IAEA Safety Standards

for protecting people and the environment

Seismic Design for Nuclear Installations

Specific Safety Guide

No. SSG-67

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

https://www.iaea.org/resources/safety-standards

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.

RELATED PUBLICATIONS

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

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

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

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.

SEISMIC DESIGN FOR NUCLEAR INSTALLATIONS

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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-67

SEISMIC DESIGN FOR NUCLEAR INSTALLATIONS

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2021

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 26007 22529
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
www.iaea.org/publications

Printed by the IAEA in Austria
November 2021
STI/PUB/1949

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Seismic design for nuclear installations / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2021. | Series: IAEA safety standards series, ISSN 1020–525X ; no. SSG-67 | Includes bibliographical references.
Identifiers: IAEAL 21–01451 | ISBN 978–92–0–117421–5 (paperback : alk. paper) | ISBN 978–92–0–117521–2 (pdf) | ISBN 978–92–0–117721–6 (epub)
Subjects: LCSH: Nuclear facilities — Design and construction. | Nuclear facilities — Safety measures. | Earthquake resistant design. | Earthquake hazard analysis.
Classification: UDC 621.039.58 | STI/PUB/1949

FOREWORD

by Rafael Mariano Grossi
Director General

The IAEA's Statute authorizes it to "establish…standards of safety for protection of health and minimization of danger to life and property". These are standards that the IAEA must apply to its own operations, and that States can apply through their national regulations.

The IAEA started its safety standards programme in 1958 and there have been many developments since. As Director General, I am committed to ensuring that the IAEA maintains and improves upon this integrated, comprehensive and consistent set of up to date, user friendly and fit for purpose safety standards of high quality. Their proper application in the use of nuclear science and technology should offer a high level of protection for people and the environment across the world and provide the confidence necessary to allow for the ongoing use of nuclear technology for the benefit of all.

Safety is a national responsibility underpinned by a number of international conventions. The IAEA safety standards form a basis for these legal instruments and serve as a global reference to help parties meet their obligations. While safety standards are not legally binding on Member States, they are widely applied. They have become an indispensable reference point and a common denominator for the vast majority of Member States that have adopted these standards for use in national regulations to enhance safety in nuclear power generation, research reactors and fuel cycle facilities as well as in nuclear applications in medicine, industry, agriculture and research.

The IAEA safety standards are based on the practical experience of its Member States and produced through international consensus. The involvement of the members of the Safety Standards Committees, the Nuclear Security Guidance Committee and the Commission on Safety Standards is particularly important, and I am grateful to all those who contribute their knowledge and expertise to this endeavour.

The IAEA also uses these safety standards when it assists Member States through its review missions and advisory services. This helps Member States in the application of the standards and enables valuable experience and insight to be shared. Feedback from these missions and services, and lessons identified from events and experience in the use and application of the safety standards, are taken into account during their periodic revision.

I believe the IAEA safety standards and their application make an invaluable contribution to ensuring a high level of safety in the use of nuclear technology. I encourage all Member States to promote and apply these standards, and to work with the IAEA to uphold their quality now and in the future.

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

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

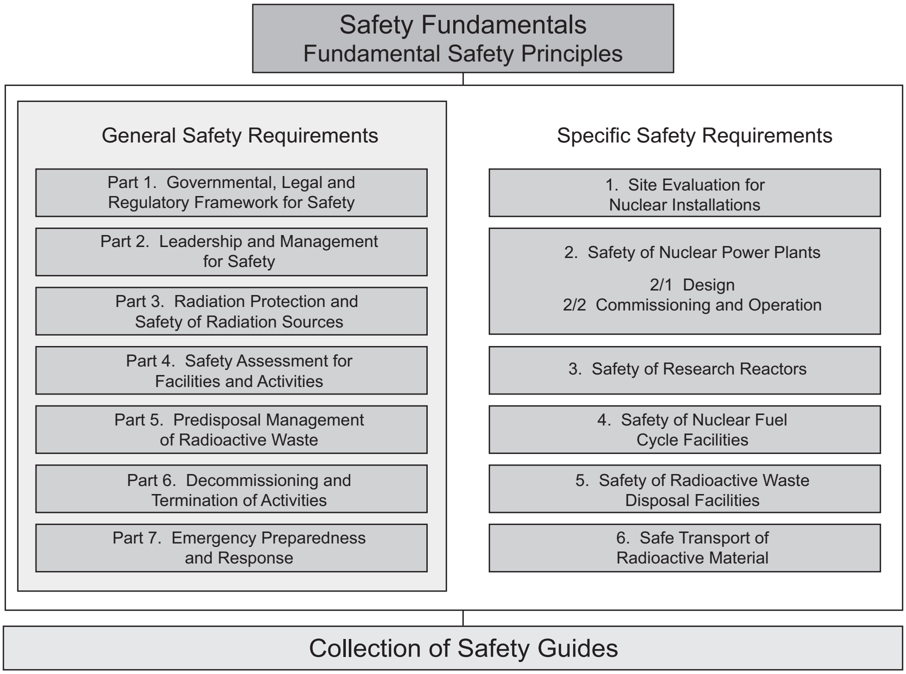

FIG. 1. The long term structure of the IAEA Safety Standards Series.

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.

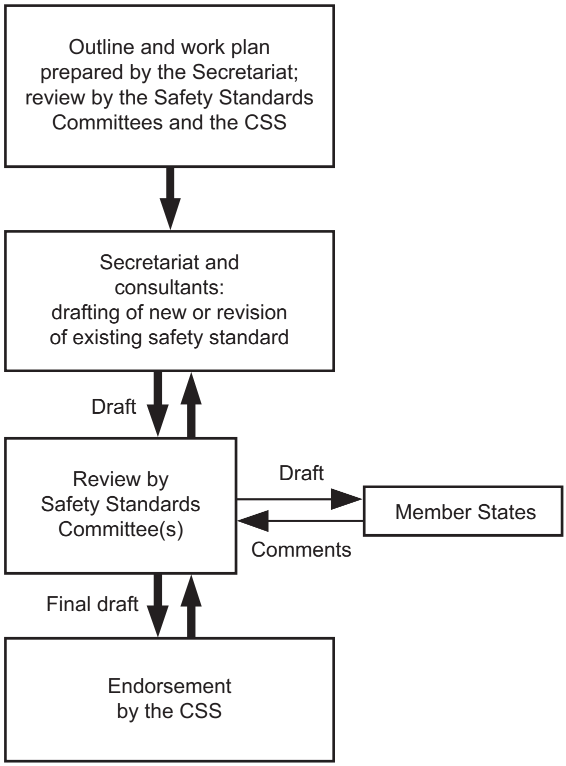

FIG. 2. The process for developing a new safety standard or revising an existing standard.

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 https://www.iaea.org/resources/safety-standards/safety-glossary). 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. IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [1], establishes design requirements for the structures, systems and components (SSCs) of a nuclear power plant. Requirements for the design of research reactors and of nuclear fuel cycle facilities are established in IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [2], and IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities [3], respectively. These publications include requirements for the design to take into account external events, including earthquakes. This Safety Guide provides specific recommendations on the design of nuclear installations to cope with the effects generated by earthquakes.

1.2. This Safety Guide incorporates the following:

(a) Progress in the design of nuclear installations and in related research, as well as regulatory practices in States, considering the lessons identified from recent strong earthquakes that have affected nuclear installations;
(b) Recent developments in regulatory practices on application of risk informed and performance based approaches for assessing the safety of nuclear installations;
(c) The experience and results from seismic design for new nuclear installations in States;
(d) A more coordinated treatment of the design of nuclear installations against seismically induced geological and geotechnical hazards and concomitant events.

1.3. This Safety Guide provides a clear distinction between (a) the process for assessing the seismic hazards at a specific site and (b) the process for defining the related basis for design and evaluation of the nuclear installations. These processes correspond to (and are performed at) different stages in the lifetime of a nuclear installation. This Safety Guide addresses the interface between these processes so as to bridge any gaps between them and avoid undue overlapping of recommendations.

1.4. Recommendations on the process for assessing the seismic hazards at a specific site, including the definition of the parameters resulting from such an assessment, are provided in IAEA Safety Standards Series No. SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations [4].

1.5. There is an important difference between the seismic design and the seismic safety evaluation of nuclear installations. Seismic design and qualification of SSCs is most often performed at the design stage of the installation, prior to its construction. Seismic safety evaluation can be conducted at the design stage (using data corresponding to the detailed design) and after the installation has been constructed (using as-built and as-operating conditions). There are some exceptions, such as the seismic design of new or replacement components after construction of the installation. Recommendations on the evaluation of existing nuclear installations are provided in IAEA Safety Standards Series No. NS-G-2.13, Evaluation of Seismic Safety for Existing Nuclear Installations [5].

1.6. In several States, designs of new nuclear reactors are being developed generically to meet the needs of many sites across a large geographical area. The intent is that each generic design uses design bases that envelop the potential seismic hazard challenges at all the candidate sites. Confirmation of this is needed when a generic design is nominated for a particular site. At this point, the site specific seismic hazards need to be assessed and compared with the generic seismic hazard design bases to ensure there is an acceptable enveloping margin between them.

1.7. This Safety Guide supersedes IAEA Safety Standards Series No. NS-G-1.6, Seismic Design and Qualification for Nuclear Power Plants[1].

OBJECTIVE

1.8. The objective of this Safety Guide is to provide recommendations on how to meet the safety requirements established in SSR-2/1 (Rev. 1) [1], SSR-3 [2] and SSR-4 [3] in relation to the design aspects of nuclear installations subjected to seismic hazards, defined in accordance with SSG-9 (Rev. 1) [4]. These recommendations focus on the consistent application of methods and procedures, in accordance with best practice, for seismic analysis, design, testing and qualification of SSCs in order that they meet the applicable safety requirements established in Refs [1–3].

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design and Qualification for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.6, IAEA, Vienna (2003).

1.9. This Safety Guide is intended for use by organizations involved in the seismic design of nuclear installations, in analysis, verification and review, and in the provision of technical support, as well as by regulatory bodies.

SCOPE

1.10. This Safety Guide addresses all types of nuclear installation, as defined in the IAEA Safety Glossary [6], as follows:

(a) Nuclear power plants;
(b) Research reactors (including subcritical and critical assemblies) and any adjoining radioisotope production facilities;
(c) Storage facilities for spent fuel;
(d) Facilities for the enrichment of uranium;
(e) Nuclear fuel fabrication facilities;
(f) Conversion facilities;
(g) Facilities for the reprocessing of spent fuel;
(h) Facilities for the predisposal management of radioactive waste arising from nuclear fuel cycle facilities;
(i) Nuclear fuel cycle related research and development facilities.

1.11. Recommendations for nuclear power plants are applicable to other nuclear installations by means of a graded approach, whereby these recommendations are applied in accordance with the potential radiological consequences of the failure of the installation when subjected to seismic loads. The recommended graded approach is to start with the recommendations relating to nuclear power plants and adjust these to installations associated with lesser radiological consequences. If no such adjustment is justified, the recommendations relating to nuclear power plants are to be applied.

1.12. This Safety Guide is intended to be applied to the design and construction of new nuclear installations. Assessment of the seismic safety of an existing nuclear installation is beyond the scope of this Safety Guide and should follow the recommendations provided in NS-G-2.13 [5].

STRUCTURE

1.13. The structure of this Safety Guide follows the general workflow of seismic design and qualification. Section 2 describes the safety requirements for addressing external hazards and the effects of seismic events and provides recommendations on general seismic design aspects. Section 3 provides recommendations relating to input for seismic design and qualification, including the design basis earthquake, the data obtained from the site characterization and the seismic categorization of SSCs. Section 4 provides recommendations on good design practices for layout, structures and different categories of component. For each category, the key seismic design issues derived from earthquake experience are identified and current best practice in seismic design is described. Section 5 provides recommendations on seismic analysis, and Section 6 provides recommendations on seismic qualification by analysis, by testing and by indirect methods. Section 7 provides recommendations on assessing the seismic margin to be ensured by the design. Section 8 provides recommendations on seismic instrumentation and suitable monitoring procedures and their relation to design assumptions and post-earthquake actions. Section 9 provides guidance on using the recommendations of this Safety Guide for nuclear installations other than nuclear power plants. Section 10 provides recommendations on the application of the management system and on project management and peer reviews. A list of definitions specific to this Safety Guide is also provided.

2. REQUIREMENTS FOR SEISMIC DESIGN AND GENERAL SEISMIC DESIGN ASPECTS

2.1. Requirements 15 and 16 of IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [7], require that the seismic hazards associated with a site for a nuclear installation be evaluated to serve as an input to the seismic design of the installation.

2.2. The requirements relevant to seismic design for nuclear power plants are established in SSR-2/1 (Rev. 1) [1]. For seismic design for research reactors and for nuclear fuel cycle facilities, relevant requirements are established in SSR-3 [2] and SSR-4 [3], respectively. All of these Safety Requirements publications stress the importance of applying a graded approach. Where no specific safety requirements for seismic design have been established for a particular type of nuclear installation, the requirements established in SSR-2/1 (Rev. 1) [1], SSR-3 [2] and SSR-4 [3] should be applied, as far as practicable, using the graded approach described in Section 9.

EXTERNAL HAZARDS

2.3. With regard to considering external hazards such as earthquakes in the design of nuclear power plants, SSR-2/1 (Rev. 1) [1] states (footnotes omitted in paras 5.17 and 5.21):

> **“Requirement 17: Internal and external hazards**
>
> **“All foreseeable internal hazards and external hazards…shall be identified and their effects shall be evaluated. Hazards shall be considered in designing the layout of the plant and in determining the postulated initiating events and generated loadings for use in the design of relevant items important to safety for the plant.**
>
> “5.15A. Items important to safety shall be designed and located, with due consideration of other implications for safety, to withstand the effects of hazards or to be protected, in accordance with their importance to safety, against hazards and against common cause failure mechanisms generated by hazards.
>
> “5.15B. For multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously.
>
> …….
>
> “5.17. The design shall include due consideration of those natural and human induced external events (i.e. events of origin external to the plant) that have been identified in the site evaluation process. Causation and likelihood shall be considered in postulating potential hazards. In the short term, the safety of the plant shall not be permitted to be dependent on the availability of off-site services such as electricity supply and firefighting services. The design shall take due account of site specific conditions to determine the maximum delay time by which off-site services need to be available.
>
> …….
>
> “5.19. Features shall be provided to minimize any interactions between buildings containing items important to safety (including power cabling and control cabling) and any other plant structure as a result of external events considered in the design.

…….

“5.21. The design of the plant shall provide for an adequate margin to protect items important to safety against levels of external hazards to be considered for design, derived from the hazard evaluation for the site, and to avoid cliff edge effects.

“5.21A. 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.”

Similar provisions for considering external hazards are established in Requirement 19 of SSR-3 [2] for the design of research reactors and in Requirement 16 of SSR-4 [3] for the design of nuclear fuel cycle facilities.

ENGINEERING DESIGN RULES

2.4. SSR-2/1 (Rev. 1) [1] states:

“Requirement 18: Engineering design rules

“The engineering design rules for items important to safety at a nuclear power plant shall be specified and shall comply with the relevant national or international codes and standards and with proven engineering practices, with due account taken of their relevance to nuclear power technology.

“5.23. Methods to ensure a robust design shall be applied, and proven engineering practices shall be adhered to in the design of a nuclear power plant to ensure that the fundamental safety functions are achieved for all operational states and for all accident conditions.”

Similar provisions for engineering design rules and proven engineering practices are established in Requirement 13 of SSR-3 [2] for the design of research reactors and in Requirement 12 of SSR-4 [3] for the design of nuclear fuel cycle facilities.

DESIGN EXTENSION CONDITIONS

2.5. SSR-2/1 (Rev. 1) [1] states:

> **“Requirement 20: Design extension conditions**
>
> **“A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant’s capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.”**

The same provisions for design extension conditions are established in Requirement 22 of SSR-3 [2] for the design of research reactors and in Requirement 21 of SSR-4 [3] for the design of nuclear fuel cycle facilities.

HEAT TRANSFER TO AN ULTIMATE HEAT SINK

2.6. SSR-2/1 (Rev. 1) [1] states that with respect to nuclear power plants:

> **“Requirement 53: Heat transfer to an ultimate heat sink**
>
> **“The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states.**
>
> …….
>
> “6.19B. The heat transfer function shall be fulfilled for levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site.”

There are no equivalent requirements in SSR-3 [2] or SSR-4 [3] in relation to the design of research reactors or nuclear fuel cycle facilities. Consequently, where the design of other nuclear installations needs to include the capability to transfer heat to an ultimate heat sink, a graded approach should be applied using the requirements established in SSR-2/1 (Rev. 1) [1] as a starting point.

CONTROL ROOM

2.7. SSR-2/1 (Rev. 1) [1] states:

> **"Requirement 65: Control room**
>
> **"A control room shall be provided at the nuclear power plant from which the plant can be safely operated in all operational states, either automatically or manually, and from which measures can be taken to maintain the plant in a safe state or to bring it back into a safe state after anticipated operational occurrences and accident conditions.**
>
> ....
>
> "6.40A. The design of the control room shall provide an adequate margin against levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site."

2.8. Similar provisions for the control room are established in Requirement 53 of SSR-3 [2] for the design of research reactors; however, there are no equivalent requirements in SSR-4 [3] for the design of nuclear fuel cycle facilities.

OTHER SEISMIC DESIGN ASPECTS

2.9. The implementation of the relevant safety requirements in the design of a nuclear installation against seismic events should ensure that Principle 8 of IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [8], on the prevention of accidents is applied.

2.10. The seismic design of items important to safety should be based on the seismic hazards determined during the site evaluation process for the nuclear installation, conducted in accordance with the requirements established in SSR-1 [7] and the recommendations provided in SSG-9 (Rev. 1) [4]. Specifically, the site specific vibratory ground motions assessed using deterministic and/or probabilistic approaches should be available and should be used to assess the adequacy of the design basis earthquake for the nuclear installation, as recommended in Section 3 of this Safety Guide.

2.11. Seismic design should consider the influence of the layout of the plant and of the detailed arrangements and layout of SSCs. Specific recommendations are provided in Section 4 of this Safety Guide.

2.12. The following specific aspects should be considered in the seismic design of nuclear installations:

(a) Protection against common cause failure of SSCs in the event of an earthquake affecting all units in a multiple unit site (seismic events can lead to serious challenges to the multiple layers of defence in depth, through common cause failures);
(b) Minimization of seismic interaction effects;
(c) Provision of adequate seismic margins and avoidance of cliff edge effects[2];
(d) Compliance with proven engineering design rules, as specified in relevant national and international codes and standards.

2.13. Special consideration should be given to the need to provide an adequate seismic margin for those SSCs ultimately required to prevent an early radioactive release or a large radioactive release in the event of an earthquake exceeding those considered for design purposes.[3] The recommendations in Section 3 of this Safety Guide are provided to determine the beyond design basis earthquake and the categorization of the SSCs to be designed or evaluated against such an event; the applicable performance criteria are considered in Sections 7 and 9.

2.14. When the recommendations of this Safety Guide are applied to the seismic design of nuclear installations other than nuclear power plants, engineering judgement and a graded approach should be used to assess the applicability of the recommendations, in accordance with the specific safety objectives defined for the type of installation concerned. Further guidance is provided in Section 9.

2.15. The design process for a nuclear installation should be well structured and should be conducted under the rules, procedures and conditions of proper

[2] A cliff edge effect is an instance of severely abnormal conditions caused by an abrupt transition from one status of a facility to another following a small deviation in a parameter or a small variation in an input value.

[3] For seismic events, it is assumed that early warnings are not possible and that there is a high probability of combinations with other seismic induced hazards (e.g. internal fires, floods).

project management. Requirements for the implementation of an integrated management system are established in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [9], and specific recommendations are provided in IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [10]. The seismic design process should be integrated into the management system (see Section 10) and should include adequate peer review.

3. INPUT FOR SEISMIC DESIGN

GENERAL CONCEPTS OF SEISMIC DESIGN

3.1. In the IAEA Safety Glossary [6], design is defined as the process and the result of developing a concept, detailed plans, supporting calculations and specifications for a facility and its parts. Equipment qualification is defined as the generation and maintenance of evidence to ensure that equipment will operate on demand, under specified service conditions, to meet system performance requirements. Seismic qualification refers to a form of equipment qualification that relates to conditions that could be encountered in the event of an earthquake.

3.2. In this Safety Guide, seismic design is the process of designing a nuclear installation to cope with the effects of the hazards generated by a seismic event, in accordance with specified performance criteria and in compliance with the requirements indicated in Section 2. Therefore, seismic qualification is part of the process of seismic design and refers to equipment qualification to comply with these objectives.

3.3. Earthquakes generate several direct and indirect phenomena. These include vibratory ground motions from associated geological and geotechnical hazards, permanent ground deformation (e.g. soil liquefaction, slope instability, tectonic and non-tectonic subsidence, cavities leading to ground collapse, settlements), and concomitant events such as seismically induced fires and floods.

3.4. If the characteristics of some geological and geotechnical hazards are such that satisfactory engineering solutions to protect against them have not been identified, the site should be deemed unsuitable, as recommended in SSG-9 (Rev. 1) [4] and IAEA Safety Standards Series No. NS-G-3.6, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants [11].

3.5. The seismic design process should consider the following steps, which highlight the major tasks involved in the design process:

(a) Defining the design basis earthquake.
(b) Establishing the seismic categorization.
(c) Selecting applicable design standards.
(d) Providing seismically resistant structural systems in accordance with the layout and the functional requirements of the nuclear installation.
(e) Evaluating the seismic demand.
(f) Determining the preliminary design of structural elements based on codes and standards, and providing adequate reinforcement detailing.
(g) Verifying that the seismic demand does not exceed the seismic capacity defined in the preliminary design and adjusting the design if necessary.
(h) The assessment of seismic margins should use realistic and best estimate assessments, and should apply different procedures from the ones used for design purposes [5].

For the design of a typical nuclear installation, each of the above steps will consist of many individual subtasks.

DESIGN BASIS EARTHQUAKE

Required input from the site evaluation process

3.6. The site evaluation process conducted before starting construction of the nuclear installation provides detailed and specific data and information for the characterization of the site and determines the external hazards that might affect the nuclear installation. If a generic seismic design basis is used, it should be shown to envelop the site specific seismic ground motion. Otherwise, the design will need to be reassessed with a design basis earthquake enveloping the site specific earthquake. After the site evaluation process, the following information relating to the need to cope with the effects of seismic events should be provided as input for the seismic design:

(a) The specific seismic hazards at the site, particularly the vibratory ground motion hazards;
(b) The detailed geological, geophysical and geotechnical characteristics of the site with the corresponding information on soil properties [11].

3.7. With regard to para. 3.6(a), the seismic hazard assessment should be available from the specific site characterization, through the application of the methods and approaches recommended in SSG-9 (Rev. 1) [4], including the determination of the parameters (spectral representations and time histories, in horizontal and vertical directions) of the vibratory ground motions at the control point established by the designer, which is usually at the free field ground surface, at the outcrop of bedrock or at any other specified depth in the soil profile.

3.8. If a deterministic approach is used, the site specific seismic parameters — such as peak ground acceleration and spectral representation — should be selected as the maximum credible values of these parameters. The spectral representation should be a smooth broadband spectrum.

3.9. If a probabilistic approach is used, the level of each relevant vibratory ground motion parameter, such as the peak ground acceleration or spectral accelerations, should include the associated annual frequencies of exceedance (e.g. 10^{-3}, 10^{-4} or 10^{-5} per year).

3.10. With regard to para. 3.6(b), the site specific static and dynamic properties of the soil parameters at the site area should be available from the geological, geophysical and geotechnical investigations and the laboratory tests and engineering studies performed during the site characterization process.

3.11. In addition to the geological, geophysical and geotechnical data and soil properties determined during the site characterization process, prior to the construction of the nuclear installation a detailed programme of geophysical and geotechnical investigations should be carried out to complete and refine the assessment of site characteristics, considering the final layout of buildings and structures and their final location within the site area. A differentiation should be made between structures important to safety and structures that are not important to safety, in accordance with their seismic category (see paras 3.32–3.39 and Table 1). A detailed subsurface exploration and testing programme should be prepared accordingly, using either a grid borehole scheme or an alternative borehole scheme suited to the site and the installation under consideration. The grid spacing may vary depending on the geometry of the subsurface characteristics. The uniform grid method can be adopted at a site with relatively uniform soil conditions. Where dissimilarities and discontinuities are present, the usual exploration process should be supplemented with boreholes at spacings small enough to permit detection of features and their proper evaluation.

3.12. As a result of the geological, geophysical and geotechnical investigations conducted at the site area and at the location of the buildings and structures of the nuclear installation, the following data should be available:

(a) Static and dynamic soil properties, for example unit weight (γ) and/or density (δ), strength capacity in drained and/or undrained conditions, low-strain shear wave (V_s) and primary wave (V_p) velocities, variation of shear modulus (G), and damping ratio as a function of shear strain levels. The data should include the variation of these properties with depth, together with an indication of the types of soil and rock encountered down to the bedrock level. A number of soil profiles should be developed to adequately represent the range of ground conditions and variations encountered at a given site. The profile is usually defined as a vertical section of horizontal layers of ground, with best estimate (mean) values of layer thickness, shear wave velocity and unit weight, and the shear modulus and damping ratio as a function of shear strain level. The use of horizontally layered soil profiles should be justified by the results of site investigations or sensitivity studies. The level or levels of the groundwater should be also determined.
(b) The variability of the thicknesses and ground layer properties to determine the following:
    (i) The best estimate, upper bound and lower bound strain compatible soil profiles, taking into account the uncertainties in soil layer geometry and soil properties; or
    (ii) The full probability distributions of the soil parameters, if the subsequent site response analysis is to be fully probabilistic.

Final site response analysis for the seismic hazard assessment

3.13. The seismic hazard assessment performed during the site evaluation process should include a preliminary site response analysis as recommended in SSG-9 (Rev. 1) [4], based on the types of soil at the site area. Later, at the design stage, a final site response analysis should be performed based on detailed data and information specific to the final location of the structures of the nuclear installation. The final vibratory ground motions should be assessed at the control point specified by the designer of the evaluation and based on the seismic hazard assessment performed at the bedrock level.

3.14. In performing seismic site response analyses as defined in NS-G-3.6 [11], the following site categorization is used:

(a) Type 1 sites: $V_s > 1100$ m/s;[4]
(b) Type 2 sites: 1100 m/s $> V_s >$ 300 m/s;
(c) Type 3 sites: $V_s < 300$ m/s

where V_s is the best estimate shear wave velocity in the foundation medium just below the foundation level of the structure in the natural condition (i.e. before any site work) for very small strains. The site categorization is valid assuming that the shear wave velocity does not decrease significantly with depth; if this is not the case, particular analyses should be carried out in accordance with best practice.

3.15. Seismic site response analysis should be performed for Type 2 and Type 3 sites. Type 1 is normally considered a rock site, and a site response analysis is not necessary if it can be demonstrated that modifying the control point of seismic motion has a negligible effect. Type 3 sites (soft soil conditions) involve detailed studies and site response analyses, as described in NS-G-3.6 [11]. [5]

3.16. As indicated in SSG-9 (Rev. 1) [4], there are two approaches to properly considering the geological and geotechnical specific soil conditions at a site as part of the estimation of the seismic vibratory ground motion. The first approach is to use ground motion prediction equations appropriate for the specific site soil conditions (i.e. equations that have been developed for subsurface conditions of the same type as at the site). The second approach is to conduct a site response analysis using the seismic input provided at bedrock or some other specified depth in the soil–rock column under the site. A site response analysis should be conducted that is compatible with the detailed and specific geotechnical and dynamic characteristics of the soil and rock layers at the site. The decision on which approach to take should therefore be based on the ground motion prediction equations used to calculate the seismic vibratory ground motion parameters at the site.

3.17. If the first approach described in para. 3.16 is used, the resulting vibratory ground motion parameters at the free surface of the top of the soil profile will also be the parameters used to define the seismic hazard design basis for the nuclear installation. If the second approach is used, a step-by-step procedure should

[4] The definition of ‘rock’ varies between States. In some States, a site is considered to be a rock site when the average shear wave velocity is larger than about 2000 m/s.

[5] For example, some States recommend not using Type 3 (soft soil) sites.

be applied to determine the final seismic vibratory ground motion at the site, including all parameters (spectral representations and time histories, in horizontal and vertical directions) at the specified control point (usually at free field ground level, at engineering rock or at another specified depth in the soil profile, such as the foundation level), as follows:

The best estimate soil profile parameters and uncertainties, based on the geophysical and geotechnical databases, should be determined for the full depth, from the bedrock to the free surface at the site. The parameters should be characterized either by best estimate, upper bound and lower bound values, or by probability distributions. This involves determining the mean values, and their uncertainties, for the following parameters for each soil layer:

(a) Low-strain shear wave velocity ($V_{S,}$ V_P);
(b) Strain dependent shear modulus reduction and hysteretic damping properties;
(c) Soil density;
(d) Layer thickness.

It should be determined whether one dimensional equivalent linear analyses should be performed or whether more complex approaches are needed.

Starting with the seismic hazard curves and associated response spectra obtained at the bedrock outcrop layer, site amplification factors should be calculated through convolution of the bedrock hazard curves for each spectral frequency of interest, so that the site amplification factors mimic the characteristics of the principal contributors to the disaggregated seismic hazard, including diffuse seismicity.

The mean uniform hazard response spectra should be developed at the identified locations of interest for the nuclear installation site and for the annual frequencies of exceedance selected to define the seismic design basis (e.g. 10^{-4} or 10^{-5} per year). The final design vibratory ground motion should be developed with margins (sufficient conservatism) to ensure that uncertainties have been properly considered.

If possible, the site response analysis results should be verified using records of observed measurements and/or microtremor surveys.

Determination of the design basis earthquake

3.18. As one of the first steps at the design stage of the nuclear installation, the design basis earthquake should be determined. The design basis earthquake is used to define the level of the seismic vibratory ground motion hazards to be taken into account in the design of the SSCs of the nuclear installation; it is based on the results of the assessment of the site specific seismic vibratory ground motion. If a generic seismic design basis is used, it should be shown to envelop the site specific seismic ground motion. In general, two levels of seismic vibratory ground motion hazard, SL-1 and SL-2, should be defined as the design basis earthquake for each nuclear installation. This is to ensure the safety of the nuclear installation in the event of a rare earthquake (i.e. SL-2) and to ensure the possibility of continued operation in the event of a less severe, but more probable, earthquake (i.e. SL-1). In some cases, depending on the site conditions (e.g. areas of low seismic activity) and national regulations, one level of seismic vibratory ground motion hazard may be defined for design purposes.

3.19. The SL-2[6] level is defined as the vibratory ground motion for which certain SSCs of the nuclear installation should be designed to perform their safety function during and/or after the occurrence of a seismic event of such intensity. For SSCs sensitive to low frequency motions (e.g. SSCs on isolators) and high frequency motions, time histories and response spectra should be examined and, if necessary, should be modified to take these effects into account.

3.20. The SL-1[7] level corresponds to a less severe, more probable earthquake than SL-2. The SL-1 earthquake level could reasonably be expected to occur and to affect the nuclear installation during its operating lifetime. As such, SSCs necessary for continued operation should be designed to remain functional in the event of an SL-1 earthquake.

3.21. The SL-2 level is defined on the basis of the results and parameters obtained from the seismic hazard assessment (see para. 3.7), in accordance with specific criteria established by the regulatory body to achieve a certain target level for the annual frequency of exceedance for SL-2. The SL-2 level should be characterized by both horizontal and vertical vibratory ground motion response spectra at the control point defined by the designer.

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[6] In some States, SL-2 corresponds to an earthquake level often denoted as the ‘safe shutdown earthquake’.

[7] In some States, SL-1 corresponds to an earthquake level often denoted as the operating basis earthquake.

3.22. If a probabilistic approach was used for the seismic hazard assessment, SL-2 typically corresponds to a level with an annual frequency of exceedance in the range of 10^{-3} to 10^{-5} (mean values), depending on the national regulatory approach. Thus, using the seismic vibratory ground motion hazard curves and uniform hazard response spectra obtained for such an annual frequency of exceedance (see para. 3.9), the SL-2 level should be calculated with due consideration of additional margins and rounding aspects.[8]

3.23. If a deterministic approach was used for the seismic hazard assessment, an estimate of the associated return period of the calculated earthquake level should be made. This estimate should be sufficient to at least allow a comparison with national standards for the design of conventional installations.

3.24. The design basis earthquake level should include adequate design conservatism. This conservatism is necessary to take into account the uncertainties associated with peak ground acceleration and spectral shape, based on the results of the seismic hazard assessment.

3.25. SL-1 typically corresponds to a level with an annual frequency of exceedance in the range of 10^{-2} to 10^{-3} per year (mean values). However, in practice, the SL-1 level is usually defined as a percentage of the SL-2 level, with appropriate consideration of its application in the design and operation stages.

3.26. Irrespective of the site specific seismic hazards, a new nuclear installation should be designed to withstand a minimum earthquake level. In this regard, considering (a) the advances in the development of the design of nuclear installations, (b) the uncertainties in the seismic hazard assessment and (c) the effectiveness in terms of cost and technical provisions of providing a high level of assurance against the seismic hazards from the conception phase of the installation, the minimum level for seismic design (SL-2) should correspond to a peak ground acceleration of 0.1g at the free field or foundation level (where g is the acceleration due to gravity) and should be not less than the values established by the national seismic codes for conventional installations. This leads to a generally more robust design of the nuclear installation, which also increases the safety margin with regard to other dynamic loads.

[8] In some States that use a performance based approach to define a site specific SL-2 level, the ground motion level is calculated by scaling the site specific mean uniform hazard spectrum by a design factor greater than 1.

BEYOND DESIGN BASIS EARTHQUAKE

3.27. In addition to the earthquake levels, SL-1 and SL-2, defined and determined for design purposes (see paras 3.18–3.26), a more severe earthquake level — derived from the hazard evaluation of the site — should be considered: see Requirements 17 and 20 of SSR-2/1 (Rev. 1) [1], Requirement 22 of SSR-3 [2] and Requirement 21 of SSR-4 [3]. For this earthquake level, referred to as the 'beyond design basis earthquake', the following applies:

(a) The design should provide adequate seismic margins for those SSCs ultimately required to prevent core damage and to prevent an early radioactive release or a large radioactive release.
(b) The design should provide adequate seismic margins to the safety classified SSCs credited for mitigatory measures for Level 4 of the defence in depth concept.
(c) It should be demonstrated that cliff edge effects are avoided within the uncertainty associated with the definition of SL-2.

3.28. A new nuclear installation should be designed against a design basis earthquake in accordance with specific design performance criteria, and it should be verified that the safety requirements quoted in para. 2.3 would be achieved in the event of a beyond design basis earthquake.

3.29. The beyond design basis earthquake and the associated loads can be determined by one of the following methods:

(a) Defining the beyond design basis earthquake level in terms of the SL-2 level multiplied by a factor[9] agreed by the regulatory body;
(b) Defining the beyond design basis earthquake level on the basis of considerations derived from the probabilistic seismic hazard assessment;[10]
(c) Defining the beyond design basis earthquake level on the basis of the maximum credible seismic hazard severity.

3.30. The beyond design basis earthquake level should be characterized by both horizontal and vertical vibratory ground motion response spectra, anchored to a

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[9] For low to moderate seismicity where the seismic margin is used to assess the robustness of the design, some States define a factor of 1.4, 1.5 or 1.67.

[10] This implies an annual frequency of exceedance lower than the one used to define the SL-2 level. In some States, mean values for the annual frequency of exceedance in the range 1×10^{-5} to 1×10^{-4} are used.

peak ground acceleration (i.e. at the zero period of the response spectrum) and at the control point defined by the seismic hazard assessment.

SEISMIC CATEGORIZATION FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.31. Seismic categorization is the process by which an item (i.e. an SSC) of the nuclear installation is assigned to a seismic category in accordance with its intended performance during and after the occurrence of an earthquake event, in addition to other classifications such as those relating to safety, quality and maintenance. The relevant acceptance criteria associated with the item are part of the categorization.

3.32. The items of the nuclear installation should be grouped into three seismic categories, as follows:

(a) Seismic category 1;
(b) Seismic category 2;
(c) Seismic category 3.

3.33. Seismic category 1 includes the items that need to remain functional during and/or after the occurrence of the SL-2 design basis earthquake. An item in seismic category 1 should maintain its functionality and/or structural integrity (depending on functional requirements) during and/or after the occurrence of the SL-2 design basis earthquake, and an adequate seismic margin should be provided to avoid cliff edge effects. Seismic category 1 should include the following items:

(a) Items whose failure could directly or indirectly cause accident conditions;
(b) Items that are necessary for shutting down a reactor and maintaining a reactor in a safe shutdown condition, including the removal of decay heat;
(c) Items that are necessary to prevent or mitigate unintended radioactive releases, including SSCs in spent fuel storage pool structures and fuel racks;
(d) Items that are necessary to mitigate the consequences of design extension conditions and whose failure would result in consequences of a high level of severity, as defined in para. 3.11 of IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [12];
(e) Items that are part of support, monitoring and actuating systems that are needed to fulfil the functions indicated in (b)–(d) above.

3.34. Physical barriers designed to protect the installation against the effects of internal or external hazards other than seismic hazards (e.g. fires, floods) should remain functional and maintain their integrity after an SL-2 earthquake.

3.35. For any items in seismic category 1, appropriate acceptance criteria[11] should be established on the basis of acceptable values for design parameters (e.g. performance targets, limit states) indicating, for example, functionality, leaktightness, maximum distortion or deformation, or maximum stress level.

3.36. Seismic category 2 includes those items whose failure to perform their intended functions would impede or affect any of the safety functions performed by seismic category 1 items. Seismic category 2 should include the following items:

(a) Items that might have spatial interactions (e.g. due to collapse, falling or dislodgement) or any other earthquake induced interactions (e.g. earthquake break of a pipe that is not important to safety resulting in spraying water onto electrical equipment that is important to safety) with items in seismic category 1, including effects on any safety related action by personnel at the installation;
(b) Items not included in seismic category 1 that are necessary to mitigate design extension conditions;
(c) Items relating to the infrastructure needed for the implementation of the emergency evacuation plan.

3.37. Items in seismic category 2 should be designed to withstand the effects of an SL-2 earthquake. Alternatively, a technical basis demonstrating that spatial interactions or other reactions would not impede or affect any of the safety functions performed by seismic category 1 items should be provided.

3.38. Seismic category 3 should include all items that are not in seismic category 1 or seismic category 2. The items in seismic category 3 should, at a minimum, be designed in accordance with the national approach to the seismic design of high risk conventional (i.e. non-nuclear) installations. For some items in seismic category 3 that are important to the operation of the installation, it may be preferable to select a more severe seismic loading corresponding to SL-1 and adopt more stringent acceptance criteria than those for conventional installations. Such an approach

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[11] In this Safety Guide, acceptance criteria are specified bounds on the value of a functional or condition indicator for an SSC in a defined postulated initiating event (e.g. an indicator relating to functionality, leaktightness or non-interaction).

will minimize the need for shutdown, inspection and restart of the installation, thus allowing the installation to continue to operate after an earthquake.

3.39. The relationship between the safety classes defined in SSG-30 [12] and seismic categories 1–3 is shown in Table 1. The inclusion of an item in a seismic category should be based on a clear understanding of the safety functions that are required to be fulfilled during and/or after an earthquake. In accordance with their different functions and their functional safety categories, parts of the same system may belong to different seismic categories. Leaktightness, degree of damage (e.g. fatigue, wear and tear), mechanical or electrical functional capability, maximum displacement, degree of permanent distortion, and preservation of geometrical dimensions are examples of aspects that should be considered and determined as input for the seismic design to establish the limiting acceptable conditions.

TABLE 1. RELATIONSHIP BETWEEN SAFETY CLASSES AND SEISMIC CATEGORIES

Safety class [12]	Seismic category	Remarks
1 2 3	1 1 or 2 1 or 2	Seismic category 1 items need either structural integrity, leaktightness, functionality, or their combinations, as appropriate. Seismic category 2 items need either structural integrity, leaktightness, or their combinations, as appropriate. Functionality is needed only if its absence might degrade the functions of seismic category 1 items. Both SL-1 and SL-2 should be used as prescribed by applicable national regulations and relevant design codes for nuclear installations.
Not classified	3	For items that are not safety classified, it should be ensured that their seismic failure will not produce interactions that affect safety classified items. The national approach to the seismic design of non-nuclear installations should apply.

3.40. As one of the first steps in the design process, a detailed list of all items in the nuclear installation should be produced, with an indication of their safety class and seismic category and the associated acceptance criteria.

SELECTION OF SEISMIC DESIGN AND QUALIFICATION STANDARDS

3.41. Once the seismic categories of the items in the nuclear installation have been established, corresponding engineering design rules should be specified. Engineering design rules are based on relevant national or international codes, standards and proven engineering practices and should be applied, as appropriate, to the seismic design of items in each seismic category.

3.42. Experience from the design and construction of nuclear installations indicates that codes, norms and standards of different origin (i.e. different country or different type of installation) are often used. Even within a State, codes or standards for the different design disciplines (i.e. mechanical, civil and electrical) are not always based on compatible safety criteria. Therefore, consistent acceptance criteria should be established, and good engineering practices should be used, to provide consistency in the application of selected codes and standards in seismic design.

3.43. At the beginning of the design stage, an analysis and evaluation of the codes, norms and standards to be applied in the design, fabrication and construction of the nuclear installation should be performed. The results of this analysis and evaluation should be documented as part of the management system (see Section 10).

4. SEISMIC DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

4.1. All procedures for seismic design should be based on a good understanding of the consequences of past destructive earthquakes, and this knowledge should be adopted and realistically applied. The recommendations in this section are derived from past experience and the observed performance of similar items, mainly in conventional industrial installations, when affected by earthquakes. These recommendations should be considered at the preliminary design stage.

LAYOUT OF THE INSTALLATION

4.2. The layout of the installation should be established early in the design stage of the installation and should aim to achieve the most suitable solution for the seismic design.

4.3. In the preliminary design stage, seismic effects (in terms of forces and undesired torsional or rocking effects) should be minimized by applying the following criteria:

(a) The centre of mass of all structures should be located at as low an elevation as practicable.
(b) The centre of rigidity at the various elevations should be located as close as practicable to the centre of mass to minimize torsional effects.
(c) Building plans and elevation layouts should be selected that are as simple and regular as practicable, with direct and clear paths for the transmission of seismic forces to the foundation.
(d) Different embedment depths of adjacent buildings should be avoided, as far as practicable.
(e) Buildings with large plan aspect ratios should be avoided. Plan aspect ratios should be as close to 1 as practicable, and large aspect ratios should be avoided.
(f) Protruding sections (i.e. lack of symmetry) should be avoided, as far as practicable.
(g) Rigid connections should be avoided between different building structures or between equipment of different seismic categories and dynamic behaviours.[12]
(h) The diversity of SSCs belonging to redundant safety trains should be properly considered in order to reduce potential common cause failures.

4.4. Adequate gap dimensions and seismic margins should be ensured in the design of the structural joints between adjacent structural parts or between adjacent buildings to avoid pounding and hammering.

[12] An example is the containment vessel and the surrounding internal concrete structures: if they are connected, they could interact during the earthquake.

4.5. Structural systems for buildings of nuclear installations should possess adequate strength and ductility, and, where necessary, they should provide a confinement function. The following structural systems should be considered acceptable for structures in any seismic category:

(a) Structures made of reinforced concrete shear walls that provide a lateral-force-resisting system;
(b) Steel or reinforced concrete moment-resisting frames specially designed to provide ductile behaviour;
(c) Reinforced concrete slab or wall moment frames.

For structures in seismic category 1 and safety class 2 or safety class 3, adequate stiffness should be provided to limit deformation in order to avoid excessive cracking or displacement that might affect attached equipment.

4.6. The following structural systems should be avoided in structures in seismic category 1 and seismic category 2:

(a) Ordinary moment-resisting frame systems (i.e. no special design details to provide ductile behaviour);
(b) Unreinforced concrete systems;
(c) Precast concrete systems with gravity-only bearing connections;
(d) Unreinforced masonry systems;
(e) Wooden structures.

4.7. The design of structures should favour ductile failure modes rather than brittle failure modes. In this regard, the following should be considered at the design stage:

(a) In reinforced concrete structures, brittle failure in the shear and/or bond of rebars or in the compressive zones of concrete should be prevented.
(b) For reinforcement, an appropriate minimum ratio of the ultimate tensile stress to the yield tensile strength should be defined to ensure a minimum ductility.
(c) The lengths for reinforcing bar anchorage should generally be longer than the lengths for structures under static or non-reversing loads.
(d) In steel structures, brittle failure should be avoided.
(e) Structural joints, particularly in reinforced concrete structures, should be designed to accommodate ductile displacements and rotations. This

provision should be consistent with the acceptance criteria specified in the seismic categorization, and it should also take into account the need for adequate seismic behaviour in design extension conditions.

(f) Sufficiently wide gaps should be provided between structures above ground level to avoid interaction (pounding) during seismic motion. Utilities crossing the gaps should be able to accommodate differential seismic displacements. However, if such interaction between structures could occur, the structural integrity should be confirmed.

4.8. Structures in seismic category 1 should be designed to exhibit linear behaviour. Limited non-linear behaviour might be permissible, provided that the acceptance criteria for the structures are met. Ductile behaviour is needed to develop adequate seismic margins.

4.9. Structures in seismic category 2 should be designed to exhibit non-linear behaviour, especially to provide adequate seismic margin capacity. Elements of structural members, particularly joints and connections, should be consistent with the acceptance criteria.

4.10. Non-structural elements of the buildings, such as partition walls, ceilings and roofing, should be designed so that they do not collapse and fall onto seismic category 1 items.

4.11. The potential for overturning and lateral sliding of the structure during an earthquake should be assessed. The effects of waterproofing material, if any, should be considered in the evaluation of lateral sliding.

4.12. Massive mat foundations associated with nuclear buildings are generally seismically robust and should be employed to separate foundations for individual buildings.

ENGINEERED EARTH STRUCTURES AND BURIED STRUCTURES

4.13. The seismic design of engineered earth structures and buried structures should be consistent with the seismic category and should comply with the recommendations provided in SSG-30 [12].

4.14. The following engineered earth structures important to safety may be encountered at nuclear installation sites:

(a) Earth structures related to ultimate heat sinks: dams, dykes and embankments;
(b) Site protection structures: dams, dykes, breakwaters, sea walls and revetments;
(c) Site contour structures: retaining walls, natural slopes, cuts and fills.

4.15. The seismic design of earth structures and buried structures should take into account the following seismic related effects:

(a) Slope failure induced by design basis vibratory ground motions, including liquefaction;
(b) Failure of buried piping or seepage through cracks induced by ground motions;
(c) Overtopping of the structure due to tsunamis on coastal sites, seiches in reservoirs, earth slides or rock falls into reservoirs, or failure of spillway or outlet works;
(d) Overturning of retaining walls.

SEISMICALLY ISOLATED STRUCTURES

4.16. The most common application of seismic isolation is to reduce the response of a structure to horizontal ground motion through the installation of a horizontally flexible and vertically stiff layer of seismic isolation devices (e.g. isolators, bearings) between the superstructure and its substructure. As a basic rule, the horizontal stiffness of the isolators should be chosen so that the fundamental vibration frequency of the isolated structural system is significantly lower than that of the original, non-isolated structure.

4.17. Isolators should be seismically qualified using full scale testing of prototypes. At a minimum, the prototypes should be tested and subjected to the maximum displacements considered in the design and for the beyond design basis earthquake. The test should provide data on the following properties used in the structural analysis:

(a) Initial stiffness, as a function of frequency;
(b) Post-yield stiffness, as a function of frequency;
(c) Damping provided by the isolation device, as a function of frequency and/or of maximum displacement.

4.18. Regarding the superstructure, an isolated structure needs a structural diaphragm above the plane of isolation (upper basemat). This diaphragm should be stiff enough to redistribute lateral loads from the superstructure to the isolation system.

4.19. The same layout rules should be applied to an isolated building as to a fixed base building, even though the seismic demand on the superstructure is likely to be smaller in the case of the isolated building. In particular, a regular distribution of mass and stiffness should reduce torsional motions, and a continuous load path should avoid localized high seismic demands. The uplift of seismically isolated structures off the isolators should be prevented by limiting the height-to-width aspect ratio of the superstructure.

4.20. The design of isolation systems should consider the following:

(a) Ensuring uniformity of load and displacement. Ideally, all isolators should be of the same type, should be under the same gravity load and should sustain the same horizontal displacement during an earthquake.
(b) Avoiding, or at least minimizing, uplift.
(c) Avoiding ultimate deformations in isolators being exceeded during earthquakes more severe than the design basis earthquake.
(d) Allowing in-service inspection and replacement of each individual isolator.
(e) Ensuring that the qualification conditions of isolators are consistent with the anticipated operating environmental conditions.
(f) Ensuring that the environmental conditions do not present hazards such as fire at the level where isolators are located.
(g) Avoiding detrimental effects to collocated SSCs that protect against other external hazards.

4.21. The substructure, the isolator pedestals (plinths) and the common footing (lower basemat) should be designed to resist not only gravity and seismic loads but also the moments induced by lateral displacements of the isolator system, including P-Delta effects[13]. The design of the lower basemat should also take into account the effect of seismic wave propagation.

4.22. A clearance space (gap) should be provided around the perimeter of the upper basemat to allow for large lateral movements of the isolated structure. Usually, the isolation system is set below ground level and the gap takes the form

[13] The P-Delta effect is a second order bending moment equal to the force of gravity multiplied by the horizontal displacement a structure undergoes when loaded laterally.

of a moat. The width of such a moat should correspond to the ultimate allowed lateral displacement of the isolation system and be correlated with the maximum expected displacement induced by the beyond design basis earthquake.

4.23. The seismic design should allow for enough flexibility of attached distribution lines (e.g. electrical cables, piping) to accommodate expected differential displacements between the equipment item and the first support of the line. Special provisions should be made for all utility lines (umbilicals) crossing the clearance space described in para. 4.22. The lines should be flexible enough to accommodate the displacements of the isolation system in any horizontal direction.

MECHANICAL EQUIPMENT

4.24. The seismic qualification of mechanical equipment should take into account the seismic categorization (see paras 3.31–3.40). Experience from the effects of earthquakes on industrial facilities shows that most of the reported failures of mechanical equipment are associated with a lack of anchorage or with insufficient capacity at the anchorage. The positive anchorage of mechanical equipment to the main structure of the building should be considered the key aspect in seismic design.

4.25. The seismic design of the anchorage should take into account the following:

(a) The full load path from the base of the equipment to the main structure should have sufficient capacity and stiffness so that the natural frequencies[14] of the installed component are not significantly reduced.
(b) The seismic demand at each support point should be computed from the in-structure response spectra using the quasi-static method or response spectrum method, with the level of damping accepted by the design standard for each particular equipment class. Simplified conservative approaches are acceptable, provided these are justified.
(c) Nozzle loads should be considered when computing the seismic demand.
(d) Prying action at base plates should be avoided by an appropriate positioning of fastenings (e.g. to avoid large eccentricities in the load path).
(e) Any parts of the load path that are prone to brittle failure should be oversized to ensure ductile controlling failure modes (e.g. in cast-in-place bolts, the failure should take place at the bolt, not at the concrete).

[14] The natural frequency is the frequency of vibration of a linear dynamic system when it is not disturbed by any external dynamic forces.

(f) Mixing different types of fastening for the anchorage of the same component (e.g. welding and expansion anchors) is not acceptable unless it can be shown that the stiffness of the different fastenings is similar.
(g) The flexibility of base plates can significantly alter the distribution of anchor forces compared with the results computed with the common rigid plate assumption. This is especially relevant when brittle failure modes are involved (e.g. pull out of expansion anchors). In such cases, the design should give consideration to the base plate flexibility.
(h) The preferred anchorage types are the following:
    (i) Cast-in-place bolts or headed studs;
    (ii) Welding to embedded plates;
    (iii) Undercut type expansion anchors.
(i) Expansion anchors other than the undercut type normally should not be used for rotating or vibrating equipment or for sustained tension supports.

4.26. When a vibration isolation device is used to support a seismic category 1 component, the seismic capacity of the isolation device should be demonstrated. In such cases, it is good practice to install limiters (bumpers) so that the maximum allowable lateral displacements are not exceeded.

4.27. The design should allow for enough flexibility of attached lines (e.g. electrical cables, piping) to accommodate expected differential displacements between the equipment item and the first support of the line.

STORAGE TANKS

4.28. Above ground vertical storage tanks are vulnerable during earthquakes, especially when they are either unanchored or only lightly anchored. The design of this type of tank should give consideration to the following points:

(a) Calculation of seismic demand should take into account the flexibility of the tank shell and its influence on the natural frequencies of the tank.
(b) A conservative freeboard should be provided to avoid damage to the roof due to sloshing of the fluid.
(c) Unanchored tanks might have large uplifts and instability failures, which can rupture attached lines and cause a loss of contents of the tank. Consequently, unanchored tanks are not usually acceptable as seismic category 1 items.
(d) The seismic capacity of the foundations of the tank should be appropriately verified, especially for ring type foundations. The assessment should be consistent with the capacity assessment of the tank shell and the anchorage.

(e) The global stability of the tank in terms of the potential for overturning and sliding should be assessed.
(f) The design of attached lines should allow for differential displacements between the tank and the first support, consistent with the design of the anchorage (i.e. the placing of supports very close to the tank should be avoided).

PIPING

4.29. In accordance with accepted engineering practice and regulatory requirements, the seismic design of piping in nuclear installations is usually done by analysis and in accordance with national or international piping design codes. In addition to such an analysis, the seismic design should take into account the following to the extent possible:

(a) Pipe materials should be ductile at service temperatures (total elongation at rupture greater than 10%). Carbon steel and stainless steel are examples of ductile materials at the usual range of operating fluid temperatures in a nuclear installation; grey cast iron and PVC are examples of brittle materials.
(b) Joints that rely only on friction should be avoided.
(c) Vertical supports should not be excessively spaced. Guidelines from established national and/or international design codes should be followed.
(d) Pipe supports should be able to withstand a seismic event without brittle failure and without loss of the restraint of the pipe.
(e) When flexible joints (e.g. bellows) are used, the movement of the pipe at both sides of the joint should be restrained to keep relative end movements during a seismic event within vendor specified limits.
(f) Piping should be sufficiently restrained in the lateral direction.

4.30. Piping that is anchored to two different buildings (or different substructures within a building) or that enters a building from underground should be sufficiently flexible to accommodate the differential motion of the attachment points at both sides.

BURIED PIPES

4.31. Buried pipes are a special type of piping that is continuously supported by the ground. The design should follow the recommendations provided in section 6 of NS-G-3.6 [11]. The main seismic design principle for this kind of piping is to make it sufficiently flexible to follow the ground deformation during seismic shaking.

4.32. The design of buried pipes should pay attention to penetrations into buildings or other structures and should ensure that there is enough flexibility to allow for the expected differential displacements between the ground and the structures to which the piping is connected.

ELECTRICAL EQUIPMENT, CONTROL AND INSTRUMENTATION

4.33. Electrical equipment (e.g. cabinets, motors, transformers, similar equipment) should be seismically qualified by analysis, testing, a combination of analysis and testing, or similarity (see para. 6.3) if it is needed to function during and/or after an earthquake.

4.34. Qualification tests made on equipment do not always include the full load path of the anchorage to the main structure. Hence, any portion of the load path that is not covered by the test should be designed and assessed separately. The seismic design should take into account the following:

(a) The full load path from the base of the equipment to the main structure should be considered.
(b) The load path should have enough capacity and adequate stiffness.
(c) Prying action at base plates should be avoided by an appropriate positioning of fastenings (e.g. avoiding large eccentricities in the load path).
(d) The portions of the load path prone to brittle failure should be oversized to ensure ductile controlling failure modes (e.g. in cast-in-place bolts, the failure should take place at the bolt, not at the concrete).
(e) The preferred anchorage types are the following:
    (i) Cast-in-place bolts or headed studs;
    (ii) Welding to embedded plates;
    (iii) Undercut type expansion anchors.
(f) For motor control centres, transformers, inverters, switchgear and control panels, the use of top bracing or lateral ties should be considered to limit the differential displacements imposed on cables, conduits and bus ducts.

4.35. When a vibration isolation device is used to support a seismic category 1 component, the seismic capacity of the selected device should be demonstrated.[15] In such cases, it is good practice to install limiters (bumpers) in order not to exceed the maximum allowable lateral displacement.

4.36. The design should allow for enough flexibility of attached electrical cables to accommodate expected differential displacements between the equipment item and the first support of the distribution system.

4.37. Adjacent panels, cabinets and racks should be connected together or sufficiently separated to prevent pounding interactions. This is particularly important for equipment containing relays susceptible to chatter and for items sensitive to damage from impact or impulse loading.

4.38. The design should ensure functionality of the instrumentation and control devices to avoid spurious signals during seismic shaking.

4.39. The seismic design aspects relating to batteries and racks should ensure that the following are properly addressed:

(a) The lateral and transverse stiffness of the racks;
(b) Overturning stability;
(c) Anchorage to the rack supporting structure;
(d) Adequacy of spacers between the batteries and use of shims at the ends of the battery rows.

4.40. Heavy batteries and transformers should be anchored directly to the floor or mounted on independent supports inside cabinets and panels to avoid interaction with other components.

CABLE TRAYS AND CONDUITS

4.41. In accordance with accepted engineering practice, the seismic design of electrical raceway distribution systems in nuclear installations is done by analysis,

[15] Vibration isolation devices not designed for earthquake loads have failed during earthquakes affecting industrial facilities.

following a national or an international design code. In addition, the seismic design should comply with the following basic rules:

(a) Limit the span of cable trays;[16]
(b) Limit the span of conduit;
(c) For cantilever bracket-supported raceways, fasten cable trays and conduits to their supports so that they cannot slide and fall off the supports;
(d) Ensure that supports can withstand the design basis earthquake levels (SL-1 or SL-2, as applicable) with adequate margins against brittle failure.

4.42. Suspended electrical raceways (i.e. cable trays and conduits) are generally seismically adequate owing to a self-equilibrating configuration, high damping, and slip and friction at bolted connections. The amount of cable tray fill should be limited to ensure acceptable stresses in supports and connections. Cable ties should be used to limit cable movement. Floor-supported raceways should have bracing for lateral and longitudinal seismic loads.

HEATING, VENTILATION AND AIR-CONDITIONING DUCTS

4.43. In accordance with accepted engineering practice, the seismic design of heating, ventilation and air-conditioning ducts in nuclear installations is usually done by analysis, following a national or international design code. In addition, the seismic design should comply with the following basic rules:

(a) Limit the span of duct supports.[17]
(b) Fasten ducts to their supports (i.e. use duct tie downs) to preclude the possibility of displacing, falling or sliding off during a seismic event. The duct should be securely attached to the last hanger support at the terminal end of the duct run. Similarly, supports designed to limit the lateral movement of the duct system should also be attached to the duct.
(c) Ensure a positive connection at joints.[18]

[16] For the most common tray designs, it is good practice for the span of cable trays between adjacent supports not to exceed 3 m in the direction of the run, as an average. When the cable tray extends beyond the last support in a run, it is installed such that the tray does not cantilever out (overhang) beyond this support by more than 1.5 m.

[17] For the most common duct designs, it is good practice for vertical support spans not to exceed 4.5 m, for supports to be set within 1.5 m of fittings such as tees in each branch of the fitting, and for duct cantilever (overhanging) lengths to be less than 1.8 m.

[18] Ducts with slip joints without pocket locks, rivets or screws could experience joint separation due to the differential displacement between supports.

(d) Ensure positive attachment of appurtenances: accessories attached to heating, ventilation or air-conditioning ducts, such as dampers, turning vanes, registers, access doors, filters or air diffusers, should be positively attached to the duct by means of screws or rivets.
(e) Ensure against brittle failure of supports: supports should be able to withstand the design basis earthquake levels (SL-1 or SL-2, as applicable) with adequate margins against brittle failure.

SEISMIC CAPACITY

4.44. The seismic capacity[19] of an SSC depends on the limiting acceptable condition for its intended functions. The limiting condition should be defined in terms of parameters such as stress, strain, displacement and duration of electrical disturbances. The seismic capacity should be derived from this limiting condition using the appropriate design code. The capacity should be larger than the demand on the SSC (acceptance criterion).

4.45. For seismic category 1 and seismic category 2 SSCs, the acceptance criteria for load combinations should be derived from the applicable nuclear design codes.

4.46. The acceptance criteria for seismic category 3 SSCs should be as stringent as or more stringent than those established by the applicable national standards and codes for normal industrial facilities.

4.47. For seismic capacity calculations, material properties should be selected on the basis of characteristic values (e.g. 95% probability of non-exceedance), supported by appropriate quality assurance procedures.

4.48. Appropriate ageing considerations are required to be taken into account to ensure the long term safety performance of SSCs in seismic category 1 and seismic category 2: see Requirement 31 of SSR-2/1 (Rev. 1) [1], Requirement 37 of SSR-3 [2] and Requirement 32 of SSR-4 [3]. Ageing mechanisms such as radiation embrittlement, fatigue, corrosion, creep and pre-stress losses should be taken into account.

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[19] Seismic capacity is the highest seismic level for which the necessary adequacy has been verified, expressed in terms of the input or response parameter at which the structure or the component is verified to perform its intended safety function.

4.49. The seismic capacities associated with failures of the soil, such as liquefaction or seismically induced settlement, should be determined in accordance with the recommendations provided in NS-G-3.6 [11].

5. SEISMIC ANALYSIS

5.1. Once the layout of buildings and civil structures has been defined and the proportioning of structural members has been undertaken, seismic analysis of these structures should be performed. The purpose of seismic analysis is twofold. First, it provides the parameters of the structural response that are needed to verify the seismic design capacity or to assess the seismic margin (e.g. in terms of stresses, internal forces and moments, and displacements) corresponding to a beyond design basis earthquake. Second, seismic analysis of buildings and civil structures provides information on the seismic demand (e.g. in-structure response spectra, in-structure acceleration or displacement time histories) for the seismic qualification of SSCs housed by these buildings and civil structures.

SITE RESPONSE ANALYSIS

5.2. For soil and soft rock sites, ground (free field) response analysis should be performed with the purpose of obtaining the strain compatible soil profiles to be used in seismic soil–structure interaction analyses and in determining the uncertainties associated with such analyses. Recommendations on site response analysis are provided in Section 3.

5.3. For hard rock sites, it can be assumed that the strains induced by the design basis earthquake will be small, to the extent that stiffness and material damping values in the ground column will not differ from the low-strain values provided by the site investigation campaigns.

STRUCTURAL RESPONSE

5.4. Structural response should be calculated using linear equivalent static analysis, linear dynamic analysis (in time or frequency domain), non-linear static ('pushover') analysis or non-linear dynamic analysis, in accordance with applicable guidelines, codes and standards. Irrespective of the method selected, the following recommendations apply:

(a) The seismic input should be defined either by design response spectra or by acceleration time histories that are compatible with response spectra.
(b) The analysis model should adequately represent the behaviour of the structure under the seismic action and consider a realistic distribution of the mass, stiffness and damping properties of the structure.
(c) The soil–structure interaction should be considered for all safety related nuclear structures not supported by a rock or rock-like soil foundation, taking into account uncertainties in ground properties.
(d) The structural response should be obtained for the three orthogonal components of seismic motion (one vertical and two horizontal).
(e) Potential second order effects, if relevant, should be considered for all vertical load path elements (P-Delta effects). In particular, all vertical load path elements should be designed to withstand the lateral displacements induced by seismic loads.
(f) Hydrodynamic effects should be considered for SSCs containing large volumes of water, for example fuel pools and service pools.

5.5. The structural response can be calculated on the basis of the simultaneous application of the two horizontal components and one vertical component of seismic input, provided that the components of the seismic input are demonstrated to be statistically independent.

5.6. Modelling of stiffness for seismic analysis should follow national and international best practice for nuclear installations. One approach would be an iterative two step process: in the first step of such modelling, the gross area of reinforced concrete sections is used to compute stiffness using linear elastic analysis. Using the stress level identified in this step, stiffness reduction factors are then evaluated for each structural element. The updated stiffness is then used in a second iteration, if necessary.

5.7. In many cases, when soil–structure interaction is considered, the variation of soil properties taking uncertainties into account envelops the variation in structural stiffness due to cracking. Since the two phenomena are independent, the introduction of artificially large uncertainties into the analysis should be avoided by considering the simultaneous occurrence of extremes when bounding the design space.

5.8. For seismically isolated structures, stiffness values for the isolating devices should preferably come from a specific qualification programme, and the variation in the stiffness of the isolators during the design life of the structure should be considered.

5.9. The model used to compute the seismic response should include the mass of the structure, the mass of permanent equipment and the mass of the live load that is expected to be concurrent with seismic loads.

5.10. The damping values used in linear elastic analyses to compute the seismic demand should be mean or median centred. If a non-linear analysis is carried out and incorporates the hysteretic energy dissipation, the damping corresponding to the lower level of response should be used to avoid duplicating hysteretic energy loss.

5.11. For complex structures, consideration should be given to separating the seismic computational model into main structures and substructures. In such cases, major structures that are considered in conjunction with their foundation media to form a soil–structure interaction model are the main systems, whereas the systems and components attached to the main systems are the subsystems.

5.12. Well established decoupling criteria should be used to decide whether a particular subsystem should be taken into account in the analysis of the main system. The decoupling criteria should define limits on the relative mass ratio and on the frequency ratio between the subsystem and the supporting main system.

5.13. A coupled analysis of a primary structure and a secondary SSC should be performed when the effects of dynamic response interaction are significant.

5.14. For the detailed analysis of subsystems, the seismic input, including the motion of differential supports or attachments, should be obtained from the analysis of the main model.

5.15. The in-structure response spectra, typically used as the seismic input for linear or pseudo-linear seismic calculations of equipment components, should be obtained from the structural response to the design vibratory ground motion. For each soil–structure configuration, the number of analyses necessary depends on national practice, but not less than three sets of spectra-compatible acceleration time histories should be used as input for in-structure response spectra generation. Depending on the number of analyses, the resulting in-structure spectra will be either averaged or enveloped to produce the final result.

5.16. To use in-structure response spectra as design seismic input for the SSCs housed by the main structure, the calculated in-structure response spectra should be peak broadened to take into account possible uncertainties in the evaluation of the vibration characteristics of the building's components.[20]

DYNAMIC SOIL–STRUCTURE INTERACTION

5.17. When consideration of soil–structure interaction[21] effects is necessary (see para. 5.2), acceptable models and analysis procedures should first be identified from an assessment of the following aspects:

(a) The purpose of the soil–structure interaction analysis and the intended use of the results (e.g. as input for determining the seismic response of the SSCs);
(b) Relevant phenomena that need to be simulated (e.g. seismic wave fields; linear, equivalent linear and non-linear soil behaviour; linear and non-linear simulation of soil–foundation contact; wave incoherence);
(c) The methodology and software to be used, based on (a) and (b).

For structures containing pools of water large enough to impact the soil–structure interaction effects, the model should incorporate the fluid–structure interaction effect.

5.18. The non-linear constitutive behaviour of the soil should be considered in the soil–structure interaction analyses. This non-linear behaviour may be introduced by equivalent linear soil properties.

5.19. Except for specific sites where significant inclined waves or surface waves may be induced by the soil configuration, the simplifying assumption of vertically propagating seismic waves should be considered acceptable for soil–structure interaction analyses.

5.20. Two main types of method are acceptable for the analysis of soil–structure interaction: direct methods and substructuring methods. Direct methods analyse the soil–structure system in a single step. Direct methods are applicable to

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[20] Typical values used by States are ±15%.

[21] Heavy, stiff structures founded on soft ground might experience significant differences in their seismic response than the same structures founded on rock. These differences may be important even for ground with an intermediate stiffness. This effect is the result of phenomena that are jointly designated as 'soil–structure interaction'.

(equivalent) linear idealizations, and they are commonly used in cases of non-linear interactions of the soil–structure system. Substructuring methods divide the soil–structure interaction problem into a series of simpler problems, solve each problem independently and then superpose the results. Substructuring methods are typically used for linear soil–structure interaction analysis.

5.21. Uncertainties in the soil–structure interaction analyses should be considered, either by the use of probabilistic techniques or by bounding deterministic analyses that cover the expected range of variation of analysis parameters affecting response, including soil properties. In all cases, the variation of soil properties considered in soil–structure interaction analyses should be consistent with the properties used to develop the design input motion (see Section 3).

Direct methods

5.22. Soil–structure interaction analysis by direct methods should include the following steps:

(a) Developing the soil–foundation–structure model, normally using a finite element modelling method;
(b) Locating the bottom and lateral boundaries of the model and assigning appropriate boundary conditions;
(c) Defining the input motion to be applied at the boundaries, compatible with the site response analysis (Section 3);
(d) Performing the analyses and obtaining the necessary response parameters.

5.23. The lower boundary of the soil–foundation–structure model should be located far enough from the soil–foundation interface that the structural response is not affected by the boundary. This lower boundary may be assumed to be rigid.

5.24. Lateral rigid or flexible boundaries should also be located at a sufficient distance from the foundation so that the structural response is not significantly affected by those boundaries. Minimum distances to the soil–foundation interface depend on the type of boundary being selected.

5.25. Soil discretization should be fine enough to produce an accurate representation of all frequencies of interest in the structural response. In addition, at the soil–foundation interface, the level of discretization should be able to accurately model the stress distribution and, if needed, the uplift phenomena, including a consideration of component and equipment frequencies, if these might influence the structural response.

Substructuring methods

5.26. Soil–structure interaction analysis by substructuring methods should include the following steps:

(a) Conducting site response analysis (see Section 3).
(b) Developing the model for the structure, normally using finite elements.
(c) For rigid boundary methods, obtaining the foundation input motion (kinematic interaction or 'wave scattering problem'). 'Rigid boundary' refers to the interface between the foundation and the soil being rigid. The validity of the rigid base assumption, wherever it is employed, should be verified by sensitivity analysis.
(d) Obtaining the foundation impedances using continuum mechanics methods, finite element methods or impedance handbooks.
(e) Analysing the coupled soil–structure system and obtaining the necessary response parameters.

5.27. Implementation details vary depending on the type of substructuring method (e.g. rigid boundary methods, flexible boundary methods, flexible volume methods or subtraction methods). Technical justifications should be provided to demonstrate the adequacy of soil–structure interaction analysis based on the subtraction method.

5.28. For uniform soil sites or for layered soil sites with a smooth variation of properties (e.g. density, shear wave velocity) to a depth equal to the largest dimension of the foundation, the use of frequency independent impedances should be considered acceptable. Frequency dependent impedance functions, together with the natural frequencies of the structure, may be used to develop frequency independent soil springs and dashpots for use in conventional time domain dynamic analysis software. Strain compatible soil properties should be used to obtain the parameters for these springs and dashpots.

Structure–soil–structure interaction

5.29. The designer should assess the potential relevance of the effects of structure–soil–structure interaction[22], based on the following considerations:

(a) Layout of the installation and separation between independent structures.
(b) Soil stiffness and damping.
(c) Differences in footprint and total mass among adjacent buildings. Smaller buildings located close to larger, heavy buildings or underground structures (e.g. tunnels) are of particular concern.

5.30. When structure–soil–structure effects are deemed to be potentially relevant, they should be considered in the design, particularly for the development of in-structure response spectra to be used for qualification of systems and components housed by the main structures.

5.31. Since both the soil and the structures exhibit three dimensional dynamic characteristics, structure–soil–structure interaction is a three dimensional phenomenon. Consequently, to adequately represent the characteristics of both the soil and the structures of the nuclear installation, a three dimensional analysis should be performed to properly characterize this spatial behaviour.

COMBINATION OF EARTHQUAKE LOADS WITH OTHER LOADS

5.32. Design operating condition loads should be grouped as follows:

(a) L1: loads during normal operation;
(b) L2: additional loads during anticipated operational occurrences;
(c) L3: additional loads during accident conditions.

5.33. Seismic loads should be considered for all possible operational states of the nuclear installation. For seismic design, loads from earthquakes (i.e. seismic demand) should be combined with the concurrent loads as follows:

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[22] Structure–soil–structure interaction refers to a phenomenon by which the seismically induced motion of a structure is transmitted to an adjacent structure through the foundation medium. A typical effect of this phenomenon is that, in the in-structure spectra of the affected structure, peaks appear at the natural frequencies of the adjacent structure.

(a) For items in seismic category 1:
    (i) L1 loads should be combined with the demand from the design basis earthquake.
    (ii) L1 and L2 or L3 loads should be combined with the demand from the design basis earthquake if the L2 or L3 loads are caused by the earthquake and/or have a high probability of being coincident with the earthquake loads (which may be the case, for example, for L2 loads that occur sufficiently frequently, independently of an earthquake).
(b) For items in seismic category 2 that have been identified to interact with items in seismic category 1, the same combinations as for seismic category 1 should be applied, possibly associated with different acceptance criteria.
(c) For items in seismic category 3, combinations that are consistent with national practice should be applied to the relevant design basis loads.
(d) The mass of snow should also be considered for sites where design snow load is relevant (e.g. larger than 1.5 kN/m^2).

6. SEISMIC QUALIFICATION

6.1. Seismic qualification is the process of verification — through testing, analysis or other method — of the ability of an SSC to perform its intended function during and/or following the designated earthquake. Seismic qualification should be carried out for seismic category 1 and seismic category 2 components.

6.2. The in-structure design spectra should be used as input for seismic qualification. For equipment installed directly on the ground, the free field response spectra defining the design basis earthquake should be used as input.

QUALIFICATION METHODS

6.3. Seismic qualification should be performed using one or more of the following approaches:

(a) Analysis;
(b) Testing;
(c) A combination of analysis and testing;
(d) Indirect methods (e.g. similarity).

6.4. The qualification programme should ensure that the boundary conditions applied to a component of the nuclear installation correctly or conservatively simulate its behaviour and earthquake conditions. Among these boundary conditions, the most important are excitation conditions, support conditions, environmental conditions, operational conditions and functional requirements.

6.5. As part of the qualification programme for equipment, a systematic evaluation of the possible modes of failure relating to earthquakes should be carried out with reference to the acceptance criteria assigned by the seismic categorization.

6.6. Qualification by analysis should be considered acceptable for passive components and for items of a size or scale that precludes their qualification by testing. Structures, tanks, distribution systems and large items of equipment are usually qualified by analytical methods.

6.7. Seismic qualification of active components should include qualification for structural integrity[23] and qualification for functionality. Seismic qualification should be performed (a) directly on an actual or prototype component; (b) indirectly on a reduced scale model, a reduced scale prototype or a simplified component[24]; or (c) by means of similarity where this can be established between a candidate component and a reference component and direct qualification has been performed on the latter. Irrespective of the method selected, it should accurately represent the actual performance of the component when subjected to the prescribed effects. Testing is limited by the ability of the test rig, or other test conditions, to properly recreate the actual in-service conditions that a component will be subject to. When using test results to qualify components, the extent to which the test process is applicable should be made clear.

6.8. The qualification of active components by analysis is only appropriate when their potential failure modes can be identified and described in terms of stress, deformation (including clearances) or loads. Otherwise, testing or indirect methods should be used for the qualification of active components.

6.9. If numerical models are used to simulate the behaviour of active components during an earthquake, an appropriate validation of such models, and verification

[23] Structural integrity is the ability of an item, either a structural component or a structure consisting of many components, to hold together under a load, including its own weight, without breaking or deforming excessively.

[24] A simplified component in this context is one that has been reduced to just those parts necessary to deliver the safety function.

of the associated software, should be carried out by either an independent analysis or a test.

6.10. Embrittlement of non-structural materials (e.g. polymers used for insulation of electrical cables, seals and gaskets in mechanical equipment components) could limit the seismic capacity of some nuclear installation systems. The design should consider this age related degradation mechanism when defining the seismic qualification programme and the inspection or maintenance programme.

QUALIFICATION BY ANALYSIS

6.11. Qualification by analysis should follow an approach that is conceptually similar to that used for the seismic design of the main buildings. The seismic input should be the seismic loading at the location of the candidate SSC, normally expressed as in-structure response spectra or in-structure time histories. The seismic demand should then be computed using an appropriate analytical method and/or numerical analysis combined with the demand from other applicable actions. The total demand should then be compared with the available capacity, in accordance with accepted codes and standards and/or functionality specifications.

6.12. The seismic demand on SSCs may be computed using equivalent linear static analysis, linear dynamic analysis (in time or frequency domain), non-linear static ('pushover') analysis or non-linear dynamic analysis, depending on the national practice and applicable codes and standards. Irrespective of the method selected, the following recommendations apply:

(a) The input to the SSC should be defined by design spectra, by in-structure time histories or by acceleration time histories that are compatible with response spectra. If design spectra (or related time histories) are used, these need to be shown to either envelop or be conservative with respect to the in-structure loading conditions at the location of the SSC.
(b) The computational model should conservatively represent the behaviour of the candidate item under the seismic action (e.g. mass distribution, stiffness and damping characteristics).
(c) The important natural frequencies of the SSC should be estimated; alternatively, the peak of the design response spectrum multiplied by an appropriate factor greater than 1 should be used as input. Multimode effects should also be considered.
(d) A load path evaluation for seismic induced inertial forces should be performed. A continuous load path, with adequate strength and stiffness,

should be provided to transfer all inertial forces from the point of application to the main structure housing the item. The seismic demand for all the links of this path should be computed.

(e) The seismic demand should be obtained for the three orthogonal components of seismic motion (one vertical and two horizontal).

(f) Energy dissipation should be taken into account in a conservative manner (considering the uncertainties associated with dissipation mechanisms) and can be modelled for SSCs in a number of ways. If a modal analysis is being performed, modal damping values can be used for common components and materials recommended by applicable nuclear design codes.

6.13. For mechanical equipment, the following might have an effect on the damping, which should be considered in the design of the components: the isolation devices to protect against vibrations; the size, location and number of support gaps; the connection type (e.g. flanged); the frequency of response; and the use of yielding or energy absorbing support devices.

6.14. For vessels and tanks that contain liquids, the effects of sloshing and impulsive loads, including frequency effects, should be considered. The effects of liquid motion or pressure changes on submerged structures should also be considered. These effects may involve hydrodynamic loads from the fluid and a reduction of functional capability (e.g. loss of shielding efficiency of spent fuel pools, disturbance of instrument signals).

6.15. Simplified analytical or design techniques may be used in some cases.[25] All such simplified techniques should be fully validated to show their degree of conservatism in comparison with more refined modelling techniques or test results, and they should be suitably documented.

6.16. The flexibility or stiffness of elements of piping systems such as elbows, tees and nozzles should be considered in the model. Spring hangers may be ignored in the seismic analysis of piping. All added masses, including their eccentricities, such as valve actuators, pumps, liquid inside pipes and thermal insulation, should be considered.

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[25] For distribution systems (e.g. piping, cable trays, conduits, tubing and ducts, and their supports), modal response spectrum analysis may be used for the seismic design of large bore piping (e.g. diameter greater than 60 mm) for safety classified systems, while static methods are usually applied for the analysis of small bore piping. Spacing tables and charts based on generic analysis or testing are also used in the evaluation of small bore piping and are typically used to evaluate cable trays, conduits, tubing and ducts.

6.17. When distribution systems (e.g. piping, cable trays, cable conduits) are connected to two or more points that have different seismic movements and applicable response spectra, the use of a single response spectrum should be justified. To take inertial effects into account, either an envelope spectrum or multiple spectra should be applied.

6.18. In addition to inertial effects, for piping systems careful consideration should be given to the effects of differential seismic motions between supports.

QUALIFICATION BY TESTING

Types of test and typical application fields

6.19. When the integrity or functional capability of an item cannot be demonstrated with a reasonable degree of confidence by means of analysis, a testing programme should be carried out to demonstrate the seismic capability of the item or to assist directly or indirectly in qualifying the item. Types of test include the following:

(a) Acceptance test (proof test);
(b) Low impedance test (dynamic characterization test).

6.20. Acceptance (proof) tests should be used for active electrical and mechanical components to demonstrate their seismic adequacy for the design basis earthquake. This test is normally performed by manufacturers to demonstrate compliance with procurement specifications. Such testing is typically carried out using a shaking table.

6.21. Low impedance (dynamic characterization) tests should normally be carried out as a first stage of proof tests to identify the main dynamic characteristics of the item (e.g. natural frequencies).

Planning

6.22. The functional testing and integrity testing of complex items, such as control panels containing many different devices, should be performed either on a prototype of the item or on individual devices with the seismic test input scaled (via the in-cabinet transfer function) to allow for the location and attachment of each device within or on the item.

6.23. Qualification by testing is required to take into account, if necessary, ageing effects that might cause deterioration or alter the dynamic characteristics of the item during its service life: see para. 5.49 of SSR-2/1 (Rev. 1) [1], para. 6.84 of SSR-3 [2], para. 6.115 of SSR-4 [3] and should be conducted according to applicable industry standards[26].

6.24. A technical specification for qualification tests should be developed. The following should be considered in the test specification (if not already covered in an applicable seismic qualification standard):

(a) Applicable seismic test standards;
(b) Acceptance criteria;
(c) Input motion;
(d) Functional requirements;
(e) Boundary (support) conditions;
(f) Number of repetitions of testing or cycles of loading per test;
(g) Environmental conditions (e.g. pressure, temperature);
(h) Operational conditions, if functional capability has to be assessed.

6.25. Qualification tests should include the following:

(a) Functional tests intended to verify the performance of the required safety function of the component;
(b) Integrity tests aimed at proving the mechanical strength of the component.

When reduced scale testing is performed, the setting of similarity criteria associated with indirect methods of seismic qualification should be considered.

6.26. Test results should be documented in the test report. The format and content of the test report should be included in the test specification.

QUALIFICATION BY A COMBINATION OF ANALYSIS AND TESTING

6.27. When qualification by analysis or testing alone is not practicable (this may be the case for large and complex active equipment such as motors, generators or multi-bay consoles), a combination of analysis and testing, in which the results

[26] The use of industry standards will depend on national regulations. In some States, standard IEEE/IEC 60980344 [13] is used. Other national or international industry standards endorsed by the national regulatory body could also be used.

of benchmark tests are used as input to the analytical procedure or are used to validate the procedure, should be used for qualification purposes.

6.28. To aid in verifying the analytical models used for qualification by analysis of large and complex items, modal testing of a prototype should be considered.

6.29. Within a programme of qualification by testing, analysis should be considered for the following purposes:

(a) To justify the extrapolation of qualification by testing to more complex assemblies (e.g. multicabinet assemblies);
(b) To help define the testing programme by obtaining a better understanding of the dynamic behaviour of complex systems;
(c) To investigate and explain unexpected behaviour during a test;
(d) To obtain a first estimate of the response before performing tests on complex systems;
(e) To develop an analytical model with modal frequencies and damping, verified by the testing of a typical component, which enables the effects of component configuration variations to be simulated analytically.

QUALIFICATION BY INDIRECT METHODS

6.30. The indirect method of qualification relies on establishing the similarity of a candidate item to a reference item previously qualified by means of analysis or testing. The seismic input used to qualify the reference item should be equal to, or should envelop, the required input for the candidate item. The physical and support conditions, the functional characteristics for active items and the requirements for the candidate item should closely resemble those for the reference item.

6.31. The reliable application of indirect methods depends on the appropriate formulation and application of rigorous and easily verifiable similarity criteria. The validation of such criteria and the training of the review team are key for the process and should be explicitly recorded in the safety documentation.

7. SEISMIC MARGIN TO BE ACHIEVED BY THE DESIGN

CONCEPT OF SEISMIC MARGIN

7.1. The evaluation of the seismic margin is part of the safety assessment of the design. Seismic robustness is expressed by the seismic margin capacity, which defines the capability of a nuclear installation to achieve a certain performance under a seismic loading exceeding the site specific seismic hazard. The seismic margin should be provided in the design by a conservative definition of SL-2 and by acceptance criteria specified in applicable nuclear design codes.

7.2. If a seismic failure of a safety function were to occur at a hazard severity corresponding to the seismic design capacity (no margin) and, consequently, the seismic performance goal was not achieved (e.g. the seismic core damage frequency was greater than the performance target), such a scenario would correspond to a seismic induced cliff edge effect. The design is required to provide adequate seismic margin (a) to protect items important to safety and avoid cliff edge effects and (b) to protect items ultimately necessary to prevent an early radioactive release, or a large radioactive release, if levels of natural hazards greater than those considered for design occur: see Requirement 17 of SSR-2/1 (Rev. 1) [1], Requirement 19 of SSR-3 [2] and Requirement 16 of SSR-4 [3].

7.3. The seismic margin should be expressed in terms of the 'high confidence of low probability of failure' (HCLPF) capacity, which provides a link with the seismic fragility of the installation. In addition, the seismic hazard severity corresponding to the initiating of a seismic induced accident can be estimated on the basis of the mean installation fragility.

7.4. There is a correlation between the hazard level used to define SL-2, the seismic margin (HCLPF capacity) and the seismic performance goal (expressed in terms of core damage frequency[27], large release frequency or large early release

[27] The core damage frequency is an expression of the likelihood that an accident could cause the fuel in a nuclear reactor to be damaged. It is a term used in probabilistic safety assessment that indicates the likelihood of an accident that could cause severe damage to the fuel in the reactor core.

frequency[28], as applicable). In this context, the minimum seismic margin of the nuclear installation to ensure that the seismic performance goal is achieved, and that cliff edge effects are avoided, should be determined.

ADEQUATE SEISMIC MARGIN

7.5. For nuclear power plants or research reactors, both seismic margin capacities expressed as the HCLPF capacity should be assessed: the first corresponds to the prevention of significant damage to the reactor core; the second corresponds to an early radioactive release or a large radioactive release. For other types of nuclear installation, seismic margins should be commensurate with the risks associated with accident conditions at the installation.

7.6. An adequate seismic margin expressed as the minimum HCLPF capacity for the installation should be established.[29] For prevention of core damage, the minimum installation level seismic margin should be consistent with the seismic performance goal (e.g. a core damage frequency of less than 10^{-5}). For prevention of early or large releases, the minimum installation level seismic margin should be consistent with the containment seismic performance goal (e.g. a large early release frequency of less than 10^{-6}).

PROCEDURES TO ASSESS THE SEISMIC MARGIN

7.7. Procedures for quantification of seismic margins for existing nuclear installations are given in NS-G-2.13 [5]. These procedures use the as-built and as-operating conditions for SSCs; consequently, seismic walkdowns are a key element. The procedures recommended for assessing the seismic margin of existing nuclear installations should also be used at the design stage for new installations, assuming that the seismic capacity of selected SSCs is not negatively affected by seismic interactions or by any design changes.

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[28] The large early release frequency is the frequency of accidents that could lead to a radioactive release prior to the implementation of protective actions such that there is the potential for deterministic effects.

[29] To demonstrate adequate seismic margin (for nuclear power plants), the reference review level earthquake in seismic margin assessments is typically defined by a factor of 1.4, 1.5 or 1.67 based on a peak ground acceleration corresponding to SL-2.

7.8. Seismic margin assessment (i.e. using a deterministic approach) is typically performed for sites with low to moderate seismicity, whereas seismic probabilistic safety assessment is recommended for sites with high seismicity. For sites with moderate to high seismicity, seismic probabilistic safety assessment provides more insights about the seismic robustness of the design, the seismic performance and the significant contributors to seismic risk (which might include human errors).

7.9. In the probabilistic approach, the median and the mean seismic fragility of the installation and the seismic performance goal (expressed in terms of the mean seismic core damage frequency or other relevant risk parameters) should be calculated. The seismic margin for the installation should be obtained from the mean seismic fragility of the installation (see para. 7.3). The facility level HCLPF can also be determined using (sequence based) probabilistic safety analysis combined with seismic margin analysis (known as 'PSA based SMA').

7.10. In the deterministic approach (i.e. seismic margin assessment), two means for achieving a safe shutdown state should be identified and the HCLPF capacity should be evaluated for all relevant SSCs. By following this approach, both the seismic margin for the installation and the SSCs limiting this seismic margin are evaluated.

7.11. The seismic margin (HCLPF capacity) for the installation should be compared with the adequate seismic margin described in paras 7.5 and 7.6 or with values established by the regulatory body.

8. SEISMIC INSTRUMENTATION AND POST-EARTHQUAKE ACTIONS

SEISMIC INSTRUMENTATION

8.1. There are a number of reasons why seismic instrumentation[30] should be installed at nuclear installations, as follows:

(a) To provide triggering mechanisms for the automatic shutdown of the nuclear installation if the earthquake level exceeds a defined threshold;

[30] Seismic instrumentation is an array of strong motion accelerographs installed at and around the site of the installation and in defined locations in safety related structures.

(b) To provide alarms to alert operating personnel of the occurrence of the earthquake and to provide information for the decision making process defined by the operating procedures for the installation;
(c) To collect data on the dynamic behaviour of SSCs during an earthquake, to obtain realistic data on the structural response and to assess the degree of validity of the analytical methods used in the seismic design and qualification of the buildings and equipment.

8.2. The seismic categorization and safety classification of seismic instrumentation should be based on the safety relevance of the postulated seismic initiating event. In addition, the need for seismic instrumentation to support the emergency operating procedures for the nuclear installation should be taken into account.

8.3. Automatic seismic scram systems, where installed, should be safety classified in accordance with SSG-30 [12], and adequate redundancy, reliability and independence should be provided. In particular, the reliability, redundancy and independence of failure of any component or signal used in common with the reactor protection system should be considered.

8.4. The seismic instrumentation installed at the nuclear installation should be defined, specified, procured, installed, calibrated, maintained and upgraded as necessary, in accordance with the specific needs of the nuclear installation and the significance of the seismic risk to the safety of the installation.

8.5. Processing, interpretation and use of the data obtained from seismic instrumentation should be included in the operating procedures (including emergency operating procedures) for the installation and should be managed in accordance with the management system (see Section 10).

8.6. A suggested minimum amount of seismic instrumentation should be installed as follows:

(a) At all nuclear installations: one triaxial strong motion recorder installed to register the free field vibratory ground motion.

(b) At nuclear power plants:
   (i) Three triaxial strong motion recorders installed to register the vibratory motion of the basemat of the reactor building;[31]
   (ii) Two triaxial strong motion recorders installed on the most representative floors of the reactor building.
(c) At nuclear installations other than nuclear power plants: two triaxial strong motion recorders installed in the basemat of the building or structure with the largest inventory of radioactive material.

In addition to the minimum seismic instrumentation described above, additional instrumentation should be considered for sites having an SL-2 free field acceleration equal to or greater than 0.2g.

8.7. The seismic instrumentation should be able to provide damage parameters based on the integration of the acceleration record (e.g. the cumulative absolute velocity [14]), as an important tool for assessing the installation response in the event of an earthquake.

8.8. Such damage indicators should be compared with values of the same quantities derived from the free field design basis earthquake and with data from earthquake experience. Such comparisons can support post-earthquake walkdowns and therefore the decision on restarting operation.

8.9. The seismic instrumentation should allow an easy comparison of the response spectra of the actual seismic event with the design basis response spectra.

POST-EARTHQUAKE ACTIONS

8.10. Post-earthquake actions should be planned for a nuclear installation at the design stage as part of a dedicated programme of operational response to external events. The post-earthquake action programme should include a combination of pre-earthquake planning and short term and long term actions to be undertaken after the earthquake. At the seismic design stage of the nuclear installation — and in accordance with the characteristics of the design and operation of the installation — the principles and general specifications of this programme should be formulated and prepared.

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[31] Three triaxial strong motion recorders at the basemat will allow the translation motion corresponding to the horizontal and vertical directions to be evaluated and the rocking corresponding to both horizontal directions to be estimated.

8.11. The post-earthquake action programme should be based on the following:

(a) An experience based approach for determining the real damage potential of felt and significant earthquakes (see paras 8.13–8.15);
(b) A systematic methodology for assessing the need for shutdown of the installation and for assessing the readiness for restart (if the installation has been shut down), based on physical inspections and tests;
(c) Criteria for ensuring the long term integrity of the installation.

8.12. The post-earthquake action programme should be comprehensive enough to minimize the likelihood of a prolonged installation shutdown following seismic vibratory ground motion that does not damage SSCs important to safety. For earthquakes below the design basis levels (SL-1 and/or SL-2), the primary emphasis is on the physical and functional conditions of the installation, as opposed to analytical evaluations. In some cases, confirmatory analytical evaluations may be performed while the installation is in operation after a restart.

8.13. A 'felt earthquake' is any earthquake that produces vibratory ground motion at the site that is perceived by nuclear installation operators as an earthquake and that is confirmed by seismic instrumentation or other related information. The control room operators should be informed of the occurrence of an earthquake by means of the installed seismic instrumentation. Typically, seismic instrumentation installed at nuclear installations is triggered at peak ground acceleration values of 0.01–0.02g.

8.14. The initiation of actions as part of the post-earthquake action programme should be limited to those earthquakes that, having been felt at the nuclear installation, are considered to be 'significant earthquakes'. A significant earthquake is a felt earthquake that has free field surface vibratory ground motion characteristics approaching the threshold for damage or malfunction of non-seismically designed SSCs. Typical definitions of significant earthquakes are earthquakes with a free field surface vibratory ground motion greater than 0.05g or a standardized cumulative absolute velocity greater than a threshold (e.g. 0.16g/s based on Ref. [14]) or other damage indicators agreed by the regulatory body.

8.15. The definition of a significant earthquake also depends on the site and the seismic design basis of the nuclear installation, since this definition may determine the actions to be taken by the operating organization and by the regulatory body. The definition of the significant earthquake is the responsibility of the licensee and, where relevant, requires agreement or approval by the regulatory body.

8.16. The objective of the post-earthquake action programme is to provide guidance as well as specific and detailed procedures to the operating organization, covering the complete range of seismic vibratory ground motion, ranging from values lower than those corresponding to the SL-1 level to values higher than those corresponding to the SL-2 level.

8.17. There are two basic stages of the post-earthquake action programme:

(a) Planning: steps taken before an earthquake occurs to prepare an appropriate post-earthquake action plan. Many of these activities will be performed at the design stage.
(b) Response: implementation of the post-earthquake action plan — on the basis of the earthquake felt or the vibratory ground motion recorded at the site and the observed consequences to the installation — as part of the operational response.

The basic principles of such a programme should be as follows:

(a) The post-earthquake actions will facilitate timely decision making concerning the present or future state of the nuclear power plant, for example the need to shut down, to continue operation or to restart.
(b) Communication to all stakeholders will be timely and transparent with regard to the status of the installation, actions taken and actions to be taken.
(c) A tiered approach will be employed starting with overall evaluations and proceeding to very detailed evaluations only when required by the situation.

Specific guidance on establishing a post-earthquake action programme is provided in Ref. [15].

9. SEISMIC DESIGN FOR NUCLEAR INSTALLATIONS OTHER THAN NUCLEAR POWER PLANTS

9.1. A graded approach should be taken to ensure that seismic design criteria are commensurate with the magnitude of the seismic hazard, including associated radiological hazards, non-radiological hazards and other relevant factors.

9.2. Simplified methods for seismic hazard assessment, based on a more restrictive data set associated with a lower earthquake return period and applicable to medium and low hazard facilities, should be considered. The level of effort, the complexity of analysis and the thoroughness of documentation should be commensurate with the magnitude of the hazards presented by the installation, the complexity of the facility and the stage in the lifetime of the installation.

9.3. The likelihood that a seismic event will give rise to radiological consequences depends on the characteristics of the nuclear installation (e.g. its use, design, construction, operation and layout) and on the event itself. Such characteristics include the following:

(a) The amount, type and status of the radioactive inventory (e.g. solid, liquid or gaseous; processed or stored);
(b) The intrinsic hazard (e.g. criticality) associated with the physical processes and chemical processes that take place at the installation;
(c) The thermal power of the nuclear installation, if applicable;
(d) The configuration of the installation for activities of different kinds;
(e) The distribution of radioactive sources within the installation (e.g. in research reactors, most of the radioactive inventory will be in the reactor core and fuel storage pool, while in processing and storage facilities it may be distributed throughout the facility);
(f) The changing nature of the configuration and layout of installations designed for experiments;
(g) The engineered safety features necessary for preventing accidents and for mitigating the consequences of accidents, including the need for active safety systems and/or operator actions to prevent accidents and to mitigate the consequences of postulated accidents;
(h) The characteristics of the structures of the nuclear installations and the means of confinement of radioactive material;
(i) Any characteristics of the process or of the engineered safety features that might lead to a cliff edge effect in the event of an accident;
(j) The potential for on-site and off-site contamination.

9.4. The nuclear installations should be categorized in accordance with the intended design objective of the installation (i.e. the performance goal) and the risk associated with a failure of an SSC important to safety. On the basis of these criteria, each nuclear installation should be assigned to one of the following four seismic design categories:

(a) Seismic design category 1: high hazard nuclear installations;
(b) Seismic design category 2: medium hazard nuclear installations;

(c) Seismic design category 3: low hazard nuclear installations;
(d) Seismic design category 4: conventional installations.

The relationship between these seismic design categories and the consequences of seismic induced failure of the nuclear installation is shown in Table 2.

TABLE 2. SEISMIC DESIGN CATEGORY BASED ON THE CONSEQUENCES OF SEISMIC INDUCED FAILURE OF THE NUCLEAR INSTALLATION

Seismic design category (SDC)	Consequences on the site	Consequences off the site	Engineering and safety analysis
SDC1: High hazard nuclear installations	Radiological or other exposures that might cause loss of life of workers in the facility.	Potential for significant off-site radiological and/or non-radiological consequences.	Similar rules as used for nuclear power plants apply. Engineering and safety analyses are needed to determine the preventive and mitigating features and to determine if safety objectives are met.
SDC2: Medium hazard nuclear installations	Potential for significant on-site consequences. Unmitigated release would necessitate on-site evacuation.	Small potential for off-site radiological or non-radiological consequences.	Engineering and safety analyses are needed to determine if safety objectives are met.
SDC3: Low hazard nuclear installations	Potential for only localized consequences (within 30–100 m of the point of release).	No off-site radiological or non-radiological consequences.	Limited engineering safety analyses are needed to determine if safety objectives are met.
SDC4: Conventional installations	No radiological or chemical release, but failure of the structure, system or component could place workers at risk of physical injury.	No off-site radiological or non-radiological consequences.	Conventional design codes.

9.5. SSCs should be seismically designed to take into account the following:

(a) The seismic design category of the nuclear installation and the need to perform in the event of an SL-2 level hazard.
(b) The appropriate limit state[32] in the event of an SL-2 level hazard (specifying the analysis methodology, design procedures and acceptance criteria).
(c) SSCs whose seismic failures do not have any interactions with the performance of safety functions; these should correspond to seismic category 3. National codes and standards for seismic design of conventional installations apply (see Table 3).

9.6. SSCs should be seismically designed and qualified in accordance with the seismic design categories and target seismic performance goals[33] presented in Table 3.

10. APPLICATION OF THE MANAGEMENT SYSTEM

10.1. The management system to be established, applied and maintained by the operating organization is required to ensure the quality and the control of processes and activities performed as part of the seismic design: see Requirement 10 of GSR Part 2 [9].

10.2. The design processes for the development of the concept, detailed plans, supporting calculations and specifications for a nuclear installation and its SSCs should be established and applied in accordance with the recommendations provided in paras 5.84–5.140 of GS-G-3.5 [10].

[32] The limit state defines the limiting acceptable deformation, displacement or stress that an SSC might experience during, or following, an earthquake and still perform its safety function. SSCs are graded on the basis of the unmitigated consequences of SSC failure or of an SSC's reaching its limit state. Deformation related failures resulting from other, non-seismic natural phenomena hazards are defined by the design codes and criteria used to design the SSCs.

[33] In this section, the term 'performance goal' is used instead of typical reactor based risk parameters (e.g. core damage frequency, large release frequency) since nuclear installations include a large variety of non-reactor facilities. Therefore, the performance goal is associated with the definition of accident conditions for these facilities (mainly losing barriers and controls of the confined nuclear materials).

TABLE 3. RELATIONSHIP BETWEEN SEISMIC DESIGN CATEGORY, SEISMIC HAZARD LEVEL AND DESIGN CODES FOR ACHIEVING THE TARGET PERFORMANCE GOAL (VALUES DERIVED FROM REF. [16])

Seismic design category (SDC)	Design codes and standards	Seismic hazard level	Target seismic performance goal
SDC1: High hazard nuclear installations	Nuclear	SL-2 / 1.0E-4	<1.0E-5
SDC2: Medium hazard nuclear installations	Nuclear	SL-2 / 1.0E-3	<1.0E-4
SDC3: Low hazard nuclear installations	Conventional	1.5 × national seismic code	<5.0E-4
SDC4: Conventional installations[a]	Conventional	National seismic code	<1.0E-3

[a] Some high hazard non-nuclear industrial facilities may be seismically designed in a manner similar to SDC3 installations.

10.3. Seismic design inputs, requirements, outputs, changes, control and records should all be established in the design processes. The seismic design outputs include specifications, drawings, procedures and instructions, including any information necessary to install or implement the designed SSCs or other safety measures.

10.4. Seismic design inputs, processes, outputs and changes should be verified. The extent of this verification should be based on the complexity of the nuclear installation, the associated hazards and the uniqueness of the design. Seismic design records, including the final design, calculations, analyses and computer programs, as well as sources of design input that support design output, are normally used as supporting evidence that the design has been properly accomplished [9].

10.5. Computer programs used in seismic design should be validated through testing or simulation prior to use if they have not already been proven through previous use [9].

10.6. Any interfaces between the organizations involved in the design should be identified, coordinated and controlled. The control of interfaces includes the assignment of responsibilities among, and the establishment of procedures for use by, participating internal and external organizations [9].

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DEFINITIONS

The following definitions apply for the purposes of this Safety Guide. Further definitions are provided in the IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection: 2018 Edition:

https://www.iaea.org/publications/11098/iaea-safety-glossary-2018-edition

beyond design basis earthquake. The seismic ground motion (represented by acceleration time history or ground motion response spectra) corresponding to an earthquake severity higher than the one used for design. It is derived from the hazard evaluation of the site and is used in seismic margin assessment or seismic probabilistic safety assessment.

control point. The depth at which the seismic ground motion response spectrum is defined by the seismic hazard assessment. Typical control point locations are at free field ground surface, at the outcrop of bedrock or at any other specified depth in the soil profile.

high confidence of low probability of failure. The earthquake level for which there is 95% confidence that the probability of failure of a structure, system or component is less than 5%. It also represents the acceleration corresponding to the mean fragility of the 1% conditional probability of failure. High confidence of low probability of failure is a measure of the seismic margin capacity of a structure, system or component.

in-structure response spectrum. The seismic response spectrum at a particular elevation of a building for a given input ground motion.

seismic demand. The applicable seismic load for a structure, system or component. Typically, the seismic demand is expressed in terms of acceleration time history, acceleration response spectra, and seismic induced forces and/or displacements.

CONTRIBUTORS TO DRAFTING AND REVIEW

Altinyollar, A.	International Atomic Energy Agency
Beltran, F.	Consultant, Spain
Blahhoianu, A.	Consultant, Canada
Coman, O.	International Atomic Energy Agency
Ford, P.	Consultant, United Kingdom
Fukushima, Y.	International Atomic Energy Agency
Godoy, A.	Consultant, Argentina
Moreno, A.	Consultant, Spain
Morita, S.	International Atomic Energy Agency
Petre-Lazar, E.	Électricité de France, France
Rangelow, P.	AREVA GmbH, Germany
Sollogoub, P.	Consultant, France

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