TSR-DS Review Guidelines

REFERENCE DOCUMENT FOR THE IAEA TECHNICAL SAFETY REVIEW (TSR) – CONCEPTUAL DESIGN SAFETY (DS)

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FOREWORD

The IAEA's Technical Safety Review (TSR) review service supports the enhancement of nuclear safety for nuclear power plants and is based entirely on the IAEA safety standards. The service addresses the needs of Member States at most stages of development and implementation of a nuclear power programme, including the conceptual design, pre-licensing and licensing phases, nuclear power plant construction, operation and plant modifications including periodic safety reviews and lifetime extension. The IAEA Service Series 41, published in 2019, provides a generic consolidated basis for the conducting TSR services, which encompasses six technical subject areas: accident management, design safety, national safety requirements, generic reactor safety, periodic safety review and probabilistic safety assessment.

The TSR review service provides assistance to regulatory bodies, plant operating organizations, vendors and technical support organizations in their technical evaluations as well as in the development of national safety requirements. After a formal request to the IAEA, the TSR review service is prepared and provides a tailored, independent evaluation of the conceptual design documentation submitted to the IAEA. A major outcome is recommendations to enhance nuclear safety in areas that may need improvements to adhere to the IAEA safety standards. This publication is intended to make Member States aware of the possibility of a service through which they can have a better appreciation of the overall design safety of a facility.

The present publication covers the topic of a specific design safety review for a conceptual design that is particularly suited to novel advanced reactors such as small modular reactors (SMR), including non-water cooled reactors against the Safety Requirements on Safety of Nuclear Power Plants: Design (IAEA Safety Standards Series No. SSR-2/1 (Rev. 1)) and the Safety Requirements on Safety Assessment for Facilities and Activities, IAEA General Safety Requirements GSR Part 4 (Rev. 1), supported by the selected Safety Guides. This review can be focused to specific technical areas.

This publication is intended to be used mainly in the preparation and execution of a TSR- design safety (DS) review service by the IAEA and to provide information to potential recipients of the service regarding the topics that can be covered. It is also expected to be useful if Member States decide to conduct such reviews themselves either through regulatory authorities or as part self-assessment activities by plant management or vendors.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. BACKGROUND

The International Atomic Energy Agency (IAEA) provides Technical Safety Review (TSR) service as an element of its regular, extrabudgetary and technical assistance programmes to assess the safety of nuclear facilities. Reviews are conducted in response to requests from Member States, with whom the scope, objectives and technical disciplines to be covered by the review are agreed and as a part of an overall project for assessing the safety of the nuclear facility. Upon Member State request, the IAEA provides, through the Safety Assessment Section (SAS), TSR service to address the needs of Member States at different stages of development and deployment of nuclear power programmes.

Since 1988, the IAEA has been providing safety review services in six technical subject areas (design safety, generic reactor safety, national safety requirements, probabilistic safety assessment, accident management, and periodic safety review) to support Member States in the application of the IAEA safety standards. The service had been primarily conducted at nuclear power plants (NPPs). Owing to recent development of small modular reactors (SMR), the IAEA recognizes the need to extend the scope of the TSR design safety (DS) review service also to include novel advanced reactors. The assistance can be provided to regulatory bodies, plant operating organizations, vendors and technical support organizations (requesting parties) in their technical evaluations

Novel advanced reactors have often a novel system design or novel application of the nuclear energy, that might make design safety review against existing IAEA safety standards primarily developed for large light water reactors inappropriate. In addition, in order to undertake the review considering variety of design and its maturity there is a need to undertake the specific review at an early stage of design development.

The present publication covers the main issues of a design safety review at a level of conceptual design that is particularly suited to novel advanced reactors such as small modular reactors (SMR) including non-water cooled reactors against the General and Specific Safety Requirements provided in IAEA Safety Standards. Depending on the requests of a Member State, the scope can be tailored on a case-by-case basis with possible limitation to specific technical areas. These review guidelines are intended to help the requesting party to provide information on the technical contents of a submission needed for the review, as well as the reviewers, to perform the review in a structured way following applicable IAEA Safety Standards.

1.2. OBJECTIVES

The objective of TSR-DS review service is to review the conceptual design documentation developed for a specific novel advanced reactor, or its application against the Fundamental Safety Principles SF-1 [1] and Safety Requirements as follows:

– Safety of Nuclear Power Plants: Design, SSR-2/1 (Rev. 1) [2];

– Safety Assessment for Facilities and Activities, GSR Part 4 (Rev. 1) [3];

– Predisposal Management of Radioactive Waste, GSR Part 5 [4];

The TSR-DS review service will evaluate and document whether:

– General and specific safety requirements provided in IAEA Safety Standards listed above have been correctly addressed. Their applicability should either be confirmed or the novel and specific design features that might affect their applicability or make irrelevant their applicability be documented

– The conceptual design documentation comprehensively documents the topics that would be expected to evaluate the safety of the specific reactor technology, or specific application for which the advanced novel reactor is designed;

– There are generic areas that may need further substantiation by the vendor;

– All specific safety issues raised by the reviewed reactor technology type have been identified as adequately documented.

The outcome of the review will be an evaluation of the safety claims provided in conceptual design documentation that will show how specific safety requirements in relevant IAEA Safety Standards have been considered. The review results will be commensurate with the level of technical details provided in the conceptual design documentation. For example, if few details regarding the design are provided, then the review will likely not be able to make detailed observations about whether the safety standards would be met.

It should be noted that a TSR does not constitute any kind of design certification or licensing activity as this is not a function of the IAEA; rather, it is the responsibility of the Member States.

1.3. SCOPE

The TSR-DS review service to review conceptual design documentation developed for a specific novel advanced reactor or its application, is generally based on the IAEA safety standards in force at the time of the request. However, due to potential limitations and gaps in existing IAEA safety standards, which were primarily developed for large water-cooled reactors, the review may not be able to fully address some specific design aspects, especially for non-water-cooled novel advanced reactors. Other sources of information (e.g. specific design information developed by the vendor, Generation IV (GIF[1]) publications, scientific publications, papers, any general design information) could contribute to the evaluation performed by the review team.

In addition to general design safety aspects, the review may also focus on specific systems, for which the applicability of existing IAEA Safety Standards has been proven, and specific safety requirements requiring adaptation identified. The examples of the systems that could be covered include the reactor core, reactor coolant system and systems designed to mitigate accident conditions.

1.4. PREPARATION OF THE REVIEW SERVICE

The IAEA Service Series 41 [6], published in 2019, provides a basic structure and common approach to development of the terms of reference across the various technical subject areas covered by TSR review services and provides general guidance on how to prepare for and conduct a TSR review service. The IAEA Service Series 41 [6] is addressed to the Member State and/or the Requesting Party, as well as to the IAEA staff and external experts forming the technical team performing of the TSR review. The process required for a conceptual design review should be sufficiently flexible to accommodate the need to stage the review to allow the development of the documentation by the Requesting Party if necessary.

1.5. STRUCTURE

Section 2 of these guidelines provides a description of the documentation, which is expected to be submitted by the requesting party as an input for the IAEA review. Section 3 describes individual items to be reviewed based on the conceptual design documentation. Section 4 provides information on a content of documentation summarizing review results. Appendix I provides a standardized format to describe the design of structures, systems and components and plant equipment in conceptual design documentation.

1 Generation IV International Forum (GIF) was created as a co-operative international endeavor seeking to develop the research necessary to test the feasibility and performance of fourth generation nuclear systems, and to make them available for industrial deployment by 2030.

2. CONTENT AND STRUCTURE OF CONCEPTUAL DESING DOCUMENTATION

2.1. SCOPE AND STRUCTURE OF THE CONCEPTUAL DESIGN DOCUMENTATION

A conceptual design is a set of disciplines that contribute to the identification of the basic design layout and nominal operating conditions of industrial processes. In the field of nuclear engineering, conceptual design aims at evaluating the best design variables and operating conditions that maximize the performance and ensures high level of nuclear safety.

A conceptual design is developed in early phase of the design process, in which the broad outlines of process functions, identified safety functions and form of these, are articulated.

The conceptual design documentation that is anticipated for the TSR-DS service should:

– Outline the safety design basis (or claims) of the advanced novel reactor;

– Provide sufficient level of information to understand the approach that will be taken to support and demonstrate the future licensing basis;

– Give a description how the safety issues will be managed;

– Provide confidence that the safety implications associated with the design novelties are properly addressed;

– Provide confidence in the safety of the novel advanced reactor.

The conceptual design documentation should provide adequate information that the advanced novel reactor plant meets applicable safety requirements presented in the IAEA Safety Standards. The conceptual design conceptual design documentation that will be subject to TSR-DS review service should include a structured information assessing the adequacy of plant safety.

It is recommended that the conceptual design documentation follows the structure of the IAEA SSG-61 on Format and Content of Safety Analysis Report [5].

In the description of the structures, systems and components, a unified format of the information provided in Appendix I should be followed to the extent possible.

2.2. CHAPTER 1: INTRODUCTION AND GENERAL CONSIDERATIONS

The conceptual design documentation should provide an introduction that includes:

– Identification of the purpose (or application) of the novel advanced reactor, making reference to the use of specific reactor technology, or specific reactor application;

– Information about the process of preparation of the conceptual design documentation, the major contributors to the preparation, such as vendors, and the use of information that has been previously reviewed by the regulatory body, if applicable;

– The research programs in support of the qualification of the new design and the new operating conditions implemented;

– A description of the structure of the conceptual design documentation, the objectives and scope of each of its chapters and the connections between them;

– A description of the national and international guidance applied in the preparation of the conceptual design documentation.

2.3. CHAPTER 2: SITE CHARACTERISTICS

For a land based SMR, this chapter should provide a description of the reference site characteristics that will be used for the design of the plant, and for its safety assessment. Design should be adequately resistant against potential geological, seismological, volcanic, hydrological, meteorological and geotechnical hazards of the future sites. Robustness of the design should also consider potential human-induced hazards of the future sites.

For other than land based SMR, this chapter should provide a justification of the external hazards considered for design.

2.4. CHAPTER 3: SAFETY OBJECTIVES AND DESIGN RULES OF STRUCTURES, SYSTEMS AND COMPONENTS

This chapter should outline the general design concepts, requirements for different types of structures, systems and components present in the plant, and the approach adopted to meet the safety objectives. This chapter interfaces practically with all system chapters of the conceptual design documentation; it should provide inputs on general design, safety principles, loads and hazards as well as qualification requirements to be considered in the design. The overall safety philosophy and general approaches for ensuring safety should be presented. These approaches should be based on the requirements for the design of nuclear power plants established in SSR-2/1 (Rev. 1) [2] with adaptation for non-water cooled reactors. This chapter should present to the extent possible the lists of events which will be considered to design the plant and demonstrate its safety: postulated initiating events, hazards (hazards considered in the design basis and more severe hazards), design extension conditions, and practically eliminated situations.

2.5. CHAPTER 4: REACTOR

This chapter should provide relevant information on the reactor to show its capability to fulfil relevant safety functions throughout design life in all plant states. This chapter should include a description of the reactor, fuel design, core components, neutronic and thermohydraulic design (if applicable), and design of reactivity control and shutdown. This chapter should outline the approach of how the reactor design will meet the specific Requirements 43–46 of SSR 2/1 (Rev. 1) or, where relevant, will justify their non-applicability or some deviations. The specific fuel or core characteristics driving the core response to the postulated initiating events should be documented.

2.6. CHAPTER 5: REACTOR COOLANT SYSTEM AND ASSOCIATED SYSTEMS

This chapter should provide relevant information on the reactor coolant system and its associated systems, where possible in accordance with the scope and format described in Appendix I. The chapter should outline the approach of how the reactor coolant systems and associated systems will meet the specific Requirements 47–50 of SSR-2/1 (Rev. 1) [2] or, where relevant, will justify their non-applicability or some deviations. The specific fuel or core characteristics as well as those of the primary coolant driving the thermohydraulic consequences to the postulated initiating events should be documented.

2.7. CHAPTER 6: ENGINEERED SAFETY FEATURES

This chapter should provide relevant information on the engineered safety features and associated systems. The engineered safety features are those structures, systems and components that are necessary to fulfil safety functions in the case of design basis accidents, design extension conditions (including, if relevant, design extension conditions with significant fuel damage), and for some anticipated operational occurrences. This chapter should describe the capability of engineered safety features to mitigate the consequences of accidents and to bring the nuclear power plant to a controlled state, and finally to reach a safe state, in accordance with Requirements 51–58 and 65–67 of SSR-2/1 (Rev. 1) [2] or, where relevant, will justify their non-applicability or some deviations. For both active or passive engineered safety features, design provisions implemented to minimize their failures (including common cause failures between redundancies and systems) should be documented.

If relevant, systems shared by different modules should be indicated and the consequences for safety (propagation of consequences to other modules, adequate capabilities and autonomy of the ESFAS in the event of a multiple unit accident) should be documented.

2.8. CHAPTER 7: INSTRUMENTATION AND CONTROL

This chapter should provide an overall description of I&C system design basis and the overall I&C architecture in support of the concept of defence in depth applied in the design of the plant systems. SMRs may have multiple, separate modules and supporting systems but are also likely to have common systems between modules. In the case of multi-module unit with mutualised operation and / or mutualised plant systems, it might be worthwhile to describe the overall I&C architecture of the unit, and the overall I&C architecture of individual modules. This chapter should describe the approach of how the specific Requirements 59–67 of SSR-2/1 (Rev. 1) [2] are considered in the proposed I&C design.

2.9. CHAPTER 8: ELECTRICAL POWER SYSTEMS

This chapter should describe relevant information on the electrical power systems. These include electrical power systems both on-site and off-site power systems that supports the defence in depth concept of the plant. Consideration should be given to the passive nature of safe shutdown systems of novel advanced reactor, which determines the system architecture of the electrical power supply system if the safety systems required for response to a design basis event are powered from battery systems. This chapter should describe the approach how the Requirement 68 of SSR-2/1 (Rev. 1) [2] on withstanding the loss of off-site power is considered for the single module, or for the multi-module plant.

2.10. CHAPTER 9A: AUXILIARY SYSTEMS AND CIVIL STRUCTURES

This chapter should provide information about the auxiliary systems, such as systems that are essential for the safe shutdown of the plant or for the protection of the public. The description of auxiliary systems should be sufficient to show how the specific Requirements 69, 71–74, 76 and 80 of SSR-2/1 (Rev. 1) [2] will be met, or to justify their non-applicability or some deviations.

2.11. CHAPTER 9B: CIVIL ENGINEERING WORKS AND STRUCTURES

This chapter should describe how the general design requirements have been complied with in the design of specific structures for a single module or in a multi module plant. Different categories of civil structures should be considered, such as the reactor building, and other civil structures.

2.12. CHAPTER 10: POWER CONVERSION SYSTEMS

This chapter should provide information on the design of the power conversion systems for electricity production or other specific application, for which the advanced novel reactor is designed. This chapter should describe how the specific Requirement 77 of SSR-2/1 (Rev. 1) [2] will be met, or will justify its non-applicability or the deviations.

2.13. CHAPTER 11: MANAGEMENT OF RADIOACTIVE WASTE

This chapter should describe measures proposed for the safe management of radioactive waste potentially combined with dangerous toxic waste that will be generated throughout the lifetime of the plant. Relevant safety requirements include those regarding waste minimization (see para. 4.8 of SSR-2/1), treatment of radioactive waste (see Requirements 78 and 79 of SSR-2/1 (Rev. 1) [2] and programmes for the management of radioactive waste (see Requirement 21 of SSR-2/2 (Rev. 1) [2]). Further requirements are provided in IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [4].

2.14. CHAPTER 12: RADIATION PROTECTION

This chapter should provide information on the policy, strategy, methods and provisions for radiation protection with due consideration of novel design options, such as highly integrated design of the reactor module (for example integral designs with components inside the same vessel), long fuel cycle, and their implication on refueling and maintenance. An estimate of the expected occupational exposures during operational states, and the design measures taken to make occupational exposures and doses ALARA should also be described.

2.15. CHAPTER 13: CONDUCT OF OPERATIONS

This chapter should describe specific options for operation that reflect the novelty of the design e.g. operation of a multi-module plant; the impact to safety when a multi-module plant has modules in different phases of a life cycle (i.e. installation / commissioning, operational, or decommissioned), and remote control of the plant.

2.16. CHAPTER 14: CONSTRUCTION AND COMMISSIONING

This chapter should briefly describe the strategy for the construction (e.g. pre-fabrication at shops, transport) and the feasibility of the commissioning when modules are in different phases of a life cycle, (i.e. installation / commissioning, operational, or decommissioned) by considering Requirement 11 of SSR-2/1 (Rev. 1) [2] and paras 6.14 and 6.15 of SSR-2/2 (Rev. 1) [7].

2.17. CHAPTER 15: SAFETY ANALYSIS

This chapter should describe the approach to safety analyses to assess the safety of the plant in response to postulated initiating events and accident scenarios based on established acceptance criteria. These analyses include deterministic safety analyses of normal operation, anticipated operational occurrences, design basis accidents and design extension conditions, including considerations relating to 'practical elimination'. The level of details provided in this chapter should be commensurate to the project design stage, e.g. conceptual, basic design, detailed design up to the commissioning and operation stages. The scope of information provided should reflect the requirements on safety analysis relevant for nuclear power plant design, in particular Requirements 16, 17, 19, 20 and 42 of SSR-2/1 (Rev. 1) [2] and Requirements 14–21 of GSR Part 4 (Rev. 1) [3].

2.18. CHAPTER 16: OPERATIONAL LIMITS AND CONDITIONS FOR SAFE OPERATION

This chapter should outline the operational limits and conditions (OLCs). It should show the approach how OLCs will ensure compliance with Requirement 6 of SSR-2/1 (Rev. 1) [2], and that they include all the required components described in para. 5.44 of SSR-2/1 (Rev. 1) [2].

2.19. CHAPTER 17: MANAGEMENT FOR SAFETY

This chapter does not have to be documented at the conceptual design stage.

2.20. CHAPTER 18: HUMAN FACTORS ENGINEERING

This chapter should outline the human factors engineering programme and its application to the plant design, to meet Requirement 32 of SSR-2/1 (Rev. 1) [2]. This programme should apply to all operational states and accident conditions and to all plant locations where such interactions are anticipated.

This chapter should include a description of operational strategies and staffing models for the multi-module unit human system interface (HSI) design; the impact to HSI design when multi-module unit has modules in different phases of a life cycle (i.e. installation / commissioning, operational, or decommissioned). Multiple operational modules can be in different operational states (e.g. refueling, outage, AOO, emergency), and the impact of operational strategies on HSI design (e.g. when one (or more) module is operational and one (or more) is in a different life cycle or operational state).

The scalability of a shared control room should also be addressed. This includes a description how HSI technology, as well as visual and control space, can be designated, augmented, or adapted to optimize control of added modules while minimizing confusion between modules. Limits to scalability should also be addressed so that human performance is not negatively affected by physical or mental oversaturation. Such changes should align with human limitations (i.e. the number of reactor modules that can be monitored by one operator should be finite).

In line with scalability, the flexibility of a control room to support outages, refueling, AOOs and emergencies requires additional guidance. Addressing the affected or incident module while also not impacting other operating modules may be a challenge. There may be consideration for dedicated management or response facilities that can accommodate any module.

2.21. CHAPTER 19: EMERGENCY PREPAREDNESS

This chapter should briefly outline the strategy for the protection of the public and workers in case of nuclear emergency as well as a justification for limiting the emergency planning zone.

2.22. CHAPTER 20: ENVIRONMENTAL ASPECTS

This chapter should provide a brief description of the approach taken to assess the impact on the environment of the construction, operation (for operational states as well as for all accident conditions) and decommissioning of the plant.

3. REVIEW ITEMS OF CONCEPTUAL DESIGN DOCUMENTATION

3.1. GENERAL SAFETY DESIGN BASIS ASPECTS

The conceptual design documentation should describe the overall safety philosophy and general approaches for ensuring the safety of the plant. In addition to any national requirements and associated regulatory guidance, these approaches may be based on the fundamental safety principles established in IAEA SF-1 [1] and requirements for the design established in IAEA SSR–2/1 (Rev. 1) [2] adapted for novel advanced reactors (e.g. high temperature gas cooled reactors (HTGR), lead fast reactors (LFR), sodium fast reactors (SFR), molten salt reactors (MSRs), or specific application (e.g. electricity generation, hydrogen generation, combined production of heat and electricity, district heating) for which the advanced novel reactor is designed.

Although the level of detail available in the conceptual design stage may vary for different advanced novel reactors, the review should confirm comprehensiveness, clarity and adequacy of the following information:

(a) Safety objectives; (SSG–61, Clause 3.3.3)

The documentation should summarize the overall safety philosophy, safety objectives and high–level principles applied in the design, based on the relevant safety principles set out in SF–1 [1].

(b) Radiation protection and radiological acceptance criteria; (SSG–61, Clauses 3.3.6 and 3.3.7)

Conceptual design documentation should describe in general terms the design approach adopted to meet the fundamental safety objective (see Clause 2.1(a) of SF–1 [1]) ensuring that:

– In all plant states, radiation doses due to direct radiation and radioactive release are kept below authorized limits and as low as reasonably achievable (ALARA) (see also Clauses 2.6 and 2.7 of SSR–2/1 (Rev. 1) [2]).

– The design for safety of a novel advanced reactor applies the safety principle that practical measures must be taken to mitigate the consequences for human life and health and for the environment of nuclear or radiation accidents (Principle 8 of the Fundamental Safety Principles SF-1 [1]).

– Plant event sequences that could result in high radiation doses or in a large radioactive release are 'practically eliminated' and plant event sequences with a significant frequency of occurrence have no, or only minor, potential radiological consequences.

An essential objective is that the necessity for off-site protective actions to mitigate radiological consequences be limited or even eliminated in technical terms, although such measures might still be required by the responsible authorities.

The documentation should outline established radiological acceptance criteria for the public to meet safety objectives stated above.

Radiological aspects of the design should consider a multi module site, as well as specific application of novel reactors, particularly in case of location close to habitation centres. The emergency planning zone should be proposed and justified.

(c) <u>General design basis for plant states considered in the design</u>; (SSG–61, Clauses 3.3.8 through 3.3.10)

The documentation should describe the general approach to defining the design basis, with account taken of operational regimes (power operation, shutdown states, refuelling), plant states (operational states, anticipated operational occurrences, accident conditions) and impacts from both external and internal hazards.

The description should include:

– The basis for the categorization of plant states (typically with expected frequencies, or explanation of other associated characteristics).

– The plant states and hazard loads under which a given structure, system or component will need to fulfil its safety functions.

– Postulated initiating events (PIEs), whether of internal origin or caused by internal and external hazards, if relevant, as considered in the design, including credible combination of failures and events, without neglecting low frequency high consequence accidents.

– A justification of events not considered as a PIEs (e.g. avoidance of some usual PIEs due to simplification of the design, unlikely combination of multiple failures).

– A list of conditions practically eliminated with a justification, how the claim of practical elimination will be supported, and how the conceptual design justifies these claims.

– The independence of provisions for different levels of defence applied in the design.

– Autonomy of the module, or modules in a multi-module plant in terms of need for electrical power sources or off-site support (not only electrical but also for other aspects, cooling).

(d) <u>Design measures for prevention and mitigation of accident</u> conditions; (SSG–61, Clause 3.3.31)

The documentation should describe measures taken to prevent accidents, to mitigate their consequences, and to ensure that the likelihood of an accident having harmful consequences is extremely low (see Clauses 3.30 and 3.31 of SF–1 [1]), including measures taken to ensure that early or large radioactive releases are practically eliminated.

(e) <u>Defence in depth</u>; (SSG–61, Clauses 3.3.12 through 3.3.15)

The documentation should describe the approach adopted to incorporate the defence in depth concept into the design, in accordance with Clauses 2.12–2.18 of SSR–2/1 (Rev.1) [2].

Description should address measures taken for adequate individual robustness and mutual independence of levels, including a description of:

– Protection of fission product barriers to the release of radioactive material and systems to ensure the protection at each level of defence in depth.

– Any envisaged operator actions necessary to assist in the fulfilment of the safety functions essential for defence in depth.

– Provisions to facilitate off–site support.

(f) <u>Safety functions</u>; (SSG–61, Clauses 3.3.4 and 3.3.5)

The documentation should identify the specific safety functions and the systems to perform the functions for the given design and its application by means of the inherent features passive or active systems and operator actions, in accordance with Requirement 4 of SSR–2/1 (Rev. 1) [2].

(g) <u>Application of general design requirements and technical acceptance criteria</u>; (SSG–61, Clauses 3.3.16 through 3.3.20)

This section should provide a high-level description of the following design principles considered in the conceptual design:

– The deterministic design principles;

– The scope of implementation of the single failure criterion and how compliance with this criterion is achieved in the design;

– Any other relevant approaches aimed at ensuring safety, such as:

  o Simplification of the design;

  o Passive safety features;

  o Gradually responding plant systems;

  o Fault tolerant plant and systems.

(h) <u>Practical elimination of the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release</u>; (SSG–61, Clauses 3.3.21 and 3.3.22)

The documentation should describe the design and operational provisions to ensure that the possibility of conditions leading to an early radioactive release or to a large radioactive release arising will be ‘practically eliminated’ (see Clause 5.31 of SSR–2/1 (Rev. 1) [2]).

(i) <u>Safety margins and avoidance of cliff edge effects</u>; (SSG–61, Clauses 3.3.23 through 3.3.25)

The documentation should provide a summary of the approach taken to ensure adequate margins to prevent cliff edge effects relating to damage to barriers against releases of radioactive material; see para. 5.73 of SSR–2/1 (Rev. 1) [2].

The documentation should also outline the approach used for demonstration of safety margins for internal or external hazards. For natural hazards, it should be shown how to protect items ultimately necessary to prevent early or large radioactive releases against hazards with severities that exceed those considered in the design basis (see Clause 5.21A of SSR–2/1 (Rev. 1) [2]).

(j) Design approaches for fuel storage and fuel handling, any radioactive material storage. (SSG–61, Clause 3.3.26)

The documentation should outline the design approaches adopted to demonstrate the performance of the safety functions in the fuel storage areas and, if relevant, during fuel handling, in particular in case of innovative refuelling options. These design approaches could imply specificities in implementation of defence in depth, different specification of derived safety functions, different monitoring means and substantial differences in the time evolution of accidents.

(k) Considerations of interactions between multiple modules; (SSG–61, Clause 3.3.27 through 3.3.29)

For multi-module plant, the documentation should describe components and systems shared between the multi modules as well as any interconnections between the units/modules. This includes:

– Any interconnections between modules in a multi-module plant to enhance safety;

– Positive and negative effects of such interconnections

– Any interconnections or services provided by shared systems that will be severed when one or more modules are shut down for an extended period and kept in a safe storage state (e.g. in preparation for future decommissioning).

– The potential impact on other operating modules in different operating modes and accident conditions of severing the interconnections and shared services.

(l) Design provisions for ageing management; (SSG–61, Clause 3.3.30 through 3.3.32)

The documentation should outline a design life of items important to safety, with description how anticipated ageing degradation mechanisms and in-service wear were considered in the conceptual design. This includes:

– How adequate margins will be maintained, with account of anticipated ageing degradation mechanisms, including those caused by testing and maintenance, by plant states during a postulated initiating event and by plant states following a postulated initiating event.

– How ageing effects caused by in-service ageing and environmental factors (e.g. corrosion, vibration, irradiation, humidity, temperature, pressure) will be managed (e.g. in-service inspection, maintenance programmes, equipment qualification programmes) over the expected life of items important to safety.

(m) Provisions for waste management and decommissioning; SSG-61, Clause 3.11.1. through 3.11.18. (waste management); Clause 3.21.1. through 3.21.10. (decommissioning))

In case of production of a unique kind of waste, the documentation should outline the strategy for the management of all radioactive waste including packaging, safe transport and storage (assuming that there is no need of special strategy for standard radioactive waste).

Furthermore, the description should outline the decommissioning options, in particular if individual modules will be decommissioned in different times (e.g. in terms of the optimization of protection and safety, the protection of the environment, and minimizing the generation of waste. Description should address treatment of materials potentially causing an excessive contamination, an increase of collective radiation dose to operating personnel, and be a challenge for dismantling and waste disposal.

(n) Human factors engineering (SSG-61, Clause 3.18.1. through 3.18.39)

The documentation should describe how the human factors including human-machine interface have been considered in the design of the advanced novel reactor (Requirement 32 of SSR-2/1 (Rev. 1)) [2] to support operating personnel in the fulfilment of their responsibilities and in the performance of their tasks, and limit the likelihood and the effects of operating errors on safety in all operational states and accident conditions and to all plant locations where such interactions are anticipated.

3.1.1. **Review items**

The review of the conceptual design documentation should verify whether the information describing general safety design basis in items (a) to (n) above of the given novel advanced reactor is sufficiently comprehensive, clear and adequate to justify the safety of the specific reactor technology at a conceptual design level, as well as the safety of the specific application for which the advanced novel reactor is designed.

3.1.2. **Possible references for review**

– IAEA SF–1, all chapters;

– IAEA SSR–2/1 (Rev. 1), Requirements 4, 21, 23 through 26 and 33; Chapters 2 and 5;

– IAEA GSR Part 5 (Predisposal Management of Radioactive Waste).

3.2. SAFETY ANALYSIS

Conceptual design of novel advanced reactors is developed in the early phase of the design process. Nevertheless, the conceptual design should formulate the objectives to protect the public and the environment from radiation harm and explain how to achieve the objective. It is understood that safety analysis at the conceptual design stage is limited and focused on main aspects such as whether the acceptance criteria for integrity of barriers and radiological consequences doses are provided and met, and the main technical features envisaged for performance of the safety functions in different plant states are adequate.

The information provided in the documentation should be comprehensive enough to justify the intended design basis for items important to safety, and to ensure that the overall conceptual design is capable of meeting the acceptance criteria associated with each plant state.

For a plant designed with multiple modules being interconnected and having shared systems, the safety analysis should consider the possibility for a PIE to affect several modules and acknowledge that the operation of a safety system would be necessary to mitigate the consequences at several modules simultaneously.

3.2.1. **Review items**

The documentation should provide an overview of safety analyses made to support the conceptual design, covering both deterministic and probabilistic analyses. This should include a description of the scope of the safety analysis and the approach adopted (e.g. conservative or realistic, as appropriate) for each state, from normal operation up to design extension conditions. At the same time, it is understood, that the analysis (in particular probabilistic analysis) can only be performed in the scope corresponding to the maturity of the design.

The documentation should also explain how any previously identified generic issues and relevant operating experience have been used to enhance the quality of the safety analysis, as indicated in, for example, paras 4.7, 4.27 and 4.52 and as required in Requirement 19 of GSR Part 4 (Rev. 1) [3].

Any applicable reference documents on the methodology used in the safety analysis may be introduced in this section.

The list of review items follows the scope and structure of safety analysis as provided in SSG-61 [5]. Not all these items may be available in the full scope at the conceptual design phase. The review of safety analysis should therefore determine whether the safety analysis at this stage of the design is comprehensive enough to support sufficient confidence in licensing of the novel advanced reactor.

3.2.1.1. *Identification, categorization and grouping of postulated initiating events and accident scenarios for modules/facilities*

The review should evaluate

- The approach used to identify postulated initiating events and accident scenarios for the analyses;
- At the conceptual design the basis for the selection and categorization of postulated initiating events including a list of the most significant scenarios considered in safety analyses to cover anticipated operational occurrences, design basis accidents and design extension conditions;
- Attention paid to specific for the given design events and accident scenarios of all types (both internal and external), considering specifics of the plant configuration, systems and physical phenomena, based on information available in relevant background documents
- Interactions between the electric grid and the plant, and interactions between different modules in a multi-module plant and other facilities on the same site (including coupled facilities) considered as sources of initiating events;
- The list of conditions to be practically eliminated, for which analytical demonstration is applicable, with the basis for the demonstration for each situation as required by para. 5.31 of SSR–2/1 (Rev. 1) [2][1].

3.2.2. **References for review:**

- GSR Part 4 (Rev. 1) Requirement 4, 6, 7, 14;
- SSR–2/1 (Rev. 1) Requirement 4, 6, 7, 14, 16, 17, 20, 33, 41;
- SSG–2 (Section 3).

3.2.3. **Safety objectives and acceptance criteria**

The review should evaluate:

- Adequacy of the radiological acceptance criteria relating to radiological consequences and the technical acceptance criteria relating to the integrity of barriers for different categories of plant states and types of analysis;
- Adequacy of probabilistic values such as fuel damage frequency or large releases frequency selected as acceptance criteria or safety objectives.

3.2.4. **References for review:**

- GSR Part 4 (Rev. 1) Requirement 16;
- SSR–2/1 (Rev. 1) Requirement 42;
- SSG–2 (Rev.1) Section 4.

3.2.5. **Human actions**

The documentation should describe the approach adopted to take into account human actions and the methods selected to model these actions in the analyses; see Requirement 11 of GSR Part 4 (Rev. 1) [3].

The review should evaluate whether:

– Credited human actions could be accomplished with the authorized minimum shift, in particular in scenarios involving external hazards affecting multiple modules (if relevant);

– Credited human actions can be executed with due consideration regarding the provision of adequate instrumentation to monitor the status of the plant, and adequate controls for the manual operation of equipment;

– The coping time for the implementation of a human action is adequate for its success;

– Credited human actions associated with operation of a multiple modules in a multi-module plant can be reliably executed.

3.2.6. **References for review:**

– GSR Part 4 (Rev. 1) Requirement 11;

– SSR–2/1 (Rev. 1) Requirement 16, 32.

3.2.7. **Deterministic safety analyses**

3.2.7.1. *General description of the approach*

General description of the approach should address the safety analyses performed to assess the safety of the plant in response to selected postulated initiating events and accident scenarios with reference to established acceptance criteria. The analyses include deterministic safety analyses of normal operation, anticipated operational occurrences, design basis accidents and design extension conditions, including in the necessary scope the deterministic demonstration relating to ‘practical elimination’ of relevant conditions potentially leading to early or large radioactive releases. Analyses to justify specific operator actions as necessary can also be included in this chapter.

The scope of information should reflect the requirements on safety analysis relevant for the conceptual stage of the design, taking into account Requirements 16, 17, 19, 20 and 42 of SSR–2/1 (Rev. 1) [2] and Requirements 14–21 of GSR Part 4 (Rev. 1) [3]. Recommendations and guidance on deterministic safety analysis are provided in IAEA Safety Standards Series No. SSG–2 (Rev. 1)[9].

The review should evaluate whether:

– Sufficient margins can be demonstrated using a deterministic safety analysis in which acceptable approaches (i.e. conservative or best estimate; see SSG–2 (Rev. 1) [9])

– Justification of the selection of systems credited in the analysis (based on their safety classification) and choice of their operating states in deterministic safety analysis with due consideration of single failure, multiple failures and consequential failures of plant systems was provided.

– Any additional assumptions (such as on the choice of operating states of systems and/or support systems, conservative time delays and operator actions) for the development of the plant models were described.

3.2.7.2. *References for review:*

- GSR Part 4 (Rev. 1) Requirement 14, 15, 18;
- SSR–2/1 (Rev. 1) Requirement 16, 17, 19, 20, 25, 42;
- SSG–2 (Rev.1) (Section 9).

3.2.7.3. *Analysis of normal operation*

The review should evaluate whether normal operation can be carried out in a safe manner in all operating modes of the novel advanced reactor, for example:

- Parameters in normal operation are maintained within the boundaries specified by the relevant operational limits and conditions, and that a reactor trip or initiation of the control and limitation systems and safety systems would be prevented;
- All possible modes of normal operation are covered in this description, with particular attention to transient operational regimes such as changes in reactor power, reactor shutdown from power operation, reactor cooling down, handling of irradiated fuel, and off–loading and transfer of irradiated fuel from the reactor module, as applicable;
- Measures are taken to ensure that radiation doses to members of the public due to planned discharges and/or releases of radioactive material from the reactor are below the dose limits and kept as low as reasonably achievable, as required by para. 2.6 of SSR–2/1 (Rev. 1) [2][1].

3.2.7.4. *References for review:*

- SSG–2 (Rev.1) Section 3.

3.2.7.5. *Analysis of anticipated operational occurrences and design basis accidents*

The documentation should describe the assumptions to be used in the analyses of selected postulated initiating events belonging to the categories of anticipated operational occurrences and design basis accidents. Sufficient information should be provided to make a judgment on the adequacy of the design of systems and components, and of the envisaged operator actions.

For each group of postulated initiating events it may be sufficient to present preliminary analyses for a limited number of scenarios that represent a bounding response for a group of events.

The parameters important to the outcome of the safety analysis should be presented, including, as a minimum, all parameters important for the assessment of compliance with the selected acceptance criteria.

The functional response of systems to the postulated initiating events, including the operating conditions in which a system is actuated, and the associated time delays and capacity after actuation, should be presented.

The information should be provided confirming that the acceptance criteria for both anticipated operational occurrences and design basis accidents will be met, and the safe stable conditions will be reached.

The review should evaluate whether for each individual group of postulated initiating events analysed sufficient information is provided including:

- A description of the postulated initiating event, the category to which it belongs and the applicable acceptance criteria to be met;
- A description of the computer codes and models used for the preliminary analysis;
- The specific values of important parameters and initial conditions used in the analysis, with an indication of the reference (nominal) values and the uncertainties associated with the parameters;

- Availability of systems (control and limitation systems, active and passive safety systems) and operator actions credited in the analysis;
- Information on any additional failures in reactor systems and components postulated to occur in the specific accident scenario and any other conservative assumptions;
- A description of the progression of the event, highlighting the key phenomena, actuation of systems, reaching a safe long term stable state;
- Assessment of compliance with relevant acceptance criteria, description of the status of fission product barriers and the performance the safety functions;
- Preliminary results of assessment of radiological consequences (in terms of radioactive releases);

Sensitivity studies as appropriate.

3.2.7.6. *References for review:*

- SSG–2 (Rev. 1) paras 7.17 – 7.44, 8.1 – 8.13.

3.2.7.7. *Analysis of design extension conditions challenging the integrity of fission product barriers*

The documentation should present the assumptions used and the preliminary results obtained from the analyses of design extension conditions that may challenge the integrity of fission product barriers.

The review should evaluate whether:

- The preliminary analyses presented in this section performed in a best-estimate way assesses that damage of the fission product barriers is limited to minimize the radioactive releases and that a safe state can be reached and maintained and that there are adequate margins to avoid cliff edge effects;
- The required protective actions following severe accident scenarios, if relevant, are limited in the area and time.

3.2.7.8. *References for review:*

- SSG–2 (Rev. 1) paras 7.17 – 7.44, 8.1 – 8.13.
- SSG–2 (Rev. 1) paras 7.56 – 7.72, 8.1 – 8.13.

3.2.7.9. *Analysis of radioactive releases from a subsystem or component*

The documentation should present the safety analysis performed for postulated initiating events caused by the release of radioactive material from a subsystem or component, typically from systems for treatment or storage of radioactive waste. Postulated initiating events range from minor leakage from a radioactive waste system up to the overheating of, or damage to, spent fuel in transit or storage, or a large break in a gaseous or liquid waste treatment system. Special consideration is needed of potential releases from the retention systems used in molten salt reactors with continuous removal of the fission products during normal operation. The scope and content of the information provided should be similar as for other design basis accidents.

3.2.7.10. *References for review:*

- SSG–2 (Rev. 1) para 7.17 – 7.44, 8.1 – 8.13.

3.3. CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

3.3.1. **Review items**

The documentation should outline the approach taken for the categorization of safety functions, for the identification of the main items (structures, systems and components) necessary to fulfil these safety functions and for the safety classification of these items: see Requirement 22 of SSR–2/1 (Rev.1) [2] and SSG–30 [10] to the extent applicable. Preferably, the conceptual design documentation should include the list of main structures, systems and components important to safety, together with the intended safety functions, safety classification, seismic categorization and the associated safety requirements. (SSG–61, Clause 3.3.35).

The review should evaluate:

– Adequacy of the methodology and criteria applied for safety classification;

– The categorization of the safety functions;

– The safety classification of the main structures, systems and components;

– The associated engineering and design rules proposed for different safety classes of structures, systems and components (e.g. quality requirements, power supply requirements, environmental qualification, seismic categorization);

3.3.2. **References for review**

– IAEA SSR–2/1 Rev. 1, Requirement 22;

– IAEA SSG–30.

3.4. PROTECTION AGAINST EXTERNAL HAZARDS

Because external hazards are site specific, the conceptual design documentation may not include details on applicable external hazards. Nevertheless, the designs of novel advanced reactors should sufficiently be robust to be deployed to sites with different site characteristics. The reference design should therefore consider sufficient margins to the external hazards in general.

The documentation should provide an overview of external hazards specifically considered in the conceptual design, including:

– Intensity of individual external hazards (loads resulting from external hazards) considered in the design;

– The approach used to ensure adequate margin over the reference design parameters characterizing the natural hazards (SSR–2/1, Clauses 5.21 and 5.21A);

– In case of a multi-module plant or a plant with coupled production facilities (e.g. for production of hydrogen) on the same site consideration of specific external hazards resulting from other modules or coupled facilities;

– Approach to consider combination of external hazards (independent or causational) or combination of external and internal hazards (SSG–61, Clauses 3.3.37 and 3.3.38);

– For multi–unit/module site, consideration of the potential for specific hazards to affect several modules at the multi-module site at the same time (SSR–2/1. Clause 5.15–B);

- Consideration of specific hazards resulting from unusual locations of the plant, such as floating plants, underground plants, extreme geological or meteorological conditions, or increased risk of human actions.

Both hazards of natural origin as well as man induced external hazards should be addressed, including at least

- Seismic hazard; (SSG–61, Clause 3.3.40);
- Extreme weather conditions; (SSG–61, Clauses 3.3.41 and 3.3.42);
- Extreme hydrological conditions; (SSG–61, Clauses 3.3.43 and 3.3.44);
- Aircraft crash; (SSG–61, Clauses 3.3.45);
- Missiles; (SSG–61, Clause 3.3.46);
- External fires, explosion and toxic gases; (SSG–61, Clauses 3.3.47);
- Other external hazards (such as electromagnetic interference, biological phenomena, collision of floating bodies or geotechnical hazards, other site and design specific hazards); (SSG–61, Clauses 3.3.48).

Information about codes and standards applicable for the design, the methodologies with basic assumptions, and any other design criteria regarding loads and load combinations considered in the conceptual design should be provided, as applicable. Attention should be paid at the level of the conceptual design to identification of certain structures, system and components (e.g. containment) to be sufficiently resistant against external hazards. The review should evaluate the adequacy of the approach and application of the design principles for protection against the other external hazards.

3.4.1. **References for review**

- IAEA SSR–2/1 Rev. 1, Requirements 14, 17 and Clause 5.32.

3.5. PROTECTION AGAINST INTERNAL HAZARDS

The documentation should provide general information on internal hazards considered in the design. This section may include (as available) a description of the quantitative design parameters of individual hazards; the relevant design criteria, codes and standards; the methods of assessment; and the general design measures provided to ensure that the essential structures, systems and components important to safety are adequately protected against the effects of all the hazards considered in the conceptual design, in order to ensure safe shutdown and availability for the mitigation of the accident conditions.

The general assumption is that internal hazards should not lead to an accident condition to the extent practicable. The consequences of internal hazards should be justified and documented.

The review should evaluate whether the design approach provides adequate protection of items important to safety and, as applicable, of personnel performing actions to protect against internal hazards. This includes:

(a) Identification of internal hazards and credible combinations of hazards, and characterization of the effects of the hazard(s) (SSG–61, Clause 3.3.49);

(b) Design provisions for preventing occurrence and propagation of internal hazards;

(c) Design of means for mitigating or limiting the adverse effects of internal hazards on items important to safety;

(d) Plausible combinations of loads derived from internal hazards with other types of loads, including loads from other internal hazards, loads from randomly occurring individual events or loads from external hazards, have to be considered. Flooding due to an internal missile is an example of such a combination;

(e) Demonstration of the effectiveness of the remaining capability of items important to safety to accomplish the safety function.

The list of applicable internal hazards considered in the design typically includes consideration of the following hazards, as required by each particular reactor design, for example:

– Internal fire, explosion and toxic gases; (SSG–61, Clause 3.3.51);

– Internal flooding; (SSG–61, Clause 3.3.52);

– Internal missiles; (SSG–61, Clause 3.3.53);

– Pipe whip following pipe ruptures and dynamic effects associated with high energy pipe ruptures; (SSG–61, Clauses 3.3.54);

– Internal missiles such as those originating from rotating machinery;

– Failures of pressurized components, supports or any other structures;

– Release of hazardous chemical species;

– Other internal hazards, such as heavy load drops, electromagnetic interference; (SSG–61, Clauses 3.3.55).

In addition, specific internal hazards applicable for different technologies should be addressed.

3.5.1. **References for review**

– IAEA SSR–2/1 Rev. 1, Requirements 14, 17, 74 and Clause 5.32.

3.6. GENERAL DESIGN ASPECTS FOR CIVIL ENGINEERING WORKS OF SAFETY CLASSIFIED BUILDINGS AND CIVIL ENGINEERING STRUCTURES

The documentation should outline the design principles applied for buildings and civil structures, including their foundations. These general aspects should discuss margins that have been considered the design of buildings and structures that are relevant to safety. (SSG–61, Clause 3.3.56)

3.6.1. **Review items**

The review should evaluate the following design principles applied for buildings and civil structures have been addressed[2] in conceptual design documentation:

(a) List of safety–relevant buildings and civil structures, including seismic categorisation. The list may include the following structures[3]:

– The reactor building;

2 The level of details may vary depending on the conceptual design phase.

3 These are examples for water cooled SMRs; other technologies may have different list of safety classified buildings and civil structures.

- The high strength steel containment immersed in the cooling pool: [4] (if provided);
- The fuel building;
- The nuclear auxiliary building;
- The electrical building;
- The control room area;
- The safety system auxiliary building and the waste treatment building.

(b) General design bases for safety–relevant civil engineering works and structures, including the following items: (SSG–61, Clauses 3.3.57 and 3.3.60)

- Applicable codes, standards and other specifications;
- Loads and load combinations;
- Assumptions made in the design (i.e. loadings, site characteristic, modelling assumptions);
- Structural acceptance criteria;
- High strength steel containment[5] immersed in the cooling pool water, if this option is retained, retaining capability and leak detection;
- The way(s) in which margins have been introduced the design of structures;
- Materials, quality control, and special or novel construction techniques and approach to future decommissioning;
- Testing and in–service inspection requirements;
- Treatment of design extension conditions and cliff edge effects, as appropriate;
- General site layout and interactions between the civil engineering structures.

(c) Any general information specific for the design of foundations and buried structures, such as overall subsoil conditions in the site and geotechnical profile considered for design; (SSG–61, Clause 3.3.58)

(d) Specific requirements for the design of the containment design, as applicable (including both design basis and design extension conditions), such as leak tightness, mechanical strength, resistance to hazards, and resistance to accident conditions. (SSG–61, Clause 3.3.59)

(e) Overall description of the containment building, including its internal structures, as applicable. Detailed descriptions of these structures, including the general layout, sections and principal features of major internal structures; (SSG–61, Clause 3.3.59)

(f) Overall description of the other safety–classified buildings and civil structures, including considerations of specific design features for novel reactors (e.g. concrete behavior at high temperature). (SSG–61, Clause 3.3.61)

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4 Some reactors are enclosed in a high strength steel containment vessel immersed in the cooling pool that acts as a heat exchanger to provide the means to transfer reactor heat to the reactor pool water to limit containment pressure.

5 Applies to some reactor designs enclosed in a high strength steel containment vessel immersed in the cooling pool.

3.6.2. References for review

– IAEA SSR–2/1 Rev. 1, Requirements 18 and 54 through 58.

3.7. GENERAL DESIGN ASPECTS FOR METALLIC STRUCTURES AND MECHANICAL SYSTEMS AND COMPONENTS

The documentation should outline general design aspects applied for mechanical systems and components. This may include design principles, codes and standards to be used in the design of mechanical components, information on physical separation, and structural analysis (information concerning the design loads and load combinations and outlining the appropriate design and service limits for components and supports). (SSG–61, Clause 3.3.62)

3.7.1. Review items

The review should evaluate adequacy of description of the following general design aspects applied for mechanical systems and components:

(a) A preliminary list of systems and their functions.

(b) Design basis of the systems.

(c) Adequacy of engineering practices.

(d) Issues associated with implementation of novel design provisions.

(e) Anticipated loads and load combinations considered in the design, and the appropriate design and service limits for safety–classified components and supports; (SSG–61, Clause 3.3.62-3.3.63).

(f) A preliminary list of operational conditions considered in the fatigue and fracture analyses of all components of the reactor coolant system and the core support components, other supporting components and reactor internals and other systems that fulfil a safety function; (SSG–61, Clause 3.3.64).

(g) Overall description of the approach and engineering design rules, codes and standards for the design and manufacturing of mechanical components, including the vessels, piping system, and associated supports.

(h) Provisions for ensuring the structural integrity of the main safety components with their component supports and core support structures, and applicable codes and standards (SSR–2/1 (Rev.1), Requirement 18 [2]).

3.7.2. References for review

– IAEA SSR–2/1 Rev. 1, Requirements 18.

3.8. GENERAL DESIGN ASPECTS FOR INSTRUMENTATION AND CONTROL SYSTEMS AND COMPONENTS

The documentation should outline design principles applied for instrumentation and control systems and components. (SSG–61, Clause 3.3.67)

3.8.1. Review items

The review should evaluate information provided in the conceptual design documentation as follows:

(a) Overall information on architecture and the principles used in the design of instrumentation and control systems and components in order to ensure defence in depth, for example: (SSG–61, Clause 3.3.67)

- Design bases;
- Application of the single failure criterion;
- Independence of provisions for the different defence in depth levels;
- Consideration of common cause failure and diversity;
- Equipment qualification;
- Maintainability, testing and testability.

(b) An information of anticipated functional and non–functional requirements for the overall instrumentation and control and for each individual system, indicating how this information will be used to categorize the functions and to assign them to systems of the appropriate safety class in accordance with SSG–30 [109]. (SSG–61, Clause 3.3.68)

3.8.2. **References for review**

- IAEA SSR–2/1 Rev. 1, Requirement 18.

3.9. GENERAL DESIGN ASPECTS FOR ELECTRICAL SYSTEMS AND COMPONENTS

The documentation should outline the design principles applied for electrical power systems and components. (SSG–61, Clause 3.3.69)

3.9.1. **Review items**

The review should evaluate information provided in the conceptual design documentation as follows:

(a) Information on the overall design principles and criteria used in the design of electrical systems and components, for example: (SSG–61, Clause 3.3.69)

- Design bases;
- Application of the single failure criterion;
- Independence of provisions for the different defence-in-depth levels;
- Consideration of common cause failure and diversity;
- Equipment qualification;
- Maintainability, testing and testability.

(b) Information on the functional adequacy of the offsite and onsite power systems and onsite electric power systems important to safety, as applicable to a novel advanced reactor design.

(c) A description how the passive nature of safety systems determines the system architecture of the electrical power supply system as the safety systems required for response to a design basis event are powered from battery systems (e.g. autonomy of a module to withstand loss of AC power supply).

(d) Anticipated functional and non–functional requirements for the overall electrical power supply and for each individual electrical system. The description of functional and non–functional requirements indicating how this information will be used to categorize the functions and to assign them to systems of the appropriate safety class in accordance with SSG–30 [10].

(SSG–61, Clause 3.3.68)

3.9.2. **References for review:**

– IAEA SSR–2/1 Rev. 1, Requirement 18.

3.10. GENERAL DESIGN ASPECTS FOR OTHER MAIN SUPPORTING SYSTEMS

The conceptual design documentation may provide, in addition to the general design aspects of the systems associated to the reactor and the nuclear supporting systems identified in sections above, a general description of other main supporting systems. This information may follow the structure provided in Appendix I.

3.11. EQUIPMENT QUALIFICATION

The documentation should outline the scope of the qualification programme to show that items important to safety, are capable of meeting design requirements and remaining fit for purpose in the range of anticipated environmental challenges under which they are required to perform. (SSG–61, Clause 3.3.71)

3.11.1. **Review items**

The review should evaluate information items provided in the conceptual design documentation as follows:

(a) Scope of the equipment qualification to meet Requirement 30 of SSR 2/1 (Rev.1) [2]; (SSG–61, Clauses 3.3.71, 3.3.73 and 3.3.75)

(b) A preliminary list of items important to safety requiring qualification.

(c) Description of the way in which the equipment qualification programme takes account anticipated and potentially disruptive environmental conditions, under which SSCs are performing, including events associated with internal and external hazards. (SSG–61, Clause 3.3.72)

3.11.2. **References for review**

– IAEA SSR–2/1 Rev. 1, Requirement 30.

3.12. PROVISIONS FOR IN–SERVICE MONITORING, TESTS, MAINTENANCE AND INSPECTIONS

3.12.1. **Review items**

The documentation should outline the feasibility of in-service inspection on the main reactor coolant components and on other items important to safety

The documentation addressing in-service inspection, periodic testing and maintenance activities should be reviewed with due account for little access and space for SMRs designed with a high level of equipment integration.

The review should check whether the intent of Requirement 31 of SSR–2/2 (Rev.1) [7] has been met, for example:

(a) A preliminary list of components subject to In-service inspection (ISI) programme, periodicity and feasibility of inspection activities.

(b) Overview of the regulations, codes and standards applicable to the areas of in–service monitoring, tests, maintenance and inspections. (SSG–61, Clause 3.3.77)

3.12.2. **References for review**

- IAEA SSR–2/2 Rev. 1, Requirement 31.

3.13. COMPLIANCE WITH NATIONAL AND INTERNATIONAL STANDARDS

The documentation should outline the design principles and criteria, established in applicable codes and standards adopted in the plant design. (SSG–61, Clause 3.3.77)

The review should evaluate whether the intent of Requirement 9 of SSR–2/1 (Rev.1) [2][1] has been met, for example:

- Items important to safety will be designed in accordance with the relevant national and international codes and standards;

- Items important to safety have preferably been of a design that has previously been proven in equivalent applications, and if not, they will be items of high quality and of a technology that will be qualified and tested;

- National and international codes and standards that will be used as design rules for items important to safety have been identified and evaluated to determine their applicability, adequacy and sufficiency, and they will be supplemented or modified as necessary to ensure that the quality of the design is commensurate with the associated safety function;

- Where an unproven design or feature is introduced or where there is a departure from an established engineering practice, safety will 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.

References for review:

- IAEA SSR–2/1 (Rev. 1), Requirement 9.

4. DOCUMENTATION OF THE REVIEW RESULTS

The final deliverable of a TSR–DS review will be the Final Report. It will consist of an executive summary of observations made by the reviewers regarding the conceptual design documentation and a set of review sheets. As contribution to the Final Report, the review of the conceptual design documentation will produce an overall evaluation of the conceptual design documentation, and briefly summarize the observations and issues that were identified during the review.

The level of detail included in the conceptual design documentation provided for the review will greatly impact the level of detail of the observations identified in the review and the Final Report. However, these observations will be beneficial for the Requesting Party to improve the quality and content of the conceptual design documentation and to ultimately enhance nuclear safety aspects based on the IAEA safety standards.

Detailed information on the format and contents of the associated documentation to be produced during the review process will be provided in Terms of References, tailor made for each TSR-DS review service.

APPENDIX I:
STANDARDIZED FORMAT TO DESCRIBE THE DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS AND PLANT EQUIPMENT IN CONCEPTUAL DESIGN DOCUMENTATION

I.1. The conceptual design documentation should provide a description of the design of the main structures, systems and components and plant equipment needed for the safe and reliable operation of novel design reactors in all plant states.

I.2. A proposed common format for each section dealing with SSCs (in particular systems) and plant equipment is given below. This section should also cover a description of the design of the main support systems of the reactor module (e.g. water, gas, molten salt, liquid metal treatment), fuel handling, spent fuel storage, nuclear steam supply systems (e.g. steam generators, heat exchangers) or other equipment interacting with the reactor installation.

Functions of each structure, system and component, and item of equipment

I.3. The safety and non-safety functions of the main structures, systems or components, should be described here.

Design basis

I.4. This section should include the information on safety design, criteria, and associated rules intended to apply to the structure, system or component, to show how the design meets the intent of the following specific safety requirements provided in SSR 2/1(Rev.1) [2]:

(a) Design basis of items important to safety (Requirement 14);

(b) Design limits (Requirement 15);

(c) Internal and external hazards (Requirement 17);

(d) Engineering design rules (Requirement 18);

(e) Safety classification (Requirement 22);

(f) Reliability of items important to safety (Requirement 23);

(g) Common cause failure (Requirement 24).

The specificity corresponds to the maturity of the design. It is understood that usually not all components will be sufficiently specified or not at the full level of details. The comprehensiveness of the list and the level of description may be adjusted according the status of the design.

Materials

I.5.In this section, adequate and sufficient information should be provided regarding the materials used in components, the behaviour of these materials under irradiation (when applicable), and the material interactions with fluids that could potentially impair the operation of engineered safety feature systems. The purpose of the information included in this section of the conceptual design documentation safety analysis report is to demonstrate compatibility of the materials with the specific fluids to which the materials are subjected. Their specific properties, quality and chemistry requirements should be described.

Monitoring, inspection, testing and maintenance

I.6. This section should present the anticipated monitoring, inspection, testing and maintenance (including ageing management) that will help show that:

– The status of the equipment or system is in accordance with the design intent;

– There is adequate assurance that the equipment or system is available and reliable to operate as necessary;

– There has been no significant deterioration in the availability, performance and integrity of the equipment or system since the last test.

Radiation protection aspects

I.7. This section should describe the design approach to ensure that occupational exposures arising from the operation or maintenance of the equipment or system, are as low as reasonably achievable in operational states and in accident or post-accident conditions.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Fundamental Safety Principles, IAEA Safety Fundamentals SF–1, Vienna, 2006.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power plants: Design, IAEA Specific Safety Requirements SSR–2/1 (Rev. 1), Vienna, 2016.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA General Safety Requirements GSR Part 4 (Rev. 1), Vienna, 2016.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSR Part 5, IAEA, Vienna (2009).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Format and Content of the Safety Analysis Report for Nuclear Power plants, Draft Specific Safety Guide SSG–61, Vienna, April 2020.

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Technical Safety Review (TSR) Service Guidelines IAEA Services Series 41, Vienna, April 2019.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power plants: Commissioning and Operation, IAEA Specific Safety Requirements SSR–2/2 (Rev. 1), Vienna, 2016.

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

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Classification of Structures, Systems and Components in Nuclear Power plants, IAEA Specific Safety Guide SSG–30, Vienna, 2014.