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

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

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 site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

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Security related publications are issued in the **IAEA Nuclear Security Series**.

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
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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 measures[1] 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

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

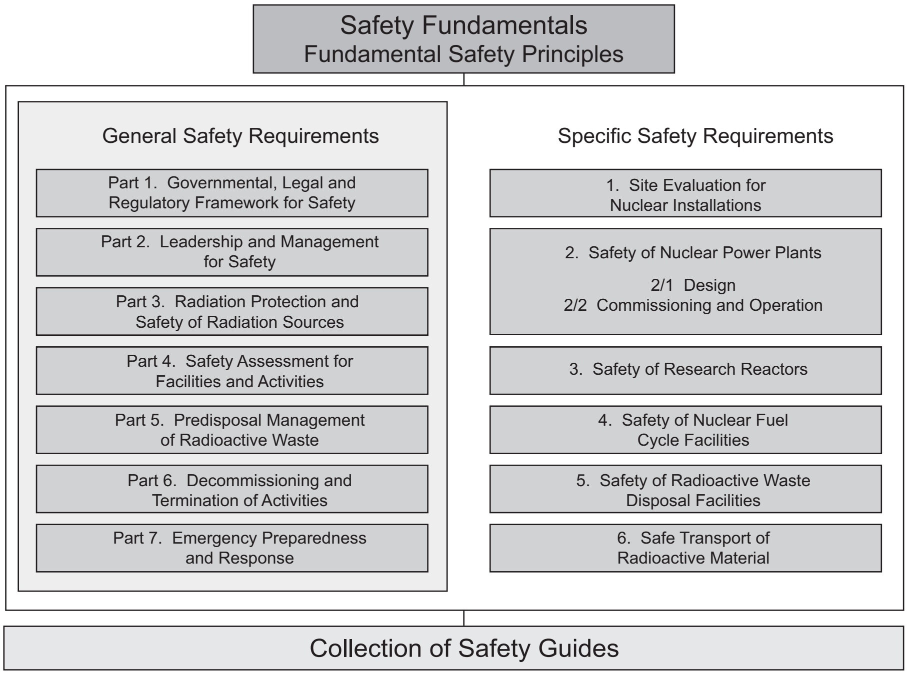

FIG. 1. The long term structure of the IAEA Safety Standards Series.

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.

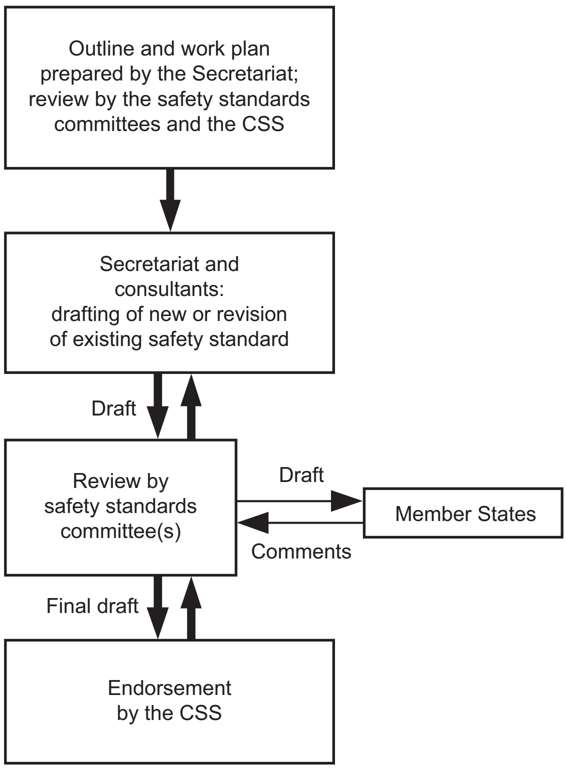

FIG. 2. The process for developing a new safety standard or revising an existing standard.

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.

CONTENTS

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 Plants[1], 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.

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[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 envelope[2]. 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 accepted[3] 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.

3.49. Design limits should also be specified for each containment structure and component.

3.50. Operational limits should be applied to operating parameters (e.g. maximum coolant temperature, minimum flow rate for air coolers) and to performance indicators (e.g. maximum closing time for isolation valves, penetration air leakage).

RELIABILITY

3.51. Paragraphs 3.52–3.62 provide recommendations on meeting Requirements 17, 21–26, 29, 30 and 68 of SSR-2/1 (Rev. 1) [1].

3.52. To achieve adequate reliability of the systems necessary to control energy, radioactive material and combustible gases released inside the containment, the following factors should be considered:

(a) Safety classification and the associated engineered requirements for design and manufacturing;
(b) Design criteria relevant for the systems (number of redundant trains, seismic qualification, qualification in relation to harsh environmental conditions, power supplies);
(c) Consideration of vulnerabilities for common cause failures (diversity, separation, independence);
(d) Layout provisions to protect the system against the effects of internal and external hazards;
(e) Periodic testing and inspection;
(f) Maintenance;
(g) Use of equipment designed to fail in a safe direction.

Systems designed to mitigate design basis accidents

3.53. In the event of design basis accidents, mass and energy release and management, and the control of radioactive releases should remain possible despite consequential failures caused by the postulated initiating event and a single failure postulated in any system necessary to accomplish these safety functions. The unavailability of systems that are undergoing maintenance or repair should also be considered.

3.54. The emergency power source should have adequate capability to supply power to electrical equipment necessary for mass and energy release and management and for the control of radioactive releases in the event of design basis accidents.

3.55. Vulnerabilities for common cause failures between the redundancies of the safety systems should be identified, and design or layout provisions should be implemented to make the redundancies independent as far as is practicable.

3.56. Recommendations relating to the protection of these systems with regard to the effects of internal hazards, external hazards and environmental conditions are addressed in paras 3.10–3.12, 3.14–3.22 and 3.77–3.84, respectively.

Safety features for design extension conditions without significant fuel degradation

3.57. The need for additional safety features is reactor technology and design dependent. A reliability analysis of the safety systems designed for mass and energy release and management should be conducted to identify the need for additional safety features to preserve containment integrity.

3.58. The more likely combinations of postulated initiating events and common cause failure between the redundancies of the safety systems should be analysed. If the consequences exceed the limits given for design basis accidents, the vulnerabilities for common cause failure should be removed or additional design features should be implemented to cope with such situations. The additional features for mass and energy release and management should be designed and installed such that they are unlikely to fail due to the same common cause.

3.59. Additional safety features should be sufficiently reliable to contribute to the practical elimination of conditions that could lead to an early radioactive release or to a large radioactive release. Recommendations similar to those provided in paras 3.53–3.56 for systems designed to mitigate the consequences of design basis accidents should be applied to safety features for design extension conditions without significant fuel degradation, while taking into account that meeting the single failure criterion is not required. Additional safety features for design extension conditions without significant fuel degradation should be supplied from a different and diverse power source (e.g. by the alternate power source installed at the nuclear power plant).

Safety features for design extension conditions with core melting

3.60. The dedicated safety features should be sufficiently reliable to accomplish their safety function.

3.61. Paragraph 6.44B of SSR-2/1 (Rev. 1) [1] states that "Equipment that is necessary to mitigate the consequences of melting of the reactor core shall be capable of being supplied by any of the available power sources."

3.62. Recommendations relating to the protection of dedicated safety features with regard to the effects of internal hazards, external hazards and environmental conditions are addressed in paras 3.10–3.12, 3.14–3.22 and 3.77–3.84.

DEFENCE IN DEPTH

3.63. Paragraphs 3.64 and 3.65 provide recommendations on meeting Requirement 7 of SSR-2/1 (Rev. 1) [1].

3.64. Different systems should be implemented for mass and energy release and management, for pressure and temperature control and for containment heat removal in the different plant states.

3.65. The following recommendations contribute to the implementation of independence:

(a) Successive items belonging to different levels of defence in depth necessary to control the pressure inside the containment or to remove energy from the containment should be identified.
(b) Vulnerabilities for common cause failure between such items should be identified, and the consequences should be assessed. The vulnerabilities for common cause failure should be removed to the extent possible where the consequences for the integrity of the containment and for radioactive releases are judged to be not acceptable. In particular, dedicated safety features designed to mitigate the consequences of design extension conditions with core melting should be sufficiently independent from equipment designed to mitigate the conditions inside the containment caused by design basis accidents.
(c) Independence implemented between systems should not be compromised by vulnerabilities for common cause failure in instrumentation and control systems necessary for the safety actuation of the systems or the

monitoring of the containment conditions (see paras 4.204–4.241 for more recommendations for instrumentation and control systems).

PRACTICAL ELIMINATION OF CONDITIONS THAT COULD LEAD TO AN EARLY RADIOACTIVE RELEASE OR A LARGE RADIOACTIVE RELEASE

3.66. As stated in para. 5.31 of SSR-2/1 (Rev. 1) [1], "The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated'."

3.67. In terms of the scope of this Safety Guide, such possibilities should include the following:

(a) Conditions involving highly energetic phenomena the consequences of which cannot be mitigated by the implementation of reasonable technical means;
(b) A core melt accident combined with a containment bypass.

3.68. Typical examples of conditions that are widely required to be practically eliminated include the following:

(a) Severe accident conditions that could damage the containment in an early phase as a result of direct containment heating, steam explosion or hydrogen detonation;
(b) Severe accident conditions that could jeopardize the integrity of the containment in a late phase as a result of a basemat or containment boundary melt-through;[4]
(c) Severe accident conditions with an open containment, notably in shutdown modes;
(d) Severe accident conditions with unintentional containment bypass.

3.69. The dedicated safety features should have adequate reliability to contribute to the practical elimination of conditions that could lead to an early radioactive release or to a large radioactive release.

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[4] These conditions should be analysed during the identification of situations to be practically eliminated, even though their consequences can generally be mitigated with implementation of reasonable technical means.

SAFETY CLASSIFICATION

3.70. Paragraphs 3.71–3.75 provide recommendations on meeting Requirement 22 of SSR-2/1 (Rev. 1) [1]. More detailed guidance is given in IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [14].

3.71. The consequences of the failure of a structure, system or component should be considered both in terms of the accomplishment of the function, and in terms of the radioactive release. For items for which both types of consequence are relevant, the safety class and the associated quality requirements necessary to achieve the expected reliability should be defined with due account taken of these two types of consequence. For items that do not contribute to the confinement of radioactive material, the safety class and the quality requirements are directly derived from the consequences, assuming the function is not accomplished.

3.72. The engineering design rules applicable to an entire system (e.g. single failure criterion, physical and electrical separation, functional independence, emergency power supply, periodic tests) should be derived from the safety class assigned to the system, assuming the function performed by the system is not accomplished.

3.73. The safety classification should be established in a consistent manner such that all systems necessary for the accomplishment of one safety function, including the associated support service systems, are assigned to the same safety class.

3.74. In accordance with Requirement 9 of SSR-2/1 (Rev. 1) [1], pressure retaining equipment that is safety classified is required to be designed and manufactured in accordance with proven codes and standards widely used by the nuclear industry (see, for example, Refs [15–17]). The engineering design and manufacturing rules applicable to each individual component should be selected with due account taken of the two effects resulting from its failure (i.e. in terms of the function(s) not accomplished and in terms of the radioactive release).[5]

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[5] In accordance with international practices, the pressure retaining boundary of components that are part of the reactor coolant pressure boundary should be designed and manufactured in compliance with the highest codes and standards defined by the industry for nuclear applications, except for parts of the reactor coolant pressure boundary whose failure would result in leakage that can be compensated by the normal water make-up system.

3.75. With regard to implementing the recommendations in paras 3.71–3.74, and in terms of the safety classification described in SSG-30 [14]:

(a) In the event of a design basis accident, systems necessary for containment isolation, for the control of pressure buildup inside the containment (e.g. containment spray system), for removal of heat from the containment, or for the transport of heat from the containment to the ultimate heat sink should be assigned to safety class 1.
(b) Systems implemented to provide a backup to safety class 1 systems for design extension conditions should be assigned to at least safety class 2.
(c) Systems necessary to preserve the containment integrity in the event of an accident with core melting (e.g. ex-vessel core cooling system, reactor coolant system depressurization system, containment spray system, containment heat removal system, venting and filtering system, systems to prevent hydrogen detonation, heat transport chain) should be assigned to at least safety class 3.
(d) The containment designed as the last physical barrier against releases should be assigned to safety class 1.

ENVIRONMENTAL QUALIFICATION

3.76. Paragraphs 3.77–3.84 provide recommendations on meeting Requirement 30 of SSR-2/1 (Rev. 1) [1].

3.77. The structures, systems and components should be qualified to perform their functions throughout their design life in the entire range of environmental conditions that might prevail before or during their operation, or should otherwise be adequately protected from those environmental conditions.

3.78. The environmental and seismic conditions that might prevail before, during and after an accident; the ageing of structures, systems and components throughout the lifetime of the plant; synergistic effects; and margins should all be taken into account in the environmental qualification. More detailed recommendations are provided in NS-G-1.6 [13] and IAEA Safety Standards Series No. SSG-48, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants [18].

3.79. Environmental qualification should be performed by means of testing, analysis and the use of engineering judgement, or by a combination of these.

3.80. Environmental qualification should include the consideration of such factors as temperature, pressure, humidity, radiation levels, chemical aspects, local accumulation of radioactive aerosols, vibration, water spray, steam impingement and flooding. Margins and synergistic effects (in which the damage due to the superposition or combination of effects might exceed the total damage due to the effects separately) should also be considered. In cases where synergistic effects are possible, materials should be qualified for the most severe effect or for the most severe combination or sequence of effects.

3.81. Techniques to accelerate the testing for ageing and qualification may be used, provided there is an adequate justification.

3.82. For components subject to the effects of ageing by various mechanisms, a design life and, if necessary, a replacement frequency should be established. In the qualification process for such components, samples should be aged to simulate the end of their design lives before being tested under design basis accident conditions.

3.83. Components that have been used for qualification testing should generally not be used for construction purposes, unless it can be shown that the conditions and methods of testing do not themselves lead to an unacceptable degradation of safety performance.

3.84. Qualification data and results should be documented as part of the design documentation.

CODES AND STANDARDS

3.85. Paragraphs 3.86 and 3.87 provide recommendations on meeting the requirements of para. 4.15 of SSR-2/1 (Rev. 1) [1].

3.86. For the design of the structures and systems of the containment, national or international codes and standards may be used, provided that the applicability and suitability of these codes and standards are demonstrated. The selected codes and standards should have the following attributes:

(a) They should be applicable to the particular concept of the design;
(b) They should form an integrated and comprehensive set of standards and criteria;

(c) They should preferably be the newest edition of the design and construction codes and standards (another edition might be used, provided that its use is adequately justified).

3.87. Relevant codes and standards have been developed by various national and international organizations, covering areas such as the following:

(a) Quality assurance;
(b) Materials;
(c) Structural design of pressurized components;
(d) Civil structures;
(e) Instrumentation and control;
(f) Environmental and seismic qualification;
(g) Pre-service and in-service inspection and testing;
(h) Fire protection.

USE OF PROBABILISTIC SAFETY ANALYSES IN DESIGN

3.88. Probabilistic safety analyses should complement the deterministic approach by identifying additional features to achieve a balanced design. The use of probabilistic analyses should not be considered as a substitute to a design approach based on deterministic requirements, but as a part of the process to identify potential safety enhancements and judge their effectiveness.

3.89. Probabilistic safety analysis should be used in support of demonstrating the practical elimination of some conditions that could lead to an early radioactive release or a large radioactive release. In particular, probabilistic safety analysis may be used to analyse the containment isolation provisions for preventing containment bypass and the total failure of the mass and energy release and management systems.

3.90. Probabilistic safety assessment should be used to confirm that the means for mitigating design extension conditions with core melting have a very low probability of failure. This should include an analysis of the reliability of the associated systems (e.g. the containment cooling system, containment filtered venting and other aspects that have traditionally been considered in Level 2 probabilistic safety assessment).

4. DESIGN OF THE CONTAINMENT AND ITS ASSOCIATED SYSTEMS

GENERAL

4.1. A number of systems are design dependent and might be different in their design principles (e.g. the use of active or passive systems for mass and energy release and management and in-vessel or ex-vessel core cooling in the event of an accident with core melting). Nevertheless, structures or systems that accomplish the same safety function in different technologies should be designed in compliance with the same design requirements.

4.2. Irrespective of the permanent design provisions for design basis accidents and for design extension conditions, as stated in para. 6.28B of SSR-2/1 (Rev. 1) [1], "The design shall also include features to enable the safe use of non-permanent equipment for restoring the capability to remove heat from the containment."

Layout and configuration of the containment and its associated systems

4.3. The layout and configuration of the containment and its associated systems are design dependent and are significantly different for reactor technologies relying on containment with a large dry volume compared with those for technologies relying on containment with a suppression pool.

4.4. The following factors should be taken into account when considering the layout and configuration of the containment:

(a) The need to accommodate and facilitate a large energy and mass release inside the containment (see paras 4.59–4.89);
(b) The provision of adequate separation between divisions of safety systems and between redundant safety features for design extension conditions, where relevant;
(c) The location of, and provisions to protect, items important to safety against the effects of internal hazards;
(d) The provision of sufficient space and shielding to ensure that planned maintenance and operations can be performed without causing undue radiation exposure of personnel;
(e) The provision of the necessary space for personnel access and for the monitoring, testing, control, maintenance and movement of equipment;

(f) Optimization of the number and location of containment penetrations so as to prevent unfiltered leaks and ensure accessibility for inspection and testing;
(g) Provisions to facilitate the replacement of equipment that is foreseen during the lifetime of the plant;
(h) Minimization of water retention to facilitate water and condensates flowing back to the containment sumps;
(i) Design of the lower part of the containment to facilitate the collection and identification of liquid leaks.

Maintenance and accessibility

4.5. Paragraphs 4.6–4.10 provide recommendations on meeting the relevant parts of Requirements 6, 32 and 81, and para. 5.15, of SSR-2/1 (Rev. 1) [1]. Nuclear security recommendations on the prevention of access by non-authorized persons to systems important to safety should also be considered and implemented in an integrated manner with the recommendations for safety, as described in paras 2.15 and 3.2 and Ref. [10].

4.6. The design should take into account the potential occupational exposure due to the following:

(a) Implementing actions in the emergency operating procedures or in severe accident management guidelines:
(b) Connecting non-permanent equipment;
(c) Performing maintenance on systems operated over a long period of time after the onset of the accident.

More guidance on occupational radiation protection is provided in IAEA Safety Standards Series No. NS-G-1.13, Radiation Protection Aspects of Design for Nuclear Power Plants [19], and in IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [20].

4.7. Maintenance related factors that should be considered in the design include the following:

(a) The provision of adequate working space, shielding, lighting, air for breathing, and working and access platforms;
(b) The provision and control of adequate environmental conditions for workers;
(c) The provision of hazard signs;
(d) The provision of visual and audible alarms;
(e) The provision of communication systems.

4.8. The accessibility of both the containment and its associated systems should be considered for all operational states. The ability to ensure that radiation doses to workers remain within dose constraints will determine whether access can be allowed during power operation or whether plant shutdown is required for such access to be permitted.

4.9. If entry into the containment during power operation is envisaged, provisions should be made to ensure that the necessary radiation protection arrangements and adequate working conditions for the workers are in place.

4.10. At least one emergency escape route from the containment should be provided that can be used without compromising the integrity of the containment.

Operator actions

4.11. In the event of an accident, there should be no need for any action to be taken by the operator within a certain grace period. For any necessary manual intervention, the operator should have adequate information available, as well as sufficient time to diagnose and assess the conditions in the plant before taking any action.

Sharing of parts of the containment systems between units

4.12. Paragraphs 4.13 and 4.14 provide recommendations on meeting Requirement 33 of SSR-2/1 (Rev. 1) [1].

4.13. As stated in Requirement 33 of SSR-2/1 (Rev. 1) [1], **"Each unit of a multiple unit nuclear power plant shall have its own safety systems and shall have its own safety features for design extension conditions."** As an example of meeting this requirement, the gas treatment system, including the exhaust line operated in accident conditions, should not be shared.

4.14. Means allowing interconnections between units of a multiple unit nuclear power plant should be installed to facilitate the management of accidents not considered in the design. An example of this is connections to refill containment water storage tanks.

Ageing effects

4.15. Paragraphs 4.16 and 4.17 provide recommendations on meeting Requirement 31 of SSR-2/1 (Rev. 1) [1].

4.16. Ageing mechanisms affecting the containment and its associated systems should be identified, taken into account in the design and incorporated into an ageing management programme. The containment may be subject to several ageing phenomena, such as the corrosion of metallic components, the creep of tendons and the reduction of prestressing in prestressed containment, the reduction of resilience in elastomeric seals, the shrinkage and cracking of concrete, and the carbonization of concrete.

4.17. Provision should be made for controlling the ageing of the containment, identifying unanticipated degradation or containment behaviour, testing and inspecting components where possible, and periodically replacing items whose safety characteristics are susceptible to age related degradation. More detailed guidance is provided in SSG-48 [18].

Decommissioning

4.18. The design of a nuclear power plant is required to incorporate features to facilitate the decommissioning and dismantling of equipment and minimize the generation of radioactive waste (see Requirement 12 of SSR-2/1 (Rev. 1) [1]). More detailed recommendations on these aspects are provided in IAEA Safety Standards Series No. SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [21].

STRUCTURAL DESIGN OF CONTAINMENT

General design process

4.19. The pressures and temperatures that can be reached in all the different plant states are fundamental parameters used for the design of the containment and its associated systems. The values of these two parameters to be used for design are derived from an assessment of the containment conditions for each of the plant states, taking into account relevant assessment methodologies and rules.

4.20. The design pressure should be defined to be higher than the value of the peak pressure that would be generated by the design basis accident with the most severe mass and energy release (i.e. the peak pressure associated with design basis accidents and the margin).

4.21. The design temperature should be defined as the value of the highest temperature that would be generated by the design basis accident with the most severe mass and energy release, calculated with conservatism.

4.22. All values of pressure and temperature used in the load combinations should be determined with adequate margins to avoid cliff edge effects and should take into account the following:

(a) Uncertainties in the amounts of fluids released and in the release rates in terms of both mass and energy, including chemical energy from metal–water reactions;
(b) Structural tolerances;
(c) Uncertainties in relation to the decay heat;
(d) The heat stored in components;
(e) The heat transferred in heat exchangers;
(f) Uncertainties in the correlations of heat transfer rates;
(g) Conservative initial conditions.

4.23. Designing for a specific maximum leak rate is not a straightforward or purely quantitative process. A number of factors should be taken into account, including the limitation of stresses in accident conditions, the adequate choice of components (e.g. isolation valves), the adequate choice of sealing materials, limitation of the number of containment penetrations, and control of the construction quality. Existing operational data, experience and practices should be used to the extent possible.

4.24. The mechanical behaviour (stresses and deformations) of the containment, as tentatively determined on the basis of the design pressure and the design temperature, should be verified for all load combinations and should comply with the corresponding engineering criteria for the integrity and leaktightness of the containment.

4.25. The mechanical resistance of the containment and its associated systems should be assessed in relation to the expected range of events and their anticipated probability over the plant lifetime, including the effects of periodic tests.

4.26. In steel containments, the load bearing and leaktightness functions are fulfilled by the steel structure. The metallic structure should be protected against fires and missiles generated inside and outside the containment as a result of internal and external hazards that affect the plant.

Loads and load combinations

4.27. Loads (static and dynamic) that are foreseen (see Table 1) should be quantified and grouped in accordance with their probability of occurrence on the basis of operating experience and engineering judgement.

4.28. Loads and load combinations should be identified with account taken of:

(a) The load type (i.e. static or dynamic, global or local);
(b) Whether loads are consequential or simultaneous (e.g. pressure and temperature loads for a loss of coolant accident);
(c) Physical barriers that protect equipment from the effects of hazards;
(d) The timing of each load (to avoid the unrealistic superposition of load peaks if they cannot occur coincidently).

4.29. At the end of the analysis, the number of loads and load combinations may be reduced by grouping them appropriately. The analysis should be performed only for the most severe cases.

4.30. The steel liner of the containment (where applicable) should be able to withstand the effects of imposed loads and accommodate relative movements of the liner and the concrete of the containment without jeopardizing its leaktightness. The liner should not be credited in the structural evaluation for the resistance of the containment.

4.31. Any additional pressure load on the concrete containment due to the instantaneous temperature rise of the liner during an accident condition should be considered.

4.32. The metallic liner, structures, penetrations and isolation valves of the containment should be protected against the effects of the internal hazards or, if not, should be designed to withstand the loads.

4.33. For a containment design with double walls, the pressurization of the space between the two walls caused by a high energy piping break should be considered, unless such a break is precluded by the design.

Engineering criteria

4.34. Engineering criteria for leaktightness and integrity of the containment and its appurtenances (penetrations, isolation systems, doors and hatches), as

TABLE 1. TYPICAL SET OF LOADS ON THE CONTAINMENT TO BE CONSIDERED AT THE DESIGN STAGE

Load category	Load	Remarks
Pre-service loads	Dead	Loads associated with the masses of structures or components, including the effects of shrinkage and creep (for concrete containment)
	Live	Loads associated with, for example, component restraints
	Prestressing, creep effects	For prestressed concrete structures only
	Loads in construction and maintenance	Temporary loads due to construction equipment or the storage of major components
	Test pressure	See Section 5
	Test temperature	See Section 5
Normal or service loads	Actuation of safety relief valve	Boiling water reactors only
	Lifting of relief valve	Boiling water reactors only
	Air cleaning of safety relief valve	Boiling water reactors only
	Operating pressure	In normal operation, including transient conditions and shutdown
	Operating temperature	In normal operation, including transient conditions and shutdown
	Pipe reactions	In normal operation, including transient conditions and shutdown
	Wind	Maximum wind speed assumed to occur over the lifetime of the plant; see NS-G-1.5 [12]

TABLE 1. TYPICAL SET OF LOADS ON THE CONTAINMENT TO BE CONSIDERED AT THE DESIGN STAGE (cont.)

Load category	Load	Remarks
Normal or service loads	Environmental and site related loads	For example, snow load, buoyant forces due to the water table and extremes in atmospheric temperature
	External pressure	Loads resulting from pressure variations both inside and outside the primary containment
	Extreme wind speeds	Loads generated by extreme wind speeds (i.e. maximum wind speed associated with the site)
Loads due to extreme external events	Design basis earthquake	See NS-G-1.6 [13]
	Loads associated with extreme wind speeds	Associated missiles to be considered
	Aircraft crash	See NS-G-1.5 [12]
	External explosion	See NS-G-1.5 [12]
Loads due to accidents	Design basis accident pressure	Calculated peak pressure in a design basis accident
	Design basis accident temperature	Calculated peak temperature in a design basis accident
	Design pressure	Design basis accident pressure plus margins
	Design temperature	Design basis accident temperature plus margins (to be applied as a uniform value)
	Design basis accident pipe reactions	See NS-G-1.7 [11]
	Jet impact and/or pipe whip	See NS-G-1.7 [11]

TABLE 1. TYPICAL SET OF LOADS ON THE CONTAINMENT TO BE CONSIDERED AT THE DESIGN STAGE (cont.)

Load category	Load	Remarks
Loads due to accidents	Local effects consequential to a design basis accident	See NS-G-1.7 [11]
	Dynamic loads, associated with a design basis accident	Loads are design dependent (e.g. for a boiling water reactor: discharge line clearing loads, pool swell, condensation oscillation, discharge line ‘chugging’)
	Design extension conditions pressure	Calculated peak pressure in the most severe condition (peak and time dependent profile)
	Design extension conditions temperature	Calculated peak temperature in the most severe condition (peak and time dependent profile)
	Actuation of the depressurization system	Depressurization of the primary circuit (where applicable)
	Internal flooding	See NS-G-1.7 [11]

proposed in paras 4.35 and 4.36, should be established on the basis of stress and deformation limits for different load combinations. Meeting the criteria given by internationally recognized codes and standards provides reasonable assurance that structures and components are capable of performing their intended functions.

4.35. It should be demonstrated that the engineering criteria for structural integrity and leaktightness are met with an adequate margin to allow for uncertainties and to avoid cliff edge effects. Margins should generally be provided by the methodologies used to determine design basis accidents and design extension conditions and by the use of proven codes for determining the limiting stresses in the structures.

4.36. Design limits should be defined in accordance with the expected performance (see paras 3.46–3.50). Design margins should be provided by one or both of the following:

(a) Limiting stresses and deformations to a specific fraction of the ultimate limit for that material;
(b) Using the load factor approach (i.e. increasing the applied loads by a certain factor).

4.37. For the design of structural integrity of the containment, the following levels should be considered:

(a) Level I: Elastic range. No permanent deformation of, or damage to, the containment structure occurs. Structural integrity is ensured with large margins.
(b) Level II: Small permanent deformations. Local permanent deformations are possible. Structural integrity is ensured, although with margins smaller than those for Level I.

4.38. For the design of leaktightness, the following levels should be considered:

(a) Level I: Leaktight structure. Leakages from the containment are below the design value[6] and can be correlated with the internal pressure on the basis of analysis, experience and test results.
(b) Level II: Possible limited increase of leak rate. The leak rate may exceed the design value, but the leaktightness can be adequately estimated on the basis of analysis, experience and test results.

4.39. The detailed load combinations are design dependent. Table 2 presents a minimum set of recommended load combinations and engineering criteria for a typical containment of a pressurized water reactor.

4.40. To provide margins, the loads resulting from an SL-2 earthquake[7] and design basis accidents should be combined using an adequate statistical combination of the loads (e.g. using the square root of the sum of the squares), even though one

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[6] In this context, 'design value' is the value of the leakage rate established as a target of the design and used in the safety analysis to determine the radioactive releases under design pressure and design temperature.

[7] An 'SL-2 earthquake' denotes the level of ground motion associated with the maximum earthquake to be considered for design, often denoted as the 'safe shutdown earthquake'.

TABLE 2. LOAD COMBINATIONS AND ENGINEERING CRITERIA FOR A TYPICAL CONTAINMENT OF A PRESSURIZED WATER REACTOR

Load description	Design	Test	Normal operation	Normal operation plus extreme wind speed	SL-2 earthquake [a]	External pressure	DBA	SL-2 plus DBA	Aircraft crash	Fire	External explosion	DEC without significant fuel damage	DEC with core melting
Dead	x [b]	x	x	x	x	x	x	x	x	x	x	x	x
Live	x	x	x	x	x	x	x	x	x	x	x	x	x
Prestressing (if applicable)	x	x	x	x	x	x	x	x	x	x	x	x	x
Test pressure		x											
Test temperature		x											
Sustained loads		x	x	x	x	x	x	x	x	x	x	x	x
Operating loads			x	x	x	x	x	x	x	x	x	x	x
Operating temperature			x	x	x	x			x		x		

TABLE 2. LOAD COMBINATIONS AND ENGINEERING CRITERIA FOR A TYPICAL CONTAINMENT OF A PRESSURIZED WATER REACTOR (cont.)

Load description	Design	Test	Normal operation	Normal operation plus extreme wind speed	SL-2 earthquake [a]	External pressure	DBA	SL-2 plus DBA	Aircraft crash	Fire	External explosion	DEC without significant fuel damage	DEC with core melting
Pipe reactions			x	x	x	x			x	x	x		
Extreme wind				x									
External pressure						x							
SL-2 earthquake					x			x					
Design pressure	x												
Design temperature	x												
DBA pressure							x	x					
DBA temperature							x	x					

TABLE 2. LOAD COMBINATIONS AND ENGINEERING CRITERIA FOR A TYPICAL CONTAINMENT OF A PRESSURIZED WATER REACTOR (cont.)

Load description	Design	Test	Normal operation	Normal operation plus extreme wind speed	SL-2 earthquake [a]	External pressure	DBA	SL-2 plus DBA	Aircraft crash	Fire	External explosion	DEC without significant fuel damage	DEC with core melting
DBA pipe reactions							x	x					
Aircraft crash									x				
Fire										x			
External explosion											x		
DEC with core melting (pressure)													x
DEC with core melting (temperature)													x
DEC without significant fuel damage (pressure)												x	

TABLE 2. LOAD COMBINATIONS AND ENGINEERING CRITERIA FOR A TYPICAL CONTAINMENT OF A PRESSURIZED WATER REACTOR (cont.)

Load description	Design	Test	Normal operation	Normal operation plus extreme wind speed	SL-2 earthquake [a]	External pressure	DBA	SL-2 plus DBA	Aircraft crash	Fire	External explosion	DEC without significant fuel damage	DEC with core melting
DEC without significant fuel damage (temperature)												x	
Engineering criteria for steel containment:													
Structural integrity	I [c]	I	I	I	II [d]	II	I	II	n.a. [e]	II	n.a.	II	II
Leaktightness	I	I	I	I	n.a.	II	I	II	n.a.	II	n.a.	I	II
Engineering criteria for prestressed containment:													
Structural integrity	I	I	I	I	II	n.a.	I	II	II	II	II	II	II
Leaktightness	I	I	I	I	n.a.	n.a.	I	n.a.	n.a.	n.a.	n.a.	I	II

TABLE 2. LOAD COMBINATIONS AND ENGINEERING CRITERIA FOR A TYPICAL CONTAINMENT OF A PRESSURIZED WATER REACTOR (cont.)

Load description	Design	Test	Normal operation	Normal operation plus extreme wind speed	SL-2 earthquake [a]	External pressure	DBA	SL-2 plus DBA	Aircraft crash	Fire	External explosion	DEC without significant fuel damage	DEC with core melting
Engineering criteria for a liner on prestressed concrete wall	I	I	I	I	II	n.a.	I	n.a.	n.a.	II	n.a.	II	II

Note: DBA: design basis accident; DEC: design extension condition; SL-2: seismic level 2.

[a] Level of ground motion associated with the maximum earthquake to be considered for design, often denoted as the ‘safe shutdown earthquake’.

[b] x: Load should be considered.

[c] I: Level I criteria should be applied.

[d] II: Level II criteria should be applied.

[e] n.a.: not applicable.

cannot realistically be a consequence of the other since the pressure boundary of the reactor coolant system is designed to withstand seismic loads caused by an SL-2 earthquake (see NS-G-1.6 [13]).

Local stresses and fatigue

4.41. Local stresses — including those at welding regions, near supports and at regions with changing geometry — and their effects on the mechanical performance of structures, including leak rates, should be evaluated.

4.42. For prestressed concrete containments, particular attention should be paid to the following:

(a) Areas of low prestressing, such as areas surrounding large penetrations and transition zones between the cylinder and the containment basemat;
(b) The concentration of stresses near penetrations and near anchorages of the tendons;
(c) The tensioning sequence during construction.

4.43. For containments provided with a metallic liner, the zones of anchorage of the liner to the concrete and the connections of the liner to other metallic structures, such as penetrations, are also critical areas. Local effects of stress in these zones should be analysed and taken into account.

Ultimate capability and failure mode

4.44. To determine the ultimate load bearing capacity and confinement capacity, a global evaluation of the structural behaviour of the containment should be undertaken. This evaluation should consider static loads (pressure, temperature and actions of pipes) and dynamic loads (seismic) and identify the most limiting parts so as to evaluate margins.

4.45. Local effects, thermal gradients and design details should also be considered so as to identify possible mechanisms for large leaks. In this regard, special attention should be paid to the behaviour of piping penetrations, soft sealing materials, electrical penetrations and structural singularities.

4.46. Various failure modes, such as liner tearing, penetration failures, rebar failure, local concrete failure and tendon failures, should be analysed. To the extent possible, a failure should not be catastrophic and should not cause additional damage to systems and components for retaining radioactive material.

STRUCTURAL DESIGN OF STRUCTURES WITHIN THE CONTAINMENT

4.47. Consideration should be given to the possibility of large releases of mass and energy inside the containment and the need for the internal structures to withstand the pressure differentials that could arise between different compartments. For each compartment, the most unfavourable location for a break should be considered. Openings between compartments should be considered by means of a conservative approach at the design stage and should be verified to be free of unintended obstructions after construction has been completed.

4.48. Consideration should be given to the need for the internal structures to withstand the loads associated with accident conditions, and so to withstand the dynamic loads that are caused by high energy discharges or pipe breaks (e.g. water flowing from the discharge line of the safety valves and the relief valves into the suppression pool, the swelling of the pool water, the oscillation of condensate water, chugging and other relevant hydraulic phenomena).

4.49. In the case of design extension conditions with core melting, the loads on the structures inside the containment depend on the strategy to cope with the molten core that is adopted in the specific design.

4.50. The load combinations and engineering criteria for leaktightness and integrity given in Table 2 should be met in the event of design extension conditions with core melting, and conditions for a containment boundary or basemat melt-through should be practically eliminated (see para. 3.68) for either of the design strategies for retention of the molten core (i.e. in-vessel retention or ex-vessel retention).

In-vessel retention strategy

4.51. In this strategy, the heat from the molten core is removed through the wall of the reactor pressure vessel. This requires the reactor cavity to be flooded to enable external cooling of the reactor pressure vessel. Mechanical and thermal loads in the walls of the cavity should be considered. Features should be included to remove the heat from the cavity and avoid pressurization of the cavity and the containment.

4.52. The structures of the cavity and the systems used for the in-vessel retention strategy should be considered as items ultimately necessary to avoid large releases; consequently, they should be designed such that the design margins are adequate to deal with seismic loads exceeding an SL-2 earthquake (see para. 5.21A of SSR-2/1 (Rev. 1) [1]).

Ex-vessel retention strategy

4.53. In this strategy, the containment should be equipped with an ex-vessel retention structure dedicated to containing and cooling the molten core outside the reactor pressure vessel.

4.54. The ex-vessel retention structure should be designed to minimize the production of combustible gases from the interaction between concrete and the molten core.

4.55. The structures and the cooling system necessary for the ex-vessel retention strategy should be appropriate and designed for stabilizing and confining the molten core inside.

4.56. The structures, components and materials used for the ex-vessel retention strategy should be appropriate to withstand the different loads and effects caused by the ingress of the molten core into the different elements of the ex-vessel retention structure.

4.57. The structures and components necessary for the ex-vessel retention strategy should be considered as items ultimately necessary to avoid large releases; consequently, they should be designed such that the design margins are adequate to deal with seismic loads exceeding an SL-2 earthquake (see para. 5.21A of SSR-2/1 (Rev. 1) [1]).

STRUCTURAL DESIGN OF SYSTEMS

4.58. For containment systems, a set of representative loads and load combinations, as well as a set of adequate engineering criteria, should be established by an approach similar to that used for the structural design of the containment, with account taken of all the relevant accident conditions.

MASS AND ENERGY RELEASE AND MANAGEMENT

4.59. ‘Mass and energy release and management’ is a term used to describe the management of those design features of the containment that affect the energy balance within the containment and thereby play a part in maintaining pressure and temperature within acceptable limits.

Control of pressure and temperature in operational states

4.60. During normal plant operation, a ventilation system should be operated to maintain the pressure and temperature inside the containment within the operational limits and conditions specified for the operation of the nuclear power plant.

Control of pressure and temperature in accident conditions

4.61. The design performance of the systems for mass and energy release and management should be established so that in the event of an accident, the pressure and temperature inside the containment can be controlled within the specified limits and a stable state can be reached, with the containment depressurized, within a reasonable period of time (typically a few days) after the onset of the accident.

4.62. The design of these structures, systems and components should comply with the specifications relevant for the plant state category for which they are designed to operate. The applicable recommendations for their design are provided in Section 3 of this Safety Guide.

4.63. Strategies for pressure and temperature control in accident conditions rely on the use of inherent safety features, active or passive safety systems or safety features, or a combination of these design options. Typical design options are described in paras 4.64–4.89.

Inherent mass and energy release and management features (containment with a large dry space)

4.64. The free volume of the space within the containment is the primary physical parameter determining peak pressures after postulated pipe rupture events. The free volume can thus be used as an inherent safety feature designed to accommodate a large energy and mass release inside the containment. If the volume of the containment is subdivided into compartments, collapsing panels or louvres should be implemented. These collapsing panels or louvres should be designed to open quickly in the event of energy released at a predetermined pressure so as to achieve fast equalization of the pressures in the various compartments and utilize the full free volume of the containment.

4.65. The containment and its internal structures, as well as the water stored within the containment, act as a passive heat sink. In the postulated conditions of a pipe rupture accident, the heat transfer rate and heat capacity of structures and components of structures are important parameters in determining pressures and temperatures. The primary mechanism for heat transfer is the condensation of steam on exposed surfaces, and the thermal conductivity of the structure plays an important part in determining the rate of heat transfer. All conditions that could affect the transfer of heat to the structures, such as the effects of coatings or gaps, should be considered in a conservative manner in the design, and adequate margins should be applied.

Spray systems

4.66. With regard to mass and energy release and management, a spray system should be designed to achieve the following:

(a) Limit the peak pressure and the duration of high pressures inside the containment in accident conditions for containment with a large dry space;
(b) Limit the duration of high pressure inside the 'dry well' and 'wet well' (see para. 4.71) for containment with a supression pool system;
(c) Control the temperature inside the dry well for containment with a supression pool system.

4.67. The spray system should be designed so that a major fraction of the free volume of the containment, into which the steam could escape in an accident, can be sprayed with water.

4.68. The spray headers and nozzles should be designed to provide an even distribution of water droplets, which should be small enough to reach thermal equilibrium with the atmosphere quickly during their fall.

4.69. The initial source of water for the containment spray system is usually a large storage tank or the suppression pool. Later, the spray system may operate in a recirculation mode and take water from appropriate collection points in the containment sump or the suppression pool.

4.70. For a spray system designed to operate in a recirculation mode, the spray nozzles should be designed to prevent clogging by pieces of debris that can reach them through the intake screens and filters.

Pressure suppression pool systems

4.71. Containments that are designed to have a suppression pool system are divided into two separate compartments: the dry well and the wet well. The two compartments are normally isolated from one another. When the pressure in the dry well is sufficiently higher than the pressure in the wet well, steam and gases flow from the dry well to the wet well and the steam condenses into the pool of water. In some designs, interaction between the dry well and the wet well can also occur if the pressure in the wet well is higher than the pressure in the dry well. In some containment designs, the suppression pools are also used to collect the steam discharged from the safety valves or the relief valves or to provide water for recirculation in the emergency core cooling system, decay heat removal system and containment spray system. Complex hydraulic and pressure transients occur when steam and gases are vented into the suppression pool water.

4.72. With regard to mass and energy release and management, the suppression pool should be designed such that the design pressure of both the dry well and the wet well is not exceeded in the event of a design basis accident. In practice, the following should be taken into account:

(a) The vent flow area between the dry well and the suppression pool should be sized to limit the maximum pressure during blowdown.
(b) The amount of water in the suppression pool should be sufficient to condense all the steam released during design basis accidents (e.g. in the event of a loss of coolant accident) and to allow for the absorption of residual and latent heat from the reactor for a sufficient time period until the normal, emergency or backup residual heat removal systems are capable of restoring a heat balance.

4.73. The design of the dry well and wet well and the connection features should be such that the hydraulic responses and the dynamic loads can be reliably determined by analysis and tests.

4.74. The hydraulic response of, and loads imposed on, the pressure suppression pool in the different plant states should be determined and considered in the design.

4.75. The structural design of the pressure suppression pool system should be such as to ensure that the pool, as well as the containment as a whole, and other associated systems remain functional in all plant states, including all postulated accident conditions.

4.76. The pressure suppression pool system should be designed in such a way that the pathway for steam and gases from the dry well to the wet well in the event of a postulated accident condition is through submerged vents in the wet well water pool.

4.77. Any leakage between the dry well and the wet well that bypasses the submerged venting lines should be minimized and should be taken into account in the design.

4.78. The use of the water inventory in the pressure suppression pool system for other functions should not impair the performance of its main function of providing a means of controlling the pressure in the dry well in case of accident conditions.

4.79. The dry well should be designed to withstand, or should be protected (e.g. by automatic vacuum breaker valves) from excessive underpressure caused by steam condensation inside the dry well operation of the spray system, either on purpose or inadvertently.

Containment heat removal system

4.80. Containment heat removal systems should be designed to remove heat from the containment and to transfer heat to a cooling chain or directly to the ultimate heat sink (e.g. the atmosphere, the sea, a river).

4.81. Piping crossing the containment wall should be considered an extension of the containment and should be subject to specifications for structural integrity and leaktightness that are comparable to those applied to the containment structure itself.

Systems operating in a recirculation mode in accident conditions

4.82. A minimum and adequate net pump suction head should be available to the recirculation pumps under any accident conditions for which the operation of the pumps is necessary. The minimum net pump suction should be calculated taking into account the potential accumulation of debris on the surface of the strainer filters.

4.83. Suction devices should be designed to minimize cavitation and prevent the ingress of foreign material (e.g. thermal insulation) that could block or damage the recirculation system.

4.84. To avoid the clogging of sump screens or strainer filters, special care should be taken in the design of piping, component insulation and the intake sump screens or strainer filters themselves. In addition, consideration should be given to chemical effects as determined by the sump or suppression pool water chemistry and temperature, as well as to the corrosion or erosion of metallic components and their interaction with the debris. The material used inside the containment (e.g. thermal insulation materials, paints) should also be carefully considered. The design should avoid certain combinations of these materials that could produce increased clogging at sump screens or strainer filters: see paras 4.195–4.202.

4.85. With regard to core cooling, the effects of debris bypassing the sump screens or strainer filters on the potential for blockage of flow channels in fuel assemblies should be taken into account.

4.86. Piping crossing the containment walls should be equipped with containment isolation devices and devices necessary to isolate leaks in the external recirculation loops to maintain sufficient water inventory for cooling. Non-isolatable leakage (e.g. between the containment penetration and the isolation valve) should be prevented by design (e.g. by the provision of a guard pipe).

Containment heat removal systems operating with passive features

4.87. For containment with a steel shell, heat released in the containment under accident conditions can be removed passively through the steel shell. A secondary external structure that is designed to remove heat by providing a natural circulation path for air (the chimney effect) is also necessary.

4.88. Heat can also be removed from the containment by the installation of a number of heat exchangers on the inner walls of the containment that transfer heat outside the containment by natural circulation to passive cooling condensers.

4.89. Where passive containment cooling is adopted, the following aspects should be considered:

(a) The design should ensure that the area of the cooling surface is adequate to transfer the heat generated in the containment and to cool down the atmosphere and the structures inside the containment. The heat transfer coefficient should be conservatively determined.
(b) The necessary natural circulation within the containment and also that to the outside heat sink should be ensured for all relevant plant states and for

any environmental conditions (e.g. atmosphere temperature, humidity) identified in the site evaluation for which such passive transfer is necessary.
(c) The possibility of freezing outside conditions should be considered for all plant states.
(d) A thorough analysis should be conducted to identify and eliminate possible harmful effects and failure modes to achieve a high degree of confidence that the safety functions will be fulfilled.

CONTROL AND LIMITATION OF RADIOACTIVE RELEASES

4.90. The containment and its associated systems are required to be designed to meet the objectives for preventing and limiting the radioactive releases specified for the different plant states, in accordance with Requirement 55 of SSR-2/1 (Rev. 1) [1].

4.91. Compliance with the relevant limits for radioactive releases should be demonstrated by only crediting the provisions designed for the relevant plant state. The demonstration should be conducted using models and analysis rules that are applicable to the plant state category.

4.92. Design provisions necessary to minimize radiation exposures and radioactive releases should take account of the different source terms that are specific to each plant state (in terms of the magnitude of the radioactive release, the isotopic composition of radionuclides and their physicochemical forms).

4.93. An assessment of potential radioactive releases from the containment should be made for design basis accidents and for design extension conditions to identify any potential weaknesses with regard to the leaktightness of the containment and to determine ways to eliminate any such weaknesses.

Containment source term

4.94. To design the overall containment performance, and in particular the measures for radionuclide management, the amount and isotopic composition of the radionuclides postulated to be released from the containment (i.e. the source term) should be estimated for the various accident conditions to be considered.

4.95. For design basis accidents, the source term should be estimated by means of a conservative analysis of the expected behaviour of the core and of the safety systems. The initial conditions for the relevant parameters (e.g. for the inventory of radionuclides in systems and for leak rates) should be based on less favourable values within the framework of the operational limits and conditions specified for the operation of the nuclear power plant.

4.96. The anticipated evolution of the physicochemical forms of the radionuclides in the containment should be assessed, with account taken of the latest knowledge (e.g. it is known that certain paints enhance the production of organic iodine).

4.97. Once iodine is trapped in water pools inside the containment, it could volatilize again in the medium to long term if appropriate pH conditions are not maintained. Therefore, all conditions that could change the pH of the water pools during an accident should be assessed and, if necessary, a means of keeping the pH of the water pools alkaline should be provided.

Leaktightness of the containment

4.98. The containment and its associated systems should be designed to minimize leaks and avoid, to the extent possible, the creation of unfiltered leak paths to the environment.

4.99. An effective way to restrict radioactive releases to the environment is to maintain the leak rate below conservative specified limits throughout the plant operating lifetime.[8] Leak rates should be small enough to ensure that the objectives stated in Requirement 55 of SSR-2/1 (Rev. 1) [1] are met.

4.100. At the design stage, a target leak rate should be set that is well below the safety limit leak rate (i.e. well below the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions). This margin should be established to reduce the likelihood that unforeseen modifications made at the design stage or at the construction stage could cause an actual leak rate to approach the safety limit leak rate.

4.101. To limit the number of leak paths, the number of penetrations of the containment wall should be optimized, as indicated in para. 4.4(f). The external extensions of the penetrations should be installed in a confined building, at least

[8] Examples of such limits that are applied in States are 0.25–0.5% overall leakage of the contained mass of free gas and steam per day at design pressure for steel containments or concrete containments with a steel liner, and 1.0–1.5% per day overall leakage for prestressed concrete containments without a steel liner.

until the first isolation valve, to collect and filter any leaks before a radioactive release occurs.

4.102. Leak rates of isolation devices, air locks and penetrations should be specified, with account taken of their importance to safety and to the integral leaktightness of the containment.

4.103. The design should include appropriate isolation devices to ensure the isolation of the containment in the event of an accident, as described in paras 4.154–4.166.

Secondary confinement building

4.104. Some designs for nuclear power plants include a secondary confinement building, which is an arrangement in which the primary containment is completely or partially enclosed within a secondary envelope. The purpose of the secondary envelope in such designs is not to take over the functions of the primary containment should it fail, but to allow for the potential collection of leaks from the primary containment and for a filtered release via the vent stack. When such a design option is implemented, the secondary confinement structure can also be designed as the shielding structure of the containment.

4.105. When a secondary confinement building is provided, direct leaks (i.e. leak paths from the containment directly to the outside) should be prevented to the extent possible.

4.106. When employing a partial secondary confinement building (i.e. one that does not completely enclose the primary containment), the envelope should enclose those areas of the primary containment that are more prone to leakage (e.g. the penetration areas).

4.107. Criteria should be set for the control of direct leaks and for the leaktightness of the secondary confinement envelope. It should be verified periodically by testing that these criteria are being met.

4.108. Systems associated with the secondary confinement building should be designed to collect, filter and discharge gases containing radioactive substances that have leaked from the primary containment in accident conditions, and to pump leaked liquids back into the primary containment.

4.109. To maximize the efficiency of the secondary confinement building, a filtered ventilation system should be provided and designed to maintain a negative gauge pressure in design basis accidents. For design extension conditions, if a negative gauge pressure cannot be achieved and maintained in the confinement volume, the resulting unfiltered leakage to the environment should be taken into account in the calculations of the radiological consequences.

4.110. The confinement volume should be kept at a negative gauge pressure in normal operation to enable the leaktightness of the secondary confinement building to be monitored.

Containment bypass

4.111. Containment bypass events arise when primary coolant and any accompanying fission products escape to the outside atmosphere without being processed.

4.112. Appropriate design provisions should be taken to demonstrate that conditions involving a containment bypass and leading to an early radioactive release or a large radioactive release have been practically eliminated.

4.113. Any piping outside the containment that circulates highly contaminated liquids or gases should be designed to be leaktight under accident conditions. Loads and process conditions should be properly considered and combined.

4.114. Conditions for the opening of the containment (e.g. equipment hatch, fuel transfer tube) should be specified and should be adequate to prevent accidents with a release of activity to the atmosphere of the containment from arising. Alternatively, the containment should be capable of being quickly closed.

4.115. Possible paths for loss of coolant accidents in interfacing systems should be prevented as far as possible, either by relocating the system in the containment or by increasing the design pressure of the low pressure system to a value that is above the pressure of the reactor coolant system. For any remaining possible paths for loss of coolant accidents in interfacing systems, reliable provisions for preventing or stopping leaks outside the containment should be implemented.

4.116. In pressurized water reactors, a steam generator tube rupture is considered as a potential containment bypass event that could lead to a radioactive release. Preventive design features should be implemented to ensure that such events have a low frequency of occurrence. The design of the plant should facilitate a fast isolation of the affected steam generator in order to minimize the radioactive release, which should not exceed the release limit defined for the relevant plant state.

4.117. Many containment designs include systems to recirculate water from collection points inside the containment, either directly or through heat exchangers, for reinjection into the reactor vessel or for the long term operation of the spray or heat removal systems in accident conditions. Parts of these recirculation systems could be located outside the containment, giving rise to a potential for radioactive release from pumps, valves or heat exchangers outside the containment. Where a design of this type is used, provisions should be made to (i) minimize any uncontrolled radioactive release to the environment that results from such leakage, (ii) periodically test the leaktightness of the various components and (iii) detect and isolate accidental leaks by qualified means.

Reduction of radioactive material in the containment atmosphere

General

4.118. As an application of the defence in depth concept, and in addition to the measures taken to ensure the leaktightness of the containment, measures should be taken to reduce the inventory of radioactive material in the containment atmosphere.

4.119. In general, a single system is not sufficient to reduce the concentration of airborne radioactive material, and multiple systems should be employed. Methods used for the reduction of airborne radioactive material in water cooled reactors (of existing and new designs) are as follows:

(a) Deposition on surfaces;
(b) Containment spray systems;
(c) Pressure suppression pools;
(d) Ventilation and venting systems.

4.120. Active systems for the reduction of the concentrations of airborne radioactive material should be capable of being tested while they are in standby mode during normal operation of the plant.

Deposition on surfaces

4.121. The containment and its internal components provide the first mechanisms for the removal of airborne radioactive material, since they present a large surface area for deposition. The plate-out and desorption factors ascribed to the containment structure should be conservatively based on the best available knowledge of the deposition of radionuclides on surfaces. The surfaces of the containment and its internal structures should be capable of being decontaminated as far as is possible.

Containment spray system

4.122. In the context of the control of radioactive releases, the containment spray system is intended to reduce the amount of airborne radioactive material by removing it from the containment atmosphere and retaining it in the water of the containment sump or the suppression pool. This serves to limit radiological consequences resulting from leakages from the containment to the atmosphere.

4.123. Important parameters that should be considered in the design of the containment spray system include spray coverage, spray drop size, drop residence time and the chemical composition of the spray medium. In addition, the following recommendations should be noted:

(a) Chemicals should typically be added to the spray water to enhance the removal of radionuclides from the atmosphere. Radioiodine is of particular importance because of its potential consequences in terms of individual doses. The system for the addition of chemicals should be designed to maximize the dissolution of radioiodine and maintain the sump chemistry or the suppression pool chemistry such that radioiodine will not be released from solution in the long term following an accident.
(b) Any chemicals added to the spray water should be non-corrosive with regard to the materials present in the containment, both in the short term and in the long term after an accident. Corrosion might not only reduce the strength of vital structural components and impair the operation of safety systems but might also generate combustible gases or other undesirable substances.

Pressure suppression pools

4.124. Water pools or tanks through which the containment atmosphere is bubbled for steam condensation should be considered a valuable means for the removal of radioactive material. However, care should be taken in evaluating the efficiency of such a process, since it is dependent on the thermodynamic conditions of water and steam. For example, the degree of subcooling of the water and the consequent efficiency of steam condensation have a significant effect on the scrubbing efficiency of a suppression pool.

Ventilation and venting systems

4.125. Where ventilation systems are used for cleaning exhaust air to reduce occupational exposure and public exposure in accident conditions, the filters should be designed and maintained so as to preclude any overloading of the filters with pollutants before their use in relation to an accident.

4.126. The ventilation system should, if necessary, be provided with equipment (e.g. moisture separators and preheaters before the filters) to prevent the temperature from dropping below the dew point at the air filter inlet.

4.127. The efficiency of the absorption material in iodine filters should be demonstrated in laboratory tests under simulated accident conditions, as deemed appropriate. Provisions should be made to test periodically the filter system in situ.

4.128. Ventilation systems are often used to collect, filter and discharge air from a secondary confinement building, which could become contaminated with airborne radioactive material in accident conditions as a result of leakage from the containment. For such cases, the recommendations in paras 4.154–4.166 apply.

4.129. Where a containment venting system is installed, the system should be designed to minimize the radioactive release to the environment. The system design could include a filtering system such as sand, multiventuri scrubber systems, high efficiency particulate air or charcoal filters, or a combination of these. High efficiency particulate air, sand or charcoal filters may not be necessary if the released gas flow is scrubbed in a water pool.

4.130. Noble gases cannot be filtered out, but consideration should be given to the use of systems to delay their release until further radioactive decay has occurred.

MANAGEMENT OF COMBUSTIBLE GASES

4.131. Paragraphs 4.132–4.150 provide recommendations on meeting Requirement 58 of SSR-2/1 (Rev. 1) [1].

Generation of combustible gases

4.132. Sources for the potential release of combustible gases and the associated threats to the containment and to systems necessary for the mitigation of the relevant accident conditions posed by such gases should be identified for the different plant states.

4.133. The sources of combustible gases should be identified with account taken of the following phenomena:

(a) Radiolysis of the water in the core;
(b) Radiolysis of the water in the sump or in the suppression pool;
(c) Metal–water reactions of core components and reactor pressure vessel internals;
(d) Chemical reactions with materials in the containment;
(e) Degassing of hydrogen dissolved in the primary coolant;
(f) Releases from the hydrogen tanks used for control of the primary coolant chemistry;
(g) Interactions between the molten core and concrete producing hydrogen and carbon monoxide.

4.134. The generation of combustible gases and the generated volume as a function of time should be calculated for design basis accidents and design extension conditions. The uncertainties in the various mechanisms for the generation of gases should be taken into account by the use of adequate margins for each mechanism. For design extension conditions with core melting, the uncertainties relating to hydrogen production are linked to phenomena such as flooding of a partially damaged core at high temperatures, the late phase of core degradation, the slumping of molten core material into residual water in the lower head of the reactor pressure vessel, and the long term interactions between molten core material and concrete.

4.135. The possible effects of the combustion of gases on the containment and on systems necessary for the mitigation of the relevant accident conditions should be evaluated. Such effects should be prevented to the extent possible, or should be limited, or the conditions for combustion to occur should be practically eliminated when it is not possible to mitigate these effects.

Threats due to combustible gases in design extension conditions with core melting

4.136. Threats to the containment are dependent on the reactor technology and the design but usually are caused by high pressure and thermal loads arising from the production of large quantities of non-condensable gases and by various regimes of combustion of the combustible gases. Both of these causes should be considered, and their effects on the containment and on systems necessary for the mitigation of such conditions should be assessed.

4.137. Even if it can be demonstrated that conditions for gas mixture flammability are not met (e.g. in cases of low hydrogen concentration, high steam concentration or low oxygen concentration), overpressurization due to non-condensable gases is nevertheless relevant. For example, for inert containment, the probability of hydrogen combustion is low because of the presence of inert gas and the absence of oxygen in normal power operation; for such a type of containment, the primary threat is the fast overpressurization caused by a large production of non-condensable gases in a small volume.

4.138. The global and local effects of combustion (static pressure loads, dynamic pressure loads and thermal loads) on the containment and the safety features necessary for the mitigation of the consequences of design extension conditions with core melting should be considered.

4.139. The general approach to designing the performance and efficiency of the various means necessary for the management of combustible gases should be based on gas concentration limits, taking into account the following recommendations:

(a) Hydrogen combustion should be postulated when conditions for flammability are exceeded (e.g. a hydrogen concentration greater than 4% by volume in dry air).
(b) As long as conditions for flame acceleration phenomena and for high dynamic pressure loads are not reached, the adiabatic–isochoric complete combustion pressure curve calculated for all the hydrogen combustions at a slow flame regime should be used to define the global and local pressure bounding loads.
(c) Conditions for flame acceleration phenomena that could lead to a deflagration to detonation transition or to a detonation should be prevented to the extent possible in areas where hydrogen accumulation is possible. For areas where such conditions could possibly be reached, detailed analyses and calculations should be conducted with the aim of demonstrating that detonation, deflagration to detonation transition, or a fast combustion regime would not lead to a challenge to the structural integrity of the containment or its associated systems.
(d) To reach safe conditions inside the containment, the performance and efficiency of the means of removing combustible gases should be designed to reduce the average concentration of such gases in the free volume of the

containment below the gas flammability limit in dry air (e.g. below 4% for hydrogen).

4.140. Calculations and analyses should cover gas generation, gas production time history, and gas concentration distribution to assess the possibility of occurrence of the various regimes of combustion: combustion at a slow flame regime, a fast combustion regime with flame acceleration, or a deflagration to detonation transition regime.

4.141. The threat of hydrogen combustion while the steam concentration is decreasing should be understood and considered with regard to the operation of the containment heat removal system.

4.142. Leaks and releases of combustible gases from the containment should also be taken into account when evaluating the threat of combustion.

Measures for the mitigation of hydrogen combustion and for the prevention of hydrogen combustion challenging the containment integrity

4.143. A variety of measures, such as the selection of materials, free space inside the containment, removal, transport, homogenization and venting, should be taken to minimize hydrogen production, to mitigate hydrogen combustion and to practically eliminate combustion regimes that could challenge the containment integrity.

4.144. Where means are necessary to limit and remove hydrogen, the means necessary to limit hydrogen concentration in the event of design extension conditions with core melting should be designed to be independent of those necessary for design basis accidents. The performance and efficiency of the measures described in para. 4.143 should be designed to ensure compliance with the concentration limits indicated in para. 4.139. In addition, the performance and the layout of these measures should be such that the containment integrity and leaktightness are maintained within the limits considered in the safety demonstration.

Removal

4.145. An adequate number of passive means (e.g. autocatalytic recombiners) and active means (e.g. igniters) should be provided. These passive and active means should be suitably distributed inside the containment with regard to their efficacy in reducing the concentration of combustible gases (e.g. in the vicinity of the release location, near expected convection flow paths between inner containment rooms, in the dome area as well as the containment periphery, and at different heights in large rooms).

4.146. The number and positioning of recombiners or igniters should be justified on the basis of adequately detailed analyses of the distribution of combustible gases.

4.147. Layout provisions should be implemented so that thermal loads (due to combustion flames or hot off-gases from recombiners) are not capable of damaging the containment liner (or the containment steel shell), the containment penetrations, or any components and cables necessary for the mitigation and monitoring of accidents with core melting.

Homogenization

4.148. The design should either incorporate active means (e.g. sprays and mixing fans qualified for operation in a combustible gas mixture) or should facilitate natural circulation throughout the containment to enhance hydrogen homogenization of the atmosphere within and between compartments, by ensuring the presence of adequate openings, and to prevent dead-end zones to the extent possible.

Inerting

4.149. One possible way to avoid combustion is to maintain an inert atmosphere (usually with nitrogen) inside the containment during reactor operation. This approach is mainly applicable to small containments.

4.150. The ingress of oxygen into the inert containment atmosphere should be prevented, for example by maintaining an overpressure in the containment, by limiting depressurization or by the provision of an additional nitrogen supply.

MECHANICAL FEATURES OF THE CONTAINMENT

4.151. The mechanical features of the containment comprise the mechanical components of the outermost barrier and the mechanical parts of the extensions of this barrier (i.e. piping, valves, ducts and penetrations). Together with the containment structure, these features form the containment envelope.

4.152. The leaktightness criteria for mechanical features of the containment and its extensions should be consistent with the assumptions used in the radiological analyses for accident conditions.

Provisions for containment isolation of piping and ducting systems

4.153. Paragraphs 4.154–4.166 provide recommendations on meeting Requirement 56 of SSR-2/1 (Rev. 1) [1].

4.154. Each line that penetrates the containment that is not part of a closed loop[9] and that either (a) directly communicates with the reactor coolant during normal operation or in accident conditions or (b) directly communicates with the containment atmosphere during normal operation or in accident conditions should be provided with two isolation valves in series. Each valve should normally be closed or should have provisions to close automatically. If the line communicates directly with the reactor coolant or the containment atmosphere, one valve should be provided inside the containment and one valve outside. Each valve should be reliably and independently actuated. Isolation valves should be located as close as practicable to the containment.

4.155. Loops that are closed either inside or outside the containment should have at least one isolation valve outside the containment at each penetration. This valve should be an automatic valve, a normally closed valve or a remotely operated valve.[10] If the failure of a closed loop is assumed as a postulated initiating event or as a consequence of a postulated initiating event, this recommendation will apply to each line of the closed loop.

4.156. Loops that are closed both inside and outside the containment envelope should have at least one isolation valve, an automatic valve, a normally closed

[9] A ‘closed loop’ is a piping or ducting system that penetrates the containment envelope and that is designed to form a closed circuit either inside or outside the containment, or inside and outside the containment in operational states and in accident conditions.

[10] An ‘automatic valve’ is a valve or damper that can be actuated either by the protection system or by other instrumentation and control without action by the operator, or by the process medium itself. For example, certain types of check valve are considered automatic valves. A ‘normally closed valve’ is a valve that is closed under active administrative control (e.g. being locked closed or continuously monitored to show that the valve is in the closed position) except for intermittent opening for specific purposes such as monitoring, testing or sampling. A ‘remotely operated valve’ is a valve or damper that can be actuated by an operator from the control room and in some cases from the supplementary control points.

valve or a remotely operated valve located outside and as close as practicable to the containment at each penetration.

4.157. Small dead-ended instrumentation lines that penetrate the containment should have at least one isolation valve outside the containment.

4.158. Containment isolation valves for instrumentation lines that are closed (i.e. not in communication with the atmosphere) are not necessary, provided that the lines are designed to withstand the accident conditions for which confinement is necessary. The rooms where these lines emerge should be equipped with a filtration–ventilation system to maintain subatmospheric pressure. Such rooms and the equipment within them should be designed to withstand increased levels of temperature and humidity due to possible leakage from these lines.

4.159. The need for an automatic isolation of the containment in accident conditions should not prevent the systems necessary to mitigate those accidents from accomplishing their intended functions.

4.160. Overpressure protection should be provided for closed systems that penetrate the containment and for isolated parts of piping that might be overpressurized by an increase of the temperature inside the containment atmosphere in accident conditions.

4.161. The extensions of the containment should be designed and constructed to levels of performance that are at least equivalent to those of the containment itself.

4.162. For specific operational conditions (e.g. conditions with an open containment or inhibited containment automatic isolation), the risk to safety should be assessed and temporary provisions should be implemented as necessary to ensure the containment isolation function can be accomplished in a timely manner.

4.163. Particular consideration should be given to the containment isolation features of the following systems that potentially could create a bypass of the containment:

(a) Systems designed for removing heat from the core, from the core debris or from the containment that can transport radioactive material outside the containment in accident conditions;
(b) Systems that can transport airborne radioactive material from the containment atmosphere to outside the containment in accident conditions

(e.g. systems used in some designs to mix the atmosphere inside the containment to prevent the ignition of hydrogen);

(c) Supporting systems or auxiliary systems (inside the containment) for which, in the event of leakage, fluids with a high activity might be released outside the containment (in some designs, the component cooling water system, the containment sump purge system or the sampling systems).

4.164. Systems connected to the primary circuit in normal operations (i.e. primary circuit filtration systems or, in some designs, the chemical and volume control system) and systems connected to the containment atmosphere should be automatically isolated in accident conditions when they are not necessary for safety.

Isolation valves

4.165. To achieve the objective of limiting any radioactive release outside the containment, the isolation devices should be designed with a specified leaktightness and closure time.

4.166. Design provisions for leakage tests (e.g. nozzles, instrumentation test lines) should be made such that each isolation valve can be tested.

Penetrations

4.167. Paragraphs 4.168 and 4.169 provide recommendations on meeting the requirement in para. 6.21 of SSR-2/1 (Rev. 1) [1].

4.168. Containment penetrations should be designed to withstand at least the same loads and load combinations as the containment.

4.169. Containment penetrations should be accessible so that leaks from individual penetrations can be detected in the leaktightness tests.

Piping penetrations

4.170. In the mechanical design of piping penetrations, including isolation valves, the loads originating from the piping system as well as loads originating from the containment should be taken into account.

Electrical penetrations

4.171. Penetrations through the containment for electrical power cables and instrument cables should be leaktight. Means for ensuring the leaktightness of these penetrations should be based on the following:

(a) Pressure glass penetrations: The pressure glass design consists of studs embedded in a pressurized glass disc flanged to the containment. Cables are connected to the studs, which extend on both sides of the glass disc and provide continuity for the electric power. The glass ensures electrical isolation between the studs and acts as a sealant. The design should include double seals on the flange to ensure the leaktightness of the assembly. These penetrations should be removable and individually testable for leaktightness at the design pressure.
(b) Pressurized and continuously pressure monitored penetrations: For pressurized penetrations, the pressurization should normally be higher than the internal pressure that could occur in the containment in accident conditions, so that leaktightness can be tested continuously. In any case, the pressure should not be lower than the pressure used in the containment leak rate test. The effects of increase in temperature on the design pressure of the fluid inside the penetrations should be assessed and taken into account in the design of the penetrations.
(c) Injected sealant penetrations: Penetrations of this type should be leak testable in integrated leak tests.

4.172. Preference should be given to designs of electrical penetrations that allow each penetration to be tested individually.

4.173. Heat produced by the electrical cables should be taken into account in selecting the materials for electrical penetrations. The materials used should be heat resistant and non-flammable. Penetrations using sealant injection should be at least flame retardant.

Air locks, doors and hatches

4.174. Paragraphs 4.175–4.180 provide recommendations on meeting Requirement 57 of SSR-2/1 (Rev. 1) [1].

4.175. Penetrations for access by personnel or equipment to the containment (containment air locks) should have air locks equipped with doors that are interlocked to ensure that at least one of the doors is closed during reactor operations and in accident conditions. In addition, such penetrations should be designed to prevent any undue exposure of workers to radiation in operational states of the plant.

4.176. The two air lock doors should be designed to withstand the same plant conditions as the containment. Local transient internal effects, such as exposure to open flames caused by hydrogen burning, need not be considered for the outer door.

4.177. The chamber between the two air lock doors should be sized so as to allow the passage of necessary maintenance equipment and a sufficient number of personnel, so as to avoid having to open the air lock too frequently during plant shutdown and maintenance.

4.178. The inner door of the air locks should be of a pressure sealing type. Double seals should be provided on each door, and there should be provisions for testing the leaktightness of the doors and the inter-seal space. Low pressure alarms should be provided if inflatable seals are used.

4.179. Equipment hatches are large openings in the containment that are normally closed. They are usually designed with a bolted flange, whose leaktightness is ensured by means of soft elastomeric seals. Loads and deformations due to pressure and temperature effects should be taken into account in the design of equipment hatches.

4.180. Containment openings (i.e. penetrations, air locks and hatches) should normally be closed. Exceptions are allowed if they are necessary for operational reasons and provided that the openings can be closed quickly and reliably to comply with established engineering criteria that apply to accident conditions. Conditions allowing equipment hatches to be opened should be specified and met before a hatch is opened.

MATERIALS

Concrete

4.181. Concrete should have quality and performance characteristics (strength, density and porosity) that are consistent with its use. The quality of the concrete used for containment structures should be correspondingly high, consistent with the safety function of the containment. Design considerations will depend on the containment concept. For example, a prestressed concrete containment can provide both structural support and leaktightness, whereas a reinforced concrete containment structure provides structural support but relies on a steel liner for leaktightness.

4.182. Consideration should be given to the design capacity of the concrete to cope with the loads (pressure loads and thermal loads) and environmental conditions (heat, moisture and radiation) generated by accident conditions. This should lead to strict specifications for the concrete in terms of strength and leaktightness.

4.183. Concrete specifications should also ensure that measures are taken to avoid material vulnerabilities that could lead to ageing effects (e.g. chloride attack, alkali–aggregate reaction, delayed ettringite formation).

4.184. Concrete with appropriate stiffness, thermal expansion and resistance to compression should be used for all electrical penetrations, large penetrations such as equipment hatches, and the joint with the basemat.

4.185. In a prestressed containment that is not sealed with a metallic liner, the concrete should remain in a prestressed condition even in accident conditions. Concrete materials that limit creep or shrinkage over the years and have low porosity should be used. The possible loss of prestress of the containment tendons over the operating lifetime of the plant should be evaluated and considered in the design.

4.186. Sleeve–concrete interfaces should be designed to minimize leaks by avoiding direct paths through the interface.

4.187. Design and construction processes should be such as to prevent the development of cracks or high leak zones.

4.188. Ageing effects should be evaluated in the selection and design of types of concrete, and a programme for monitoring the effects of ageing over time should be developed: see SSG-48 [18] and Requirement 31 of SSR-2/1 (Rev. 1) [1].

Metallic materials

4.189. Metallic materials used for the containment and its associated systems, including welds, should be of high quality; qualified and certified materials that meet national safety standards should be used.

4.190. In the selection of metallic materials, the following should be considered:

(a) Thermal and mechanical loads;
(b) Chemical interactions, including those with chemicals used in containment spray systems;
(c) Sensitivity to ageing effects;
(d) Resistance to brittle fracture;
(e) Resistance to corrosion.

4.191. Metallic materials that have the potential to generate hydrogen on contact with water or steam, such as zinc and aluminium, should not be used inside the containment. If such materials are essential to the design, their use should be limited and the effects of hydrogen generation should be analysed.

Soft sealing materials

4.192. Soft sealing materials are commonly used in multiple confinement applications, such as in the sealing of ventilation valves or the inflatable sealing of air locks. Although these materials contribute to a very high leaktightness of the containment under normal conditions, their behaviour in accident conditions should be properly demonstrated. Potentially damaging effects for soft sealing materials include embrittlement and cracking due to high temperatures and irradiation, dissolution due to moisture and steam, and swelling or shrinkage due to temperature fluctuations. Specific consideration should be given to the protection of these materials from the direct effects of hydrogen burning and the accumulation of radioactive aerosols. In extreme conditions, such materials might degrade to the extent that their mechanical properties are altered.

4.193. The anticipated lifetimes of soft sealing materials and the ageing mechanisms that affect their performance should be assessed, and appropriate replacement intervals should be established. Sealing components should be designed to be easily inspected and replaced.

Covering, cushioning, thermal insulation and coating materials

4.194. Paragraphs 4.195–4.202 provide recommendations on meeting the requirements of para. 6.30 of SSR-2/1 (Rev. 1) [1].

4.195. Covering, cushioning, thermal insulation and coating materials should not compromise any safety functions in the event of their deterioration. They should be installed and affixed so as to prevent their becoming loose and possibly clogging sieves and valves.

4.196. In particular, materials used to insulate pipes and tanks inside the containment should be selected and designed to achieve the following:

(a) To minimize the production of debris that can accumulate on containment floors and clog the sump screen or strainer filter or damage recirculation pumps;
(b) To ensure easy decontamination if the need arises;
(c) To avoid giving rise to fire hazards;
(d) To minimize the release of toxic gases during the heating of such materials at the plant start up.

4.197. The amount of debris generated in the event of high energy pipe breaks and transported to the sumps should be assessed, and the surface of filters should be sized accordingly in order not to compromise the normal operation of the pumps necessary to mitigate the consequences of accidents.

4.198. A cleaning system for the filters should be installed, taking into account the large uncertainties about the types and amount of debris that could clog the filters.

4.199. If organic liners are applied to increase the leaktightness of the containment, they should be selected to provide good adhesion and a low air (gas) permeability, and to withstand the thermal loads and pressure loads, as well as the environmental conditions in the containment, without losing their safety function (e.g. the organic material should have a good ability to span cracks and a resistance to blistering after thermal ageing). Provision for managing the ageing of these organic liners should be made, including provision for maintenance and surveillance.

4.200. Painting and coating materials should be selected so as not to pose a fire hazard, and to avoid clogging of the containment sump.

4.201. In the selection of painting and coating materials, the effect of the solvents used in such materials becoming dissolved in the sump (e.g. the effect on the volatility of iodine) should be considered.

4.202. Ageing mechanisms that affect covering, cushioning, thermal insulation and coating materials should be assessed, and appropriate replacement intervals should be established.

INSTRUMENTATION

4.203. Paragraphs 4.204–4.241 provide recommendations on meeting Requirement 59 of SSR-2/1 (Rev. 1) [1].

4.204. Adequate instrumentation should be provided for the following purposes:

(a) Monitoring the stability of the containment;
(b) Detection of deviations from normal operation;
(c) Periodic testing;
(d) Monitoring the availability of the associated systems;
(e) Initiation of automatic operation of systems;
(f) Post-accident monitoring.

4.205. The different purposes of instrumentation can result in measurements of the same parameters for different levels of defence. The consequences of sharing sensors for different purposes should be considered in order to preserve adequate independence between the different levels of defence in depth. The following recommendations should be implemented to the extent possible:

(a) Separate sensors should be provided for the automatic actuation of the systems and for the accident monitoring of the plant.
(b) Separate sensors should be provided for the automatic actuation of the reactor scrams and the operation of the safety systems (including their backup systems) implemented to reinforce the prevention of accidents with core melting.
(c) Different and dedicated sensors should be provided for the mitigation of accidents with core melting.

4.206. Instrumentation should be qualified for seismic loads and environmental conditions that might prevail before or during its operation.

4.207. The test sequences for equipment qualification should be consistent with well proven international practices. More detailed recommendations are provided in IAEA Safety Standards Series No. SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants [22].

Monitoring the stability of the containment

4.208. Deformation (radial, vertical or circumferential) or movement of the containment structures or the containment walls should be monitored (e.g. monitoring of settlement and differential settlement of the buildings) throughout the lifetime of the containment.

4.209. For prestressed concrete walls, means to detect loss of the prestressing should be provided. The concrete compression and stiffness parameters (such as Young's modulus) should be defined, and they should be verified by such means as acoustic measurements. The temperature in concrete discontinuities should also be measured to aid the interpretation of the results of proof pressure tests.

4.210. Measurements to monitor the containment stability and deformations over time should be recorded to show trends.

4.211. Appropriate instrumentation for measurements relating to earthquakes should be installed at suitable locations (e.g. on the basemat of the containment and on suitable floors).

Detection of deviations from normal operation

4.212. Appropriate instrumentation should be incorporated inside the containment for an early detection of deviations from normal operation, including the following:

(a) Leaks of radioactive material;
(b) Abnormal radiation levels;
(c) High energy leaks;
(d) Leaks of primary coolant;
(e) Fire;
(f) Failure of components.

4.213. The necessary instrumentation sensitivity and measurement range to detect a developing deviation should be estimated by appropriate analytical methods.

4.214. For the adequate detection of different abnormal conditions, information can be provided by individual instrumentation or by a combination of instruments. The parameters typically monitored are described in paras 4.215–4.229.

Containment atmosphere temperature

4.215. Monitoring of the containment atmosphere temperature is necessary to check whether it is within the range specified for normal operation, as follows:

(a) A sufficient number of temperature sensors should be installed to measure the containment atmosphere temperature.
(b) Measurements from containment air coolers may be used to estimate temperatures inside the containment.

4.216. The results of the measurements of containment atmosphere temperatures should be recorded to show trends.

Containment pressure

4.217. Monitoring of the containment pressure should be established to check whether the pressure is within the range specified for normal operation (small variations of the pressure could be caused by the operation of the air operated valves, by changes in the containment temperature or by leakages of fluids such as compressed air or nitrogen).

4.218. For the secondary confinement building, or for a containment with double walls, monitoring of the pressure inside the secondary confinement building or in the annulus[11] should be established to check whether the pressure is within the range specified for normal operation (a small negative pressure should be maintained).

4.219. The results of the measurements of containment pressure should be recorded to show trends.

Containment atmosphere gas composition

4.220. The gas composition of the containment atmosphere should be monitored at locations of potential high concentration of combustible gases.

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[11] ‘Annulus’ indicates the free volume between the two walls of the containment.

Humidity at different locations

4.221. Humidity is a highly significant factor for the detection of leaks in operational states. The following parameters can be used as a basis for measuring the humidity:

(a) The dew point temperature of the containment atmosphere;
(b) Electrical parameters such as the impedance or resistance of sensors;
(c) The amount of condensate in the air coolers of the containment.

4.222. The results of the measurements should be recorded to show trends.

Water levels in the drain storage tanks and sumps

4.223. Drain storage tanks and the sumps of each safety system, as well as the condensate collector of each air cooler, should be provided with a water level indicator.

Radiation levels and radioactivity measurements

4.224. Radiation levels at different locations inside the containment should be measured for the radiation protection of the workers and for an early detection of any anomalies.

4.225. Measurements of the levels of radioactivity in the containment atmosphere and in water (drain storage and sumps) should be made as a complementary means for the detection of leaks.

Visible abnormalities

4.226. A video surveillance system should be installed inside the containment to detect anomalies at relevant locations where leaks or other malfunctions can be expected or where personnel access is difficult (e.g. reactor coolant pumps, equipment hatch, personnel air locks, reactor pools).

4.227. Mobile cameras should be available for use, as necessary.

Noise and vibration

4.228. The measurement and analysis of audio signals from the containment for the detection of abnormalities should be considered (e.g. the use of spectral and Fourier transform analyses of acoustic noise signals).

Fire

4.229. Smoke and flame detectors should be installed as additional means of early detection of a fire in each compartment where there could be a risk of fire.

Periodic testing of the containment leak rate

4.230. Appropriate instrumentation for conducting periodic leak tests should be incorporated inside the containment. Measurements of temperature, pressure and humidity, as well as flow rates, should be combined for the periodic calculation of the mass of the containment atmosphere and for the estimation of the leak rate. For steel containments, the temperature of the steel should also be measured. More details are given in Section 5.

Monitoring of the availability of systems

4.231. Appropriate instrumentation should be used to monitor the availability of the systems used for mass and energy release and management, the control of radioactive releases and the management of combustible gases.

4.232. The availability of the systems should be verified by means of the following:

(a) Continuous monitoring, and display in the main control room, of the main parameters important to safety (a single integrated monitor for critical safety parameters is recommended);
(b) Periodic testing and inspections as required;
(c) For the systems for mass and energy release and management, monitoring of the positions of valves, the status of components in operational states, and flow rates;
(d) For the systems for management of radioactive material, monitoring of the positions of isolation valves, air locks and doors; the pressure of inflatable air lock seals; and water levels in the different water tanks necessary to the operation of those systems.

Initiation of automatic operation of systems

4.233. In the event of a significant release of mass and energy or of radioactive material into the containment, different types of information need to be considered to ensure the complete and effective management of the mass and energy, radioactive material and combustible gases released inside the containment. This management process should be initiated automatically or could be initiated by the operator, provided that there is sufficient time available for implementing operator actions.

4.234. Information from the monitoring of a variety of parameters should provide evidence that a large release of energy or a significant release of radioactive material has occurred inside the containment. Depending on the reactor technology or the design, the following factors might be relevant:

(a) High pressure inside the containment;
(b) High radiation levels inside the containment atmosphere;
(c) Low pressure in the reactor coolant system;
(d) Small subcooling margin in the reactor coolant system (for pressurized water reactors);
(e) Low water level in the reactor pressure vessel.

4.235. In addition to conditions that require a complete and effective management of the mass and energy, gases, and radioactive material released inside the containment, there are other events for which only the individual isolation of the affected lines is necessary to limit the release of radioactive material from the containment to the environment.[12] The actuation conditions of the isolation devices should be derived from the values of appropriate parameters, such as the following:

(a) Levels of radiation and levels of airborne radioactive contamination;
(b) Pressure changes in the affected system;
(c) Temperature changes in the affected system;
(d) Water level in the affected system.

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[12] Such an event might be a break occurring outside the containment of a pipe crossing the containment and carrying radioactive material, or for the failure of an interface between two associated systems (e.g. rupture of a heat exchanger tube of the component cooling water system) that leads to a release of radioactive material from a system inside the containment to a system outside.

Accident and post-accident monitoring

4.236. For the determination of the plant status in the event of an accident and for the management of accidents, appropriate instrumentation displays and records should be available in the main control room and the emergency response facility to allow personnel to diagnose the situation and implement the actions specified in the emergency operating procedures or in the severe accident management guidelines. The information provided by such instrumentation should include the following:

(a) Conditions and gas composition inside the containment (containment pressure and temperatures, radiation levels, airborne activity levels, steam, oxygen or hydrogen concentration if relevant).
(b) Process parameters to check that the required safety actions are in progress and to indicate the operations of the required safety systems and safety features for design extension conditions (e.g. flow rates, water levels in tanks and sumps, operating pressures in the systems).
(c) Process parameters to indicate the potential for degradation or loss of containment leaktightness (e.g. the position of containment isolation valves, status of hatches and doors, containment pressure, airborne activity in the surrounding buildings).
(d) Process parameters to implement actions specified in the emergency procedures or severe accident management guidelines (process parameters to control the pressure and to maintain the conditions inside the containment below the specified limits).
(e) Information for assessing the radiological consequences in a timely manner and for assisting in decisions on long term actions for the protection of the public (off-site emergency measures). Instrumentation for assessing radiological consequences could include the following:
    (i) Dose rate monitoring instruments and detectors of airborne activity in the containment and in peripheral buildings;
    (ii) Sensors for monitoring conditions in the containment sump water (e.g. temperature, pH);
    (iii) Radioactivity monitors for noble gases, radioiodine and aerosols in the stacks and in the containment venting line;
    (iv) Position indicators of valves for containment venting.

4.237. Dedicated instrumentation should be provided to allow personnel in the main control room to initiate long term actions necessary to maintain the containment integrity in the event of an accident with core melting. Such instrumentation should provide information about the following process parameters:

(a) Parameters to initiate the fast depressurization of the reactor coolant system (before core melting) and to confirm the open position of the depressurization valves;
(b) Parameters to confirm the flooding of the reactor cavity (for in-vessel strategy) or the flooding of the ex-vessel retention structure (for ex-vessel retention strategy);
(c) Parameters for the localization of the molten core (for ex-vessel retention strategy);
(d) Parameters to initiate and confirm the operation of the containment spray;
(e) Parameters to initiate and confirm the operation of the containment heat removal system;
(f) Parameters to initiate the venting of the containment (if relevant);
(g) Parameters for hydrogen risk management.

4.238. A monitoring or sampling system should be provided inside the containment to enable an assessment of the risks of explosion from combustible gases. The design of the system should take into account the following factors:

(a) Possible sources of combustible gases, such as interaction between clad material and water, or interaction between the molten core and concrete, or due to radiolysis;
(b) The presence or absence of oxygen and inert gases;
(c) The presence of noble gases and aerosols;
(d) The presence of devices aimed at recombining hydrogen, and the types of device (passive or active);
(e) Sufficient mixing of the containment atmosphere to avoid local hydrogen accumulation.

4.239. The monitoring can be achieved by direct gas concentration measurement or by sampling. An alternative is to assess the recombination activity of the recombiners by temperature measurement.

4.240. Provisions should be made in the design for sampling of the containment atmosphere and the sump water at suitable locations. The sampling devices should be qualified for the expected containment conditions and should be installed so as to avoid a containment bypass in the event of their rupture. The sampling devices should be designed to ensure that the dose constraints for occupational exposure are not exceeded for the workers who operate them.

4.241. Monitoring or sampling lines that could transport radioactive material outside the containment should be considered as extensions of the containment and should be subject to specifications for structural integrity and leaktightness comparable to those applied to the containment structure itself.

5. TESTS AND INSPECTIONS

5.1. To demonstrate that the containment and its associated systems meet design and safety requirements, tests and inspections should be conducted during construction, commissioning and operation in accordance with proven codes and standards and taking into account underpinning recommendations. The recommendations given in IAEA Safety Standards Series No. NS-G-2.6, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants [23], should also be considered.

5.2. Paragraphs 5.3–5.30 provide recommendations on meeting Requirement 29 of SSR-2/1 (Rev. 1) [1].

INSPECTION DURING CONSTRUCTION

5.3. Inspections should be performed at different stages of the construction to ensure conformity to design and construction specifications. Deficiencies, deviations from standards and non-conformances should be tracked and reported. Typical examples of structures, systems and components that should be subject to inspections performed during construction are as follows:

(a) The vertical tendon anchorage area;
(b) The basemat rebar installation and concrete work;
(c) The horizontal tendon anchorage area;
(d) The tendon duct arrangement;
(e) The liner plate work;
(f) The rebar arrangement around large openings.

5.4. Construction works and inspections should be performed by qualified personnel.

COMMISSIONING TESTS

5.5. Commissioning tests for the containment and its associated systems should be performed before the first criticality of the reactor to demonstrate the containment structural integrity, to determine the leak rate of the containment envelope and confirm the performance of the systems and equipment.

Structural integrity test

5.6. A pressure test should be conducted to demonstrate the structural integrity of the containment, the envelope and the pressure retaining boundary of the associated systems.

5.7. The pressure test should be conducted at a specified pressure that is at least the design pressure, and for which account is taken of the applicable codes for the materials used. The test temperature should not be close to the ductile brittle transition temperature for metallic material.

Integrated leak rate tests of the containment envelope

5.8. An integrated leak rate test should be conducted to demonstrate that the leak rate of the containment envelope does not exceed the specified maximum leak rate. The test should be conducted with the components in a state representative (to the extent practicable) of the conditions that would prevail under accident conditions in order to demonstrate that the specified leak rate would not be exceeded under such conditions.

5.9. To establish a point of reference for future in-service leak tests, the leak rate test performed during commissioning should be conducted at a test pressure or pressures consistent with the pressure selected for the in-service leak tests, in accordance with the following recommendations:

(a) If the in-service tests are to be conducted at a pressure lower than the design pressure, the leak rate test during commissioning should be conducted at pressures between the pressure selected for in-service leak testing and the positive design pressure.
(b) If the in-service tests are to be conducted at the design pressure, the leak rate test during commissioning of the containment should be conducted at this pressure.

5.10. The need to reliably validate the leak rate assumed in the safety analysis over the entire plant operating lifetime for the entire range of pressures calculated should be taken into consideration in the choice of test pressure. There are two methods of validation, as follows:

(a) Absolute method: The leak rate can be validated by measuring the decrease in pressure or the dry air mass as a function of time. In this method, the temperature and pressure of the containment atmosphere, the external atmospheric temperature and pressure, and the humidity of the containment atmosphere should be measured continuously and factored into the evaluation. Means should be provided to ensure that the temperature and humidity of the containment atmosphere are uniform.
(b) Reference vessel method: The reference vessel method determines the air mass from the pressure differential between the containment atmosphere and the reference vessel atmosphere. The pressure differential is determined from a manometer, one leg of which is open to the pressurized (and leaking) containment, while the other leg is connected to a leaktight pressurized system of tubing placed throughout the containment. The reference vessel temperature and the containment temperature are assumed to be equal.

5.11. The need for initial and periodic testing should be considered in the design, and all the components that might be damaged during testing should be identified. The necessary means to pressurize and depressurize the containment and appropriate instrumentation for testing should be included in the design.

5.12. Appropriate instrumentation should be provided in the containment. To determine representative atmospheric conditions in the different zones of the containment, this instrumentation should be appropriately positioned and installed either permanently or when necessary.

5.13. For double wall containments, one way to determine the direct leak rate from the containment to the environment (i.e. if the leaked water or gas does not collect in the annular space between the inner and the outer containment walls) is by calculation. This calculation should determine the difference between (a) the total leak rate from the containment as determined by the leak test for the inner containment (this consists of both flow from the primary containment into the annulus and flow from the primary containment to the atmosphere) and (b) the leak rate from the primary containment wall to the annulus, obtained after ventilation of the annulus has been stopped (this is typically calculated by subtracting the normal flow out of the annulus vent from the flow out of the annulus vent during the leak test).

Local leak rate tests of isolation devices, air locks and penetrations

5.14. Local leak rate tests should be performed to establish a baseline leakage measurement for each isolation device, air lock and penetration. The following components are the most sensitive parts of the containment envelope, and special attention should be paid to them:

(a) Isolation devices in systems open to the containment atmosphere;
(b) Isolation devices in fluid system lines penetrating the containment;
(c) Penetrations that have resilient or inflatable seals and expansion bellows, such as the following:
  — Personnel air locks;
  — Equipment air locks;
  — Equipment hatches;
  — Fuel transfer tube;
  — Spare penetrations with bolted closures;
  — Cable penetrations with resilient seals;
  — Pipe penetrations with flexible expansion bellows in the connections to the containment.

5.15. The design should permit leak rate tests of isolation devices, air locks, penetrations and containment extensions.

5.16. The design should facilitate local testing by providing access to penetrations and incorporating necessary connections and isolation valves.

5.17. To permit greater precision in measuring the leak rate and to improve the detection of leaking valves, the capability to test individual valves should be provided.

Functional tests of equipment and wiring in the containment

5.18. Tests should be performed to verify that the performance of the associated systems complies with the design specifications, unless the tests would have a detrimental effect on safety.

5.19. Tests should be performed on all electrical wiring of the associated systems to demonstrate that there are no deviations from the design and that all connections are in accordance with the design.

IN-SERVICE TESTS AND INSPECTIONS

5.20. In-service integrated leak rate and local leak rate tests and inspections should be periodically performed to demonstrate that the associated systems continue to meet the requirements for design and safety throughout the operating lifetime of the plant.

5.21. The test methods and intervals for in-service tests should be specified so as to reflect the importance to safety of the items concerned. In devising test methods and determining the frequency of testing, consideration should be given to the necessary levels of performance and reliability of the systems individually and as a whole.

5.22. Appropriate features should be provided for performing commissioning and in-service testing for containment pressure and leaktightness, and the correlated loads should be considered for the purposes of structural design.

5.23. General guidance on in-service inspection is provided in GS-G-3.5 [9].

Structural integrity tests

5.24. Periodic structural tests should be conducted to demonstrate that the containment continues to perform as intended in the design. The test pressure should be the same as in the commissioning test and as required by the applicable design codes. In the design, attention should be paid to the additional stresses imposed by the tests, and test pressures should be established to prevent the tests from causing excessive stresses to the containment. A leak test should be performed during any structural integrity test. In some States, a tendon monitoring programme could be used instead of a pressure test for prestressed concrete containments equipped with unbounded tendons, although leak testing would still be necessary.

Integrated leak rate tests of the containment envelope

5.25. The design should provide the capability for periodic in-service testing of the leak rate to verify that the leak rate assumed in the safety analysis is maintained throughout the operating lifetime of the plant. The in-service leak rate tests may be made at either of the following:

(a) A pressure that permits a sufficiently accurate extrapolation of the measured leak rate to the leak rates at the pressures under accident conditions considered in the safety analysis;
(b) The containment design pressure.

5.26. There are also methods available to provide a continuous estimate of the overall containment leak rate during plant operation and to derive approximate indications of containment leak rates in accident conditions. Such approaches are generally based on variations in the containment pressure or the mass balance during normal operation of the plant. In some cases, the use of these methods together with extensive local leak rate tests during shutdown for refuelling could justify a reduction in the frequency of the integrated leak rate tests.

5.27. In a containment with a pressure suppression pool, to ensure that the bypass rate of the pool is consistent with the value considered in the safety analysis, features should be provided for periodically assessing any leakage that might lead to bypassing of the pool.

Visual inspection

5.28. Visual inspections are important for monitoring and detecting ageing effects and for detecting cracks and monitoring their evolution. Visual inspections may augment the results from structural monitoring and instrumentation.

5.29. Where it is technically feasible, the design should provide for a complete visual inspection of containment structures (including the tendons for prestressed concrete containments), penetrations and isolation devices.

5.30. Visual inspection of the containment envelope should be made in conjunction with each of the tests specified in paras 5.24 and 5.25. A visual inspection technique should be employed that is specifically qualified for detecting the type and size of cracks and other defects that are determined to be important to leakage and structural integrity.

Appendix

PLANTS DESIGNED TO EARLIER STANDARDS

A.1. Paragraph 1.3 of SSR-2/1 (Rev. 1) [1] states:

> "It might not be practicable to apply all the requirements of this Safety Requirements publication to nuclear power plants that are already in operation or under construction. In addition, it might not be feasible to modify designs that have already been approved by regulatory bodies. For the safety analysis of such designs, it is expected that a comparison will be made with the current standards, for example as part of the periodic safety review for the plant, to determine whether the safe operation of the plant could be further enhanced by means of reasonably practicable safety improvements."

A.2. This implies that the capability of existing plants to accommodate accident conditions not considered in their original design basis should be systematically assessed with the further objective of improving the current level of safety and, in particular, the overall efficiency of the containment and its associated systems.

A.3. Most of the containment and associated systems of existing plants were designed for design basis accidents (e.g. large loss of coolant accidents), without account being taken of the possibility for more severe accidents to occur. However, safety assessments have shown that the conservative deterministic approaches originally followed in the design have resulted in the capability to withstand situations more severe than those originally included in the design basis for existing plants.

A.4. The assessment should be conducted on the basis of a set of design extension conditions whose consequences should be analysed with the purpose of further improving the safety of the nuclear power plant by achieving the following:

(a) Enhancing the plant capability to withstand more challenging events, conditions and hazards than those considered in the design basis;
(b) Minimizing radioactive releases harmful to the public and the environment as far as reasonably practicable in such events or conditions.

A.5. Although the design extension conditions to be assessed are reactor technology and design dependent, the selected set of design extension conditions should systematically include core melt situations, phenomena that could lead to the loss of the containment integrity and external events that exceed the original design basis.

A.6. The assessment for potential backfitting should utilize a holistic approach that considers the safety contributions of installed equipment, non-permanent equipment and emergency arrangements to protect the public.

A.7. The assessment should aim to justify, with a reasonable level of confidence, that the relevant equipment would be available to perform the expected function. The assessment may use realistic models and assumptions, as well as different acceptance criteria to those for design basis accidents, provided that the absence of a cliff edge effect can still be justified.

A.8. The assessment of the robustness of a structure or mechanical equipment may be performed by applying deterministic methods, probabilistic methods or a combination of the two.

A.9. The assessment of leaktightness, integrity or operability of structures and components should be performed with account taken of reasonable uncertainties in the loads and in the response of the structure or component.

A.10. The backfitting measures for preventing early radioactive releases or for implementing actions in the short term should not rely on the use of off-site mobile equipment.

A.11. Although the use of permanent equipment for avoiding large radioactive releases should be preferred (as for new plants), a more flexible approach with regard to the use of non-permanent equipment may be acceptable, where the plant is provided with adequate connection features.

A.12. All natural hazards that are addressed in the design basis should be re-evaluated on the basis of up to date methodologies and meteorological and geological data. Hazards not yet evaluated in the design basis that could have an impact on the containment should be considered, and their effects should be evaluated. The design of the containment and its associated systems for accident conditions beyond the original design basis conditions should be assessed to know whether they would be capable of performing their function with adequate margins under the new conditions.

A.13. The resistance of structures and components necessary to avoid radioactive releases that would require long term protective actions should be evaluated with regard to natural hazards that exceed the severity considered in the design.

A.14. With regard to challenges for the containment integrity, the following should be achieved:

(a) Conditions leading to direct containment heating should be identified and reliably prevented.
(b) Possibilities for steam explosions should be identified, and the effects of such explosions should be evaluated.
(c) Different and diverse means should be implemented to control pressure buildup inside the containment in the different plant states.
(d) Multiple means should be implemented to remove heat from the containment in the different plant states.
(e) If a containment venting system is necessary for certain events that are beyond the original design basis, it should be reliable and robust enough to withstand loads from hazards (e.g. earthquakes) and from accident conditions, as well as to withstand the dynamic and static pressure loads that exist when the containment venting line is operating.
(f) Specific safety features and systems should be implemented to ensure the cooling and stabilization of the molten core.

A.15. With regard to the control of radioactive releases, the following should be achieved:

(a) All piping penetrating the containment should be isolated, except for piping belonging to systems necessary for the mitigation of accident conditions.
(b) The containment should be kept leaktight to the extent possible under accident conditions with core melting.
(c) Different means should be implemented to reduce the amount of radioactive material in the containment atmosphere in accident conditions.
(d) Mechanisms and potential paths for unintentional containment bypass should be identified and the consequences evaluated.
(e) If venting of the containment atmosphere is necessary, it should be possible to close the containment venting lines reliably.
(f) With regard to intentional releases (e.g. containment venting) in the event of a severe accident, consideration should be given to providing filtration with high efficiency filters before discharge to the environment.

A.16. With regard to the management of combustible gases, the risks of various hydrogen combustions should be evaluated and adequate provisions should be implemented, if necessary, to prevent hydrogen combustions challenging the containment integrity and to control the concentration of combustible gases inside the containment.

A.17. With regard to instrumentation, the following should be achieved:

(a) The operability, reliability and adequacy of instrumentation should be evaluated (e.g. for measurement ranges, environmental qualification and power supply) to ensure that operating personnel obtain essential and reliable information about the containment status in the different plant states.
(b) The containment should be equipped with measuring and monitoring instrumentation that provides sufficient information on the progress of core melt accidents and threats to containment integrity and by which the operating personnel can undertake the necessary actions in accordance with the severe accident management guidelines. Any new instrumentation for monitoring the progression of severe accidents should be qualified for the relevant accident conditions with core melting.

A.18. With regard to non-permanent equipment, the following should be achieved:

(a) Non-permanent equipment that would be necessary to minimize the consequences of events that cannot be mitigated by the installed plant capabilities should be stored and protected to ensure its timely availability when necessary, with account taken of possible restricted access due to external events (e.g. flooding, damaged roads).
(b) Relying on non-permanent equipment may be adequate provided there is a justification to demonstrate that the coping time to avoid containment failure is long enough to make use of the equipment.

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