

IAEA Safety Standards

for protecting people and the environment

Design of the Reactor Containment and Associated Systems for Nuclear Power Plants

Specific Safety Guide

No. SSG-53



IAEA

International Atomic Energy Agency



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IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available on the IAEA Internet site

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The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

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The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

DESIGN OF THE REACTOR
CONTAINMENT AND
ASSOCIATED SYSTEMS FOR
NUCLEAR POWER PLANTS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA SAFETY STANDARDS SERIES No. SSG-53

DESIGN OF THE REACTOR CONTAINMENT AND ASSOCIATED SYSTEMS FOR NUCLEAR POWER PLANTS

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2019

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FOREWORD

The IAEA's Statute authorizes the Agency to “establish or adopt... standards of safety for protection of health and minimization of danger to life and property” — standards that the IAEA must use in its own operations, and which States can apply by means of their regulatory provisions for nuclear and radiation safety. The IAEA does this in consultation with the competent organs of the United Nations and with the specialized agencies concerned. A comprehensive set of high quality standards under regular review is a key element of a stable and sustainable global safety regime, as is the IAEA's assistance in their application.

The IAEA commenced its safety standards programme in 1958. The emphasis placed on quality, fitness for purpose and continuous improvement has led to the widespread use of the IAEA standards throughout the world. The Safety Standards Series now includes unified Fundamental Safety Principles, which represent an international consensus on what must constitute a high level of protection and safety. With the strong support of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its standards.

Standards are only effective if they are properly applied in practice. The IAEA's safety services encompass design, siting and engineering safety, operational safety, radiation safety, safe transport of radioactive material and safe management of radioactive waste, as well as governmental organization, regulatory matters and safety culture in organizations. These safety services assist Member States in the application of the standards and enable valuable experience and insights to be shared.

Regulating safety is a national responsibility, and many States have decided to adopt the IAEA's standards for use in their national regulations. For parties to the various international safety conventions, IAEA standards provide a consistent, reliable means of ensuring the effective fulfilment of obligations under the conventions. The standards are also applied by regulatory bodies and operators around the world to enhance safety in nuclear power generation and in nuclear applications in medicine, industry, agriculture and research.

Safety is not an end in itself but a prerequisite for the purpose of the protection of people in all States and of the environment — now and in the future. The risks associated with ionizing radiation must be assessed and controlled without unduly limiting the contribution of nuclear energy to equitable and sustainable development. Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures¹ have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered ‘overarching’ requirements, are expressed as ‘shall’ statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it

¹ See also publications issued in the IAEA Nuclear Security Series.

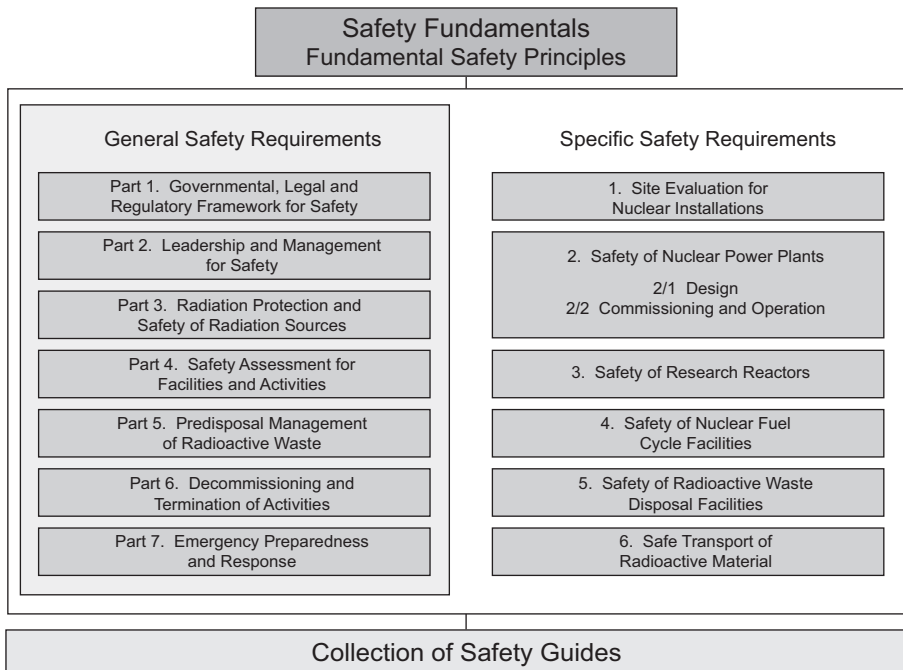


FIG. 1. The long term structure of the IAEA Safety Standards Series.

is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be

used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five safety standards committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards.

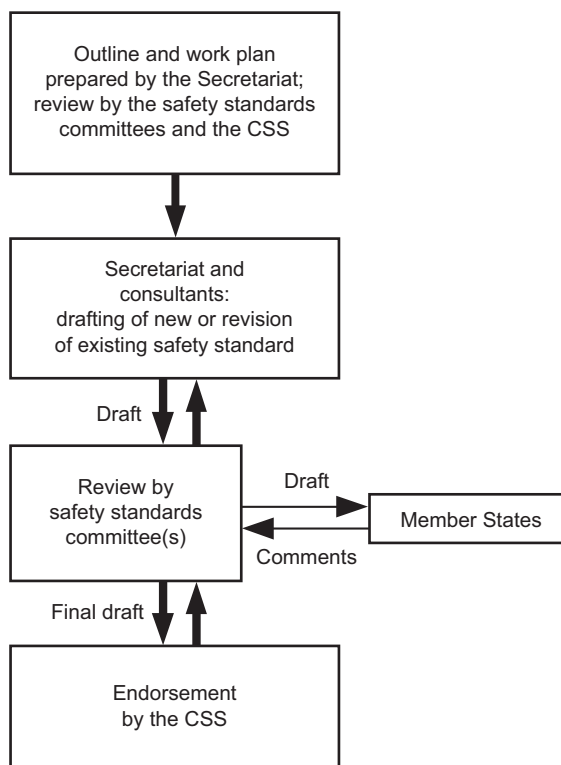


FIG. 2. The process for developing a new safety standard or revising an existing standard.

It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Safety Glossary (see <http://www-ns.iaea.org/standards/safety-glossary.htm>). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

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1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide on the design of the reactor containment and associated systems for nuclear power plants provides recommendations on how to meet the requirements of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [1], in relation to the containment structures and systems for nuclear power plants. This Safety Guide is a revision of IAEA Safety Standards Series No. NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants¹, which it supersedes.

1.2. In accordance with Requirement 4 of SSR-2/1 (Rev. 1) [1], the confinement of radioactive material in a nuclear power plant, including the control of discharges and the minimization of radioactive releases into the environment, is a fundamental safety function to be ensured in any operational state and accident condition. In accordance with the concept of defence in depth, this fundamental safety function is achieved by means of several barriers and levels of defence. For nuclear power plants, a strong structure surrounding the reactor (known as the ‘containment’) is designed to prevent or control and limit the release and the dispersion of radioactive substances [2]. Moreover, taking into account the mass and energy and the combustible gases that can be released in the event of an accident, systems designed to preserve the integrity of the containment or to avoid a bypass of the containment are necessary. Systems necessary for normal operation or to minimize radioactive releases, to remove energy, or to preserve the structural integrity of the containment in accident conditions are referred to as ‘associated systems’, or simply as ‘systems’, in this Safety Guide.

1.3. The containment and its associated systems referred to in this Safety Guide comprise the containment structure and the systems with the functions of isolation, control and management of mass and energy releases, control and limitation of radioactive releases, and control and management of combustible gases. This definition also applies to double wall containments.

¹ INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Reactor Containment Systems for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.10, IAEA, Vienna (2004).

OBJECTIVE

1.4. The objective of this Safety Guide is to provide recommendations on meeting the requirements of SSR-2/1 (Rev. 1) [1] relevant to the containment and its associated systems.

1.5. This Safety Guide is intended for use primarily with land based, stationary nuclear power plants with water cooled reactors designed for electricity generation or for other heat generating applications (e.g. district heating, desalination). It is recognized that for other reactor types, including future plant systems featuring innovative developments, some of the recommendations might not be appropriate or might need some judgement to be applied in their interpretation.

1.6. This Safety Guide is intended for use by organizations responsible for designing, manufacturing, constructing and operating nuclear power plants, as well as by regulatory bodies.

SCOPE

1.7. The recommendations provided in this Safety Guide are targeted primarily at new nuclear power plants. For nuclear power plants designed with earlier standards, it is expected that in the safety assessment of such designs, a comparison will be made with the current standards (e.g. as part of the periodic reassessment of the plant) to determine whether the safe operation of the plant could be further enhanced by means of reasonably practicable safety improvements: see para. 1.3 of SSR-2/1 (Rev. 1) [1]. Further guidance on the application of the recommendations to existing nuclear power plants is provided in the Appendix.

1.8. This Safety Guide addresses the functional aspects of the containment and its associated systems for the management of the mass and energy, radioactive material and combustible gases of the reactor for the plant states considered in the plant design envelope². In particular, recommendations relevant to the design of equipment and systems necessary for the mitigation of design extension conditions without significant fuel degradation and for design extension conditions with core melting have been added. Consideration is also given to the definition of the design bases for the containment and its associated systems, in particular to

² The phrase 'plant design envelope' is used to refer, in a simplified way, to all conditions postulated in the design of a nuclear power plant.

aspects affecting the structural design, the reliability and the independence of systems that form part of different levels of defence.

1.9. Recommendations are also provided on the tests and inspections that are necessary to ensure that the containment and its associated systems for nuclear power plants are capable of accomplishing their intended functions throughout the operating lifetime of the nuclear power plant.

1.10. Design limits and engineering criteria, together with the system parameters that should be used to verify them, are specific to individual designs for nuclear power plants and to individual States and are therefore outside the scope of this Safety Guide. However, general recommendations on these topics are provided.

1.11. Issues relating to the confinement of spent fuel are outside the scope of this Safety Guide. Recommendations on these issues are provided in IAEA Safety Standards Series Nos NS-G-1.4, Design of Fuel Handling and Storage Systems for Nuclear Power Plants [3], and SSG-15, Storage of Spent Nuclear Fuel [4]. Issues relating to the confinement of radioactive substances in buildings such as the radioactive effluent treatment or storage building or the auxiliary building are also outside the scope of this Safety Guide.

STRUCTURE

1.12. In Section 2, the safety functions relating to the containment and its associated systems are described and the main requirements of SSR-2/1 (Rev. 1) [1] that need to be considered are addressed. Section 3 provides recommendations on the design basis of the containment structure and its components and associated systems. Section 4 provides specific recommendations for the design of the containment and its associated systems. Section 5 covers tests and inspections and provides recommendations for commissioning tests and for in-service tests and inspections. General guidance on the application of the recommendations to existing nuclear power plants is provided in the Appendix.

2. CONTAINMENT SAFETY FUNCTIONS AND THE DESIGN APPROACH

2.1. This section addresses the application of the principal technical requirements established in SSR-2/1 (Rev. 1) [1] for the design of the containment and its associated systems.

SAFETY FUNCTIONS

2.2. As stated in Requirement 54 of SSR-2/1 (Rev. 1) [1]:

“A containment system shall be provided to ensure, or to contribute to, the fulfilment of the following safety functions at the nuclear power plant: (i) confinement of radioactive substances in operational states and in accident conditions; (ii) protection of the reactor against natural external events and human induced events; and (iii) radiation shielding in operational states and in accident conditions.”

2.3. The conditions under which these safety functions have to be accomplished are required to be identified and characterized to define the different elements of the design bases of the relevant structures, systems and components (see Requirement 14 of SSR-2/1 (Rev. 1) [1]).

CONFINEMENT OF RADIOACTIVE MATERIAL

2.4. As stated in Requirement 55 of SSR-2/1 (Rev. 1) [1]:

“The design of the containment shall be such as to ensure that any radioactive release from the nuclear power plant to the environment is as low as reasonably achievable, is below the authorized limits on discharges in operational states and is below acceptable limits in accident conditions.”

2.5. For operational states, the annual dose received by people living in the vicinity of a nuclear installation is expected to be comparable to the effective dose due to natural background levels of radiation (i.e. the levels that originally existed at the site). For public exposure in planned exposure situations, the proposed range of values for the dose constraint indicated in IAEA Safety

Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [5], represents an increase of up to 1 mSv in a year over the dose received from exposure due to naturally occurring radiation sources.

2.6. The approach to radioactive releases in accident conditions is required to be as follows:

- (a) For design basis accidents and design extension conditions without significant fuel degradation, releases are minimized such that off-site protective actions (e.g. evacuation, sheltering, iodine thyroid blocking) are not necessary (see para. 5.25 of SSR-2/1 (Rev. 1) [1] and Ref. [6]).
- (b) For design extension conditions with core melting, releases are minimized such that only off-site protective actions limited in terms of lengths of time and areas of application are necessary, and sufficient time should be available to take such measures (see para. 5.31A of SSR-2/1 (Rev. 1) [1] and Ref. [6]).
- (c) Accident sequences that might lead to an early radioactive release or a large radioactive release are ‘practically eliminated’ by appropriate design provisions (see paras 2.11, 2.13 and 2.14 of SSR-2/1 (Rev. 1) [1]).
- (d) In addition, the containment and its associated systems are designed so that any radioactive release is as low as reasonably achievable, is below the authorized limits on discharges in operational states and is below acceptable limits in accident conditions (see Requirement 55 of SSR-2/1 (Rev. 1) [1]).

2.7. The leaktightness of the containment is essential to confine radioactive material and to minimize radioactive releases. Leaktightness is generally characterized by specified maximum leak rates (overall leak rate and specific leak rates for containment penetrations, air locks, hatches and containment isolation valves) that are not expected to be exceeded under accident conditions. Equipment intended to ensure the functions of the containment are performed is required to be designed and qualified to ensure that the containment keeps its integrity and leaktightness before and during the prevailing environmental conditions for which the equipment is necessary (see Requirements 30 and 55 of SSR-2/1 (Rev. 1) [1]).

2.8. Isolation of the containment is necessary to confine radioactive releases into the containment atmosphere caused by accident conditions (see Requirement 56 of SSR-2/1 (Rev. 1) [1]).

2.9. In accordance with Requirement 58 of SSR-2/1 (Rev. 1) [1], the systems designed to ensure that the specified design limits of the containment (e.g. in

relation to pressure, temperature and combustible gases) will not be exceeded are required to be implemented, as necessary, to preserve the structural integrity of the containment in accident conditions. Multiple means are required to be implemented to remove heat from the containment in accident conditions. The systems specifically dedicated to addressing design extension conditions with core melting are required to be independent of safety systems as far as practicable (see para. 4.13A of SSR-2/1 (Rev. 1) [1]).

2.10. The structural integrity of the civil structures of the containment and of the systems necessary for the mitigation of accident conditions is required to be ensured with appropriate margins, taking into account the loads or combinations of loads originating from the hazards or prevailing in the plant states during which such structures are required to operate (see Requirement 42 of SSR-2/1 (Rev. 1) [1]).

2.11. Irrespective of the multiplicity of design provisions taken to prevent an accident from escalating to a plant state involving significant core damage, a set of the most likely representative core melting conditions is required to be postulated. For such conditions, additional safety features are required to be implemented to minimize the radioactive releases (see Requirement 20 of SSR-2/1 (Rev. 1) [1]).

2.12. In addition to the design provisions implemented to mitigate the consequences of the postulated accident conditions, the use of non-permanent equipment is also to be considered, and adequate connection points and interfaces with the plant are required to be installed with the objective of avoiding large releases of radioactive material and unacceptable off-site contamination in accident conditions exceeding those considered in the design (see Requirement 58 of SSR-2/1 (Rev. 1) [1]).

2.13. In accident conditions, highly energetic phenomena that could jeopardize the structural integrity and the leaktightness of the containment are required to be dealt with by incorporating adequate features to ensure that the possibility of such phenomena may be considered to have been ‘practically eliminated’ (see Requirements 20 and 58 of SSR-2/1 (Rev. 1) [1]).

PROTECTION AGAINST EXTERNAL AND INTERNAL HAZARDS

2.14. In accordance with Requirement 17 of SSR-2/1 (Rev. 1) [1], the containment or a shielding structure is required to be designed to protect items important to safety housed inside the containment against the effects of natural and human induced external hazards identified by the hazard evaluation for the site, and

against the effects of internal hazards originating from equipment installed at the site. Causation and the likelihood of hazard combination should be considered.

2.15. The containment or the shielding structure also provides protection against the effects of possible malicious acts directed against the facility. Recommendations and guidance on security measures are provided in publications in the IAEA Nuclear Security Series.

RADIATION SHIELDING

2.16. In operational states and in accident conditions, the containment contributes to the protection of plant personnel and the public from undue exposure due to direct radiation from radioactive material within the containment. In accordance with Requirement 5 of SSR-2/1 (Rev. 1) [1], the composition and thickness of the concrete, steel and other materials:

“shall be such as to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits, that they are kept as low as reasonably achievable in operational states for the entire lifetime of the plant, and that they remain below acceptable limits and as low as reasonably achievable in, and following, accident conditions.”

2.17. Dose limits for workers and for the public in planned exposure situations are established in GSR Part 3 [5].

3. DESIGN BASIS OF THE CONTAINMENT STRUCTURE AND ITS COMPONENTS AND ASSOCIATED SYSTEMS

GENERAL

3.1. The design of the containment and its associated systems should be conducted taking into account the requirements of IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [7], and Requirements 1–3 of SSR- 2/1 (Rev. 1) [1]. The recommendations of IAEA Safety Standards Series Nos GS-G-3.1, Application of the Management System

for Facilities and Activities [8], and GS-G-3.5, The Management System for Nuclear Installations [9], should also be taken into account.

3.2. The design of the containment and its associated systems should take into account requirements, recommendations and guidance for both safety and security. Safety measures and security measures should be designed and applied in an integrated manner, and as far as possible in a complementary manner, so that security measures do not compromise safety and safety measures do not compromise security. Recommendations for nuclear security are provided in IAEA Nuclear Security Series No. 13, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5) [10].

3.3. The design basis for the containment and its associated systems should consider all plant states (i.e. any condition arising in normal operation, anticipated operational occurrences, design basis accidents and design extension conditions). Load combinations created by internal and external hazards should also be included in the design basis for the relevant structures, systems and components.

3.4. Design conditions and design loads should be calculated taking into account bounding conditions determined for each of the relevant plant states or hazards.

3.5. The necessary performance of structures, systems and components for operational states should be derived on the basis of the following needs:

- (a) To confine radioactive material;
- (b) To minimize radioactive releases;
- (c) To contribute to radiation shielding;
- (d) To maintain pressure and temperature within the range specified for operational states;
- (e) To establish and maintain adequate environmental conditions in work areas;
- (f) To provide for the necessary access and egress of personnel and materials;
- (g) To perform containment structural and leaktightness tests;
- (h) To accommodate the loads that occur during operational transients (e.g. loads due to differential thermal expansion and variation of outside environmental temperature).

Other factors, including nuclear security considerations (see paras 2.15 and 3.2), also need to be taken into account.

POSTULATED INITIATING EVENTS

3.6. Paragraphs 3.7 and 3.8 provide recommendations on meeting Requirement 16 of SSR-2/1 (Rev. 1) [1].

3.7. The postulated initiating events relevant to the containment and its associated systems should include equipment failures and errors potentially leading to accident conditions with a significant release of radioactive material or with a significant release of mass and energy inside the containment. Postulated initiating events occurring in shutdown modes, with an open containment or when some systems are disabled for maintenance, should also be considered.

3.8. The following postulated initiating events should be considered in the design of the containment and its associated systems:

- (a) Large, medium and small breaks in the reactor coolant system;
- (b) Large, medium and small breaks in the main steam or feedwater system;
- (c) Equipment failure in systems carrying radioactive liquid or gas within the containment;
- (d) Fuel handling accidents inside the containment.

INTERNAL HAZARDS

3.9. Paragraphs 3.10–3.12 provide recommendations on meeting Requirement 17 of SSR-2/1 (Rev. 1) [1] in relation to internal hazards. More detailed recommendations are provided in IAEA Safety Standards Series No. NS-G-1.7, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants [11].

3.10. Internal hazards that should be considered in the design are those that could jeopardize the performance of the containment and its associated systems. A list of internal hazards that usually need to be considered is given below for guidance. This list should be supplemented as necessary to include specific hazards relevant to the design:

- (a) Breaks in high energy systems located inside the containment or inside the buildings that house the systems to mitigate the consequences of accident conditions;
- (b) Breaks in systems or components containing radioactive material located in the containment;

- (c) Failure of fuel handling equipment;
- (d) Heavy load drop;
- (e) Internal missiles;
- (f) Fires and explosions;
- (g) Flooding.

3.11. Layout and design provisions should be taken to protect the containment and its associated systems against the effects of internal hazards, as follows:

- (a) The containment and its associated systems should be protected against high energy impacts (e.g. internal missiles, pipe whipping, jet impingement, heavy loads) or should be designed to withstand the loads generated by such impacts, as well as the loads caused by explosions.
- (b) The redundancies of the systems should be segregated to the extent possible, or adequately separated, and should be protected as necessary to prevent the loss of the safety function performed by the system (prevention of common cause failures initiated by the effects of the internal hazards; see Requirement 24 of SSR-2/1 (Rev. 1) [1]).
- (c) The design measures implemented in respect of segregation, separation and protection should also be adequate to ensure that the response of the systems, as described in the analysis of the postulated initiating events, remains valid when considering the effects of the hazard.
- (d) A single hazard should not result in a common cause failure between safety systems designed to control design basis accidents and safety features required for design extension conditions with core melting.

3.12. The design methods and construction codes used should provide adequate margins to avoid cliff edge effects in the event of a slight increase in the severity of the internal hazards (see also Requirements 9 and 11 of SSR-2/1 (Rev. 1) [1]).

EXTERNAL HAZARDS

3.13. Paragraphs 3.14–3.22 provide recommendations on meeting Requirement 17 of SSR-2/1 (Rev. 1) [1] in relation to external hazards. More detailed recommendations are provided in IAEA Safety Standards Series No. NS-G-1.5, External Events Excluding Earthquakes in the Design of Nuclear Power Plants [12].

3.14. Guidance on typical external hazards, and their combination as appropriate, that usually need to be considered is provided in NS-G-1.5 [12] and IAEA Safety

Standards Series No. NS-G-1.6, Seismic Design and Qualification for Nuclear Power Plants [13]. The list of external hazards contained in NS-G-1.5 [12] should be adapted or supplemented as necessary to include site specific hazards.

3.15. The containment and the buildings that house systems to mitigate the consequences of accident conditions should be designed to withstand the loads imposed by external hazards and protected against any effects caused by neighbouring buildings that are not designed to withstand loads from external hazards.

3.16. Systems required for mass and energy release and management, the control of radioactive releases and the management of combustible gases in accident conditions should be protected against the effects of external hazards or be designed to withstand the loads caused by the external hazards. For each hazard, all components that need to retain their operability or integrity during or after the hazard should be identified and specified in the design basis of the component.

3.17. Design methodologies should contain measures to verify that adequate margins exist to avoid cliff edge effects in the event of a slight increase in the severity of the external hazards (see Requirements 9 and 11, and para. 5.21, of SSR-2/1 (Rev. 1) [1]).

3.18. Short term actions necessary to meet the engineering criteria established for the containment in the event of design basis accidents or design extension conditions should be accomplished by permanent systems (see also para. 5.17 of SSR-2/1 (Rev. 1) [1]).

3.19. The autonomy of systems designed for mass and energy release and management, the control of radioactive releases and the management of combustible gases inside the containment during accident conditions should be such as to remain operational for longer than the time necessary before crediting off-site support services. The autonomy can be based on crediting the provisions taken at the unit and at the site, provided that the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously has been considered (see para. 5.15B of SSR-2/1 (Rev. 1) [1]).

3.20. As stated in para. 5.21A of SSR-2/1 (Rev. 1) [1]:

“The design of the plant shall also provide for an adequate margin to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those

considered for design, derived from the hazard evaluation for the site.”

Paragraphs 3.21 and 3.22 provide recommendations on meeting this requirement.

3.21. Margins provided by the design of the structures, systems and components that are ultimately necessary to avoid an early radioactive release or a large radioactive release should be adequate, such that the integrity and operability of those systems would be preserved in the event that natural hazards cause loads exceeding those derived from the hazard evaluation of the site. A detailed list of these structures, systems and components is dependent on the specific design. The list below provides typical examples of structures, systems and components that could be considered:

- (a) Containment structure;
- (b) Equipment or structures necessary to contain the molten core;
- (c) Systems necessary to remove heat from the molten core;
- (d) Systems necessary to remove heat from the containment and transfer heat to the ultimate heat sink in design extension conditions;
- (e) Systems to prevent gas combustion regimes from challenging the containment integrity;
- (f) Containment venting system (if it exists);
- (g) Systems for containment isolation.

3.22. In the case of external flooding, all the structures hosting the systems listed in para. 3.21 should be located at an elevation higher than the one derived from the site hazard evaluation, otherwise adequate engineered safety features (e.g. watertight doors) should be in place to protect these structures and ensure that mitigating actions can be maintained.

ACCIDENT CONDITIONS

General

3.23. Accident conditions that should be considered relevant for the design of the containment and its associated systems are those with the potential to cause excessive mechanical loads or to jeopardize the capability to limit radioactive releases to the environment.

3.24. Accident conditions should be used in determining capabilities, loads and environmental conditions in the design of the containment and its associated

systems. The determination of capabilities, loads and environmental conditions should be based on, but not necessarily limited to, the following:

- (a) The mass and energy released to the containment as a whole as a function of time;
- (b) Maintaining an adequate coolant inventory;
- (c) The heat transfer to the containment structures, and to and from components;
- (d) The mechanical loading, both static and dynamic, on the containment structure and its subcompartments;
- (e) Releases of radioactive material inside the containment;
- (f) The amount of radioactive material released to the environment;
- (g) Cooling, stabilization and localization of the molten core (for the ex-vessel retention strategy);
- (h) The rate of generation and amount of combustible gases released inside the containment.

3.25. Paragraphs 3.26–3.28 provide recommendations on meeting Requirement 18 of SSR-2/1 (Rev. 1) [1].

3.26. To the extent practicable, codes and engineering rules that are used for design should be documented, validated and, in the case of new design codes, developed using up to date knowledge and in accordance with recognized standards for quality assurance. Users of the design codes should be qualified and trained in the operation and limits of the codes and the assumptions made in the design.

3.27. Calculations of boundary conditions for design basis accidents and design extension conditions should be documented, indicating the relevant assumptions for the evaluation of parameters, the engineering criteria and the computer codes that are used.

3.28. Computer codes should not be used beyond their identified and documented domain of validation.

Design basis accidents

3.29. Paragraphs 3.30 and 3.31 provide recommendations on meeting Requirement 19 of SSR-2/1 (Rev. 1) [1].

3.30. For the performance of the containment and its associated systems, conditions retained as design basis accident conditions should be calculated taking into account the less favourable initial conditions and equipment performance and

the single failure that has the largest impact on the performance of the safety systems. When introducing adequate conservatism, the following should be taken into account:

- (a) For the same event, an approach considered conservative for designing one specific system could be non-conservative for another.
- (b) The adoption of excessively conservative assumptions could lead to an unrepresentative analysis and consideration of undue stresses on components and structures.

3.31. The containment and its associated systems should be designed such that venting is not necessary in design basis accidents.

Design extension conditions

3.32. Paragraphs 3.33–3.38 provide recommendations on meeting Requirement 20 of SSR-2/1 (Rev. 1) [1].

3.33. In addition to the design basis conditions, relevant design extension conditions should also be identified and used to establish the design bases of the containment and its associated systems necessary to meet the objectives established for that category of accident. For design extension conditions without significant fuel degradation, the radiological consequences should be comparable to those established for design basis accidents. For accident conditions with core melting, the radioactive release should be such that the necessary off-site protective actions remain limited in terms of lengths of time and areas of application.

3.34. The calculations performed to assess design extension conditions may be less conservative than those imposed by design basis accidents, provided that the margins necessary to avoid cliff edge effects are still sufficient to cover uncertainties. Performing sensitivity analyses is also useful as a means of identifying the key parameters.

3.35. Design extension conditions relevant to the design of the containment and its associated systems should be identified on the basis of a deterministic approach supplemented by probabilistic analyses and engineering judgement (see Requirement 42 of SSR-2/1 (Rev. 1) [1]).

3.36. For design extension conditions without significant fuel degradation, in general the following three types of failure should be considered:

- (a) Equipment failures leading to a release of mass and energy greater than that postulated for design basis accidents (e.g. from a loss of coolant accident or main steam line break);
- (b) Multiple failures (e.g. common cause failures in redundant trains) in the containment and its associated systems that prevent the safety systems from performing their intended function when requested;
- (c) Multiple failures that cause the loss of a safety system that is being used to fulfil the fundamental safety functions in normal operation (e.g. residual heat transport systems).

3.37. As multiple failures are likely to be caused by the occurrence of dependent failures that could then lead to the failure of the safety systems, an analysis of the dependencies between redundant trains of systems that are installed to control the pressure buildup in, or to remove energy from, the containment in the event of a design basis accident should be conducted to identify relevant possibilities for design extension conditions.

3.38. The following conditions relevant to the design of the containment and its associated systems should normally be considered for design extension conditions:

- (a) Station blackout;
- (b) Loss of the means designed to limit the pressure buildup in the event of a design basis accident;
- (c) Loss of the heat transfer chain removing heat from the containment to the ultimate heat sink;
- (d) A faulty pressure suppression function (boiling water reactor);
- (e) Loss of the ultimate heat sink.

3.39. Paragraphs 3.40–3.45 provide recommendations on meeting Requirements 20 and 68 of SSR-2/1 (Rev. 1) [1] in relation to design extension condition with core melting.

3.40. A set of the most likely representative conditions of an accident with core melting should be used to provide inputs to the design of the containment and the safety features necessary to mitigate the consequences of an accident with core melting. Conditions with core melting, retained as boundary conditions for the design of the containment and its associated systems, should be justified on the basis of Level 2 probabilistic safety analyses, supplemented by engineering

judgement, to allow the selection of appropriate conditions that are representative and most probable.

3.41. Accidents involving core melting should be postulated as design extension conditions, irrespective of the design provisions taken to prevent such conditions and irrespective of their estimated probability of occurrence.

3.42. Design extension conditions with core melting need to be considered as bounding conditions in the design of safety features for such conditions. Aspects that affect the accident progression and that influence the containment response and the source term should be taken into account. These aspects include the following:

- (a) The containment status (containment open or bypassed);
- (b) The amount of radioactive material initially released to the containment;
- (c) The pressure at the onset of core damage;
- (d) The amount and concentration of combustible gases released inside the containment;
- (e) The timing of core damage (early emergency core cooling system failure (injection phase) vs. long term cooling failure);
- (f) The status of containment safety features (containment cooling, spray, fan coolers, suppression pool);
- (g) The status of AC and DC power sources;
- (h) The status of instrument air systems;
- (i) The status of the spent fuel pool systems if they are inside the containment.

3.43. Design provisions should be implemented to prevent a containment failure in the event of design extension conditions. These provisions should aim to prevent significant overpressurization of the containment, stabilize the molten core, remove the heat from the containment and prevent gas combustion regimes from challenging the containment integrity.

3.44. Multiple means to control the pressure buildup in accident conditions inside the containment should be implemented, and venting (if any) should be used only as a last resort.

3.45. For design extension conditions for which venting the containment would be necessary to preserve the integrity of the containment, the use of the venting

system should not lead to an early radioactive release or a large radioactive release (see para. 6.28A of SSR-2/1 (Rev. 1) [1]). To ensure that this is the case:

- (a) The containment venting system should be equipped with filters of adequate capacity and high efficacy.
- (b) The containment venting system should be designed to withstand loads from external hazards (including natural hazards exceeding those considered for design, derived from the hazard evaluation for the site) and the static and dynamic pressure loads existing when the containment venting system is operated.
- (c) It should be possible to reliably open and close the vent line valves.
- (d) Measures should be taken to prevent the design limit from being exceeded in the event of subatmospheric containment pressure.

DESIGN LIMITS

3.46. Paragraphs 3.47–3.50 provide recommendations on meeting Requirements 15 and 28 of SSR-2/1 (Rev. 1) [1] in relation to design limits and to operational limits and conditions for safe operation.

3.47. The performance of the containment and its associated systems is required to be assessed against a well defined and accepted³ set of design limits and criteria.

3.48. A set of primary design limits for the containment and its associated systems should be established as a means of ensuring the overall safety functions of the containment in all operational states and in accident conditions. These primary design limits are usually expressed in terms of the following:

- (a) Overall containment leak rate at design pressure;
- (b) Direct (unfiltered) leakages;
- (c) Dose limits and dose constraints for the public (see GSR Part 3 [5]) and limits on radioactive releases specified for operational states and accident conditions;
- (d) Dose limits and dose constraints for workers (see GSR Part 3 [5]), and maximum dose rates for shielding purposes.

³ 'Well defined and accepted' generally means either widely accepted by regulatory bodies or proposed by international organizations.