The necessary judgement needs to ensure that "the safety assessment shall be consistent with the magnitude of the possible radiation risks arising from the facility or activity" (GSR Part 4 (Rev. 1), para 3.3 [2]). The safety assessment of such an innovative technology should be made applying a graded approach commensurate with their possible radiation risks.

3.6. To ensure that an adequate safety assessment is undertaken, it is a systematic and comprehensive identification of the potential issues and knowledge gaps associated with the innovative technology should be performed and documented. The identification of issues and knowledge gaps should consider the full service life and all applicable uses of the technology, taking into account the following:

(a) Potential issues or knowledge gaps might be associated with design, manufacturing qualification, installation, commissioning, operation, including the ageing management, and issues related to decommissioning. All modes of operation (including those to enable examination, maintenance, inspection and testing) and failure modes should be considered when applicable.
(b) Any new hazards associated with the innovation should be identified and their impact on safety should be evaluated.
(c) The extent to which an innovative technology affects the design safety or operational safety of a nuclear power plant depends on its role in the safety architecture of the plant and its significance in the safety case. The implications of changing parts of an established design should be considered to prevent unintended consequences.

3.7. Innovative technology may present a range of uncertainties and unknowns, which should be addressed. Uncertainties should be identified systematically and to the fullest extent practical to ensure that the widest range of information is available to assess the safety implications of the innovations. Specific uncertainties will be dependent on the innovation and its application but may be associated with, for example:

(a) Limitations in the understanding of behaviour (e.g. issues when modelling and predicting the behaviour of materials);
(b) New or unfamiliar failure modes, postulated initiating events, fault conditions or hazards;
(c) Availability and completeness of test data, and whether or not it has been benchmarked;
(d) Availability of reliability data;
(e) Understanding of the limits of safe operation and margins;
(f) Operational experience;
(g) Interfaces and connections between the innovative technology and other technologies used in the design, including incompatibility or conflicting actions;
(h) The human-machine-interface (HMI) used in the design.

3.8. An adequate and proportionate research programme to establish the state of knowledge on issues related to safety such as those mentioned above to identify potential gaps in this knowledge, and to embrace relevant good practices and available information should be implemented.

3.9. The research programme should consider the need to using the following tools:

(a) Testing: establishing a testing programme which can be replicated and repeated, providing recorded results under a range of conditions relevant to the innovation;
(b) Prototyping: producing a functional model of the innovation sufficiently similar to the final product to test theories and claims made;
(c) Modelling: utilizing a modelling technique which can be replicated and repeated, providing recorded results under variable conditions. Also, if the innovation includes connected

technologies, these connections or interfaces should also be modelled to prove that the technology behaves as desired and expected;

3.10. To meet Requirement 6 of GSR Part 2 [13], an integrated management system needs to be put in place that includes a quality management system. The full development process of innovative systems, from design, manufacturing, testing and safety assessment, should meet the quality requirements. In particular a configuration management process should be implemented where all the changes in the design of the systems are documented. The safety assessment related to an innovative system should explicitly refer to a frozen system configuration and the impact on the safety assessment of any further change should be evaluated.

3.11. Innovations might build on or improve existing technologies or practices and adapt them for a new use. In this case, there should be a strong understanding of the intended performance. Where an innovation departs from existing technologies or practices, the vendor, operating organization, and regulatory body should be convinced that any remaining uncertainties do not introduce an unacceptable risk, in which case the innovative technology is not to be implemented.

3.12. While many innovations offer potential safety and operational benefits, improvements to certain aspects of the design or operation are likely to introduce challenges to safety in other areas that need to be managed. Therefore, all relevant benefits and detriments of the innovation should be identified and their impact on safety should be assessed, considering the need for coordination between safety, security and safeguards (see Section 6)

3.13. Following a comprehensive understanding of the innovation and identification of uncertainties and unknowns, the potential safety implications should be evaluated, and, through the safety assessment, it should be ensured that the design is sufficiently robust.

ACTIONS TO MANAGE KNOWLEDGE GAPS AND UNCERTAINTIES FOR SAFETY DEMONSTRATION

3.14. The introduction of innovation into an NPP design will require an understanding of any remaining uncertainties that may affect safety. Several approaches can be employed to reduce these uncertainties and demonstrate the adequacy of the design. These approaches are introduced in the following paragraphs, and further recommendations on specific strategies to address challenges for the safety demonstration of innovative technology are given in Sections 4 and 5.

3.15. The recommendations provided in IAEA Safety Standards Series Nos SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [8], SSG-3 (Rev. 1), Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [9] and SSG-4 (Rev. 1), Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [10] on conservative, best-estimate and realistic approaches for deterministic and probabilistic safety assessment are applicable to innovative nuclear power plant designs. To demonstrate that there are sufficient margins incorporated into the design to account for uncertainties and unknowns, the safety assessment should include a combination of deterministic and probabilistic approaches, with appropriate application of the relevant recommendations from SSG-2 (Rev. 1) [8], SSG-3 (Rev. 1) [9] and SSG-4 (Rev. 1) [10].

3.16. Sensitivity studies should be performed to understand the impact of uncertainties on the safety assessment and to demonstrate sufficient margins. These studies should help to identify the parameters having a relevant impact on the safety and show that a conservative approach has been followed when accounting for their uncertainties, based on the safety significance of the innovation. The sensitivity studies should test the limits of current knowledge and demonstrate the absence of cliff edge effects leading to unacceptable consequences.

3.17. To meet Requirements 11 and 23 of GSR Part 4 (Rev. 1) [2], the safety assessment should be used to underpin operating procedures and rules that define the safe operating envelope of the nuclear power plant at all relevant levels of defence in depth. Where uncertainties associated with innovative technology remain, the safety assessment should investigate the need for conservative limits for operation, an extended commissioning phase or additional tests to mitigate these uncertainties. Based on this analysis further actions should be established, including, where necessary, changes in the nuclear power plant design.

3.18. The role played by safety provisions and the subsequent reliability assessment should consider the cases where there are significant uncertainties regarding the safety performance of a plant design that incorporates innovative technology. The safety assessment should determine if it is appropriate to include compensatory measures and/or features into the design, which will allow to address the uncertainties mentioned above. Such compensatory measures or features may incorporate redundant and/or diverse means for achieving the safety function to ensure that the plant can be returned to a safe, stable state following a fault condition.

3.19. To meet the requirements of SSR-2/1 (Rev. 1) [3], and in line with the recommendations 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 [11], the safety assessment during the design of a nuclear power plant should be used to demonstrate that conditions leading to an early radioactive release or large radioactive release are practically eliminated. Where there are uncertainties that affect the performance or reliability of an innovation relevant to a claim of practical elimination, the safety assessment should demonstrate that the specific phenomena affecting the functionality of an innovation are physically impossible or that the safety measures and design margins are sufficient in order to accommodate these uncertainties and provide high confidence in that demonstration.

3.20. Consideration should be given to incorporating enhanced measures that will enable monitoring of specific parameters to improve knowledge and operating experience and to complement the validation of models used to predict the performance of the innovative technology.

3.21. Consistent with good engineering practice, independent review should be incorporated into the safety assessment. It should be based on a plant configuration and any further plant configuration changes should be assessed, where applicable (see also para. 3.10). Given the use of innovative technology, there may be instances where there is a lack of relevant expertise within vendors, operating organizations, or regulatory bodies to undertake an independent review, for example if the innovation is being deployed for the first time. Independent review and subsequent questioning are important for gaining confidence in the safety assessment. In such cases expert advice from a range of organizations with related experience should be elicited to develop the necessary levels of confidence.

3.22. Expert elicitation should be used, however, in a manner that minimizes bias in expert judgement, which might affect safety conclusions (see also para. 4.147).

3.23. Independent verification and validation of safety claims related to innovative designs should be conducted by experts who are not involved in the development of the design. These experts should comprehensively review novel safety demonstration methods. Special attention should be given to validating assumptions used in safety assessment. This should extend to the results of computer codes and to models used to support the safety assessment.

3.24. Paragraph 3.15 of SF-1 [7] states that "Safety has to be assessed for all facilities and activities, consistent with a graded approach." The graded approach is defined in the IAEA Nuclear Safety and Security Glossary 2022 [12][4], and GSR Part 1 (Rev. 1) [6] and GSR Part 4 (Rev. 1) [2] establish requirements for the application of such an approach.

Graded approach to safety assessment by designers and operating organizations

3.25. With regard to the application of a graded approach for safety assessment of an innovative technology, Requirement 1 of GSR Part 4 (Rev. 1) [2] states:

> "**A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out at a particular stage for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity.**"

3.26. In accordance with paras 3.3–3.4 of GSR Part 4 (Rev. 1) [2], the graded approach is required to be consistent with the magnitude of the possible radiation risks and to take into account other factors as appropriate. Where an innovative technology is associated with a reduced potential source term[5] (e.g. a micro-reactor or small modular reactor with a small power level or lower potential radiation risks), this should be reflected in the safety assessment. For innovative designs, the application of a graded approach should consider potential radiological releases and the level of uncertainty and lack of knowledge related to the innovation. Depending on the case, in the context of large uncertainties and unknowns further specific efforts might be necessary for safety assessment compared to approaches for conventional nuclear power plants.

3.27. The graded approach to safety assessment for an innovative technology should include an identification of those areas and issues where less rigour and scrutiny may be appropriate, as well as those areas and issues where additional effort and analysis might be needed. It is expected that this will be implemented with proper consideration of the level of knowledge and uncertainties connected with above mentioned areas and issues.

Graded approach to review and assessment by regulatory bodies

3.28. GSR Part 1 (Rev. 1) [6] establishes requirements on the use of a graded approach in the regulatory framework and management system, respectively. In particular, Requirement 26 of GSR Part 1 (Rev. 1), states that "**Review and assessment of a facility or an activity shall be**

[4] Two definitions are given for 'graded approach':

1. For a system of control, such as a regulatory system or a safety system, a process or method in which the stringency of the control measures and conditions to be applied is commensurate, to the extent practicable, with the likelihood and possible consequences of, and the level of risk associated with, a loss of control.
2. An application of safety requirements that is commensurate with the characteristics of the facilities and activities or the source and with the magnitude and likelihood of the exposures.

[5] The magnitude of the possible radiation risks may not simply scale with the core inventory as different designs may have differing radionuclide retention characteristics owing to differing physical barriers and release modes. Application of a graded approach may include that greater attention is paid to the uncertainties associated with the source term.

commensurate with the radiation risks associated with the facility or activity, in accordance with a graded approach."

3.29. The approach utilized by the regulatory body should be selected so as to ensure that sufficient confidence is provided by the designer or operating organization, as appropriate, in the safety demonstration of innovative technology, and for decisions involving societal concerns that fall within the regulatory body's legal mandate, without unduly limiting the utilization of these technologies. The approach should also take into account the knowledge gaps and level of uncertainties in terms of innovative technology. In the case of large uncertainties, more resources and efforts might be needed for the review and assessment by regulatory bodies. Finally, the approach chosen for the innovative technology should be consistent with, and an integral part of, the graded approach for the review and assessment of the overall safety case for a nuclear power plant.

3.30. Regulation and licensing of innovative technology should be carried out using an appropriately graded approach, following a structured methodology that takes into account the safety significance of the innovation, its potential impact on the radiological consequences, potential knowledge gaps and uncertainties as well as activities and plans to close them. For that, regulators should develop a sufficient understanding of the innovative technology and its implications on safety.

3.31. Multiple regulatory bodies might need to review and assess an innovative technology either simultaneously or consecutively. As each regulatory body can benefit from the results of the review and assessment by other regulatory bodies, they should seek timely exchange of information and cooperation on innovative technologies of mutual interest.

3.32. Sections 4 and 5 provide further recommendations and/or insights on the development of a safety assessment approach taking into account the implications of innovative technology on the safety demonstration and describing the aspects that should be considered during the regulatory review and assessment. These recommendations should be used to derive an adequate graded approach to the safety assessment of an innovative technology.

4. SPECIFIC STRATEGIES TO ADDRESS CHALLENGES IN THE SAFETY DEMONSTRATION OF INNOVATIVE TECHNOLOGY IN NUCLEAR POWER PLANTS

LIMITED APPLICABILITY OF ESTABLISHED SAFETY ASSESSMENT APPROACHES FOR NUCLEAR POWER PLANTS

4.1. The safety assessment of a nuclear power plant design which uses the innovative technology should include assessment of defence in depth, with due consideration of assessment of engineering aspects, assessment of human factor as well as deterministic safety analysis and probabilistic safety assessment (see also SSG-2 (Rev. 1) [8], SSG-3 (Rev. 1) [9] and SSG-4 (Rev. 1) [10]). However, established approaches might have limited or challenging applicability for a number of reasons, as listed in para. 2.9 and para. 2.10, including lack of knowledge on relevant phenomena or large uncertainties on phenomena and processes, lack of sufficiently predictive computer codes, limited relevant data and operating experience, and lack of applicable regulations, codes and standards. Therefore, the approaches applied for safety assessment of nuclear power plant design which uses the innovative technology should adequately address above listed limitations.

4.2. Where established safety assessment approaches are not fully applicable, alternative approaches should be used to demonstrate compliance with safety requirements. Alternative approaches for safety assessment should be demonstrated to be sufficiently predictive and robust as well as, validated against data, experiments and testing, or applicable operating experience.

4.3. The performance and reliability of innovative features of a technology should be demonstrated through either analysis, appropriate test programmes, operating experience, or a combination of these following a graded approach. This demonstration should be based on a comprehensive identification of failure modes and an investigation into the susceptibility to common cause failure mechanisms for safety relevant SSCs using innovative technology.

4.4. The safety assessment should consider and provide input to the identification of ageing effects and degradation mechanisms for SSCs of the innovative technology. The lifetime management for inspection, repair or replacement of SSCs using innovative technology should be supported by, and reflected in, the safety assessment.

4.5. An innovative technology might result in new initiating events, fault sequences or plant conditions, which should be systematically identified and considered in the safety assessment, including both deterministic safety analysis and probabilistic safety assessment.

4.6. Interdependent effects among the items important to safety should be assessed for adverse system interactions associated with the innovative technology.

4.7. For the purposes of the safety analyses of a nuclear power plant design which uses the innovative technology there might be situations where the application of the concept of design extension conditions without significant fuel degradation and those with core melting is challenging. This might be the case for technologies where the notion of core melting is not relevant. Practically eliminated situations for innovative technologies might also not be consistent with the recommendations provided in SSG-88 [11], for example the general types of plant event sequence that should be considered as described in para 4.13 of SSG-88 [11]. In such situations, alternative approaches for design extension conditions (including severe accidents), and for practical elimination endorsed on a national level by the regulatory authority, should be based on the specific safety characteristics of an innovative technology[6].

4.8. The safety assessment should support an appropriate interpretation of the requirements for mitigating postulated severe accidents, such as those established in Requirement 20 and paras 5.27–5.32 of SSR-2/1 (Rev. 1) [3].

4.9. The safety assessment should include evaluation of internal and external hazards [14], [15]. For innovative technologies, special attention should be given to internal hazards associated with innovative technologies (see paras 4.24-4.28). For external hazards the safety assessment should consider that impacts from such hazards might result in unique postulated initiating events or initiating events for NPPs with innovative technology. For example, fast reactors may be susceptible to transient over-power events as a result of reactor internal movements following ground motion following a seismic event.

[6] For instance for reactors with liquid fuel the core melt can't be considered as design extension condition. Another example is potential design extension condition where freezing of liquid metal coolants or liquid fuel occurs.

4.10. Following a graded approach, the safety assessment for a nuclear power plant design which uses an innovative technology might consider a reduced potential radiation risk or include additional margins due to uncertainties or lack of knowledge for aspects relied upon in the safety assessment. The assessment should ensure that the approach taken is not optimistic (e.g. through inappropriately relaxed requirements). Where a graded approach affects multiple aspects of the safety assessment (e.g. maximum design fuel temperature, maximum reactor vessel design pressure, required heat transfer capacity of a passive residual heat removal system, design time window for an innovative accident management provision to become effective, minimum design pressure of the last reliable confinement barrier), a balanced approach should be chosen that maintains sufficient caution with respect to any residual uncertainties and knowledge gaps.

4.11. In the safety assessment process for an innovative technology, new information should be considered as it becomes available. Depending on the nature of the new information, the need to update the safety assessment in a timely manner should be considered, and if needed, the determination of fault sequences, the derivation of initiating events and the determination and analysis of design extension conditions should be updated.

4.12. The safety assessment, including probabilistic safety assessment and deterministic safety analysis, should be commenced early in the design process for an innovative technology and should be updated as the design evolves, accounting for the innovative design features that affect the analysis approach and results.

Identification of postulated initiating events and initiating events for safety analyses

4.13. Following the recommendations in para 3.13 of SSG-2 (Rev. 1) [8] and para. 5.13 of SSG-3 (Rev. 1) [9], postulated initiating events and initiating events[7] in the safety assessment for an innovative technology should be identified considering any new accident sequences and initiating events specific to the design under consideration. All foreseeable events with the potential for significant consequences and events with a non-negligible frequency of occurrence should be systematically identified and further considered in safety assessment.

4.14. The identification of postulated initiating events and initiating events for the safety assessment of innovative technology should include events with subsequent failures. Specific consequential failures, interactions and internal hazards associated with an innovative technology should be identified and taken into account. For example, a sodium or lead-cooled fast reactor may utilize air cooling as the preferred method for decay heat removal; postulated initiating events and design extension conditions that can lead to restricting or reducing air flow in the air cooling system should be identified and included in the list of postulated initiating events and design extension conditions for these reactor technologies. An example of the relevant failure could be a hole in the duct work for the air cooling system that results in an interconnection between the inlet and outlet duct work (e.g. bypassing the natural circulation flow path).

4.15. The process of identifying postulated initiating events and initiating events should consider all anticipated modes of operation of the nuclear power plant. New modes of operation relevant to the innovative technology should be identified and taken into account. For instance, for molten salt reactors with the fuel dissolved in the salt, some maintenance modes require the transfer of

[7] The term postulated initiating events is used to describe those used in the deterministic safety analysis and the term initiating events is used to describe the term used in the PSA.

salt from the fuel circuit to storage tanks, where residual heat still has to be removed from the salt. Postulated initiating events in this configuration should be considered in the safety assessment and categorized in accordance with their frequency.

4.16. SSG-2 (Rev. 1) [8] and SSG-3 (Rev. 1) [9] provide recommendations on the development of the comprehensive lists of postulated initiating events and initiating events. In the context of innovative technologies, application of some of the recommended approaches might be challenging (i.e. depending on the available operating experience and availability of similar nuclear power plants). In such cases still the comprehensiveness of the lists of postulated initiating events and initiating events should be ensured by applying measures to overcome above mentioned limitations. For instance, it might require the use of operating experience from non-nuclear applications in addition to extensive use of analytical methods such as master logic diagrams, hazard and operability studies or failure mode and effects analysis could be needed to identify failures that could lead to an initiating event.

4.17. The systematic analysis used to develop lists of initiating events for safety analyses (deterministic and probabilistic) should include an evaluation of unique or innovative features whose failure or degraded operation might lead to a radioactive release. For example, for small modular reactors where either liquid or solid fuel is circulated outside of the reactor vessel, initiating events related to this fuel process should be identified. Another example is an initiating event involving the sodium processing system in a sodium cooled reactor, where a leak might lead to a release from a component such as a trap or filter.

4.18. Uncertainties and lack of knowledge regarding potential challenges to safety functions should be taken into account when grouping new postulated initiating events identified for an innovative technology and assigned to anticipated operational occurrences, design basis accidents and design extension conditions for deterministic safety analysis, and into initiating events for probabilistic safety assessment.

4.19. Innovative technology can introduce unique phenomena that might result in unique postulated initiating events and initiating events. The safety assessment for an innovative design and technology should identify such events considering the new accident sequences specific to the design under consideration using a systematic approach.

Severe accident considerations for innovative technology

4.20. The IAEA Nuclear Safety and Security Glossary [12] defines a severe accident as an “Accident more severe than a design basis accident and involving significant core degradation.” SSR-2/1 (Rev. 1) [3] requires measures to prevent, respond to and mitigate a severe accident. For PWRs, the term “significant core degradation” is used to describe a situation with core melt” where the first barriers have lost their function and the last remaining barrier for confining the radioactive inventory of the nuclear power reactor is challenged, and which, upon release, would require protective off-site emergency measures that are not limited in terms of lengths of time and areas of application. Thus, maintaining the integrity of this barrier against such releases becomes imperative. If the notion of core melt is not relevant for a given innovative technology (see para. 4.7), during the implementation of the safety analyses an alternative approach for severe accidents should be defined for the nuclear power plant. This approach should consider both the source of radioactive releases, which may stem from the fuel or from other sources (e.g. coolant activation, construction materials activation), and the state of the barriers to such releases.

4.21. Design extension conditions and practically eliminated situations should be identified, considering that in certain accident sequences, which could be more severe than design basis accidents or involve additional failures, design basis safety provisions cannot prevent or control the accident sequence, making additional safety provisions necessary. The recommendations provided in SSG-88 [11] should be applied and interpreted accordingly for the specific innovative technology.

4.22. The demonstration of prevention of severe accidents should be achieved based on the accident analysis considering both single initiating events and events with subsequent failures following the recommendations in SSG-2 (Rev. 1) [8], SSG-3 (Rev. 1) [9], SSG-4 (Rev. 1) [10] and SSG-88 [11], as applicable.

4.23. Among the severe accident scenarios related to innovative technology in nuclear power plants, there might be specific situations for which off-site radiological consequences cannot be reasonably mitigated. If such situations are likely to result in unacceptable consequences, these situations should be explicitly listed and should be demonstrated to be practically eliminated following the recommendations in SSG-88 [11]. In these cases, the safety assessment should prove with high confidence that such consequences have a very low frequency of occurrence or are physically impossible. Accident sequences that involve considerable uncertainties about phenomena and processes, which could challenge plant safety and lead to an early release or a large release into the environment, should not be claimed as practically eliminated unless all relevant knowledge gaps have been closed.

Internal hazards associated with innovative designs

4.24. Internal hazards should be analysed as part of the safety assessment. The following paragraphs provide recommendations on specific aspects potentially associated with an innovative technology.

4.25. Innovative features may include materials or working media that cause new types of internal hazard, increase the frequency of an existing internal hazard, or are more susceptible to well-known internal hazards like internal flooding. The susceptibility of innovative features to existing internal hazards and to the generation of new internal hazards should be carefully analysed. For instance, sodium is prone to react with air and water and that can increase the risk of fire in sodium cooled reactors.

4.26. Innovative designs may involve compact layouts, for instance when an integral reactor is located inside a compact steel containment. A compact layout can make it more difficult to implement physical separation to prevent failures from hazard that degrade multiple redundancies or safety provisions on multiple levels of defence in depth. The capability of the nuclear power plant design to implement sufficient provisions for physical separation of components important to safety should be analysed at an early stage of the design to make sure that the planned arrangement is compatible with the safety assessment. For instance, fire zoning could be difficult to implement in a compact steel containment.

4.27. For designs with a multi-module layout or modules sharing SSCs, both the possible propagation of internal hazards from one module to another or possible common cause initiators that may affect every module, and the potential generation of new kinds of internal hazard should be considered in the safety assessment.

4.28. Some internal hazards with the potential for significant consequences could be eliminated by design. The safety assessment should justify that the elimination of such hazards is adequately implemented. The potential for cliff edge effects leading to unacceptable consequences should be analysed, as appropriate.

Unique plant operational states for innovative designs

4.29. An innovative technology may involve unique plant operational states owing to, for example, unique fuel cycle arrangements, configurations related to multi-modularity, or modes of operation associated with non-electrical applications and co-generation. Similarly, the definition of safe shutdown condition for an innovative technology might be different to that for conventional NPPs. The safety assessment should identify all plant operational states, including those associated with planned refuelling, outages, and transport periods (where applicable). Plant operational states should be based on either actual plant experience or planned practices or procedures for the design. The identified plant operational states should be considered both in deterministic safety analysis and probabilistic safety assessment. One example of unique plant operational states is in multi-module nuclear power plants, where the refuelling of one reactor module might be implemented while the other interconnected reactor modules are in operation. The possibility of different operational states of different modules should be taken into consideration in the analysis and management of internal and external hazards in multi-module.

4.30. Paragraph 4.36A of GSR Part 4 (Rev. 1) [2] states:

> "For sites with multiple facilities or multiple activities, account shall be taken in the safety assessment of the effects of external events on all facilities and activities, including the possibility of concurrent events affecting different facilities and activities, and of the potential hazards presented by each facility or activity to the others."

When an innovative nuclear power plant is planned to be added to an existing site already housing other nuclear facilities, the safety assessment should therefore take into account the potential impact of the existing facilities on the innovative plant.

Addressing passive and inherent features in safety assessment

4.31. Passive systems and inherent features that are based on innovative technology may need specialized analysis to determine their capability and reliability to deliver their safety functions under specific operational or accident conditions. This analysis should be undertaken as part of the safety assessment to demonstrate adequate implementation of defence in depth. The recommendations provided in paras 5.123–5.129 of SSG-3 (Rev. 1) [9] relating to probabilistic safety assessment for passive system can also be used for deterministic safety analysis, where applicable.

4.32. Where the safety assessment takes credit for passive systems or inherent features, especially first-of-a-kind systems, the reliability and performance assumptions should be based on sufficient operating experience, where available, or testing at representative scales (e.g. bench or scale model testing).

4.33. The safety demonstration for a passive system should foresee specific pre-operational or startup testing to confirm the assumptions in the safety assessment regarding the reliability and performance, where appropriate. Similarly, code predictions should be checked against such tests where available.

4.34. The safety assessment of passive system should provide evidence regarding its performance within system's mission time foreseen by design. Depending on the scenario and the passive system characteristics, additional safety provisions might be required when the safety function provided by the passive or inherent feature is required after the system's mission time is exceeded.

Human factors evaluation for innovative technology

4.35. The evaluation of human factors within the safety assessment for an innovative technology should follow the recommendations provided in IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [16], where applicable. While innovative technologies may reduce the need for operator actions, it remains essential to demonstrate that operators can effectively monitor the plant and respond to events. While the reliance on human factors may be reduced, a graded approach should still be used to assess operator roles in monitoring, decision-making, and emergency response.

Feedback from safety assessment to safety classification and reliability of SSCs

4.36. The safety assessment should inform the classification of SSCs based on their relevance to safety. The recommendations provided in IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [17] are applicable to innovative technology. If alternative approaches are endorsed on a national level by the regulatory authority, then these approaches should be based on the specific safety characteristics of an innovative technology, and should use insights from deterministic safety analysis and probabilistic safety assessment.

4.37. The safety classification of SSCs that use innovative technology should also consider in an integrated manner the possible radiation risks that the SSCs protect against, and the level of uncertainty and potential knowledge gaps associated with the innovative technology. The relative importance of the SSCs using innovative technology should be identified through a combination of deterministic safety analysis and probabilistic safety assessment.

4.38. The safety classification of innovative SSCs should start early in the design process and should be updated, when necessary, as the design and its safety assessment mature and throughout the service life of the innovative technology as more information becomes available. The safety classification of the SSCs should be reconsidered in the case where new information is gained during the design, manufacturing, or commissioning phase. Early engagement with relevant national regulatory authorities should be sought on the safety classification of SSCs using innovative technology.

Evaluation of performance characteristics for Innovative Technology

4.39. Requirement 23 of SSR-2/1 (Rev. 1) [3] requires the reliability of SSCs to be commensurate with their safety significance. Nuclear power plant designs that utilize innovative technology can include SSCs whose reliability is difficult to estimate and demonstrate (e.g. for passive systems, see paras 4.31-4.34).

4.40. Reliability targets are related to the need to ensure SSC performance. Paragraph 7.29 of SSG-2 (Rev. 1) [8] states:

> "The safety analysis should establish the performance characteristics and set points of the safety systems and operating procedures to ensure that the fundamental safety functions are always maintained. The analysis provides the basis for the design of the reactivity control systems, the reactor coolant system and the engineered safety features (e.g., the emergency core cooling systems and the containment heat removal systems)."

The safety assessment should demonstrate that the adequate performance characteristics are established for the SSCs for all relevant operating conditions. Sufficient margins to reflect any residual lack of knowledge and uncertainties related to the innovative technology should be demonstrated.

4.41. It might be difficult to test all aspects of the innovative technologies for the full range of conditions for which they need to fulfil their safety function. In this case, simulations with validated computer codes should be performed complementary to testing in order to support the reliability evaluation over the range of operational states and accident conditions. Suitable sensitivity, scaling and uncertainty studies should be performed to support a robust safety demonstration. It should be justified that the set of simulations and sensitivity analyses is sufficient to address the testing gaps mentioned above.

Deterministic safety analysis for NPP design which uses the innovative technology

4.42. Deterministic safety analysis should be performed following the recommendations provided in SSG-2 (Rev. 1) [8] and SSG-88 [11], as applicable, applying a graded approach. Additional recommendations specific to innovative technology are provided below.

4.43. Safety acceptance criteria for innovative technology should be defined considering the specifics of the technology, following a graded approach. The margins included in safety acceptance criteria should be proportionate to the level of uncertainty and knowledge gaps relevant to safety for the innovative technology.

4.44. According to Req. 13 of SSR-2/1(Rev.1) [3] the plant states are grouped into a limited number of categories primarily on the basis of their frequency of occurrence[8] (e.g. anticipated operational occurrences, design basis accidents, design extension conditions). Depending on the degree of innovation and commensurate with the lack of knowledge and level of uncertainty, additional conservatism should be used relative to the lower frequency cut-off values. Such additional considerations can be beneficial in case that new information leads to a re-evaluation of a postulated initiating event or fault sequence likelihood during the lifetime of the technology or if new postulated initiating events are identified.

4.45. The categorization of plant states should follow Requirement 13 of SSR-2/1 (Rev.1) [3] and the recommendations provided in section 3 of SSG-2 (Rev. 1) [8] and section 3 of SSG-88 [11]. The grouping of postulated initiating events and design extension conditions associated with an innovative technology should also consider any lack of knowledge and the level of uncertainty relevant to safety when grouping them with postulated initiating events or design extension conditions that are associated with proven technologies. Potential contributors to a group of postulated initiating events and design extension conditions, respectively, should not be grouped together for deterministic safety analysis if they are qualitatively different with regard to knowledge gaps or levels of uncertainty.

4.46. Paragraphs 7.27–7.29 of SSG-2 (Rev. 1) [8] provide recommendations on using conservative methods for the deterministic safety analysis of anticipated operational occurrences and design basis accidents. Approaches to ensure safety margins should be implemented following the recommendations provided in section 6 of SSG-2 (Rev. 1) [8]. The degree of conservatism and the size of safety margins for an innovative technology should be commensurate with the degree of innovation and to the level of uncertainties and lack of knowledge relevant to safety. This approach should specifically be applied to uncertainty ranges in The Best-Estimate Plus Uncertainty analysis

[8] Typical ranges are given in Annex II of SSG-2 (Rev. 1) [8].

and sensitivity cases used when applying best estimate methods. Uncertainty ranges and enveloping margins used in the deterministic safety analysis should be adequately justified, based on experimental data and operating experience, whenever possible.

4.47. The safety assessment should show that should show that "adequate provisions have been made at each level of defence in depth", that necessary "layers of protection and physical barrier are independent of another as far as practicable" and that the "reliability and effectiveness of the required levels of defence" is ensured (IAEA GSR Part 4 (Rev. 1), Requirement 13 and para. 4.45 [2]), thus reliably preventing unacceptable consequences for each postulated initiating event. Deterministic safety analysis should be performed in line with the recommendations provided in paras 4.7 and 4.19, commensurate with the degree of innovation, and suitable criteria for assigning safety provisions to levels of defence in depth should be defined. In addition, the safety assessment should take into account the level of uncertainty and knowledge gaps associated with the innovative technology when demonstrating that there is a sufficient number of layers of protection and physical barriers and that the safety provisions on the different levels of defence in depth are sufficiently independent.

4.48. For innovative technology, relevant possible radiation risks might be present in systems or locations that are not considered for conventional water cooled reactors. The deterministic safety analysis should systematically identify and assess such locations. For instance, for molten salt reactors where the fuel is dissolved in the salt, a large proportion of the radiological source term can be located in the off-gas system.

4.49. Where planned discharges from the last effective confinement barrier of the reactor for certain fault sequences are part of the safety concept for an innovative technology, the safety assessment should investigate reasonably practicable filtered discharge routes.

4.50. For innovative technology, radioactive inventories might not be located mainly within a fuel matrix but could be more dispersed and more susceptible to transport. The deterministic safety analysis should demonstrate that adequate reliable confinement barriers are provided for all locations in the plant where the radiological source term might be transported to during operational states and accident conditions, including postulated severe accidents. For instance, for molten salt reactors in which the fuel is dissolved in the salt, a large amount of the radiological source term is in the coolant and one safety feature is the drainage of fuel salt from the reactor loop into a holding tank. The holding tank should therefore be included in the confinement function.

4.51. The analysis of design extension conditions (including postulated severe accidents) within the deterministic safety analysis should be tailored to the specifics of the innovative technology, using the recommendations provided in paras 4.7 and 4.20-4.23.

4.52. As established in para. 5.73 of SSR-2/1 (Rev. 1) [3], deterministic safety analysis is required to demonstrate prevention of severe accidents and avoidance of cliff edge effects and unacceptable off-site consequences. For an innovative technology, the level of uncertainty and knowledge gaps should be reflected in adequate margins to any cliff edge effects and should be supported by sensitivity analyses. Where a best estimate or realistic analysis approach to design extension conditions is used (see paras 7.54-7.55 and 7.67 of SSG-2 (Rev. 1) [8]), the level of uncertainty and lack of knowledge associated with an innovative technology should be adequately considered with increased margins and more conservative assumptions.

Probabilistic safety assessment for NPP design which uses the innovative technology

Use of alternative risk metrics in probabilistic safety assessment

4.53. Innovative technologies may need alternative risk metrics (and associated end states) to be defined for the purposes of probabilistic safety assessment (see also para. 2.11 of SSG-3 (Rev. 1) [9]) considering the specifics of different plant states and severe accidents for a given technology. Available guidance and technical insights relevant to the innovative technology should be considered for determining suitable risk metrics and associated end states for probabilistic safety assessment.

4.54. Where the probabilistic safety assessment for an innovative technology is judged against off-site consequence criteria, it should have a Level 2 PSA and an appropriate approach to compare with the acceptance criteria, such as, for instance, Level 3 PSA. and. Insights should inform and should be combined with relevant results from the deterministic safety analysis to meet Requirements 5 and 20 of SSR-2/1 (Rev. 1) [3] (see also SSG-88 [11]).

4.55. Where the probabilistic safety assessment for an innovative technology is based on off-site consequence criteria, it should be ensured that the intermediate results can be used to characterize an adequate implementation of defence in depth, in line with the recommendations of SSG-3 (Rev. 1) [9] and SSG-4 (Rev. 1) [10]. Specifically, the risk of failure of the design basis and preventative design extension provisions, and the risk of off-site releases after failure of the final confinement barrier (see paras 4.20–4.21) should be characterized and considered in the probabilistic safety assessment.

Use of existing human reliability analysis methods

4.56. The use of innovative technology may involve the analysis of operator actions that are different from operator actions analysed in probabilistic safety assessments for conventional nuclear power plants. The analysis of operator actions should follow the applicable recommendations in SSG-3 (Rev. 1) [9] and SSG-4 (Rev 1) [10]. The implementation of human reliability analysis for an innovative technology might involve the adaptation of existing human reliability analysis methods or the application of new methods. This should be justified in terms of adequate consideration of relevant operator actions and relevant contextual factors related to innovative technology. One example is the long time frames (sometimes several days) available for operator actions for some innovative technologies, in which cases, although human reliability analysis techniques developed for conventional NPPs may be used, longer time frames may challenge the modelling approach for both the considered recovery actions and for the estimated lower bound for the human error probabilities. However, if external recovery measures are credited in the safety analysis, possible damage to the plant site should be reflected in the reliability quantification.

4.57. Another challenge involves the use of digital instrumentation and controls, and digital displays involving touch screen controls. Commonly used human reliability analysis approaches for probabilistic safety assessment were developed, among other things. on the basis of data for analogue controls including manual switches. Where human reliability analysis results used in the probabilistic safety assessment are obtained by applying existing human reliability analysis approaches to settings and conditions, as well as human error modes for which those methods were not developed or qualified, appropriate uncertainties should be considered in the probabilistic safety assessment. More details on human reliability analysis methods and their characteristics in this context can be found in Ref. [19].

4.58. In addition, when considering screen-based interactions for an innovative technology, modelling of the interface management should be performed, and negative cognitive workload and other factors that can trigger human failure events should be considered.

Reliability data for the probabilistic safety assessment

4.59. One common issue with innovative nuclear power plants is the lack of failure rate data for SSCs, or for existing SSCs used in a new setting, used in the probabilistic safety assessment. For example, there is significant failure data for motor operated valve failure modes in water, but limited data for similar valves used in molten salt systems. Justification should be provided for the data to be used, and it is good practice to compare data from a number of different sources, including both nuclear and conventional applications, and determine whether any differences can be explained.

4.60. In general, judgements should be made and documented in selecting data sources considered to be suitable for innovative technology. For initiating events with a low frequency of occurrence or for equipment with a low probability of failure, even generic data is usually sparse or non-existent, so the values to be used in the probabilistic safety assessment should be assigned by informed judgement. The reasoning on which such judgements are based should follow the recommendations in para. 4.147.

4.61. A related issue is the probabilistic safety assessment modelling of unique or first-of-a-kind SSCs in an innovative technology. Estimates of the failure probabilities for such SSCs should be justified. Some unique SSCs include modelling of unique instrumentation and control architecture, including hybrid analogue/digital control systems, which can be developed to minimize the likelihood of software common cause failure.

Computer codes used for supporting analyses to probabilistic safety assessment

4.62. The probabilistic safety assessment of an innovative technology may involve the use of unique or newly developed software. Paragraphs 5.55–5.58 of SSG-3 (Rev. 1) [9] and paras 3.15–3.17 of SSG-4 (Rev. 1) [10] recommend that codes for performing supporting analyses to probabilistic safety assessment should be validated and used within their range of applicability. For an innovative technology, there might be a lack of suitable simulations tools, and particularly of simulation tools that adequately support uncertainty and sensitivity analyses. In this context, the challenges with regard to computer codes for supporting analyses should be identified and the recommendations provided in paras 4.89–4.109 should be followed to ensure that such computer codes are available as and when they are needed for the probabilistic safety assessment.

4.63. The recommendations in para 4.62 are not only applicable for software supporting success criteria determination or reliability analysis, but may also be applied to computer codes used for the evaluation of specific phenomena or processes for the probabilistic safety assessment. For example, for the probabilistic safety assessment of a sodium cooled reactor it might be necessary to use a specific simulation tool for investigating the absorption rate of radionuclides in the sodium coolant and for predictions of increased release rates for high temperature of the sodium.

Uncertainties associated with innovative technology

4.64. The probabilistic safety assessment study should support the evaluation of the potential impact of uncertainties associated with the innovative technology on the probabilistic safety assessment insights and further risk informed decision making.

Simplified safety assessment approaches

4.65. The para. 3.24 of IAEA SF-1 [7] states that

> *"The resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the radiation risks and their amenability to control."*

4.66. Some innovative nuclear power plant designs might pose very limited potential off-site radiation risks with regard to accidents (e.g. microreactors). Following a graded approach, it should be considered whether a more simplified, consequence-oriented safety assessment approach might be adequate for such designs [7]. The application of such an approach should be thoroughly justified and consequence-oriented criteria should be developed with a supporting methodology to define the bounding or maximum credible design basis events, with systematic consideration of uncertainties.

4.67. When using simplified safety assessment approaches, bounding postulated events should be systematically evaluated compatible with the deterministic safety analysis recommendations provided in SSG-2 (Rev. 1) [8].

4.68. Based on the results of the safety assessment, SSCs important to safety should be identified and classified following the guidance in paras 4.36-4.38, ensuring that the SSC classification addresses the uncertainties connected with the application of simplified safety assessment techniques (e.g. the conservative approach is expected to be applied where needed).

4.69. In any case, the safety demonstration for NPPs with innovative technologies should consider all levels of defence in depth defined IAEA SSR-2/1 (Rev. 1) [3] para. 4.13 to ensure that the necessary safety layers are foreseen in the design.

LIMITED KNOWLEDGE OF RELEVANT PHENOMENA AND MATERIAL PROPERTIES

4.70. Paragraphs 4.71–4.88 provide recommendations on specific strategies to address challenges in the safety assessment of innovative technology related to the use of working media, materials and nuclear fuels. For these, there can be a lack of knowledge about relevant physical or chemical phenomena and related material properties during operational states or accident conditions over the lifetime of the plant. This applies to working media, materials and fuels not previously used in nuclear power reactors or used in a different way. It can also apply if such media, materials and fuels are used outside of the range of conditions where sufficient knowledge is available and new phenomena or emergent behaviour might become relevant to safety.

4.71. For an innovative technology, it should be assessed if there is a lack of knowledge of the following aspects, resulting in a potential impact on safety:

(a) Physical, chemical or biological phenomena related to the working media, materials and nuclear fuels;
(b) Properties of the working media, materials and nuclear fuels;
(c) Changes in the phenomena and properties for operational states and accident conditions;
(d) Changes in properties over the lifetime of the plant;
(e) Degradation mechanisms for working media, materials and nuclear fuels, including ageing, radiation exposure, thermal and mechanical stresses, and chemical reactions.

4.72. The safety assessment should also consider the interactions between different working media, materials and nuclear fuels related to the innovative technology. This should extend to all categories of plant states from normal operation up to design extension conditions (see Requirement 13 of SSR-2/1 (Rev. 1) [3]).

4.73. It should be determined if the working media, materials or nuclear fuels introduce new postulated initiating events and initiating events for the reactor or new failure modes for the SSCs.

4.74. It should be determined if the working media, materials or nuclear fuels introduce new internal hazards relevant to safety.

4.75. If the assessment finds relevant knowledge gaps, including those related to the determination of postulated initiating events and initiating events, failure modes of the SSC, or effects of hazards, the impact on safety and on the safety demonstration should be assessed and ranked based on safety significance of the identified gaps.

4.76. Where knowledge gaps relevant to safety or safety demonstration have been identified, which are related to phenomena or material properties, specific actions to close those gaps or mitigate their potential effects on safety should be specified and implemented. These actions should be proportionate to the risk associated with the knowledge gaps and their ranking based on the safety significance of potentially affected items important to safety and in line with the requirements established in para 4.16 of SSR-2/1 (Rev. 1) [3].

4.77. An effective and efficient strategy for addressing relevant knowledge gaps should consider the recommendations in paras 4.78–4.82, to the extent applicable for the innovative technology.

4.78. Knowledge gaps related to relevant phenomena and material properties should be closed by dedicated experimental research and tests. This includes building dedicated test facilities to the extent necessary. Scaling effects applicable to test facilities should be considered. Some considerations regarding different types of knowledge gap are listed below:

(a) Knowledge gaps related to specific phenomena or material properties might require dedicated separate effect tests.
(b) Knowledge gaps related to the interactions of phenomena or the interaction of working media, materials and nuclear fuels, and the possibility of synergetic behaviour, might require integral effect tests.
(c) Knowledge gaps related to scaling effects and emergent behaviour might make building large scale test facilities, demonstration plants or prototype reactors necessary.

4.79. Available experimental and test data from non-nuclear applications should be identified and used to close knowledge gaps, if applicable. Where necessary, methods to transfer data from applications similar to the specific innovative technology should be developed and verified.

4.80. For closing identified knowledge gaps, independent research (e.g. in academic institutes or research centres), should be performed to the extent practicable. Opening access to some data relevant to the innovative technology can be an asset in this regard if it fosters diversity in research relevant to safety.

4.81. If experiments and simulations cannot close knowledge gaps or will take up too much time and resources for an appropriate action, expert panels should be used to identify and assess knowledge gaps and estimate uncertainties and the potential impact relevant to safety. Expert panels should follow the guidance in para 4.147 and involve independent experts to ensure a diversity of viewpoints and to protect against bias in expert judgement. Expert panels should consider if and when measurement or operating experience data will be required for closing knowledge gaps for the safety assessment throughout the life cycle of the innovative technology. Expert judgements should be updated as relevant new experimental data or operating experience become available. It needs to be noted that the experiments and simulations cannot be substituted by expert judgement in all cases. They are expected to be mostly aimed to reduce the knowledge gaps and to provide essential input for tailoring further research and development efforts in the corresponding areas.

4.82. Where appropriate, extended periods during construction and commissioning should be foreseen for a first-of-a-kind system or reactor and/or a prototype reactor to acquire data for closing identified knowledge gaps and improving the safety assessment.

4.83. The impact of remaining knowledge gaps and uncertainties with regard to the safety performance of an innovative technology should be reduced by specifying enveloping margins on acceptance criteria in the safety assessment.

4.84. Safety margins should be set considering the impact of any provisions for condition monitoring, examination and testing of working media, material and nuclear fuels used in an innovative technology. Provisions should only be credited in the safety assessment if they are demonstrated to be effective. These provisions include:

(a) Irradiation and neutron flux monitoring;
(b) Chemistry control and monitoring;
(c) Temperature, pressure and vibratory load measurements;
(d) Leakage detection;
(e) Inspections;
(f) Degraded performance monitoring to warn before a failure.

4.85. Where there are knowledge gaps related to working media, materials or nuclear fuels, the impact assessment should also consider if there are adequate methods to produce a safety case for the processing, storage and disposal of these materials and any parts of the reactor made from or in contact with such materials.

4.86. The assessment of the impact of knowledge gaps and uncertainties on the safety demonstration and the strategy to address such knowledge gaps and uncertainties should be updated throughout the service life of the innovative technology as relevant new information about phenomena and material properties becomes available.

4.87. The strategy to address knowledge gaps and uncertainties relevant to safety demonstration may conclude that such knowledge gaps and uncertainties cannot be addressed as planned. Such information should also be used as an input for the design process. In this case, changes to the design of the innovative technology to address this concern should be considered.

4.88. Finally, the strategy for addressing knowledge gaps should also consider specific measures to ensure that existing knowledge relevant to the innovative technology and its safety assessment is not lost throughout the service life of the technology.

LACK OF COMPUTER CODES FOR SAFETY ANALYSIS

4.89. Paragraph 4.18 of GSR Part 4 (Rev. 1) [2] states that

> "The necessary preparations shall be made to ensure that … The necessary tools for carrying out the safety assessment are available, including the necessary computer codes for carrying out the safety analysis."

4.90. Furthermore, Requirement 18 of GSR Part 4 (Rev. 1), [2] states that "**Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation.**" The associated para. 4.60 of GSR Part 4 (Rev. 1) [2] states:

> "Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation to a sufficient degree. Model verification is the process of determining that a computational model correctly implements the intended conceptual model or mathematical model; that is, whether the controlling physical equations and data have been correctly translated into the computer codes. System code verification is the review of source coding in relation to its description in the system code documentation. Model validation is the process of determining whether a mathematical model is an adequate representation of the real system being modelled, by comparing the predictions of the model with observations of the real system or with experimental data. System code validation is the assessment of the accuracy of values predicted by the system code against relevant experimental data for the important phenomena expected to occur. The uncertainties, approximations made in the models, and shortcomings in the models and the underlying basis of data, and how these are to be taken into account in the safety analysis, shall all be identified and specified in the validation process. In addition, it shall be ensured that users of the code have sufficient experience in the application of the code to the type of facility or activity to be analysed."

4.91. Further recommendations on the use of computer codes for safety assessment are provided in SSG-2 (Rev. 1) [8], SSG-3 (Rev. 1) [9] and SSG-4 (Rev. 1) [10]. These recommendations can be applied with judgment to the safety assessment of innovative technology.

4.92. The recommendations provided in paras 5.1–5.5 of SSG-2 (Rev. 1) [8] on the selection of computer codes for deterministic safety analysis are fully applicable to innovative technology and can be applied to other fields relevant to safety assessment. Specifically, computer codes for demonstrating the performance of an innovative technology should be validated and should use models and calculational methods suitable for their intended use in the safety assessment. The models in the code should describe the relevant phenomena and processes and their interactions relevant to the innovative technology with adequate predictiveness. Properties of working media, materials, nuclear fuels, and other physical or chemical properties should be provided with adequate accuracy for the full range of conditions relevant to the use of the code.

4.93. The computer codes needed to support the safety assessment of an innovative technology should be identified at an early stage during the design of an innovative technology. This assessment should be updated regularly as the safety assessment for the technology matures throughout its service life and the needs and roles for computer codes in the safety assessment are better understood.

4.94. Where computer codes are foreseen to support the safety demonstration, it should be assessed if they can be used as available, or if one or several of the following challenges need to be addressed:

(a) Lack of models or methods for phenomena and processes: the code lacks models for phenomena or methods to describe processes relevant to the performance of the innovative technology so that its results are not adequate.

(b) Limited computer code predictiveness: the code is capable of simulating relevant phenomena and processes for the innovative technology, but only with simplified approaches. Code results are not sufficiently predictive to show that the safety acceptance criteria are met with reasonably practicable safety margins.
(c) Lack of demonstrated validation: the code is capable, in principle, of simulating relevant phenomena and processes with adequate predictiveness. However, it has not been validated for the range of conditions it needs to be used for.
(d) Lack of simulation code performance: the tool can adequately predict relevant phenomena and processes and has been validated. However, simulations take so much time that the tool cannot be used, for example, to demonstrate adequate performance of the innovative technology throughout its mission time.
(e) Lack of uncertainty quantification: the code can adequately perform simulations of the innovative technology and its performance. However, there is a lack of information on model uncertainties, or uncertainty analyses cannot be performed with the code due to missing code capabilities and/or performance constraints.
(f) Lack of integral simulation capabilities: several computer codes can adequately simulate specific aspects of an innovative technology, however there is no means of simulating the interaction of multiple systems in an integral manner and there are no appropriate decoupling criteria for the safety analyses.
(g) Other challenges in code performance or application: the code suffers from other challenges not falling in the categories above, for which strategies and actions need to be defined.

4.95. If the computer codes needed to support the safety assessment are subject to any of the challenges in para. 4.94 and they cannot be resolved by using other available computer codes, a strategy with specific actions should be developed for delivering computer codes adequate for safety assessment of the innovative technology at the time when they are needed.

4.96. Safety assessment for an innovative technology might rely more on computer codes for some aspects because, for example, there is a lack of experimental data, proven engineering practice, or operating experience. Paragraph 5.4 of SSG-2 (Rev. 1) [8] recommends in part, comparing the code predictions with outputs from other independently developed codes and standard problems and/or benchmarks, where available. If there is only one computer code suitable and available for the innovative technology, this can increase the risk of a single point of failure for the design and this should be reflected in the safety assessment.

4.97. Section 9 of SSG-2 (Rev. 1) [8] provides recommendations on the independent verification of deterministic safety analysis by the licensee, including the use of independent verification calculations. In some Member States, independent confirmatory analyses are performed by, or on behalf of, the regulatory body. In both cases, using an independent computer code is considered good practice. Where possible, calculations with an independent computer code should be used to provide additional confidence that challenges related to limited applicability of computer codes have been adequately resolved.

4.98. The identification of suitable computer codes should therefore include consideration of whether independent computer codes will be needed to support the safety demonstration or for external confirmatory analysis. In this case, the strategy should be extended to identify a suitable set of available computer codes and define specific actions to address and close relevant gaps in the performance of such codes for their intended role in the overall safety demonstration. To this end, early interaction with relevant regulatory bodies should be sought.

4.99. Gaps relating to the availability, predictiveness and validation status of computer codes are often linked to knowledge gaps on relevant phenomena for the innovative technology. The methods and strategies to address this issue recommended in paras 4.70–4.88 are generally applicable. These methods and strategies should be used where suitable to either close such gaps or address uncertainties in the safety assessment. Where knowledge gaps can be closed in this way, the selected computer codes should be improved and validated as necessary and proportionate to support the safety demonstration.

4.100. The following paragraphs provide recommendations on how to address different gaps identified for computer codes supporting the safety demonstration if adequate models or methods relevant to an innovative technology cannot be added to computer codes, or if this would not be a proportionate solution.

4.101. If there are gaps due to a lack of models or computer code capabilities, it should be considered if additional actions are applicable and effective. These include:

(a) Assessing the impact of the phenomenon or material properties on computer code results by performing uncertainty analyses or sensitivity analyses that adjust existing code model parameters to appropriately enveloping values;
(b) Implementing and validating an ersatz model using existing computer code capabilities to determine the impact of the phenomenon or material properties on code results needed for safety demonstration;
(c) Investigating the phenomenon with a separate applicable and validated code (e.g. a high-resolution simulation computer code), and determining or estimating its impact on results.
(d) Utilizing analytical methods, experimental data, and operational data to supplement or replace code capabilities
(e) Utilizing experimental justification or engineering judgement to support the safety demonstration.

4.102. Computer codes might not be sufficiently predictive even if they include suitable and validated models and methods, for example because the computer code paradigm does not allow for a high spatial or temporal resolution or because input data is not known with sufficient precision or is difficult to derive for the kinds of inputs the code expects. The following specific actions should be considered to the extent applicable and effective:

(a) Use of modelling resolutions, numerical settings or model options beyond current good practices for the code, and their validation.
(b) Coupling of the computer code to a different computer code with sufficient predictiveness for the phenomena or processes in the part of the plant or system where this is needed. If code couplings need to be added or modified, they should be verified and validated.
(c) Investigation of the impact of insufficient predictiveness on code results by performing uncertainty analyses or by sensitivity analyses varying existing code model parameters to appropriately enveloping values.
(d) Definition of additional margins to acceptance criteria in the safety assessment so that uncertainties affecting code results are adequately bounded and the computer code becomes sufficiently predictive.

4.103. If computer codes are adequate for some phenomena and processes but are not sufficiently predictive for the overall behaviour of the innovative technology under all relevant conditions, the following specific options can be considered to the extent applicable and effective:

(a) The use of separate codes for separate aspects of the assessment by defining suitable decoupling criteria and related safety acceptance criteria with sufficient margins to cover for any limitations of, and simplifications made in, this approach.

(b) The coupling of separate, sufficiently predictive, computer codes to capture integral behaviour. The code coupling should be adequately verified and validated and shown to be sufficiently predictive.

4.104. If there is a lack of validation or documentation of validation status of a computer code for an innovative technology, one or several of the following actions should be considered as part of the strategy to the extent applicable and effective (see also para. 5.21 to 5.39 of IAEA SSG-2 (Rev. 1) [8]):

(a) Dedicated validation can be performed against existing experimental data or applicable operational data (e.g. from a prototype reactor or a full-scale test of a system), which is suitable to confirm that relevant phenomena and processes are adequately predicted by the code. To this end, suitable data measured for non-nuclear purposes are often applicable.
(b) The computer code can be validated against other already validated codes with demonstrated adequate predictiveness for the phenomena and processes relevant to safety.
(c) As code validation can take considerable resources, and as validation can affect more than one code, the strategy should consider involving independent external organizations (e.g. research organizations, academic institutions) in this activity. The strategy should also assess the option of providing public access to some data suitable for the validation of codes for the safety assessment of the innovative technology.
(d) Additional validation can identify limitations in the computer code predictiveness, which should be removed, or can identify the potential for improvements of the code. Both will often involve modifications to the code. The strategy should therefore include proportionate and practicable corrective actions by code developers.

4.105. If there are limitations in computer code performance (e.g. code simulations take too long or take so many computational resources that timelines for safety demonstration become untenable) the following actions should be considered as part of the strategy:

(a) In some cases, code performance can be improved by optimizing the inputs for the computer code calculation model, switching to lower resolution or simplified calculation models for parts of the nuclear power plant design, limiting the computational domain or simulating only a certain time window of interest.
(b) Using coupled computer code calculations is another option, where a fast-running code is used for those parts of the reactor where results are less sensitive to reduced code predictiveness, and computationally costly high-fidelity codes, where a high level of detail of phenomena or processes is needed (e.g. local or temporal).
(c) Where computer codes allow parallel computation, the use of additional computational resources can be considered. This is particularly effective if large calculational models are distributed to several computing nodes, but there are diminishing returns with an increasing number of computational nodes.
(d) Instead of computationally costly computer codes, the use of fast-running codes with lower resolution models can be considered if these are sufficiently predictive. In that case, it should be checked if the margins defined for safety demonstration need to be adjusted.
(e) It should be checked if more reliance on experimental justification and engineering judgement can be pursued to support the safety demonstration.
(f) It may be possible to improve the numerical performance of the calculation code, for example, by refactoring out inefficient programming, enabling parallelization, or using more performant numeric algorithms. Considerable expertise in the computer code, its architecture and source code are needed for this, and such improvements, their verification and validation may involve significant resources and revisions by code developers. Therefore, including this option in a strategy should be done with some caution.

(g) If computer code performance does not allow for parameter uncertainty propagation approaches to uncertainty analysis, it should be considered if alternative uncertainty quantification methods should be pursued.

4.106. Defining adequate margins to acceptance criteria for safety assessment involves a sufficient understanding of uncertainties associated with the phenomena and processes associated with an innovative technology and how they translate to computer code results. Depending on the safety assessment approach, paras 2.8–2.15 of SSG-2 (Rev. 1) [8] recommend performing uncertainty and sensitivity analysis if best-estimate computer codes, models, and boundary conditions are used. Such approaches are also recommended for probabilistic safety assessment (see paras 5.55 and 5.56 of SSG-3 (Rev. 1) [9]). If there is a gap regarding uncertainty quantification, this can be due to lack of knowledge on relevant phenomena, in which case the approaches discussed for that should be explored. Otherwise, a strategy should consider the following options for computer codes to the extent applicable and effective:

(a) The safety assessment can use enveloping and bounding assessment approaches and foresee adequate margins so that the impact of uncertainties on computer code results is covered in the safety assessment.
(b) Where feasible, model uncertainties can be determined from existing experimental data or original sources.
(c) There are different approaches for uncertainty quantification of computer code results, which include uncertainty propagation, sensitivity index methods, or construction of surrogate models (e.g. that use machine learning). If computer codes need to be modified to support uncertainty quantification, they should be duly verified and validated. Where surrogate models are derived for uncertainty quantification, they should be treated as separate computer codes for the purposes of the recommendations in this Safety Guide and should be verified, validated and used accordingly.
(d) By performing targeted sensitivity cases, the impact of uncertainties related to an innovative technology on computer code results can be characterized. Insights should be used to adjust margins in the safety assessment.
(e) Comparing results of different computer codes for the same scenario against each other can provide a better understanding of the variability of computer code results. Insights should be used to adjust margins in the safety assessment.

4.107. If the options above are not feasible, the impact of uncertainties related to an innovative technology on code results could be estimated by expert elicitation. In this case, the experts involved should not only have sufficient understanding of the underlying phenomena and processes relevant to the innovative technology, but they should also be sufficiently familiar with the calculational methods and limitations of the computer codes. Additionally, input from independent experts should be sought to ensure diversity of viewpoints and protect against bias in expert judgement (see also para. 4.147).

4.108. Where computer codes need to be improved to produce an adequate safety assessment of an innovative technology, it should be ensured that these modifications are possible if and when they are needed, and it should be ensured that a review by regulatory authorities is possible. Specifically, closing gaps in computer codes will often involve access to the source code, and an in-depth understanding of the code, its models and methods, and their implementation. Unless the computer code is developed in-house, agreements with third-party computer code developers ensuring access to such data and expertise as well as securing sufficient rights to the modified computer code should be established as early as practicable.

4.109. The strategy for selecting computer codes and closing any gaps relevant to an innovative technology should consider how these tools can be maintained, updated and replaced, as applicable, over the service life of the technology. Developing new models or methods for an innovative technology, which are suitable for computer codes, and implementing, verifying, and validating such models and methods, as well as maintaining computer codes over the service life of the technology takes considerable resources. If information and data on the innovative technology that can be used for computer code development is made publicly available, and if related research activities are initiated on the national or international level, external organizations can be involved in improving independent computer codes relevant to the technology. The strategy on computer codes should consider fostering such external involvement.

LACK OF RELEVANT OPERATING EXPERIENCE

4.110. If there is limited or no relevant operating experience for an innovative technology, the safety assessment should support the determination of appropriate requirements on quality, reliability and qualification of SSCs using such technology.

4.111. Depending on the degree of innovation and the lack of relevant operating experience, a process should be established for SSCs using innovative technology that ensures confidence in the reliability of the innovative safety feature to provide its safety function. Important elements of that process are elaborated in the further paragraphs of this subsection.

General strategy for demonstration of SSC performance and reliability

4.112. A phased approach should be established to demonstrate the performance and reliability of SSCs important to safety through research and development, design controls, manufacturing, installation, initial testing, surveillance, and operational in-service examinations or testing stages. A principal objective of the strategy should be the management of uncertainties[9] associated with SSC performance and reliability because of limited applicable operating experience.

4.113. The recommendations provided in IAEA Safety Standards Series No. SSG-69, Equipment Qualification for Nuclear Installations [20] are applicable to innovative technology with informed judgment. If there is a lack of operating experience, more emphasis should be placed on the methods of equipment qualification listed in para. 4.1 of SSG-69 [20] to demonstrate that the relevant SSCs will deliver their safety functions with the required reliability.

4.114. For the derivation of a targeted test and qualification programme, the attributes and characteristics of SSCs using innovative technology relevant to safety should be defined.

4.115. A quality assurance programme or process description with key criteria should be considered as part of the safety assessment for an SSC using innovative technology.

4.116. Where possible, generic data from similar SSCs designed and operated in similar conditions in nuclear or other industries should be collected and assessed to identify the relevant failure modes. The applicability of these data should be justified, and the effects of the identified failure modes should be analysed in the safety assessment (e.g. through a failure mode and effects

[9] In this context, the management of uncertainties include, e.g. identification of the sources of the uncertainties, analysis of uncertainties, developing and implementing measures to reduce uncertainties, developing measures to mitigate remaining uncertainties (e.g. applying bigger margins)

analysis). When using such data, the analysis should consider the trustworthiness of the information, in particular if the data was not collected under a robust quality assurance programme or equivalent measures. Appropriate statistical analysis techniques should be applied when using such data along with test data for an SSC using innovative technology.

4.117. Potential vulnerabilities to common cause failure mechanisms relevant to the innovative technology (e.g. environmental conditions, manufacturing defects, design specification errors) should be identified and assessed.

4.118. As part of the safety demonstration, the design lifetime of an SSC using innovative technology should be defined. This should include adequate margins to take into account the degree of uncertainty and knowledge gaps on lifetime-limiting mechanisms (e.g. corrosion, wear and tear) for the innovative technology. Suitable approaches and provisions for replacing the SSC should be determined, using information from surveillance and condition monitoring, where appropriate. These approaches and provisions should be derived from the safety assessment and, as appropriate, time limited ageing analyses.

4.119. For an innovative technology, some SSCs that have been previously qualified and in use in existing facilities may be proposed for use in different environmental conditions. In this case, additional analysis and/or type tests should be conducted. For example, for use in an innovative technology, a passive autocatalytic recombiner could be permanently located in a high radiation field that could degrade the recombiner's capability to fulfil its safety function over time. In this case, additional radiation testing of the device should be performed to ensure its adequate qualification for its new operating environment. Periodic replacement of the device during the lifetime of the reactor might also be needed.

4.120. If an innovative design feature is not designed, fabricated, constructed and operated under a recognized nuclear quality assurance programme, the performance characteristics of the innovative component should be derived as part of the safety assessment. Typical commercial grade quality assurance programmes may be used to help identify the acceptance criteria and the process to determine the performance characteristics.

4.121. The safety assessment should define claims on critical characteristics and the processes and measures that support these claims. While defining the claims, the graded approach should be followed, proportionate to the importance of the SSC to safety and considering knowledge gaps for the innovative technology.

4.122. The process for defining SSC's critical characteristics should also consider whether the design reduces the possibility of human errors, and the potential safety implications should be considered in the safety assessment.

4.123. The safety assessment should identify those aspects of SSCs that use innovative technology that are relevant to reliability and availability claims. Specific sub-processes should be foreseen in the operating experience feedback process to allow recognition of developing or emerging problems related to these aspects, so that proactive measures can be taken before serious conditions arise (see para. 2.53 of IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [21]).

4.124. Depending on its role in the safety demonstration, an SSC that uses innovative technology may have to be qualified and perform reliably for a narrow set or a range of challenges and operating and environmental conditions (see also SSR-2/1(Rev.1) Requirement 30 [3] and SSG-69 [20]). The operational limits and conditions and performance criteria of SSCs using innovative technology should be established and reflected in the safety assessment. For an innovative technology, limiting conditions of operation and performance criteria should be established in such a way that they envelop the impact of any remaining gaps of knowledge and uncertainties relating to the innovation.

Testing to collect performance and reliability data

4.125. Testing should be conducted for SSCs using innovative technology important to safety to determine the SSC's level of reliability to perform its safety function over time and under various conditions.

4.126. In the absence of directly applicable data, the SSC's initial performance data should be collected through qualification testing as well as analysis for a range of conditions commensurate with importance of the SSC to safety and its safety classification.

4.127. For the purpose of the safety demonstration, initial test plans for SSCs that use innovative technology should be established and thoroughly described. The information on the initial tests for safety demonstration purposes should include test descriptions containing sufficient information for the testing of innovative safety features to demonstrate that they will meet their design and test acceptance criteria. The time span defined as initial operation phase will depend on the specific nature of the innovative technology. For a lead or first unit, the initial operation phase could involve supplemental or augmented maintenance, surveillance, condition monitoring, or inspections to gain baseline confirmatory performance data thereby reducing uncertainties in as-built and as-operated SSC performance. Supplemental testing or monitoring may be discontinued if acquired data or specified criteria are met to justify no observed adverse performance results.

4.128. Accelerated ageing tests may be necessary for the qualification of an SSC that uses innovative technology. Limitations in such tests should be rigorously examined, bearing in mind the SSC's potential operating conditions, and should be considered in the safety assessment. The factors affecting SSC performance in this context include material compatibility, plant chemistry, environmental conditions and effects of foreign materials.

4.129. It may be necessary to impose initial conservative limitations on SSCs using innovative technology relevant to safety until enough in-service experience has been gained to fully justify the reliability assumptions in the safety assessment. The safety assessment for such SSCs should demonstrate that the frequency of surveillance testing, the scope of non-destructive examinations and the periodic replacement strategy, as applicable, are adequate.

4.130. For an innovative technology, standardized guidance to address ageing through testing might not be available to fully address the range of environmental effects on SSC materials, including combined effects. For example, in helium cooled high temperature gas reactors various detrimental impurities in the coolant are known to degrade helium purity ultimately impacting the longevity of SSCs (e.g. heat exchangers). It has been a challenge to fully address the integrated effects of the corrosion mechanisms in such reactors. Strategies to address these challenges should be developed and included in the safety demonstration. Such strategies may include improved control of detrimental impurities, better alloying, and use of oxidation resistant coatings.

4.131. For instances where consensus-based testing approaches do not exist to adequately demonstrate that the SSCs will deliver their safety function with the required reliability, establishing appropriate design-specific lifetime models for SSCs that use innovative technology may be necessary. Over the lifetime of the nuclear power plant, these models should be periodically updated with the available data (i.e. plant specific and/or generic data) and should be addressed in the safety demonstration.

4.132. Knowledge gaps and uncertainties regarding performance and reliability for an innovative technology should be adequately closed or reduced. In this regard, the safety assessment and available operating experiences for future similar nuclear power plants should provide insights for potential revision of the operational testing programmes and justifications for potential reduction of their scope.

Performance and condition monitoring to build operating experience

4.133. Routine surveillance tests should be established as part of equipment performance and condition monitoring for items important to safety that use innovative technology, in order to build up operating experience data. The safety assessment for the innovative technology should be used to set a proportionate scope of this monitoring.

4.134. For an innovative technology, targeted surveillance measures, such as enhanced non-destructive examination, vibration monitoring, equipment performance analysis and trend analysis, should be used to support reliability assumptions in the safety assessment and to verify pre-service testing results. These measures may be useful when there is limited experience with advanced manufacturing techniques, or for confirming that adverse conditions are adequately addressed in the safety assessment.

4.135. The safety assessment should be used to establish appropriate condition monitoring measures (e.g. visual examinations, predictive maintenance techniques) that capture specific failure modes unique to the innovative feature, and deviations from the conditions necessary to maintain the feature's specified performance. For an innovative technology, this may include innovative condition monitoring techniques or devices. For example, some non-water-cooled reactor technologies (e.g. using opaque, high density coolants) may need innovative sensors and instrumentation qualified for their challenging operating environments.

4.136. The safety assessment should include a description, with justification, of how an SSC that uses innovative technology will be inspected and maintained throughout its service life to prevent the failure modes identified in the safety assessment, in particular for passive systems (e.g. seismic isolator systems) and components.

4.137. Innovative monitoring techniques, such as those using machine learning methods for data analytics, may be used to augment established predictive maintenance techniques and technologies (e.g. vibration monitoring, temperature sensing) or assessment of non-destructive examination results for identifying degradation patterns and trends in order to minimize system or component unavailability, or to optimize maintenance. Any claims in the safety assessment for an innovative technology, which are based on such innovative techniques, should be supported by justifications that these techniques are sufficiently predictive and validated for their intended use, and should take the level of uncertainty and remaining knowledge gaps into account.

Use of digital twins for safety demonstration

4.138. Digital twins[10] in complex industrial and engineering applications have, in some cases, increased operational efficiencies, enhanced safety and reliability, reduced errors, created faster information sharing, and improved prediction capability, and record keeping. Digital twin technology is rapidly evolving and being integrated into nuclear applications for existing facilities and used in the development and deployment of innovative technologies. If digital twins are used to support the safety demonstration, the following considerations should be addressed:

(a) ***Virtual prototyping***: Digital twins can enable the creation of virtual prototypes of nuclear design concepts and dynamically simulate the behaviour of various reactor components, including the core, fuel assemblies, coolant systems and safety systems. This may allow designs to be tested and refined before physical construction, reducing the risk of errors if implemented properly. The architecture and process to build a digital twin model should be developed using an internationally recognized standard[11] and justified for use in the specific nuclear application.

(b) ***Model validation:*** The use of digital twins for safety demonstration should be based on a validation and verification process that ensures the digital twin models adequately represent the innovative technology within a nuclear power plant. Demonstrating the reliability and accuracy of digital twin simulations is vital for their use in safety assessment.

(c) ***Innovative sensors and instrumentation***: To deploy digital twins for some innovative technologies, it might be necessary to use innovative instrumentation and sensors that can operate in environments (e.g. temperatures, chemistry, radiation fields) that are more challenging than current light water reactors. These sensors and instrumentation should demonstrate that the requirements for environmental qualification, performance, reliability and maintainability will be met for the service life of the equipment in the innovative nuclear power plant. If innovative sensors and instrumentation are used in experimental testing of an innovative technology to collect data, they should undergo qualification to provide confidence in the fidelity and management of the acquired data to support safety demonstration.

(d) ***Cybersecurity, data protection and integrity***: Digital twins rely on vast amounts of sensitive data related to an innovative technology in a nuclear power plant. Assessment of cyber risks to the digital twin should be performed and adequate countermeasures proposed. Digital twin data collection should be established and maintained under a quality assurance programme or equivalent, if used in experimental testing, sufficient to support the safety assessment for an innovative nuclear power plant.

[10] A digital twin is considered to be a virtual representation of an object, process or system that spans its service life, is updated using real-time data, and uses simulation, machine learning and reasoning to help decision making. A digital twin may include various types of models, data and framework to produce knowledge or insights about the represented object, process or system. Using the virtual prototype, various simulations can be performed to test how the product behaves under different conditions. For example, engineers can simulate mechanical stress, heat, or vibrations to see how the product performs under strain, or test how software and hardware components interact.

[11] For example, the International Organization for Standardisation (ISO) standard ISO 23247 [22] specifically addresses the use of digital twins in manufacturing and provides a framework for the development and use of digital twins in various industries.

(e) ***Model lifetime management***: A structured change control process should be established for the digital twin to ensure that modifications do not compromise safety or security. A digital twin model maintenance process should be defined, taking into consideration operating experience feedback. Where a digital twin model is used to support safety related uses, the lifetime management process should extend to the safety demonstration of the SSC or reactor itself, as appropriate.

4.139. The safety assessment should investigate the impacts of using digital twins on human factors (e.g. over-reliance on the model, misunderstanding of the limitations in the simulation scope).

LACK OF APPLICABLE REGULATIONS, CODES AND TECHNICAL STANDARDS

Regulatory cooperation

4.140. Depending on the degree of innovation, vendors or potential operating organizations of an innovative technology should seek early engagement with relevant regulatory bodies on applicable regulations, codes and standards. Topical reports as well as proposal or position papers (e.g. white papers) can be used to seek early regulatory feedback. In this context, conducting a regulatory gap analysis should be performed in pre-application or early application stages to ensure completeness of the safety analyses and regulatory review.

4.141. If regulations or an established regulatory approach cannot be applied to an innovative technology with reasonable judgement, or if their application would lead to disproportionate outcomes, optimized solutions should be sought (e.g. adaptation of an existing regulatory approach or exemption from existing requirements with sufficient technical justification). In this context additional regulatory approaches might be needed to address these gaps where appropriate.

4.142. It may occur that an innovative technology is introduced in a Member State where at least some of the national regulations are not applicable. In this case, the relevant regulatory authorities should seek timely exchange and early cooperation on innovative technologies of mutual interest, while systematically building their technical capabilities to cope with innovative technology in general. Where appropriate, a common understanding of regulatory positions on specific innovative technologies should be found. Innovative technology vendors, operating organizations, and other potential interested parties should foster regulatory cooperation by engaging with relevant regulatory authorities in a timely manner and by contributing to a framework that enables such cooperation. For regulatory clarity, national regulations should be updated to reflect any new generic expectations of innovative technology, where needed.

Codes and standards relevant to safety assessment

4.143. To the extent practicable, the relevant existing codes and standards should be used for the design of an SSC that uses innovative technology. Where no nuclear-specific codes and standards are available, the safety assessment should consider the use of other industry standards. These should be augmented with additional requirements to adapt them to nuclear power plants where necessary.

4.144. Gaps in existing codes and standards for an innovative technology should be identified and addressed in the safety assessment. The insights from safety assessment should be used to understand the relevance of any gaps in codes and standards.

4.145. The safety demonstration should include input from expert panels or independent subject matter experts to compensate for gaps in codes and standards. The provisions for expert input should establish the composition of panels and the qualifications and independence of the experts used, the decision making criteria, and how the recommendations and observations made are documented and transparently addressed.

4.146. An independent expert review of design specific standards to assess gaps in their applicability to SSCs important to safety that use innovative technology should be considered.

4.147. An expert elicitation process should be implemented in a manner that minimizes bias in expert judgement (e.g. confirmation bias, anchoring bias) affecting overall safety conclusions (see also para 3.22 of this Safety Guide). Several techniques may be employed to estimate quantitative parameters and qualitative knowledge extraction such as the use of a phenomena identification and ranking table (PIRT).

4.148. Consideration should be given to developing a new code or standard for the innovative technology or updating an existing code or standard, for example to make it more technology-inclusive or flexible.

4.149. To update existing codes and standards or develop new ones, valid data are needed. Consequently, vendors and their suppliers should consider early collaboration with other interested parties on producing or obtaining such data. For example, supporting the development of a new code or update of an existing code for the use of innovative materials or advanced manufacturing technologies could be a reason to manufacture a demonstrator component and to plan dedicated experiments and tests.

5. SAFETY DEMONSTRATION OF SPECIFIC INNOVATIVE TECHNOLOGIES

5.1. This section provides recommendations on specific innovative technologies to meet the challenges for safety demonstration identified in Section 3. These recommendations complement, and should be implemented in conjunction with, the recommendations provided in Section 4 on strategies to address generic challenges for safety assessment of innovative technology. Each sub-section below starts with the descriptive paragraph presenting the innovation under the discussion and specific challenges for safety demonstration connected with it.

FUEL CONCEPTS

5.2. New types of nuclear fuel, or existing fuel types used under different conditions, are often associated with a lack of knowledge regarding their behaviour in different plant states and throughout the fuel lifetime. Also, sometimes there are claims from the designers regarding inherent safety of the fuel, which need to be properly demonstrated. For an adequate safety demonstration, it is important to have a comprehensive understanding of the potential issues and knowledge gaps associated with new fuel concepts.

5.3. Innovative aspects, issues and knowledge gaps related to new fuel concepts should first be identified, and specific strategies should be defined to address the corresponding challenges for safety assessment following a graded approach.

5.4. The identification of innovative aspects, issues and knowledge gaps should be comprehensive and consider the full lifetime and all applicable uses related with new fuel concepts, including on-site transport, handling and storage, both for irradiated and unirradiated fuel. In addition, fuel cycle back end and long term management should also be considered.

5.5. To identify the novel aspects, issues and knowledge gaps for a specific new fuel concept, a good understanding should be reached of how the fuel is contributing to assuring the fundamental safety functions (i.e. reactivity control, heat removal and confinement of radioactive material).

5.6. Corresponding fuel design limits should be established within the safety assessment, taking into account the level of uncertainty and knowledge gaps for the innovative fuel. It is considered good practice to select margins in design limits that allow appropriate decoupling criteria for fuel safety assessment and other parts of the overall safety assessment to be defined.

5.7. The adequacy of fuel design limits should be demonstrated as part of the fuel qualification programme, which should include experiments and testing. The fuel qualification programme should address normal operation, anticipated operational occurrences, design basis accidents and design extension conditions. Fuel experimental data should be collected under an appropriate quality assurance programme that meets applicable regulatory requirements. As fuel performance during operational states and accident conditions is dependent on fuel manufacturing parameters, this should be covered in the fuel manufacturing specification. As fuel qualification can take a long time, reference can be made to actions such as those mentioned in paras 3.17 and 3.20 to address any temporary knowledge gaps.

5.8. In some innovative nuclear power plant designs, greater emphasis has been placed on the inherent characteristics of the fuel (or fuel/moderator combination) to retain fission products, compared to existing reactor fuels[12]. Such claims should be considered in the safety assessment and adequately demonstrated against the corresponding safety performance criteria and the defence in depth requirements. In this context, fuel safety performance criteria could include aspects such as limits on the permissible leakage of fission products from the fuel in operational states and accident conditions, or maximum fuel failure fraction (i.e. ratio of failed fuel in the core, in the sense of loss of confinement function).

5.9. Cooling of the fuel is one of the main safety functions that should be thoroughly analysed and demonstrated in accordance with the corresponding design criteria (e.g. temperature limits).

5.10. Specific internal hazards (e.g. hydrogen production, corrosivity, potential for chemical reactions) may originate in the fuel for some innovative concepts. These specific hazards related to new fuel concepts should be systematically identified and considered during the safety assessment.

5.11. The safety demonstration for new innovative fuel concepts generally needs sufficiently predictive and validated computer codes to support safety assessment. The necessary computer codes may not be readily available or not adequately verified and validated, and the addition of new fuel concepts might be a lengthy process involving significant research and development. Where such tools are not available, the recommendations provided in paras 4.89-4.109 should be followed to provide such simulation codes as and when they are needed.

[12] For example, tri-structural isotropic (TRISO) fuel credits a series of barriers (including barriers within and outside the fuel system) to prevent the release of radioactive material and is claimed to implement a function of confinement of radioactive material in all plant states. It needs to be noted that it is expected that the nuclear power plant designs which use the innovative technology will meet the containment requirements stated in IAEA Safety Requirements.

NON-WATER REACTOR COOLANTS

5.12. Innovative nuclear power plant designs may use non-water reactor coolants, such as sodium, lead, lead–bismuth, helium or molten salts. These coolants bring advantages and disadvantages compared to water, with potential impacts on safety. Importantly, non-water reactor coolants may impact all aspects of the safety demonstration throughout the lifetime of the plant. While there is some operating experience with sodium fast reactor coolant there is still limited knowledge and operating experience for these coolants, in particular considering new design concepts proposed worldwide (see, e.g. Ref. [23]).

5.13. In order to ensure an adequate safety demonstration, it is important to properly identify all challenges and knowledge gaps associated with the use of non-water reactor coolants that are relevant to safety. The novel aspects, challenges and knowledge gaps related to non-water reactor coolants should be identified as early as possible in the design process of the reactor and should be used as a basis to outline the strategies for safety demonstration in this context.

5.14. The identification of novel aspects, knowledge gaps and challenges in using non-water reactor coolants should be comprehensive and should cover the entire lifetime of the installation, including coolant quality (e.g. purity) at the beginning of operation, coolant chemistry in operation, management of impurities, coolant storage during inspection, and used coolant management and disposal.

5.15. Considering the significant differences in properties between water and other coolants, such as liquid metals, molten salts or gases, the role of the coolant in the fulfilment of the fundamental safety functions (i.e. reactivity control, residual heat removal and confinement of radioactive materials) should be properly understood and considered in the safety assessment. One example is the confinement capabilities of some non-water coolants in design basis accidents and design extension conditions.

5.16. For non-water reactor coolants an extensive characterization of the coolant's physical properties should be established by appropriate testing in all anticipated conditions, especially regarding neutronic and thermohydraulic properties, in order to correctly model, in the safety assessment, the coolant behaviour in all plant states including design extension conditions. For instance, the thermo-physical properties of a molten salt may be a function of the composition of the salt (fluoride or chloride or a mixture) and any fuel (uranium, plutonium, thorium) that is dissolved in it.

5.17. Non-water reactor coolants interact with SSCs exposed to them and can affect the reliability of such SSCs over their service life. This aspect can have a significant impact on the safety demonstration. The reliability of SSCs operating in contact with non-water reactor coolants should be demonstrated using appropriate methods, including research and development and testing.

5.18. One of the important characteristics of non-water reactor coolants is their chemistry. The impact of the coolant chemistry on the safety demonstration should be assessed and it should be demonstrated that the design provisions for chemistry control are reliable and effective to maintain the coolant at its specified conditions from normal operation up to accident conditions.

5.19. Non-water reactor coolants might raise specific considerations regarding the coolant parameters and undesirable phenomena (e.g. freezing of the liquid metal coolants) that might result in failure to fulfil safety functions. The adequacy of the coolant parameters in different plant states should be demonstrated. This should include the definition of a safe shutdown condition for safety assessment. Examples of relevant parameters for lead–bismuth cooled reactors are the minimum operational temperature (to avoid freezing) and the oxygen concentration (to control liquid metal corrosion). Freezing is also an issue for lead, sodium and molten salt coolants.

5.20. The characteristics of non-water reactor coolants may change over the operation of an NPP . This change can have a significant influence on the validity of the safety demonstration over the lifetime of the installation. The phenomena that can lead to the alteration of the coolant properties should be identified and the approaches to manage these should be demonstrated. An example of this in a lead–bismuth cooled reactor is the accumulation of corrosion products in the coolant due to exposure to oxygen, especially during maintenance operations. In a molten salt reactor where fuel is dissolved in the salt, the composition of salt evolves as fission products appear and that can affect the neutronic, physical and chemical properties of the salt.

5.21. For innovative non-water cooled reactors, the supporting systems to maintain coolant chemistry can differ significantly from those in water cooled reactors and might be associated with significant source terms or need continuous processing of contaminated fluids for operational discharges or storage. The safety assessment should consider these systems both in operating and in accident conditions.

5.22. Using non-water reactor coolants could be the cause of new initiating events, new types of accident consequences or new internal hazards. It could also lead to new locations for radioactive inventories that can contribute to an accident level source term and need to be confined. New initiating events, new internal hazards as well as the potential for the coolant to represent a significant source term should be comprehensively identified and a robust safety demonstration of the design's ability to cope with these new challenges should be made. An example of the coolant constituting a significant source term on its own is the case of a lead–bismuth cooled reactor where, through the irradiation of bismuth, polonium is generated, and in the case of an initiating event involving coolant–water interaction the polonium can became volatile. In a molten salt reactor where fuel is dissolved in the salt, the salt itself contains a part of the fission products and represents a major potential source term.

5.23. Innovative reactor coolants interact with corrosion products and fission products, and these are transported, as solutes or dispersed, with the coolant in the reactor circuit and into connecting systems, like purification systems and cover gas treatment systems. The safety assessment should identify locations where such contaminations can accumulate, such as in filters or through gassing. The safety assessment should further show that provisions to maintain safety features are effective and that radiation risks posed by such accumulations are controlled. One example is the transport of activated dust in a gas cooled reactor, which can constitute a radiation risk during maintenance.

5.24. Innovative reactor coolants can transport corrosion products and activated impurities that can significantly increase the source term in scenarios with coolant spilling or coolant interactions with other fluids. The safety assessment should include the identification of relevant corrosion products and impurities, the determination of their concentrations and their impact on the source term.

5.25. If non-water reactor coolants are used in a nuclear power plant for which there is a limited or lacking operational experience with the coolant, then it should be taken into account in the safety demonstration and appropriate tests should be performed to confirm the maintenance, surveillance, operating and emergency operating procedures. The recommendations provided in paras 4.110-4.138should be followed.

5.26. Non-water reactor coolants might pose challenges for the surveillance, inspection and maintenance of SSCs important to safety or the plant overall. Such challenges should be identified.

The safety assessment should include provisions to overcome these challenges and demonstrate that they are effective over the service life of the technology. For example, commonly used non-destructive testing, monitoring and surveillance techniques do not work reliably for opaque coolants (e.g. liquid lead).

5.27. The safety demonstration for non-water reactor coolants in an innovative technology generally needs sufficiently predictive and validated computer codes to support safety assessment. The necessary computer codes may not be readily available or not properly verified and validated, and the addition of non-water reactor coolants might be a lengthy process involving significant research and development. Where such tools are not available, the recommendations provided in paras 4.89-4.109 should be followed to provide such simulation codes as and when they are needed.

5.28. Where non-water reactor coolants or working media for items important to safety are used, codes and standards for demonstrating the safe and robust design of SSCs might not be fully applicable. Also, requirements in applicable regulations might need derogation or interpretation. These challenges should be identified in a timely manner and the recommendations provided in paras 4.140–4.149 should be followed to address these challenges.

5.29. Some non-water reactor coolants are used in high temperature ranges that exceed the scope of existing codes and standards for component design and manufacturing. Specific requirements for design, selection of material, manufacturing and control should be developed in order to properly address conditions relevant to non-water reactor coolants.

5.30. Non-water reactor coolants might pose challenges for the safe management of radioactive wastes and discharges resulting from their specific properties and characteristics and their incompatibilities with existing waste management and treatment techniques. Components operating in these coolants might also represent challenges for safety in decommissioning stage. The safety assessment should identify all radioactive waste that may be produced from the use of non-water reactor coolants and demonstrate that they can be safely managed during the operational lifecycle.

PASSIVE SAFETY FEATURES

5.31. Innovative nuclear power plant designs may rely on passive features and inherent characteristics both to simplify the design, and in an effort to lower overall plant risk. A passive safety feature is considered to be a system composed entirely of passive components and structures or a system that uses active components in a very limited way to initiate subsequent passive operation. Many passive cooling systems rely on natural phenomena, such as natural circulation, to perform their safety functions. In some cases, in particular for relatively small reactor designs, the natural circulation involves relatively small driving forces that can, in principle, be easily disrupted.

5.32. Passive safety systems relying on small driving forces may have a narrow range of conditions under which they can effectively perform the safety function. The following aspects should be recognized and, when relevant, addressed in the safety demonstration for such systems:

(a) Potential failure modes and corresponding impact on system operation (i.e. comprehensive knowledge and understanding of phenomena that could influence the performance or failure of a passive safety feature considering the driving forces involved);
(b) The impact of initial plant conditions (e.g. power operation) on passive safety features performance;

(c) The impact of environmental conditions on passive safety features performance;
(d) The dynamic behaviour of the performance of passive safety features, including possible degradation of effectiveness in the long term;
(e) Evaluation of potential adverse plant system interactions;
(f) Application of margins, to avoid cliff edge effects (see SSG-2 (Rev. 1) [8]); since the range of conditions necessary to perform the safety function might be narrow for passive safety features, a limited change of these conditions might be challenging.

5.33. An adequate safety demonstration of passive safety features involving small driving forces should consider the followingapproaches to ensure overall reliability:

(a) Scale testing of the passive safety features, including testing of possible flow disruption mechanisms;
(b) Computer code modelling of the passive safety features with codes validated using the testing;
(c) Plant startup testing and validation of the expected system response against the computer code modelling. For example, measuring the heat removal at low power to ensure that the system is responding as expected.
(d) Periodic testing in order to detect any possible degradation in time.

5.34. During the safety assessment specific attention should be paid to conditions resulting from internal and external hazards to confirm that the necessary boundary conditions to have a successful operation of the passive safety features are met. Some passive safety features (e.g. those involving relatively small driving forces) may be more sensitive to environmental changes induced by environmental conditions. The factors to be evaluated therefore include the following:

(a) Environmental conditions that change air temperature, moisture and particle concentration in the air for a system that uses the atmosphere as heat sink;
(b) Fire that could modify the necessary temperature distribution in a system that uses buoyancy for fluid circulation;
(c) Pipe deformation in case of a seismic event or load drop for a system that uses natural fluid circulation.

5.35. The safety assessment of passive safety features should be updated, when necessary, based on continuous monitoring during operation to ensure that the system remains operable. Monitoring should include trend analysis to look for potential system degradation such as flow blockage or heat transfer mechanism degradation. It should account for possible ageing mechanisms identified.

5.36. Passive safety features that utilize stored energy (e.g. pressurized injection tanks) or large volumes of water to cool the reactor vessel or fuel, are less likely to be influenced by small changes in conditions as the driving forces to provide their safety function are often not small The safety assessment should consider if the monitoring, inspection and testing provisions are adequate to ensure that the specified energy content is available for the system to provide its safety function when needed (see also Requirement 29 of SSR-2/1 (Rev. 1) [3]). In addition, the assessment should consider the adequacy of provisions to maintain the reliability of the passive safety feature and any support systems (see also Requirements 23 and 27 of SSR-2/1 (Rev. 1) [3]), and the limits on conditions placed on the system (see also Requirement 28 of SSR-2/1 (Rev. 1) [3]), taking into account the uncertainties and state of knowledge associated with the innovative technology. Moreover, a passive safety feature’s performance should be investigated with computer code modelling, using sufficiently validated and predictive codes. In addition, strategies for plant startup validation testing as well as suitable on-line testing and inspections should be developed.

5.37. For safety assessment purposes, when quantifying the reliability of passive safety features, consideration should be given to the occurrence of root causes that may prevent the safety function being delivered by the passive safety features due to the range of conditions under which it has to initiate and maintain its performance. During safety assessment, consideration should be given to those parameters that may change, and to the potential causes of these changes (e.g. impact induced deformation, ageing) with due consideration of uncertainties. As part of the defence in depth evaluation of the innovative NPPs , the possibility of a common cause failure of a passive features should be evaluated to the extent practicable.

5.38. The use of the same safety feature in several levels of defence in depthshould be avoided to the extent practicable. If not practicable, thisshould be addressed in the safety assessment and thoroughly justified (e.g.high confidence in high reliability of the passive safety feature).

5.39. For passive safety features with small driving forces, the modelling of several physical phenomena with high accuracy is often needed.The scope of the validation of computer codes used for modelling of those phenomena should take into account the results and date obtained from the adequate tests. The uncertainties associated with computer code results should be evaluated and taken into account in the safety demonstration.

5.40. Although the operator actions and active components are used in a very limited way to initiate operation of passive safety features, the sensitivity of such features to adverse human failure events or active components failures should be carefully considered. It should be demonstrated that the required measurements to assess the correct operation of the passive features are available to the operator in all relevant plant states.

INSTRUMENTATION AND CONTROL

5.41. NPPs may rely on some innovative aspects of instrumentation and control systems or innovative use of existing technologies. It includes aspects such as the use of fully digital instrumentation and control platforms, extensive use of instrumentation and control components not specifically developed for nuclear applications. The use of digital I&C in NPPs can be described as state of the art and offers many advantages for the plant operator, including self-monitoring features, a reduced number of periodic inspections, automation of complex functions, performance of complex calculations and improved operating and monitoring functions. On the other hand, the increased complexity results in new fault postulates such as a postulated software CCF of a control system or risks due to cyber-attacks. This has led to the expansion of measures for diversification (e.g. use of new analog platforms for backup I&C systems) and the allocation of I&C systems/components to security zones.

5.42. The introduction of new, innovative technologies for NPPs usually results from applications in conventional applications. While the use of industrial technology is widely accepted for non-safety-relevant functions, the use of so-called commercial-off-the-shelf products for safety functions is typically restricted to Digital Devices of Limited Functionality. In case of the use of components not specifically developed for nuclear application in the I&C architecture, the safety demonstration should include the appropriate proof of qualification, considering the safety relevance of the functions to be realized and potential failure consequences.

5.43. Where instrumentation and controls are designed and used in an innovative manner, as described in para. 5.41, this can introduce new and unique initiating events, as well as unique accident sequences or system responses to these events, which should be systematically considered in the safety assessment. For example, there might be a potential for automation to disable operator functions or provide misleading indications to the operator that can potentially result in inappropriate operator responses.

5.44. Consideration of potential common cause failures of the instrumentation and control systems is an integral part of the safety assessment (see Requirement 24 of SSR-2/1 (Rev. 1) [3] and paras 4.25–4.40 of IAEA Safety Standards Series No. SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants [24]) and is fully applicable also to innovative nuclear power plant designs. Special emphasis should be placed on the cases when diversity is being demonstrated by applying two different digital systems (without analogue systems being used in the design).

5.45. The use of innovative instrumentation and control systems can result in new or unique failure modes that should be systematically considered in the safety assessment. Prominent failure analysis methods (e.g. failure mode and effect analysis, system theoretic process analysis) can be used to systematically identify the potential failure modes, including software failures, to be considered in the safety assessment.

5.46. The introduction of innovative software-based instrumentation and control systems may lead to new vulnerabilities to cyber-attacks, which should be analysed in terms of potential impact on safety. The interfaces between safety demonstration and security related issues are further discussed in Section 6.

ARTIFICIAL INTELLIGENCE SYSTEMS

5.47. One of the innovative technologies which might be tried to be used by NPP designers is artificial intelligence. It is to be noted that the term AI covers a wide range of technologies, ranging from symbolic AI and probabilistic inference to machine learning, deep learning and hybrid techniques. Many of the recent advances in AI concern data-driven techniques which have been made possible by developments in supporting micro-electronic technologies. Due to the differing nature of AI technologies, and differences in their relative maturity levels, the AI solution needs to be chosen according to the specifics of the use case and its safety demonstration is to be adapted accordingly. The AI applications are not widely used currently and there is a lack of experience on this topic. The challenges for safety demonstration of AI are connected with factors such as the impact of data on model development and performance, algorithmic approach, integration with new and existing engineered systems, uncertainty analysis, and aspects of human interaction.

5.48. The safety demonstration should provide evidence that the potential adverse effects of AI systems failures and limitations do not affect fulfilment of the fundamental safety functions.

5.49. It should be demonstrated through a suitable approaches that the use of AI will not degrade overall level of safety. This demonstration should systematically consider the uncertainties associated with AI.

5.50. If operators are relied upon or receiving an input from AI systems to accomplish safety actions, it should be demonstrated that AI systems do not impede operator performance including the ability to perform those safety actions within the necessary response times.

HUMAN AND ORGANIZATIONAL IMPLICATIONS

5.51. Human and organizational arrangements for innovative nuclear power plant designs might significantly differ from established practices at conventional NPPs. Innovative designs may incorporate innovative human–machine interfaces, innovative modes of operation (e.g. remote operation with no human presence on site), or the use of artificial intelligence and machine learning to support operators. Technologies such as passive safety features, innovative automation systems and accident tolerant fuels can also be leveraged to extend fault response timelines before an operator needs to act. In some cases, the operator could be removed from the fault response entirely. These innovations are closely related to the safe operation of the plant and should be subject to a comprehensive safety assessment (see also SSG-51 [16]).

5.52. For modes of operation involving significantly reduced staffing or no permanent operator presence on site, the safety assessment should demonstrate that the proposed model allows for sufficient staff to be available to respond and implement the required safety functions in a timely manner in scenarios where human action is necessary, including design extension conditions. In this context, the safety assessment should include systematic identification of potential scenarios, considering all potential plant modes of operation, equipment failure modes and system configurations. The safety assessment should also consider the provisions, feasibility and time needed to transport off-site operators and other responsible staff to the site for all relevant accident scenarios.

5.53. The indirect consequences of small numbers of staff on site should be considered within the safety assessment as it reduces the capacity to detect, manage and recover from faults as a consequence of there being fewer or no people on site interacting with the plant.

5.54. Some innovative nuclear power plant technologies propose multi-module deployment in a single unit, some with a single shared control room. It should be demonstrated that effective oversight and control can be achieved during all plant states, including accidents, considering the complex interactions which could occur, and the associated operator responses. Providing appropriate evidence for the safety demonstration may involve practical demonstration of the ability to control (e.g. using data from simulators).

5.55. The small size and integral nature of many innovative nuclear power plant designs potentially impacts the ability to conduct reliable examination, maintenance, inspection and testing activities. The safety demonstration should show that the design supports reliable examination, maintenance, inspection and testing to validate the required reliability and performance of SSCs.

5.56. In the event of an internal or external hazard or an accident scenario affecting multiple reactor modules, the operators may have to prioritise their response to the most risk-significant units. The risk significance might change depending on accident progression and failures observed on site, as well as lack of information, which is typical for fault scenarios. This activity has the potential to be extremely cognitively demanding, which should be systematically addressed in the safety assessment, considering the entire spectrum of scenarios affecting multi-module designs and appropriate mitigations put in place.

5.57. Where innovative technology is used to organize the information provided to the operators, such as filtering, or consolidation of alarms / warnings, the safety assessment should extend to this interface to the operators and demonstrate that it is effective and reliable. In case operating advice is provided by the innovative technology, it should be shown that this advice is consistent with the operating or emergency procedures for all relevant conditions of the nuclear power plant.

5.58. For nuclear power plants with potential for implementation of some functions remotely, the effectiveness of such functions should be demonstrated for all plant states considered in the safety assessment. The analysis should consider all relevant factors that might affect successful operator actions.

5.59. Some nuclear power plant designs might utilize innovative human–machine interfaces for which there is a lack of operating experience, reduced knowledge compared to existing interfaces, and large uncertainties related to their functionalities. In this context, a systematic identification of potential failure modes of the innovative interface should be undertaken. It should be demonstrated that the operators can reliably bring the plant to a safe shutdown condition for a degraded and/or fully failed human–machine interface. For example, the following faults may be relevant (see Ref. [25]):

(a) Failure of automation (e.g. failure part way through an automatic sequence, missing a step, completing a step when not all parameters are met, failure to conspicuously display when a failure in automation has occurred);
(b) Failures of data display (e.g. individual data point freeze, display freeze, conflicting sensor display, failure to access information in a timely manner).

5.60. It should also be demonstrated that the design of through-life component replacement and end of life decommissioning recognizes the practical challenges related to human resources and organizational aspects that exist in relatively small reactor designs, where physical space is limited.

5.61. Existing analytical models used for human actions might not be appropriate for some innovative technologies. Any gaps in modelling capability should be identified, systematically analysed and addressed within the safety assessment (e.g. implementing enhancement of the analytical models, development of new analytical models).

INNOVATIVE MATERIALS

5.62. Innovative technologies may utilize a range of materials that have not previously been used in a nuclear context or have been used in different conditions. Such materials may be used across the nuclear power plant design, including within the fuel and core design, as a moderator, within heat transport circuits or for structural elements. There is a potential lack of knowledge of the relevant failure mechanisms and ageing mechanisms of such materials. Also, the existing codes and standards might not cover such materials or the specific conditions in which they are used. Thus, extensive use of innovative materials might impact many aspects of the safety demonstration throughout the lifetime of the plant.

5.63. The benefits and detriments of an innovative material should be evaluated as choices to improve certain properties or characteristics might introduce other negative aspects that need to be managed. Thesenegative aspects should be identified and their overall impact on safety should be demonstrated.

5.64. Reliable data related to the properties of an innovative material should be obtained through a comprehensive test campaign to support the safety demonstration.

5.65. The potential impact of the innovative material applied in SSCs on their failure modes should be identified and assessed within the safety demonstration. This should include the intrinsic material properties and any aspects arising from their incorporation in an SSC (e.g. the melting point of a material might be altered if it is in contact with another material due to the formation of an eutectic system).

5.66. Any potential hazards posed by the innovative material should be identified (e.g. fire hazards if the material is flammable), and the generation, use and storage of hazardous materials should be considered in the safety assessment. Hazard management strategies should be developed based on the safety assessment results, along with relevant accident management arrangements.

5.67. Some innovative technologies utilize highly corrosive materials (e.g. molten salt). The safety demonstration for such technologies should include comprehensive identification of ageing and degradation mechanisms, and appropriate strategies for their safe management.

5.68. Innovative materials may also have the potential for undesirable interactions with other materials (including coatings claddings, nuclear fuel, coolant or fission products). These interactions might result in hazardous conditions (e.g. production of flammable or toxic gases and materials, rapid oxidation), accelerated degradation, or ageing phenomena. All relevant interactions with other materials and with the operating environment should be identified, and the behaviour of the materials should be assessed and appropriately addressed in the design and safety demonstration. Specifically, the safety assessment should show that the performance of the SSCs important to safety is tolerant of all possible conditions for all relevant plant states.

5.69. When innovative combinations of materials are used (e.g. the addition of a coating to fuel cladding to improve its behaviour under accident conditions), appropriate testing should be conducted to demonstrate that these innovations operate as expected.

5.70. Materials should be selected with appropriate consideration of their operational lifetime, processing, transport, storage and disposal. It should be demonstrated that the nuclear power plant design incorporates adequate provisions for the safe processing and storage of materials on the site. The safety demonstration should consider the location, physical and chemical form, and hazard potential of the materials. All relevant ageing and degradation effects (including combined effects) should be identified, and appropriate management strategies developed.

5.71. The safety assessment should identify all radioactive waste that may be produced from the innovative materials and demonstrate that they can be safely managed during the operational lifecycle.

5.72. The safety demonstration should also address longer term effects from fission product interactions for susceptible materials where such impacts might be revealed only after chronic long term exposure, for example because of low concentrations of radionuclide impurities in the coolant.

5.73. Where there are remaining uncertainties in the performance of innovative materials, it should be demonstrated that the design incorporates sufficient margins to ensure the required level of safety.

5.74. It should be demonstrated that the design includes suitable and sufficient features to address uncertainties related to specific material characteristics. The specific design features may include the following:

(a) Provisions for irradiation and neutron flux monitoring;
(b) Provisions for chemistry control and monitoring, sampling and analysis;
(c) Provisions for examination and representative testing of materials and components (during manufacture, in-situ and removed from plant);
(d) Provisions for inspection (manufacturing, pre-service, in-service and post-service inspection, as appropriate);
(e) Provisions for through life material monitoring (e.g. surveillance, sampling);
(f) Provisions for condition monitoring, including ageing;
(g) Measures to give sufficient forewarning of failure.

ADVANCED MANUFACTURING TECHNOLOGIES

5.75. Advanced manufacturing technology refers to the use of innovative and cutting-edge techniques, processes and tools in the manufacturing industry. Advanced manufacturing technology often involves intensive usage of tools such as automation, robotics, powder metallurgy, 3D printing, and other innovative technologies to improve productivity, performance and quality of components, and their production.[13] Advanced manufacturing technologies have been successfully implemented in a number of non-nuclear fields such as aerospace and medicine, and are emerging in the nuclear industry. The knowledge and operating experience of the failure mechanisms, reliability and ageing of advanced manufacturing technology tools is limited and might create challenges for the safety demonstration. For the purposes of this Safety Guide, advanced manufacturing technologies include those techniques and material processing methods that are not commonly used in the nuclear industry and have yet to be formally standardized through nuclear codes and standards or regulatory approval. Advanced manufacturing technologies cover a wide range of novel and non-standardized manufacturing methods, and in some cases, also involve the use of innovative materials.

5.76. Some Member States are developing strategies and guidance to allow broader use of advanced manufacturing technology in the nuclear sector. It is recognized that existing nuclear quality assurance programmes, certain industrial codes, and regulatory requirements establish broadly applicable requirements for the design, manufacturing, fabrication and testing of components that encompass the introduction of advanced manufacturing technologies into the nuclear sector[14]. The performance of the components or fuels manufactured using advanced manufacturing technology should be analysed in terms of the potential impact on nuclear safety. In the meantime, the safety assessment should demonstrate that existing safety programmes and protocols are suitable for the advanced manufacturing technology being used.

5.77. When introducing advanced manufacturing technologies to the nuclear sector, alternative manufacturing approaches may be proposed that do not apply directly existing standards or regulations. If such alternative approaches are permitted by the provisions of existing codes, criteria for those approaches should be developed, and the safety assessment should demonstrate that the criteria are met.

5.78. The safety demonstration should address the availability of applicable codes and standards, existing utilisation experience (when applicable), the safety significance of the SSC for which the technology will be used and other aspects related to the use of advanced manufacturing technology.

5.79. The safety assessment of the advanced manufactured SSC should consider, as appropriate, impact of surface contamination on potential dose during SSC removal formaintenance or during plant decommissioning (see SSR-2/1 (Rev.1) Requirement 12 [3]). In this case, the surface finishing of 3D-printed parts may have an impact on how easily radioactive materials can adhere

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[13] This Safety Guide provides recommendations on safety demonstration with regard to advanced manufacturing technologies and techniques in general. The specifics of each type of advanced manufacturing technology are beyond the scope of this Safety Guide.

[14] For example, 3D printing is emerging in nuclear manufacturing to produce complex and optimized on-demand parts at a reduced cost. 3D printing uses additive manufacturing techniques in the process of converting a digital model into a solid object. Subcomponents for innovative fuels (e.g. fuel debris filters, grid plates) for existing water-cooled reactors have already been produced using 3D printing techniques. 3D printing is also an approach to manufacture non-light-water reactor fuels.

to or be removed from the component. For components designed to be reused or decontaminated, surface finishing and post-processing treatments should be systematically considered.

5.80. The following generic aspects should be addressed in the safety demonstration of the application of the advance manufacturing techniques in innovative NPPs:

(a) Quality assurance: an acceptable process should be followed for the use of advanced manufacturing technologies to ensure adherence to quality requirements for SSCs manufactured using such technologies.
(b) Process qualification: steps should be taken to demonstrate that the component is produced with characteristics that meet the design requirements. The critical characteristics of an item important to safety should be identified and measured to demonstrate high quality of the fabricated component. Qualification testing should be conducted to evaluate the range of acceptable material properties such as tensile strength, hardness and chemistry, and to demonstrate that the design requirements are met.
(c) Supplemental testing: testing should be conducted to demonstrate that those material and component properties necessary to meet the design requirements are acceptable in the applicable service environmental conditions, and thus the performance of the component in service will be acceptable.
(d) Production process control and verification: steps should be taken to demonstrate that each component is produced in accordance with a qualified process.
(e) Performance monitoring: performance monitoring options should be examined, and it should be demonstrated that the component will continue to reliably meet its design requirements until the end of its intended service life.

The above and below paragraphs illustrate the importance of fabrication specifications play a critical role in ensuring the consistent quality and reliability of parts produced through additive manufacturing. Precise and well-defined specifications are essential because even slight variations in process parameters, material quality, or environmental conditions can lead to significant changes in the properties of the final product. For instance, factors such as build orientation, layer thickness, and temperature control directly influence mechanical properties like strength, ductility, and fatigue resistance. Without stringent control over these specifications, it becomes difficult to maintain consistency between parts manufactured at different times or across different machines, leading to variability in performance, and possibly unacceptable from a safety standpoint.

5.81. Regarding process qualification[15], the following aspects should be specifically considered in the safety demonstration:

(a) Manufacturing process: types of defect that could result from the manufacturing process, and the related important parameters that could affect the quality of the process;
(b) Raw material: essential characteristics with an impact on the material behaviour should be identified;
(c) Interface between the machine and the material: the validity domain of the machine regarding the manufacturing process and the material involved;
(d) Component behaviour: the behaviour of the component regarding the expected loads in operation and the environmental aspects (e.g. irradiation, corrosion, fatigue);
(e) Comprehensiveness: the material compliance across all geometrical points;

[15] Examples and additional technical insights regarding the qualification of advanced manufacturing technology could be found for instance in Ref. [26]

(f) Test coupons: the representativeness of the test coupons should be demonstrated.

NON-ELECTRICAL APPLICATIONS

5.82. Many innovative nuclear power plant designs, including small modular reactors, are being designed to support non-electrical applications (e.g. hydrogen production, heat generation, desalination), often via co-generation. Overall, the safety assessment framework requirements established in GSR Part 4 (Rev. 1) [2], and the recommendations provided in SSG-2 (Rev. 1) [8], SSG-3 (Rev. 1) [9], and SSG-4 (Rev. 1) [10] are applicable, however they should be applied taking into consideration the unique conditions relevant to the innovative technology involved in non-electrical applications.

5.83. The safety assessment should identify any new types of postulated initiating event, initiating event and failure mode that might be triggered by the specific technology and site configuration under consideration. The safety assessment should consider all damage mechanisms and hazards arising from the specific design (e.g. building damage from a detonation or fire in the hydrogen production unit).

5.84. Postulated initiating events and initiating events related to the 'non-nuclear part' (i.e. related to the non-nuclear energy conversion and its utilization) of the plant should only be grouped with the postulated initiating events and initiating events occurring in the 'nuclear part' if they are similar with respect to required safety functions. If postulated initiating events and initiating events are grouped in this way, the safety assessment should still allow the impact of the 'non-nuclear part' on safety to be assessed.

5.85. The impact of new postulated initiating events and initiating events that result from the non-electrical application should be considered. For example, if the steam line is designed to supply both a turbine generator and steam supply lines (which can be used for industrial applications, district heating or desalination), postulated initiating events and initiating events related to both the turbine generator and steam supply lines should be considered in the safety assessment.

5.86. The list of postulated initiating events and initiating events for the safety analysis of nuclear power plants with non-electric applications should include, where applicable, the flow of any hazardous material from the supported industrial application back into the plant.

5.87. The plant response to new postulated initiating events and initiating events associated with non-electrical applications should be analysed in sufficient detail in the safety assessment. For example, in the case of a shared steam line between a turbine generator and the district heating lines, if the heating line is isolated, the plant response would depend on whether the turbine generator can accept the increased steam flow, or whether the reactor can run-back given the loss of load.

5.88. In applications where there is a potential spread of radioactive material from the 'nuclear part' to the 'non-nuclear part' of the plant, the potential contamination should be considered in the safety assessment.

5.89. In the case of nuclear power plants that can operate in multiple configurations of the 'nuclear part' and 'non-nuclear part', each of the configurations should be considered in the safety demonstration. It should be demonstrated that each configuration is bounding for a given context, and then it may be possible to simplify the safety assessment by providing proper justification. For example:

(a) A steam line might be designed to supply both a turbine generator and district heating lines, and the plant is designed to operate with either of the end loads isolated. In this case the safety demonstration should consider all of the possible configurations. Additionally, the potential supply of decay heat (following a reactor trip) to the district heating lines should be considered as a potential complexity to be considered for shutdown operational states.
(b) Heat storage, such as molten salt storage tanks, might be used to supply a turbine generator with increased steam flow during times when peak power is needed, with reduced steam flow during times of lower electricity demand. In this case, the range of power outputs to the turbine generator should be considered in the safety assessment, unless it can be shown that the variation in the output does not impact safety. Generally, the reactor power remains unchanged, while the turbine generator output is variable. However, changes in the turbine generator output can result in an overall increased likelihood of the failure of the turbine generator and supporting equipment, which also should be considered in the safety assessment.

5.90. Potential configurations mentioned in the previous paragraph may also include aspects related to the location of the 'nuclear part' and 'non-nuclear part' of the plant. If, for example, the hydrogen production application is nearby, a potential hydrogen detonation should be considered in the design, and the safety assessment should consider whether the hazards can affect safety functions for the different levels of defence in depth. If the hydrogen production plant is not located near the reactor site, but still supplied by steam, the implications of a long coolant or steam line should be considered in the safety assessment (e.g. a higher likelihood of steam line breaks).

5.91. For potentially hazardous applications, such as hydrogen production or support for a nearby chemical or industrial facility, all associated internal and external hazards should be systematically identified and evaluated in the safety assessment. The hazards are expected to be site- and design-specific as discussed in IAEA SSG-79 [27]. Some examples of the sources of hazards that might be relevant are:

(a) Substantial stockpiles of synthesis gas and natural gas feedstock;
(b) Potential releases of hazardous chemicals that are used in the process and recycled during thermochemical processes (these might exist in different states such as liquid, a mist, or a gas depending on the temperature and pressure);
(c) Hydrogen explosion and combustion hazards, hydrogen embrittlement;
(d) Hazards connected with flammable and toxic materials contained in the system including toxic releases affecting control room habitability;
(e) Transport of radioactive material, such as tritium, from the reactor core to the non-electrical application part;
(f) Thermal turbulences caused by issues in the co-generation applications.
(g) Transport of chemicals produced by the nearby chemical facility.

5.92. If a nuclear power plant designed for non-electrical applications is intended to be located near an industrial complex area containing a number of other industrial facilities which might or might not be supported by the plant, the potential hazards from these industrial facilities should be identified and evaluated in the safety assessment, following the relevant recommendations provided in SSG-2 (Rev. 1) [8] and SSG-3 (Rev. 1) [9].

5.93. With regard to safety assessment for external hazards, recommendations are provided in SSG-2 (Rev. 1) [8], SSG-3 (Rev. 1) [9], IAEA Safety Standards Series Nos SSG-68, Design of Nuclear Installations Against External Events Excluding Earthquakes [28], and SSG-67, Seismic Design for Nuclear Installations [29]. These safety standards recommend that off-site induced effects of the external hazards (e.g. destroyed infrastructure outside of the nuclear power plant due to external hazards) should be considered in combination with the external hazard direct impacts on the plant equipment. In this context, if potential external hazards affecting the plant under consideration can also impact nearby industrial facilities, this impact should be included in the safety assessment. For example, if a seismic event can result in a release from a nearby chemical facility, the release should be considered in the safety assessment for this event such as the impact on operator responses. The safety assessment should include evaluation of different modes of operation of both the nuclear plant and the non-electrical facility.

5.94. The safety assessment should consider the potential impact of the operational changes or events in the non-electrical application or plant (e.g. load changes in the non-electrical plant).

MULTI-MODULE DESIGNS

5.95. Some innovative designs (e.g. small modular reactors) include more than one reactor module in a single unit. Multi-module designs may have shared structures and systems, and there might be interactions or other dependencies between the modules that affect safety. The interactions between the modules could create additional hazards, but bring also opportunities in terms of enhancing safety (e.g. the use of equipment from one module for the needs of another module). This complexity should be addressed in the safety assessment in a relevant manner.

5.96. The specific safety considerations associated with multi-module designs should be systematically identified and considered in the safety assessment. The potential safety considerations include aspects such as: the potential for propagation of a fault scenario occurring within one module to others, the potential for common cause failures between modules, issues related to human interactions, and conflicting procedures for emergency preparedness and response. More detailed recommendations are provided in SSG-3 (Rev. 1) [9] and SSG-4 (Rev. 1) [10][16].

5.97. Potential combinations of operational states and configurations for different reactor modules should be considered in the safety assessment. Some combinations may be eliminated from the safety assessment with justification (e.g. operating rules forbidding certain configurations, or their non-feasibility).

5.98. Multi-module initiating events and the propagation of an initiating event or accident from one module to another should be systematically identified and considered in the safety assessment. These events might be caused by the interconnections between modules, shared SSCs, or close physical proximity of modules.

5.99. The potential for common cause failures affecting several modules at the same time (e.g. owing to shared SSCs and operators or the impact of internal or external hazards) should be systematically identified and considered in the safety assessment.

5.100. The impact of multi-module designs on human factors engineering should be considered in the safety assessment. This includes the following aspects:

(a) Human–machine interfaces and staffing, in the case of shared control rooms;
(b) Emergency or reserve shutdown panels, in the case of shared control rooms;
(c) Emergency response (considering all potential configurations);

[16] Additional technical insights can be found in Ref. [30].

(d) Examination, maintenance, inspection and test activities.

More detailed recommendations are provided in SSG-3 (Rev. 1) [9] and SSG-51 [16] and technical insights can be found in Ref. [19].

5.101. The arrangement of several reactor modules within the same unit might be associated with new kinds of internal hazards. The potential to generate internal hazards in a multi-module unit, while erecting, commissioning, operating, maintaining, transporting or dismantling a module, should be systematically considered in the safety assessment. The safety assessment should demonstrate that the provisions to prevent the hazards associated with such activities are adequate. For instance, if constructing a new module while other modules are already in operation, there is the potential for specific events such as the drop of the module during installation. Another example is an event in one module while connecting an additional module to the plant services, such as off-site power or cooling water.

5.102. Potential common issues for emergency preparedness and response during accidents involving more than one module should also be considered in the safety assessment.

5.103. Deterministic safety analysis and probabilistic safety assessment should systematically address multi-module considerations in the corresponding methodologies. The approach described in Ref. [31] is dedicated to multi-unit probabilistic safety assessment but could be directly adapted for the implementation of the multi-module probabilistic safety assessment. When adapting the safety assessment methodology, the level of detail, complexity and modelling efforts should be commensurate with the specific design phase and the potential safety impact of multi-module considerations, striking a balance between the efforts and risk insights obtained.

TRANSPORTABLE NUCLEAR POWER PLANTS

5.104. A transportable nuclear power plant, as it is characterised in IAEA Safety Report Series No. 123 [1], consists of a nuclear power plant that is designed to be geographically relocated as a complete, or near complete, system. However, it is not designed to produce energy during transportation. Floating nuclear power plants, are a subset of transportable nuclear power plants, that are constructed in one or more locations before being moved by sea and/or inland waters to another location where they operate. Specifics of the floating nuclear power plants are defined by their aquatic environment. Transportable nuclear power plants may incorporate one or several of the specific innovations presented in the previous subsections. The safety impact of innovative technology should be assessed considering the recommendations provided in the relevant subsections.

5.105. Safety assessment should be performed for all stages of the lifetime of a transportable nuclear power plant in which nuclear fuel and/or radioactive material are involved, including commissioning and reaching first criticality; transport with fresh fuel, slightly irradiated fuel, spent fuel and/or radioactive waste; operation at designated site(s); and refuelling and maintenance (which could take place at a service centre outside the site where the reactor is operating). The safety assessment should consider all the relevant locations of the plant (e.g. the initial core could be loaded at a shipyard or service centre). In addition, when a transportable NPP is being moved with irradiated fuel onboard, the appropriate safety assessments should be made from the early design stage for the route and each location of the plant, including considerations on emergency preparedness and eventually necessary cross-border cooperation.

5.106. The safety assessment for a specific stage in the lifetime of a transportable nuclear power plant should consider the surrounding environments at that time. The mode of transport and corresponding transporting means may fall under the jurisdiction of the relevant responsible organizations, which should be involved in the safety assessment as appropriate.

5.107. External and internal hazards (including hazards associated with the means of transport and structural failures of these means) should be identified and properly considered during the safety assessment for each specific stage in the lifetime of a transportable nuclear power plant.

5.108. The applicability of simulation tools and their capability to address specific factors relevant to transportable nuclear power plants (e.g. pitching, vibration) should be evaluated and considered during the safety demonstration.

5.109. In deterministic safety analysis and probabilistic safety assessment for the transport stage of transportable nuclear power plants, the following should be taken into consideration:

(a) Specifics of the transport scenario (e.g. quantity and characteristics of radioactive material onboard, characteristics of route, existence and characteristics of shelter harbours, transit and transshipment facilities).
(b) The configuration of the plant in transport could differ from that during operation:
    (i) The plant could include parts that are transported together with the nuclear reactor, and other parts that are not;
    (ii) SSCs could be in different modes during transport and during operation.

5.110. The potential of inadvertent criticality when transporting a reactor with fuel onboard should be assessed in the safety assessment. This should include an assessment of inadvertent criticality for fault and accident conditions during transport.

5.111. Potential gaps related to the design process for a transportable nuclear power plant in the regulations, codes and standards of the supplier State, transit States and recipient State should be identified and addressed in the safety demonstration. For example, for floating nuclear power plants, there might be differences in welding standards that are relevant to nuclear components in the ship's hull.

5.112. The potential overlap between nuclear, marine, shipyard, and manufacturing regulations, codes and standards in the supplier, transit and recipient States should be identified and addressed in the safety demonstration.

5.113. For floating nuclear power plants the potential of inadvertent sinking introduces entirely new factors with an impact on nuclear safety, which should be addressed in the safety demonstration. If the sinking of a transportable nuclear power plant cannot be demonstrated to be practically eliminated, the safety assessment should consider the potential for recovery of the plant or its parts containing radiation sources. Alternatively, the safety assessment should demonstrate the practical elimination of radionuclide releases requiring long-term restrictions on the use of marine resources in the vicinity of the sunken transportable nuclear power plant. A similar approach should be followed for the air transport of a plant over the sea.

5.114. For designs of transportable nuclear power plants deployed on different kinds of sites (e.g. floating barge-mounted reactor, turbine paired with a land-based power supply building) similar SSCs might be affected by the same hazard in a different way. These unique impacts should be taken into account in the safety assessment.

6. CONSIDERATION OF INTERFACES WITH SECURITY AND SAFEGUARDS WHEN DEMONSTRATING THE SAFETY OF INNOVATIVE TECHNOLOGY

6.1. Requirement 8 of SSR-2/1(Rev. 1) [3] states:

> "**Safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a nuclear power plant shall be designed and implemented in an integrated manner so that they do not compromise one another.**"

Recommendations on how to meet this requirement when demonstrating the safety of innovative technology in a nuclear power plant design are provided in paras 6.2–6.3.[17]

6.2. Starting from the early design stage of innovative technology, the safety demonstration should consider the potential interfaces between safety and security measures, with the aim to demonstrate that they do not compromise one other. These interfaces include the following aspects:

(a) Consideration of site characteristics and building layouts, taking into account potential delays, due to security measures (see IAEA Nuclear Security Series No. 40-T, Handbook on the Design of Physical Protection Systems for Nuclear Material and Nuclear Facilities [32]), in safety related on-site and off-site actions and accessibility aspects (e.g. actions of fire brigades and other emergency response teams);
(b) Potential initiating events that might be triggered by security related equipment of innovative nuclear power plants (e.g. additional fire hazard coming from electrical equipment used for security, spatial interactions between security and safety equipment;
(c) Consideration of innovative instrumentation and control architecture for computer-based systems, taking into account the potential interaction between safety related functions and cybersecurity measures (see IAEA Nuclear Security Series No. 33-T, Computer Security of Instrumentation and Control Systems at Nuclear Facilities [33] and IAEA Nuclear Security Series No. 17-T (Rev.1), Computer Security Techniques for Nuclear Facilities[34]);
(d) Consideration of cybersecurity vulnerabilities in the reliability demonstration of digital instrumentation and control and other innovative technologies;
(e) Consideration of security measures when off-site arrangements are made to bring in additional human resources or equipment to respond to external hazards at innovative nuclear power plants (which might lead to potential delays, or challenges with timing aspects);
(f) Consideration of specific security measures for non-electrical applications and multi-modular designs with potential interactions and dependencies.

6.3. Starting from the early design stage of innovative technology, the safety demonstration should consider the potential interfaces between safety measures and safeguards arrangements with the Member States, with the aim to demonstrate that they do not compromise one other. These interfaces should include the following aspects:

[17] The recommendations provided are intended to cover the main aspects of the interfaces between safety, security and safeguards, but this is not an exhaustive list. The interfaces to be systematically considered in the safety assessment are expected to be specific to each particular innovative design.

(a) Potential initiating events that might be triggered by safeguards-related equipment installed at innovative nuclear power plants (e.g. additional fire hazard coming from electrical equipment used for safeguards, spatial interactions between safeguards and safety equipment);
(b) Consideration of potential human interactions during safeguards inspections or other safeguards verification activities and measures (e.g. unattended monitoring systems for innovative plants, such as microreactors);
(c) Potential safety implications of fuel handling in relation to safeguards verification activities (e.g. unattended monitoring measures for designs with online refuelling, such as high temperature gas reactors with tri-structural isotropic fuel);
(d) Potential new requirements for safeguards verification resulting from the use of innovative technology or fuels, which can change the probability of certain initiating events and as such impact the safety demonstration of innovative technology (e.g. the need to test new safeguards techniques during outages in innovative nuclear power plants);
(e) Potential safety implications in case of joint use of equipment at innovative nuclear power plants for both safety and safeguards purposes, where applicable (e.g. sensors, other instrumentation and control equipment).

6.4. The interfaces between safety, security and safeguards could become more sophisticated, and therefore complex, as technologies advance. Judgement should therefore be used in interpreting and applying the recommendations above, taking into account the ongoing evolution of technology and Member States commitments related to safety, security and safeguards.

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