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Evaluation of Seismic Safety for Nuclear Installations

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(Front inside cover)

IAEA SAFETY RELATED PUBLICATIONS

(to be included later)

FOREWORD

(to be included later)

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. The present publication provides guidance and procedures for the evaluation of safety of nuclear installations against the effects generated by earthquakes.

1.2. This Safety Guide provides recommendations on meeting the safety requirements stated in the following safety standards:

– IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [1];

– IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [2];

– IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3];

– IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Operation [4];

– IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [5];

– IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities [6].

1.3. This Safety Guide addresses requirements both for existing and new nuclear installations. For an existing installation, safety assessments are required to be reviewed periodically and the review may consider potential changes in site seismic hazard characterization [1] [2] [4] [5] [6]. At the design stage of a new nuclear installation, it is required to be checked that the design provides for an adequate margin to protect items important to safety against levels of external hazards more severe than those selected for the design basis [3] [5] [6]. In addition, it is required to be checked that the design of nuclear power plants provides 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 [3]. Hence, seismic safety assessments described in this Safety Guide can be either a part of the design process or a completely separate procedure from the design stage.

1.4. This Safety Guide is related to a number of other IAEA Safety Guides dealing with seismic hazard and seismic design, including IAEA Safety Standards Series Nos SSG-9, Seismic Hazards in Site Evaluation for Nuclear Installations [7], NS-G-1.6, Seismic Design and Qualification for Nuclear Power Plants [8], and NS-G-3.6, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants [9]. In addition, Ref. [10] provides detailed information relevant to this Safety Guide.

1.5. Guidelines for the seismic safety evaluation of existing nuclear installations — mainly nuclear power plants — have been developed and used in many Member States since the beginning of the 1990s[1]. More recently, criteria and methods applied for seismic safety evaluation of existing installations started being used, after some adaptation, for assessing beyond design basis earthquake conditions for new designs, prior to construction. This assessment is different than the seismic design and qualification of the installation, which is carried out for the design basis earthquake following the guidelines in NS-G-1.6 [8]. The seismic safety evaluation of a new design is intended to explore beyond design basis conditions for the new design[2].

1.6. Seismic safety evaluation differs from seismic design and qualification [8]. The main difference is in the evaluation criteria. Design, as traditionally understood[3], uses conservatively defined loads and capacities for structures, systems and components (SSCs) in order to meet the limits given in the design code. Thus, these methods are aimed at meeting the limits given by the codes for the design level earthquake in every SSC. In this way, safety for the design level earthquake is demonstrated. On the other hand, in seismic safety evaluation the aim is to establish the actual capacity of the SSCs in the 'as-is' condition and use it in the evaluation of the seismic capacity of the installation as whole. In doing this, experience from exposure to past seismic events, testing, and analytical estimates of capacity are utilized, and expert judgement plays a significant role. The 'as-is' condition of the installation includes the 'as-built', 'as-operated', 'as-modified' and 'as-maintained' conditions of the installation, and its condition of ageing at the time of the assessment.

1.7. The terms used in this Safety Guide, including the definition of a graded approach, are to be understood as defined in the IAEA Safety Glossary [11]. Explanations of terms specific to this Safety Guide are provided in footnotes.

1.8. The present publication supersedes the Safety Guide on Evaluation of Seismic Safety for Existing Nuclear Installations[4].

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[1] The development and use of guidelines on the seismic safety evaluation of existing nuclear installations started in the United States of America, where such guidelines were developed and their application to all existing nuclear power plants was required.

[2] Some Member States used these methodologies as a complementary technical support and they should not be solely used to comply with Requirements 17 of SSR-2/1 or equivalent requirements from SSR-3 or SSR-4

[3] The final seismic safety evaluation to check that the design provides for an adequate margin to protect items important to safety against levels of external hazards more severe than those selected for the design basis, as required by Refs. [3] [5] [6], can now be considered as a part of the 'design process'.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13, IAEA, Vienna (2009).

OBJECTIVE

1.9. This Safety Guide provides recommendations and guidance in relation to the seismic safety evaluation of nuclear installations, meeting the applicable requirements from Refs. [1] [2] [3] [4] [5] and [6]. For existing installations, such an evaluation may be prompted by a seismic hazard perceived to be greater than that originally established in the design basis, by new regulatory requirements, by new findings on the seismic vulnerability of SSCs, or by the need to demonstrate performance for beyond design basis earthquake conditions, in line and consistent with internationally recognized good practices. For new designs of nuclear installations, the seismic safety evaluation is motivated by the need to demonstrate that safety margins above the design basis earthquake are sufficient to avoid cliff edge effects[5] and, in case of nuclear power plants, sufficient to protect items ultimately necessary to prevent radioactive releases in the event of an earthquake with a severity exceeding the one considered for design.

1.10. This Safety Guide is intended for use by regulatory bodies responsible for establishing regulatory requirements, by designers and safety analysts involved in the design of new nuclear installations and by operating organizations of existing installations directly responsible for the execution of the seismic safety evaluation and upgrading programmes, as applicable.

SCOPE

1.11. This Safety Guide addresses an extended range of new and existing nuclear installations, that is: land-based stationary nuclear power plants, research reactors and any adjoining radioisotope production facilities; storage facilities for spent fuel; facilities for the enrichment of uranium; nuclear fuel fabrication facilities; conversion facilities; facilities for the reprocessing of spent fuel; facilities for the predisposal management of radioactive waste arising from nuclear fuel cycle facilities; and nuclear fuel cycle related research and development facilities [11]. Most of the recommendations in this Safety Guide are independent of the type of nuclear installation or the reactor type, but aspects such as performance criteria and systems modelling are specific to each installation type. The recommendations for nuclear power plants are also applicable to other nuclear installations through the use of a graded approach.

1.12. For the purposes of this Safety Guide, 'existing' nuclear installations are those

[5] A 'cliff edge effect', in a nuclear power plant, is an instance of severely abnormal plant behaviour caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input [3].

installations that are either (a) at the operational stage (including long term operation and extended temporary shutdown periods) or (b) at a preoperational stage for which the construction of structures, manufacturing, installation and/or assembly of components and systems, and commissioning activities are significantly advanced or fully completed. In existing nuclear installations that are at the operational and pre-operational stages, a change of the original design bases, such as for a new seismic hazard at the site, or a change in the regulatory requirements regarding the consideration of seismic hazard and/or seismic design of the installation, may lead to important physical modifications.

1.13. For the purpose of this Safety Guide, 'new' nuclear installations are those installations for which the design has reached a level of development in which a detailed definition of SSCs is available, including the data itemized in paras 4.2– 4.5. Typically, a 'new' nuclear installation[6], as understood in this Safety Guide, is not constructed or construction is at a very early stage.

1.14. Three methodologies are discussed in detail in this Safety Guide: the deterministic approach generally represented by Seismic Margin Assessment (SMA), the Seismic Probabilistic Safety Assessment (SPSA), and a combination of SMA and SPSA known as 'PSA-based Seismic Margin Assessment'. Variations of these approaches or alternative approaches may be demonstrated to be acceptable also, as discussed in Section 3.

STRUCTURE

1.15. Section 2 itemizes the safety requirements addressed by this Safety Guide and provides general concepts and general recommendations on the seismic safety evaluation of nuclear installations. Section 3 provides recommendations on the selection of the methodology for performing the seismic safety assessment. Section 4 provides recommendations on data requirements (collection and investigations), both for new and for existing installations. Section 5 is the core of this Safety Guide. It provides recommendations on considerations in relation to the assessment of seismic hazards and with the seismic capability necessary for defence-in-depth level 4, then provides recommendations on the implementation of the SMA, PSA-based SMA and SPSA methodologies for seismic safety evaluation focused on nuclear power plants. Section 6 provides recommendations on applying a graded approach to the evaluation of nuclear installations other than nuclear power plants (with reference to Section 5 where appropriate). Section 7 presents recommendations on the use of seismic safety evaluation results, including

[6] New installations may include a standard design based on generic site parameters, for which the site has not been specified

potential seismic upgrading. Section 8 provides recommendations on the management system to be put in place for the performance of all activities, and it identifies the need for configuration management in future activities to maintain the seismic capacity as evaluated. Sections 1–4, 7, and 8 apply (in total or in part) to all nuclear installations. Section 5 is focused on nuclear power plants.

1.16. The appendix to this Safety Guide presents seismic failure mode considerations for different types of SSCs. The annex provides an example of criteria for defining seismic design classes and performance targets in a nuclear installation.

2. GENERAL CONSIDERATIONS FOR EVALUATION OF SEISMIC SAFETY FOR NUCLEAR INSTALLATIONS

SAFETY REQUIREMENTS FOR SEISMIC SAFETY EVALUATION

Safety assessment

2.1. As established in the GSR Part 4 (Rev. 1) [1], the following requirements should be applied for seismic design robustness and periodic review of seismic safety:

Requirement 10 of GSR Part 4 (Rev. 1) [1] states:

> "**It shall be determined in the safety assessment whether a facility or activity uses, to the extent practicable, structures, systems and components of robust and proven design.**"

Requirement 13 of GSR Part 4 (Rev. 1) [1] states that "**It shall be determined in the assessment of defence in depth whether adequate provisions have been made at each of the levels of defence in depth.**"

Paragraph 4.48A of GSR Part 4 (Rev. 1) [1] states that "Where practicable, the safety assessment shall confirm that there are adequate margins to avoid cliff edge effects that would have unacceptable consequences."

Requirement 15 of GSR Part 4 (Rev. 1) [1] states that "**Both deterministic and probabilistic approaches shall be included in the safety analysis.**"

Requirement 24 of GSR Part 4 (Rev. 1) [1] states that**:** "**The safety assessment shall be periodically reviewed and updated.**"

2.2. Similar provisions should be applied to research reactors and to nuclear fuel cycle facilities, as established in Requirement 5 of SSR-3 [5], and Requirement 5 of SSR-4 [6], respectively.

Hazard assessment

2.3. As established in SSR-1 [2], the following requirement should be applied to address potential changes in the perceived seismic hazard:

Requirement 29 of SSR-1 [2] states:

> "**All natural and human induced external hazards and site conditions shall be periodically reviewed by the operating organization as part of the periodic safety review and as appropriate throughout the lifetime of the nuclear installation, with due account taken of operating experience and new safety related information.**"

Design

2.4. As established in SSR-2/1 (Rev. 1) [3], the following requirements should be applied regarding the seismic margin to be provided by the design of nuclear power plants[7]:

Requirement 17 of SSR-2/1 (Rev. 1) [3] states:

> "**All foreseeable internal hazards and external hazards, including the potential for human induced events directly or indirectly to affect the safety of the nuclear power plant, 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.**"

. . .

Paragraph 5.21 of SSR-2/1 (Rev. 1) [3] states:

> "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."

Paragraph 5.21A of SSR-2/1 (Rev. 1) [3] states:

> "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 hazards evaluation for the site."

2.5. Similar provisions should be applied to research reactors and to nuclear fuel cycle facilities, as established in Requirement 19of SSR-3 [5], and Requirement 16 of SSR-4 [6], respectively.

Operation

2.6. As established in SSR-2/2 (Rev. 1) [4], the following requirements should be applied during operation of nuclear power plants to assess the consequences of changes in the perceived seismic hazard:

[7] Paragraph 1.3 of SSR-2/1 (Rev. 1) [3] acknowledges that "it might not be practicable to apply all the requirements of this Safety Requirements publication to nuclear power plants that are already in operation or under construction". Hence, for the purposes of the present Safety Guide, the requirements here may be considered applicable only to new nuclear power plants.

Requirement 12 of SSR-2/2 (Rev. 1) [4] states:

> “**Systematic safety assessments of the plant, in accordance with the regulatory requirements, shall be performed by the operating organization throughout the plant’s operating lifetime, with due account taken of operating experience and significant new safety related information from all relevant sources.”**

Paragraph 4.44 of SSR-2/2 (Rev. 1) [4] states:

> “Safety reviews such as periodic safety reviews or safety assessments under alternative arrangements shall be carried out throughout the lifetime of the plant, at regular intervals and as frequently as necessary (typically no less frequently than once in ten years). Safety reviews shall address, in an appropriate manner: the consequences of the cumulative effects of plant ageing and plant modification; equipment requalification; operating experience, including national and international operating experience; current national and international standards; technical developments; organizational and management issues; and site related aspects. Safety reviews shall be aimed at ensuring a high level of safety throughout the operating lifetime of the plant.”

GENERAL CONCEPTS FOR EVALUATION OF SEISMIC SAFETY FOR NUCLEAR INSTALLATIONS

2.7. Well designed and maintained nuclear installations, especially nuclear power plants, have an inherent capability to resist earthquakes larger than the earthquake considered in their design. This inherent capability or robustness — usually described in terms of the ‘seismic margin’ — is a direct consequence of (i) the conservatism that is present in the seismic design and qualification procedures used according to previous or current practices in earthquake engineering and (ii) the fact that in the design of nuclear power plants the seismic loads may not be the governing loads for some SSCs[8].

2.8. Typically, current criteria for seismic design and qualification applicable to nuclear power plants introduce seismic design margins, often substantial, which the traditional design process does not by itself quantify in its entirety. The process by which margins develop through the various stages of the analysis, design and construction may lead to large variations

[8] The existence of margins has been demonstrated not only through the implementation of SMA or SPSA methodologies for existing nuclear power plants in several Member States, but also by the performance of some plants in large earthquakes. Those plants have experienced large earthquakes, which exceeded their design basis, and have survived the earthquakes with little or no damage.

throughout the nuclear installation. The seismic margin typically varies from one location in the installation to another, from one SSC to another, and from one location to another in the same structure[9]. Consequently, when evaluating the seismic safety of a nuclear installation, there should be a detailed examination of the actual design methods and, for existing installations, of the ‘as-is’ condition, in order to understand the sources of conservatism and margins. It should not be automatically assumed that there is an excess of seismic capacity all over the nuclear installation since this may lead to complacency in the seismic safety evaluation.

2.9. The methodologies presented in this Safety Guide are intended for evaluating and quantifying the seismic margin over the design basis earthquake of a particular installation. The realistic seismic response of the SSCs, in terms of their safety function, should be understood. From this understanding, maximum seismic capacity of the SSCs for which there is high confidence that the safety functions are fulfilled, can be derived. High confidence capacities of the SSCs are used to assess the seismic safety margin of the installation as a whole.

2.10. The seismic safety evaluation of an existing installation strongly depends on the actual condition of the installation at the time the assessment is performed. This key condition is denoted the ‘as-is’ condition, indicating that an earthquake, when it occurs, affects the installation in its actual condition, and the response and capacity of the installation will depend on its actual physical and operating configuration. The ‘as-is’ condition typically consists of the original design, design changes during construction and operation, and ageing. That is why the upkeep of up-to-date as-built design documentation and of ageing management programme is very important. The ‘as-is’ condition of the installation should be the baseline for any seismic safety evaluation.

2.11. Seismic safety evaluation performed on the basis of the as-is condition of the installation, should emphasize pragmatic evaluations rather than using extensive complex analyses. Non-linear analyses of relatively simple structural models or the use of higher damping values and ductility factors — provided that they are used with care and are consistent with allowable deformations — may be particularly helpful in understanding post-elastic behaviour. Numerous field observations and research and development programmes have demonstrated a high seismic capacity results when the ductile behaviour of SSCs is able to accommodate large strains.

[9] One of the main reasons for this variation, as mentioned in para. 2.7, is the fact that nuclear installations are designed for a wide range of internal and external extreme loads, for example, pressure and other environmental loads due to accident conditions, aircraft crash, tornado or pipe break. Therefore, seismic loads may not be the governing loads for some SSCs. Another reason is the method of equipment qualification in which envelope-type response spectra are generally used.

2.12. When a reliable seismic hazard analysis is available for a particular site (see SSG-9 [7]), seismic safety evaluation should use a realistic definition of the hazard-dominant earthquake motion for the selected annual frequency of exceedance, in terms of amplitude, duration, directivity and frequency content. When there are several dominant seismic sources that lead to very different motion characteristics (e.g., far field and near field), the feasibility of using several motion characterizations and, therefore, assessing seismic safety (margins) against each of them, should be considered.

REASONS TO PERFORM SEISMIC SAFETY EVALUATIONS

New installations

2.13. In accordance with the requirements established in GSR Part 4 [1], SSR-2/1 (Rev. 1) [3], SSR-3 [5], and SSR-4 [6] (see paras. 2.1, 2.2, 2.4 and 2.5 of this Safety Guide), an evaluation of the seismic safety of new nuclear installations is required to be performed as a safety assessment, when the design is completed, to verify that safety margins above the design basis earthquake are sufficient to avoid cliff edge effects. In addition, in the case of a nuclear power plant, the evaluation is required to verify that margins are sufficient to protect items ultimately necessary to prevent radioactive releases in the event of an earthquake with a severity exceeding the one considered for design. This safety assessment should be reflected in the Safety Analysis Report of the installation [12]. Recommendations on the level of seismic margin to be achieved in a new installation are provided in IAEA Safety Standards Series No. DS490, Seismic Design of Nuclear Installations [13].

2.14. In connection with para. 2.13, the design of a new nuclear power plant needs to meet two requirements: (a) Adequate seismic margin for items important to safety to provide protection against seismic hazards levels exceeding those considered for design and to avoid cliff edge effects (see para. 5.21 of SSR-2/1 (Rev. 1) [3]); and (b) Adequate seismic 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 (see para. 5.21A of SSR-2/1 (Rev. 1) [3]). The seismic margin to meet (b) applies to a reduced set of SSCs and normally shows larger plant state margins than the seismic margin needed to meet (a).

Existing installations

2.15. In accordance with the requirements established in GSR Part 4 (Rev. 1) [1], SSR-1 [2], SSR-2/2 (Rev. 1) [4], SSR-3 [5] and SSR-4 (see paras. 2.1, 2.3 and 2.6 of this Safety Guide), and in line with international practice, an evaluation of the seismic safety of an existing nuclear installation is required to be performed in the event of any one of the following:

(a) Evidence of a significant increase in the seismic hazard at the site, arising from new or additional data (e.g. newly discovered seismogenic structures, newly installed seismological networks or new paleo-seismological evidence), new methods of seismic hazard assessment, and/or the occurrence of actual earthquakes that affect the installation;

(b) Regulatory requirements, such as the requirement for periodic safety reviews, that take into account the 'state of knowledge' and the actual condition of the installation;

(c) Inadequate seismic design, generally due to the vintage of the facility;

(d) New technical findings, such as vulnerability of selected structures and/or non-structural elements (e.g. masonry walls), and/or of systems or components (e.g. relays);

(e) New experience from the occurrence of actual earthquakes (e.g. better recorded ground motion data and the observed performance of SSCs);

(f) The need to address the performance of the installation for beyond design basis earthquake ground motions in order to provide confidence that there is no 'cliff edge effect', that is, to demonstrate that no significant failures would occur in the installation if an earthquake were to occur that was somewhat greater than the design basis earthquake;

(g) A programme for long-term operation, extending the life of the plant, for which such an evaluation is required.

2.16. If, for the reasons listed in para. 2.15 or for other reasons, a seismic safety evaluation of an existing nuclear installation is required, the purposes of the evaluation should be clearly established before the evaluation process is initiated. This is because there are significant differences among the available evaluation procedures and acceptance criteria, depending on the purpose of the evaluation[10]. In this regard, the objectives of the seismic safety evaluation may include one or more of the following:

(a) To demonstrate the seismic safety margin beyond the original design basis earthquake and to confirm that there are no cliff edge effects.

(b) To identify weak links in the installation and its operations with respect to seismic events.

[10] Available evaluation procedures, and the differences between them, are presented and discussed in Section 3.

(c) To evaluate a group of installations (e.g. all the installations in a region or a State), to determine their relative seismic capacity and/or their risk ranking. For this purpose, similar and comparable methodologies should be adopted.

(d) To provide input for integrated risk informed decision-making.

(e) To identify and prioritize possible upgrades.

(f) To assess risk metrics (e.g. core damage frequency and large early release frequency) against regulatory requirements, if any.

(g) To assess installation capacity metrics (e.g. systems-level and installation-level fragilities or, HCLPF[11] capacities) against regulatory expectations.

2.17. The objectives of the seismic safety evaluation of an existing installation should be established in line with the regulatory requirements, and in consultation and agreement with the regulatory body. Consequently, and in accordance with such objectives, the level of seismic input motion, the methodology for capacity assessment and the acceptance criteria to be applied, including the required end products, should be defined. In particular, for evaluating seismic safety for seismic events more severe than the event specified in the original design basis, the safety objectives should include the functions required to be ensured and the failure modes to be prevented during or after the earthquake's occurrence.

2.18. The final documentation to be produced at the end of the evaluation of an existing installation should be identified from the start in agreement with the regulatory body and should be consistent with the established purpose of the evaluation programme (see paragraph 8.6). The end products of these evaluations may be one or more of the following:

(a) Metrics of the seismic capacity of the nuclear installation in deterministic and/or probabilistic terms;

(b) Quantification of the seismic risk;

(c) Identification of SSCs with low seismic capacity, and the associated consequences for plant safety, to be used in decision-making for seismic upgrade programmes;

(d) Identification of operational modifications to improve seismic capacity;

(e) Identification of improvements to housekeeping practices (e.g. storage of maintenance equipment);

[11] The High Confidence Low Probability of Failure (HCLPF) capacity is the earthquake motion level at which there is a high confidence of a low probability of failure. HCLPF capacity is a measure of seismic margin (see Section 5).

(f) Identification of interactions with equipment and piping, including fire protection systems, high enthalpy lines and utilities;

(g) Identification of actions to be taken before, during, and after the occurrence of an earthquake that affects the installation, including arrangements for operational and management response, analysis of the obtained instrumental seismic records and performed inspections, and the integrity evaluations to be performed as a consequence;

(h) A framework to provide input to risk informed decision-making.

CONSIDERATION OF RELEVANT ASPECTS RELATED TO SEISMIC HAZARD

2.19. An initial step of any seismic safety evaluation — in parallel with the collection of related data as indicated in Section 4 — should be to identify the seismic hazards with regard to which the seismic safety of the installation will be evaluated. In this respect, the seismic hazards specific to the site should be assessed in relation to three main elements[12]:

(a) Evaluation of the geological stability of the site [7] [9], with two main objectives:

  (i) To verify the absence of any capable fault that could produce differential ground displacement phenomena underneath or in the close vicinity of buildings and structures important to safety. If there exists evidence that indicates the possibility of a capable fault in the site area or site vicinity, the fault displacement hazard should first be assessed in accordance with the guidance provided in SSG-9 [7].

  (ii) To characterize potential permanent ground deformation phenomena (i.e. liquefaction, slope instability, excessive settlement, subsidence or collapse).

(b) Characterization of the severity of the seismic ground motion at the site, that is, assessment of the vibratory ground motion parameters, taking into consideration the full scope of the seismotectonic effects at the four scales of investigation[13] and as recommended in SSG-9 [7].

(c) Evaluation of other concomitant phenomena such as earthquake induced river flooding due to dam failure, coastal flooding due to tsunami, and landslides.

2.20. In general, the seismic hazard assessment may be performed using a deterministic or a

[12] In most cases, it is foreseen that a seismic hazards assessment will be available as part of the site investigation or a periodic revaluation of the hazards. The available hazard assessments will need to be reviewed to determine if they are adequate for the purposes of the seismic safety evaluation being performed.

[13] In SSR-1 [2] and SSG-9 [7], four scales of investigations are defined: (1) 'regional' radius R about 300 km, (2) 'near region', R no less than 25 km, (3) 'site vicinity', R no less than 5 km, and (4) site area, R about 1 km.

probabilistic approach, depending on the objectives and requirements of the seismic safety assessment. In either case, both the aleatory and the epistemic uncertainties should be taken into consideration.

2.21. The evaluations recommended in paras. 2.19 (a) and 2.19 (c) of this Safety Guide should be performed in all cases for a seismic safety evaluation, regardless of the methodology used and in accordance with SSG-9 [7], NS-G-3.6 [9] and IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [14]. For evaluating the geotechnical hazards (e.g. liquefaction, slope instability, subsidence, collapse), the most current available seismic hazard parameters should be used.

2.22. With respect to para. 2.19 (b) of this Safety Guide, the recommendations on assessing the seismic hazard at the site are dependent on the objectives of the evaluation. A site-specific ground motion seismic hazard assessment is generally preferred, and is a prerequisite that should be carried out, as recommended in SSG-9 [7], when the objectives of the evaluation include the assessment of the seismic risk posed by the installation or risk-based metrics for the SSCs. On the other hand, it should not be considered a prerequisite when the objective of the evaluation is to determine the seismic margin above a predefined reference level earthquake and/or to rank the SSCs contributing to the installation-level seismic capacity to withstand that reference level earthquake for identification of seismic weak links. However, even in those cases, a seismic hazard assessment should be performed when site-specific information indicates that the ground motion characteristics (e.g. spectral shape) might differ significantly from the ones assumed for design.

2.23. A site-specific probabilistic seismic hazard assessment [7] should be performed when the objectives of the seismic safety assessment entail the following:

(a) Calculation of risk metrics (e.g. core damage frequency and large early release frequency);

(b) Establishment of a risk management tool for risk informed decision-making;

(c) Determination of the relative risk between seismic and other internal and external hazards;

(d) Provision of a tool for cost–benefit analysis for decision-making in relation to plant upgrades.

2.24. For the SMA and PSA-based SMA methodologies, the reference level earthquake[14] defines the seismic input that should be used in the seismic safety evaluation. The reference level earthquake should not be understood as a new design earthquake (see also para. 5.5). It should be understood as a tool to determine the seismic margin of the installation and its seismic 'weak links'[15]. The reference level earthquake should be sufficiently larger than the design basis earthquake to ensure that it challenges the seismic capacity of the SSCs so that an installation-level HCLPF can be determined and the 'weak links' (if any) can be identified. The seismic input for a seismic safety evaluation should not be less than a peak ground acceleration of 0.1 g at the foundation level.

2.25. For the SPSA methodology, the reference level earthquake[16] is defined using the site-specific probabilistic seismic hazard assessment results. Generally, those results include seismic hazard curves defining the annual frequency of exceedance (often referred to as the annual probability of exceedance) of ground motion parameters (e.g. spectral accelerations), associated response spectra (e.g. uniform hazard spectra) and characteristics of the dominant source parameters (e.g. magnitude and distance from the site). The reference level earthquake should be defined at an annual frequency of exceedance that corresponds to an earthquake severity that significantly contributes to the seismic risk of the installation. When there are several dominant seismic sources which lead to very different motion characteristics (e.g. far field and near field), overall seismic hazard curves may be split into multiple, mutually exclusive, contributions and multiple corresponding reference level earthquakes may be defined for the seismic safety assessment. In that case, the seismic risk computed for each contribution should be added up to obtain the total risk.

EVALUATION OF SEISMIC SAFETY FOR MULTI-FACILITY SITES

2.26. For sites with multiple nuclear installations (mainly nuclear power plants) and/or with nuclear power plants that credit for a significant number of shared systems and resources, seismic safety evaluation is required to consider potential interactions between installations. Safety evaluation of multi-facility sites provides risk insights that help minimize the risk of multiunit accidents (e.g. due to shared systems and resources) and to maximize the benefits

[14] In the literature on SMA methodology, this 'reference level earthquake' is sometimes known as the 'review level earthquake' or the 'seismic margin earthquake'.

[15] In this context, a seismic 'weak link' is a non-redundant SSC or identical redundant SSCs (affected by common cause failure) which has a smaller capacity than the majority of the other SSCs and, as such, it could be controlling the installation-level seismic capacity.

[16] The 'reference level earthquake' concept, as used in the present Safety Guide (see para. 5.5), is not to be confused with the seismic level that is used sometimes in SPSA as a threshold for explicit calculation of fragilities, when below, and for assignment of generic fragilities, when above.

associated to shared systems and resources among units. The Multiunit-PSA is an appropriate methodology for considering potential interactions in a multiunit context. The technical background of the methodology can be found in Refs. [15], [16], [17] and [18].

CONSIDERATION OF SEISMIC SAFETY EVALUATION AT THE DESIGN STAGE

2.27. At the design stage, SPSA or PSA-based SMA methodologies are typically used to address the requirements described in paras. 2.13 and 2.14 of this Safety Guide. At the design stage, methodologies are limited to information available in the design phases and cannot rely on an as-built and as-operated installation. All tasks are similar with the one used for existing installations and the differences consists only in the availability of information. Instead of as-built and as-operated information, at the design stage, methodologies should rely on as-designed information only. Seismic walkdowns cannot be conducted at the design stage.

2.28. During development of the design, seismic safety evaluation should be used to address and eliminate seismic vulnerabilities identified in the past, to check the effectiveness of the defence in depth provisions, to provide insights for setting performance targets consistent with the seismic safety goals, and to optimize the robustness of seismic design.

CONSIDERATION OF SEISMIC SAFETY EVALUATION AT THE LICENSING STAGE

2.29. At the licensing stage, the detailed design is completed, and site-specific seismic induced hazards are known. For nuclear power plants, SPSA methodology is typically used to provide input to the final safety analysis report (see Section 15 of SSG-61 [12]). The seismic safety evaluation should provide assurance that the seismic design is adequate for the site-specific seismic conditions. Particularly, the SPSA for new installations provides risk insights, in conjunction with the assumptions made, and contributes to identify and support requirements important to the seismic design of the plant.

2.30. After the plant is built and operation starts, the seismic safety evaluation performed at the licensing stage should be updated to reflect as-built and as-operated conditions.

3. SELECTION OF THE SEISMIC SAFETY ASSESSMENT METHODOLOGY

3.1. The selection of the seismic safety assessment methodology is an important decision that should be carefully considered due to its crucial consequences. This selection should satisfy the following objectives:

(a) The selected assessment methodology should be adequate for achieving the objective of the seismic safety evaluation in the context of the reasons that motivated the evaluation. Paragraphs 2.16 and 2.15 list a number of these objectives and reasons, respectively. This section provides guidance on the applicability of each methodology (i.e., SMA, PSA-based SMA, and SPSA)[17] to a number of common objectives for existing and new installations.

(b) The selected methodology and its end products should be able to meet the regulatory requirements applicable to the installation.

(c) The selected methodology should be capable of demonstrating that the installation will meet the requirements described in paras. 2.1–2.6, as applicable to the evaluation reasons and installation type. Requirement 15 of GSR Part 4 (Rev. 1) [1] indicates that both deterministic and probabilistic approaches complement one another and specifies that both approaches be included in safety analysis within a graded approach. This section discusses the capabilities and limitations of each methodology.

3.2. It is possible that more than one assessment methodology[18] can satisfy the objectives in para. 3.1. In deciding between multiple feasible methodologies, the selection should consider the following:

(a) The availability and quality of knowledge and data sources needed to support the execution of the methodology and its technical elements. For example, the SPSA methodology requires the performance of site-specific probabilistic seismic hazard analysis (PSHA) studies, which in turn require availability of specific information about seismicity rates and ground motion propagation characteristics from all potential sources within a distance range that can contribute to the seismic hazard of interest at

[17] The methodologies presented in this publication are internationally recognized approaches that reflect the current state of practice. Other methodologies may be used in individual Member States in the context of their national regulatory environment. Such latter methodologies are not covered in this publication.

[18] The scope of this document primarily focuses on safety evaluation that uses the concepts of HCLPF and/or Seismic Fragility for defining the seismic margin of the installation. Alternative methods for seismic safety evaluation that are not predicated on using the HCLPF (or Seismic Fragilities) for estimating the seismic margin of the installation are not precluded if they are justifiable. In determining the appropriate evaluation methodology to be implemented, consideration should be given to the history and characteristics of the site, the level of risk posed by the site specific seismic hazard, the basis of the key safety case claims and objectives, and the national regulatory practice.

the installation, and explicit characterization of uncertainty in these parameters. A deterministic seismic hazard analysis only needs knowledge of this information for the few rupture sources that dominate the seismic hazard at the installation and can accommodate a less explicit uncertainty characterization.

(b) The schedule requirements for executing the selected methodology.

(c) The initial and maintenance cost[19] commitments of the selected methodology.

(d) The potential added values achieved in addition to the primary safety evaluation objective, and their alignment with the longer-term strategic objectives of the installation. The added values to consider may include usability of the safety assessment methodology components or end products for other objectives, reusability or upgradeability of these components or end products in the future, and flexibility to accommodate future changes in regulatory requirements over the remaining or anticipated service life of the installation.

(e) The assessment methodology does not need to be the same for all seismic-induced hazards and potential SSC failures. For example, a SPSA methodology may be selected to perform the safety evaluation for vibratory ground motions only. Meanwhile, a screening evaluation can demonstrate that the installation has sufficiently high margin for the effects of the remaining seismic hazards, that is, that these hazards have negligible contribution to seismic risk and need not be explicitly included in the SPSA.

SEISMIC MARGIN ASSESSMENT

3.3. The SMA methodology is the least resource-intensive of the three methodologies discussed in this Safety Guide and it is used mainly for existing installations. It can be executed using as input a seismic hazard characterization developed using either probabilistic or deterministic approaches. The implementation details of this methodology should meet the guidelines presented in Section 5.

3.4. The end product of an SMA is an installation-level HCLPF capacity, which is based on the HCLPF capacity of two (or more) independent success paths .

3.5. The SMA methodology is applicable to the following safety evaluation objectives, and it should be considered of limited applicability otherwise:

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[19] The maintenance cost is in reference to the level of effort required to periodically update the SPSA or SMA to keep its results valid over time, for instance, to incorporate updates to seismic hazard, modified or replaced SSCs, facility configuration or operational changes, availability of new data, and improvement in seismic capacity evaluation methods.

(a) Determination of the seismic safety margin higher than a specified earthquake (e.g. the design basis earthquake) or an actual earthquake that affected the installation;

(b) Demonstration of seismic robustness of the installation against cliff edge effects when robustness is characterized by seismic safety margin;

(c) Demonstration of sufficient safety margin to restart operation following the occurrence of a beyond design basis earthquake that may have shut down the nuclear installation in addition to other actions defined in Ref. [19];

(d) Comparing an estimate of installation-level HCLPF capacity to regulatory expectations;

(e) Identification of weak links in the credited success paths for the nuclear installation's response to a beyond design basis earthquake event;

(f) Identification of possible upgrades for SSCs in the success paths to improve the seismic safety margin;

(g) Comparative safety assessment of a group of nuclear installations benchmarked by seismic safety margin against either (i) the same earthquake effects, (ii) the effects of a common earthquake scenario, or (iii) earthquakes that represent the same level of seismic hazard at each site;

(i) Effective communication about the robustness of the nuclear installation to stakeholders, including the public.

(j) Demonstration that regulatory seismic requirements are met for plants which were designed without seismic requirements.

PSA-BASED SEISMIC MARGIN ASSESSMENT

3.6. The PSA-based SMA methodology is a hybrid between the SMA and SPSA methodologies. It combines the typically less resource-intensive hazard assessment, fragility, and Boolean logic solution approaches of the SMA methodology with the accident sequence event tree and fault tree analysis from the SPSA. This methodology is used for both new and existing installations. The implementation details of this methodology should meet the guidelines presented in Section 5.

3.7. The end products of the PSA-based SMA should be the installation-level HCLPF capacity and HCLPF capacities for all accident sequences of interest (i.e. minimal cut-sets[20]) that can lead to an installation performance unacceptable to safety. An additional end product may be an estimate of the installation-level full fragility curve[21] in addition to its HCLPF capacity. The sequence-level HCLPF capacities are typically taken to be the highest SSC HCLPF capacity in each cut-set.

3.8. The PSA-based SMA methodology is applicable to the following safety evaluation objectives in addition to those introduced in para. 3.5, and it should be considered of limited applicability otherwise:

(a) Comparing an estimate of installation-level and accident class-level HCLPF capacities to regulatory expectations;

(b) Identification of critical accident scenarios that can undermine safety in the installation's response to a beyond design basis earthquake event and the weak link(s) in each sequence;

(c) Identification and prioritization of possible upgrades for safety-related SSCs to improve the seismic safety margin;

(d) Providing preliminary insight to risk-informed design and resource allocation decisions (e.g. safety classification of SSCs);

(e) Comparative safety assessment of a group of installations benchmarked by either (i) installation-level seismic safety margin or (ii) sequence-level safety margins against specific accident classes and/or potential consequences.

SEISMIC PROBABILISTIC SAFETY ASSESSMENT

3.9. The SPSA methodology can only be executed using as input a site-specific seismic hazard characterization developed using probabilistic approaches. The SPSA methodology discretizes the seismic hazard from PSHA into acceleration levels with corresponding annual occurrence frequencies and explicitly convolves[22] these frequencies with the installation-level fragility. The installation-level fragility should be constructed by explicitly solving the installation accident sequence. Boolean logic trees using failure probabilities obtained by

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[20] A 'minimal cut-set' is a combination of events (failures) whose sequence causes the accident to occur. Occurrence of all events in the cut-set is necessary and sufficient for the accident to take place.

[21] The installation-level fragility represents the conditional probability of facility unacceptable performance for a given value of the hazard parameter (e.g. peak ground acceleration). It is normally presented as a function of the hazard parameter in the form of a curve. It is commonly referred to as "plant-level fragility' for nuclear power plants. See Section 5 for more details.

[22] Convolution is a type of mathematical integration. Ref. [10] provides an example of the convolution integral.

quantifying accident sequencies associated to each initiating event. Non-seismic failure rates of SSC and human error probabilities are also taken into consideration in SPSA. This methodology is used for both new and existing installations. The implementation details of this methodology should meet the guidelines presented in Section 5. More guidance on the SPSA methodology can be found in IAEA Safety Standards Series No. DS523, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [15].

3.10. The end products of the SPSA should include the products of the two SMA methodologies, plus the annual frequency of the installation unacceptable performance due to seismic hazard, the installation-level fragility curve, the risk importance metrics for accident sequences and components, and the explicit quantification of uncertainties in the computed results.

3.11. The SPSA methodology is applicable to the following safety evaluation objectives in addition to those introduced in paras. 3.5 and 3.8, which should be considered in the methodology selection:

(a) Comparing the risk metrics for unacceptable performance (e.g. core damage frequency and large early release frequency) to regulatory expectations;

(b) Quantification and ranking of relative risk contributions of accident sequences and individual SSCs in the installation's as-operated condition;

(c) Evaluation of risk reduction worth of possible SSC upgrades, procedure changes, or mitigation strategy implementation;

(d) Providing quantitative input to risk-informed design and resource allocation decisions (e.g. impact to risk from safety classification of SSCs);

(e) Understanding and incorporation of uncertainty in seismic safety metrics into the safety evaluation conclusions;

(f) Enabling risk monitoring models that integrate real-time condition changes in the installation (e.g. living PSA and digital twin technologies);

(g) Comparative safety assessment of a group of installations benchmarked by either seismic safety margin or risk metrics.

CONSIDERATIONS ON APPLICATION TO NEW OR EXISTING NUCLEAR INSTALLATIONS

3.12. The methodology selection should be constrained by the objectives and available information for each nuclear installation. The objectives of the seismic safety assessment are different for a new installation (see paras 2.13 and 2.14) and for an existing installation (see paras 2.15–2.17). In addition, there may be substantial differences in the available information for new installations and for existing installations (see para. 4.1). A new installation project will typically face different challenges in collecting data (e.g. site characterization information) from those in an existing installation. Both aspects, the objectives of the assessment and the available information, should be considered when selecting the most appropriate methodology.

3.13. The selected methodology should be able to meet the applicable regulatory requirements. Regulatory requirements for existing nuclear installations and for new installations are different in several Member States[23].

3.14. The schedule and cost priorities for the seismic safety assessment should be considered in the selection between multiple feasible methodologies. These priorities and their decision-making consequences are typically distinct in a new nuclear installation from those in an existing installation, due to the constraints of applicable regulations and socio-economic factors.

3.15. The anticipated service life of a new nuclear installation may be different and will typically be significantly longer than the remaining service life of a similar existing installation. This should make the reusability and shelf life of a more rigorous methodology longer for a new installation. Accordingly, the 'return on investment' from performing the more cost-extensive SPSA methodology for a new nuclear installation typically runs longer than for an existing installation, which may be approaching the end of its service life.

[23] For example, in the United States of America, new nuclear power plant license applications are required to demonstrate a plant-level HCLPF of at least 1.67 times the ground motion response spectrum that defines the design basis earthquake. This requirement is not applicable to operating nuclear plants.

4. DATA COLLECTION AND INVESTIGATIONS FOR EVALUATION OF SEISMIC SAFETY FOR NUCLEAR INSTALLATIONS

DATA AND DOCUMENTATION ON THE DESIGN BASIS

General

4.1. The design basis data and documentation should be collected from all available sources. This compilation does not pose special difficulties for new nuclear installations. For existing installations, emphasis should be put on the collection and compilation, as far as possible, of the specific data and information on the nuclear installation that were used at the design stage. It is acknowledged that limitations on the quantity and quality of the available original design data may arise for old existing installations. However, the more complete information is collected from the design stage, the less effort and fewer resources will be required for the seismic safety evaluation.

General documentation of the installation

4.2. All available general and specific documentation for new and existing installations should be compiled, including the following:

(a) The safety analysis report, preferably the final safety analysis report.

(b) Codes and standards used for the design of the installation:

   (i) Standards adopted and procedures applied to specify the nominal properties of the materials used and their mechanical characteristics;

   (ii) Standards adopted and procedures applied to define load combinations and to calculate the seismic design parameters;

   (iii) Standards used for the design of structures, components, piping systems and other items, as appropriate;

   (iv) Standards and procedures used for the design of conventional buildings at the time of the design of the installation, which ought to have been considered minimum requirements.

(c) General arrangement and layout drawings for structures, equipment, and distribution systems (e.g. piping, cable trays, ventilation ducts).

(d) Probabilistic safety assessment (PSA) of internal (and external) events, if performed.

(e) For existing installations, data and information on results and reports of seismic qualification tests for SSCs performed during the pre-operational period, including any information available on inspection, maintenance, and non-conformance reports and corrective action reports. For new installations, specifications for seismic qualification tests (e.g. required response spectra) may be sufficient.

(f) For existing installations, quality assurance and quality control documentation, with particular emphasis on the as-built conditions for materials, geometry and configuration, for assessing the modifications during construction, fabrication, assembly and commissioning, including non-conformance reports and corrective action reports. The accuracy of the data should be assessed.

Specific documentation of the SSCs included in the seismic safety evaluation

4.3. Specific information on the original design of the installation, in particular on those SSCs included in the programme for seismic safety evaluation, should be collected, as follows:

(a) System design:

  (i) System description documents;

  (ii) Safety, quality and seismic classification;

  (iii) Design reports;

  (iv) Report on confirmation of the functionality of systems;

  (v) Instrumentation and control of the system, including the general concept, the type of devices and how they are mounted.

(b) Geotechnical design:

  (i) Excavation, structural backfill and foundation control (e.g. for settlement, heaving and dewatering);

  (ii) Construction of retaining walls, berms or artificial slopes;

  (iii) Soil–foundation–structure failure modes and design capacities (e.g. estimated settlements, sliding, overturning, uplifting, liquefaction).

(c) Structural design:

  (i) Stress analysis reports for all structures of interest;

  (ii) Structural drawings (e.g. structural steel, reinforced and/or prestressed concrete), preferably as-built documentation for existing installations;

(iii) Material properties (specified and test data);

(iv) Typical details (e.g. connections).

(d) Component design:

(i) Seismic analysis and design procedures;

(ii) Seismic qualification procedures, including test specifications and test reports;

(iii) Typical anchorage requirements and types used;

(iv) Stress analysis reports;

(v) Pre-operational test reports, if any.

(e) Distribution system design (piping, cable trays, cable conduits, ventilation ducts):

(i) Systems description documents;

(ii) Piping and instrumentation diagrams;

(iii) Layout and design drawings of piping and its supports;

(iv) Diagrams of cable trays and cable conduits and their supports;

(v) Diagrams of ventilation ducts and their supports;

(vi) Design Reports including stress analysis if available.

(f) Service and handling equipment (although some of this is non-safety-related equipment, its evaluation may be needed for analysis and study of interaction effects in operational and storage configurations):

(i) Main and secondary cranes;

(ii) Fuel handling equipment.

Seismic design basis

4.4. The characterization of the seismic input used for design should be well understood for conducting the seismic safety evaluation. Any discrepancy between the documentation of the seismic hazard assessment performed during the site evaluation studies and the design basis values finally adopted should be identified. This information is essential for determining the reference level earthquake, which will be used to assess the seismic safety margin of the installation. In this regard, the following aspects should be covered:

(a) Specification of the design earthquake level(s) as used for the design and qualification of SSCs [7].

(b) Free field ground motion parameters in terms of elastic ground response spectra, acceleration time histories or other descriptors, such as the power spectral density.

(c) Dominant earthquake source parameters used to define the seismic input motions, such as magnitude, distance, definition and duration of strong motion. Other parameters, such as the focal mechanism or the source spectral shape, might have been used as well.

(d) If some structures were designed in accordance with design codes whose design spectra have implicit reductions for inelastic behaviour, the corresponding elastic ground response spectra should be derived to provide a basis of comparison with the elastic ground response spectra typically used to define the reference level earthquake for the seismic safety evaluation.

Soil–structure interaction, structural modelling and in-structure response details

4.5. Information on soil–structure interaction analysis, modelling techniques, and techniques of structural response analysis used in the design should be collected as follows:

(a) Soil–structure interaction parameters:

  (i) The location selected for applying the seismic input ground motion — for example, free field surface on top of finished grade, foundation mat level or base rock level (control point location);

  (ii) Soil profile properties, including soil stiffness and damping properties used in the site-specific response analysis, information on the water table variation, and consideration of strain dependent properties;

  (iii) Method to account for uncertainties in soil properties and techniques of soil–structure interaction analysis, for example, envelope of three analyses for best estimate, lower bound, and upper bound soil profiles;

  (iv) Applicability and consideration of seismic wave phenomena in the definition of the input motion. Those should include: definition of seismic input motion as a vertically propagating shear wave (typical); coherency; wave passage effect.

(b) Modelling techniques:

  (i) Modelling techniques and analytical methods used to calculate the seismic response of structures and the in-structure response spectra (floor response spectra);

(ii) Material and system damping, cut-off of modal damping, frequency dependency of damping;

(iii) Allowance for inelastic behaviour, as assumed in the design phase and as implemented during construction.

(c) Structural analysis and response parameters:

(i) One- or two-stage analysis, using coupled or substructure models of soil and structures;

(ii) Equivalent static analyses of components and structures;

(ii) Dynamic analysis of components and structures;

(iii) Natural frequencies and modal shapes, if available;

(iv) Output of structural response (e.g. structure internal forces and moments, in-structure accelerations, deformations or displacements);

(v) Foundation response, including overall behaviour such as sliding or uplift;

(vi) Calculations of in-structure response spectra (floor response spectra), including:

– Damping of equipment;

– Enveloping and broadening criteria, if used.

ADDITIONAL DATA AND INVESTIGATIONS FOR EXISTING NUCLEAR INSTALLATIONS

Current (as-is) data and information

4.6. For an existing nuclear installation, after collecting as many data as is feasible in relation to the original design basis, as recommended in paras 4.2–4.5, the present state and actual conditions of the installation (i.e. the 'as-is' condition) should be identified[24]. The collection of as-is data should cover those selected SSCs that will be considered within the scope of the programme for seismic safety evaluation and that have either a direct effect on system performance or an indirect effect such as by transmitting earthquake motion from one location to another or by affecting safety related SSCs in case of a seismically induced failures. It should be also emphasized that the as-is condition should properly reflect and include the effects of

[24] Any seismic safety evaluation to be performed for an existing nuclear installation should be made by considering the state of the installation at the time the assessment is performed. This condition of the installation is denoted the 'as-is' condition. Consequently, one of the first and more important steps of the programme for seismic safety evaluation is to collect all the necessary data and information to provide a complete representation of the actual situation of the installation.

ageing degradation of the installation throughout its operational lifetime. Pending physical or operational modifications should also be recognized so that they can be taken into account in the evaluation. When applicable, a sufficient number of samples should be collected on parameters of interest (e.g. concrete strength) to adequately define the variability (e.g. mean and standard deviation).

4.7. If the nuclear installation has been subjected to periodic safety reviews, as recommended in IAEA Safety Standards Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants [20], the reports of these reviews should be made available for the purposes of the seismic safety evaluation.

4.8. If the nuclear installation has an ageing management programme, any outputs from it (e.g. condition assessment, periodic inspection reports) that identify the as-is condition should be made available for the purposes of the seismic safety evaluation. If some SSCs (e.g. active equipment) are not be covered under an ageing management programme, but under some other programme (e.g. maintenance rule programme), the related documentation should also be made available for the purposes of the seismic safety evaluation.

4.9. A critical review of all available as-built and pre-operational documentation (reports, drawings, photographs, film records, reports of non-destructive examinations) should be performed. For this purpose, a preliminary screening walkdown should be carried out to confirm the documented data and to acquire new, updated information. During this walkdown, data about any significant modifications and/or upgrading and/or repair measures that were performed over the lifetime of the nuclear installation should be collected and documented, including any reports on ageing effects. The judgement about how significant a modification would need to be in order to have an impact on the seismic response and capacity of the installation should be made by experts on the evaluation of seismic capacity.

4.10. Special attention should be paid to requirements, procedures and non-conformance reports for construction and/or assembly related to the following:

(a) Slopes, excavation and backfill;

(b) SSCs not accessible for inspection;

(c) Field-routed items (e.g. piping, buried piping, cable trays, conduits, and tubing);

(d) Installation of non-safety-related items (e.g. masonry walls, shielding blocks, room heaters, potable water lines and fire extinguishing lines, and false ceilings);

(e) Separation distances or clearances between components;

(f) Field-tested items;

(g) Anchorages.

Recommended investigations: soil data

4.11. To perform reliable and realistic site-specific seismic response analysis, data on the static and dynamic material properties of soil and rock profiles should be obtained. For an existing installation, if these data were obtained at an earlier stage (e.g. during the design stage), they should be reviewed for adequacy with regard to current methodologies. In this respect the following should be taken into account:

(a) Appropriate ranges of static values and dynamic values for the geodynamic properties, which account for site specific geotechnical characteristics, should be available for use in the programme for seismic safety evaluation.

(b) For ground materials, the density and low strain properties (normally in situ measurements of compressional, P, and shear, S, wave velocities), laboratory measurements of three-axis static properties, and, if possible, dynamic properties and material damping ratio should be available.

(c) As a function of depth, the variation of dynamic shear modulus values and damping values with increasing strain levels should be available. Strain dependent variations in ground material properties may be based on generic data if ground materials are properly correlated with the generic classifications.

(d) For hard rock layers, variation of properties with increasing strain levels may usually be disregarded.

In operating nuclear installations, the performance of soil investigation campaigns might encounter implementation difficulties. In such cases, judgement may be needed, supported by all practically achievable gathering of data. In any case, substitution of physical data by judgement should be avoided to the maximum extent possible.

4.12. Information on the location of the local water table and its variation over a typical year should be obtained.

4.13. For the various stages of site investigation, design, and construction, other data may be available from non-typical sources, such as photographs, notes, and observations recorded by operations staff or others. These data should be evaluated in the light of their source and method of documentation. To the extent possible, the collection of such data should be carried out in compliance with the recommendations provided in NS-G-3.6 [9].

4.14. All available information relating to actual earthquake experience at the site or at other industrial installations in the region should be obtained. Special attention should be paid to earthquake-induced phenomena such as river flooding due to dam failure, coastal flooding due to tsunami, landslides, and liquefaction.

Recommended investigations: data on building structures

4.15. The as-is concrete classes used for the construction of the safety related structures of the nuclear installation should be verified on the basis of existing installation-specific tests and industry standards for concrete. Destructive and non-destructive methods may be used[25]. The as-is data collected, should be used for further analyses and capacity evaluations rather than the nominal design data. If there is significant deviation from the design values, the cause of this deviation and its consequences should be investigated.

4.16. The actual material properties of the reinforcing steel should be used in the evaluation. Material properties should be available from existing test data. If not, reliable methods of destructive and non-destructive testing should be used. The information on the reinforcing steel should include both mechanical properties and detailing (e.g. size of reinforcing bars, placement, geometric characteristics, concrete cover, distances between bars). For the evaluation of the overall capacity of a structure, the properties of all significant load bearing members should be evaluated. Other cases where detailing of the reinforcement may be important include, for example, penetrations and anchorage of large components.

4.17. Although ageing effects are usually estimated in a separate project, in the seismic safety evaluation, at a minimum, the survey of a concrete building should include visual examination for cracks, effects of erosion/corrosion and surface damage, the degree of carbonization, the thickness of concrete cover, and the degree of degradation of below ground foundations due to, for example, chlorides or other corrosive contaminants present in groundwater.

4.18. A sample survey should be made to verify the geometrical characteristics of selected structural members. The number of samples collected should be statistically significant to allow for the accurate computation of sample statistics (e.g. sample mean and sample standard deviation).

4.19. An important element of the evaluation is the verification of realistic non-seismic loads (e.g., live and dead loads) and possibly the new assessment of loads, other than seismic loads, that will be used in the seismic safety evaluation. Usually, both the dead and the live loads in

[25] Non-destructive methods alone are usually not sufficient for establishing concrete strength with reliability.

the as-is condition differ from those used in the original design. The deviations should be carefully examined and documented.

Recommended investigations: data on piping and equipment

4.20. If design information is inadequate for piping, equipment, and their supporting structural systems, analysis and/or testing should be performed to establish their dynamic characteristics and behaviour. A representative sample may be sufficient.

5. SEISMIC SAFETY ASSESSMENT FOR NUCLEAR INSTALLATIONS

ASSESSMENT OF SEISMIC HAZARDS FOR NUCLEAR INSTALLATIONS

Seismic hazard assessment approach

5.1. Site specific seismic hazard should preferably be used to characterize the reference level earthquake for the seismic safety evaluation (see para. 2.22). The seismic hazard assessment may be performed using a probabilistic or a deterministic approach. A probabilistic approach should be used to develop the reference level earthquake for an SPSA. A deterministic approach may be used to develop the reference level earthquake for an SMA and PSA-based SMA.

5.2. The PSHA should include a probabilistic characterization of ground motions that can be produced at the installation site by all seismic sources within the regional seismotectonic model, in accordance with SSG-9 [7]. The ground motion characterization should be performed for the range of annual frequencies required to meet the regulatory requirements and to achieve the objectives of the safety evaluations. Deaggregation of the PSHA results should be performed at the reference level earthquake to identify the dominant seismic sources, that is, those that have the largest contributions to the hazard.

5.3. The Deterministic Seismic Hazard Analysis (DSHA) should include determination of ground motions that the dominant seismic sources within the regional seismotectonic model are capable of producing at the installation site. The ground motions should be determined in accordance with SSG-9 [7], considering the maximum potential magnitude of each source, the closest associated distance to the site, and an appropriately high confidence level to account for variability due to epistemic uncertainty and aleatory variability in the source model, ground motion prediction model, and site conditions.

5.4. The dominant seismic sources in a DSHA should be identified by careful review of the seismotectonic model, as recommended in SSG-9 [7], in the absence of deaggregation data from a PSHA. Dominant sources may not be the same for the different ground motion parameters and other seismic hazards (see para. 2.19). For sites located in a region of low to moderate seismicity, low-frequency ground motion accelerations can be dominated by distant high-magnitude sources while high-frequency ground accelerations are often dominated by diffuse seismicity, that is, nearby moderate magnitude sources. Geological failures are primarily caused by low-frequency ground motions, while the dominant sources for concomitant phenomena hazards are phenomenon specific.

Development of reference level earthquake

5.5. The reference level earthquake is the seismic hazard realization at which the responses and capacities of the SSCs identified for the seismic safety assessment should be explicitly evaluated. A reference level earthquake is necessary for technical consistency in the safety evaluation, considering that several important dynamic response parameters depend on the seismic excitation level, including the following:

(a) Damping depends on the extent of shaking-induced cracking in concrete structures and slip or other connection deformations in metallic structures;

(b) Geotechnical material properties and physical integrity exhibit degradation as the shaking level increases;

(c) The potential for the geotechnical failures whose characterization is necessary to evaluate the geological stability of the site (see para. 2.19 (a)) typically depends on the shaking level.

5.6. The reference level earthquake should be defined for the vibratory ground motion hazard, using response spectra that characterize horizontal and vertical ground acceleration components at the site. For other seismically induced hazards (e.g. fault displacement), development of reference parameters should be performed on a case-specific basis if these hazards cannot be screened out in accordance with para. 5.11.

Characterization of vibratory ground motions

5.7. For SMA and PSA-based SMA evaluations, the reference level earthquake may be set according to several criteria and should be in accordance with the objectives of the safety assessment (see paras 3.5 and 3.7) and available hazard assessment information (see paras 5.1–5.4). These criteria include the following:

(a) A scaled spectrum of the original design basis earthquake;

(b) A scaled spectrum or broadened spectrum of an earthquake that affected the installation;

(c) A generic spectrum or suite of spectra (e.g. used in certification of a standard design);

(d) A scaled site-specific spectrum for a specified earthquake scenario (e.g. para. 5.3);

(e) A site-specific spectrum for a specified uniform hazard of exceedance (e.g. para. 5.2);

(f) A generic or site-specific spectrum determined by the regulator.

5.8. When the reference level earthquake is not based on current site-specific hazard assessments, as in paras. 5.7(a)–5.7(c), the corresponding spectra should be compared to the site-specific deterministic or uniform probabilistic hazard spectra (see para. 5.1) to develop an understanding of the resulting seismic safety margin of the installation in a site specific context.

5.9. For SPSA evaluations, the reference level earthquake spectrum at each frequency should be set to spectral acceleration levels that contribute most significantly to the resulting seismic risk and have comparable, but not necessarily equal, annual probabilities of exceedance. The following considerations should be observed in the reference level earthquake for SPSA:

(a) The selected reference level earthquake spectrum shape should result in low sensitivity of the computed seismic risk to the selection of the ground motion hazard parameter for the SPSA (e.g. peak ground acceleration or spectral acceleration at selected frequencies);

(b) Because prior to performance of the SPSA, the relative contributions of ground motion levels to seismic risk can only be estimated, the appropriateness of the reference level earthquake based on this estimation should be confirmed after completion of the SPSA or addressed if found to be questionable (e.g. using sensitivity studies).

Characterization of other seismically induced hazards

5.10. Characterization of the reference level earthquake parameters for other seismically induced hazards is only necessary for those hazards that cannot be screened out of explicit evaluation in the safety assessment. Screening of non-vibratory ground motion hazards and concomitant phenomena (para. 2.19) should be individually performed for each hazard and credible phenomenon.

5.11. Screening should be performed based on one of the following two criteria:

(a) Credibility: Occurrence of the screened hazard at the site with a severity that challenges the installation safety is practically impossible or its annual probability of occurrence is too low compared to the reference level earthquake for vibratory ground motions (e.g. fault displacement hazard is screened out due to absence of capable faults in close vicinity of the nuclear installation, or soil deposits are so dense and ground water table is so low that liquefaction may only occur at incredibly high vibratory ground motions).

(b) Consequence: Potential occurrence of the screened hazard has no consequence on the safety of the nuclear installation due to physical features or reliable mitigation measures (e.g. river flooding due to upstream dam failure leads to an upper bound water line elevation at the site that does not challenge the external flood design basis of the installation).

5.12. For non-vibratory seismic hazards that cannot be screened out, the reference parameters for SMA and PSA-based SMA evaluations should be determined on a hazard-specific basis considering the criteria adopted for the reference level earthquake spectrum (see para. 5.7) and the hazard assessment approach (see para. 5.1). Options for determining these parameters include the following:

(a) Ground motion parameters developed using deterministic hazard assessment in accordance with paras 5.3 and 5.4. The reference level parameters should be scaled by an appropriate margin based on the reference level earthquake spectrum.

(b) Ground motion parameters developed using probabilistic hazard assessment in accordance with para. 5.2 and prediction equations specific to these parameters[26]. The reference level parameters should correspond to annual probabilities of exceedance similar to those of the reference level earthquake spectrum at an appropriately high confidence level to account for uncertainties in the geotechnical evaluation.

(c) Ground motion parameters developed using geotechnical evaluations of the site response at the reference level earthquake for vibratory motion (e.g. slope deformation evaluation using the reference level spectrum as input motion). The reference level parameters (e.g. slope displacement) should correspond to an appropriately high confidence level to account for uncertainties in the geotechnical evaluation.

5.13. For non-vibratory hazards that cannot be screened out, the reference earthquake parameters for SPSA evaluations should be determined using a probabilistic hazard assessment approach (see para. 5.2). The determination of ground motion parameters in the range of annual exceedance frequencies of interest may be performed by direct prediction (e.g. see para 5.12 (b)) or indirect prediction (e.g. see para. 5.12 (c)). In any case, the epistemic uncertainty and aleatory variability in the assessment approach for each hazard should be incorporated. The reference level parameters should correspond at a minimum to annual probabilities of exceedance similar to those of the reference level earthquake spectrum. However, due to typically strong nonlinearities associated with geotechnical failure modes, and their potential to cause site-wide cliff edge effects, multiple earthquake levels, especially above the reference level, should be explicitly used in developing the fragility functions associated with the corresponding SSC failures.

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[26] Ground motion prediction equations for most non-vibratory ground motion parameters are typically at an earlier stage of technical evolution than, and not as commonly available or reliable as, those for vibratory ground motions.

5.14. For concomitant phenomena that cannot be screened out in accordance with para. 5.11, the reference earthquake parameters should be determined on a case-specific basis. These phenomena may be triggered by earthquake ground motions occurring at sites with significantly different subsurface properties or located far away from the installation, and their correlation with the reference level earthquake ground motions at the site requires specific evaluation.

IMPLEMENTATION GUIDELINES COMMON TO ALL METHODOLOGIES FOR EVALUATION OF SEISMIC SAFETY FOR NUCLEAR INSTALLATIONS

Scope of the seismic safety assessment

5.15. A multidisciplinary expert team composed of systems engineers, operations personnel, and seismic capability engineers should collectively determine the scope of the seismic safety assessment. A typical assessment team should have 3–5 members[27]. The first four steps involved in this determination of the scope are described in paras 5.16 to 5.19. These steps are fundamentally the same for all three assessment methodologies discussed in Section 3 and differ only in implementation details as noted where applicable to each methodology later in this Section.

5.16. The first step in determining the scope should be identifying the safety functions to be fulfilled in order to control the progression or mitigate the consequences of an accident to an acceptable end state if the installation experiences a beyond design basis earthquake. These safety functions and acceptable accident end states should be in accordance with the regulatory framework and safety requirements for the nuclear installation[28].

5.17. The second step in determining the scope should be to establish agreement on the following defining conditions for the safety assessment:

(a) Establishing the initial conditions of the nuclear installation to be considered at the time of the earthquake. This includes, for example: (i) definition of whether the installation is in normal operating mode or in another mode (e.g. shutdown); (ii) definition of what constitutes normal operating conditions for the installation systems and their components; and (iii) determining whether a seismic-induced abnormal condition should be triggered and considered to occur concurrent with or following earthquake shaking (e.g. loss of off-site power or small loss of coolant accident).

[27] The assessment team selection process is reviewed in Ref. [10].

[28] For nuclear power plants, Requirement 4 in SSR-2/1 (Rev. 1) [3] lists the fundamental safety functions as: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel store; and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

(b) Defining the safety-related functions and corresponding systems that are credited in achieving the acceptable end states identified in para. 5.16. The SMA methodology focuses on defining a subset of functions and systems necessary to achieve a determined number of success paths (typically two) to an acceptable end state. The PSA-based SMA and SPSA methodologies broaden their focus to include systems and functions whose failure might lead to the progression of an accident to an unacceptable end state.

(c) Identifying operator actions that are credited in the safety evaluation. These actions should be established in the emergency procedures.

(d) Availability and credit to take for emergency response and mitigation systems that are not safety-related. This may include mobile alternative resources (such as supplies of water, compressed air, and mobile electrical power) stored on the site, that are located and maintained in such a way as to be functional and readily accessible when needed in postulated emergency conditions.

(e) Availability and credit to take for outside assistance. The type, response time, and conditions for availability of outside assistance should be established in the safety procedures and agreed upon with the regulatory body.

5.18. The third step in determining the scope should be to prepare a list of selected SSCs[29] for seismic capability evaluation. Paras. 5.20–5.22 provide recommendations on this process.

5.19. The final step in determining the scope should be to perform a seismic evaluation walkdown. Paragraphs 5.23–5.33 provide recommendations on this process. For a new nuclear installation, the walkdown may be replaced with a virtual review[30] (to the extent practical) followed by a confirmatory walkdown after construction of the installation is finished.

Development of the selected SSCs list

5.20. The selected SSCs list should be developed jointly by the expert multidisciplinary team. This selection should be based on the following considerations and confirmed by a systems walkdown (para. 5.21):

(a) Inclusion of SSCs necessary for the safety-related systems identified in para. 5.17(b) to fulfil their safety functions. These SSCs are not limited to front-line and support safety

[29] The term 'selected SSCs' is used in this Safety Guide to mean those SSCs that are of interest to the SMA or SPSA. Elsewhere, the terms 'safe shutdown equipment list' (SSEL) and 'seismic equipment list' (SEL) have commonly been used with a similar meaning. The term 'selected SSCs' is used here since required SSCs include more than just equipment.

[30] A virtual review is such that the 3D model of the installations is displayed directly in the VR space, and some elements of the seismic walkdowns.

systems but include instrumentation and control equipment, cable trays, passive elements, and other distribution systems.

(b) Inclusion of other SSCs whose seismic-induced response or damage could interact with one or more of the SSCs and interfere with their ability to fulfil their safety function (e.g. falling, impact, fire, flood, and spray hazards);

(c) Inclusion of SSCs whose seismic-induced damage may impede operator actions identified in para. 5.17(c) (e.g. physically injure operators or block their entry, egress, or their use of tools needed to execute actions);

(d) Inclusion of SSCs necessary for post-earthquake emergency procedures credited in achieving an acceptable end state, for example, the mitigation systems identified in para. 5.17(d);

(e) Inclusion of SSCs whose seismic-induced damage may impede the arrival or deployment of the outside help identified in para. 5.17(e);

(e) Inclusion of the structures that house or support the identified SSCs;

(g) Inclusion of SSCs that represent unique features of the installation from a seismic safety perspective (e.g. SSC related to credible and consequential concomitant phenomena described in para. 5.14).

5.21. A systems walkdown should be performed for existing nuclear installations. For new installations, a virtual review should be performed of the available design to the extent practical. This walkdown should confirm the completeness and consistency of the selected SSCs list with the as-built systems configuration, familiarize the seismic capability engineers with the as-built configuration, conditions, and apparent seismic robustness or vulnerability of the SSCs, investigate the surrounding areas to identify potential sources of seismic-induced interactions with the required SSCs, ensure that the credited operator travel paths are compatible with plant operating procedures, and verify potential assumptions used to justify including or screening elements out of the scope of the safety assessment (see para. 5.11).

5.22. The selected SSCs list prepared according to paras 5.20 and 5.21 should include all the SSCs that belong in the success path or logic tree model for the acceptable end state(s) of the nuclear installation. Several SSCs on this list may be removed from explicit seismic capability evaluation if qualitative review indicates that they have either: (i) significantly low seismic capacities and should be assumed to fail in an earthquake, or (ii) significantly high seismic capacities and can be assumed to be rugged in an earthquake[31]. These screening decisions should be confirmed by observation in the seismic evaluation walkdown (see para. 5.23). The selected SSCs list should be refined during the walkdown and finalized as part of the walkdown documentation (see para. 5.33).

Seismic evaluation walkdown

5.23. Seismic evaluation walkdowns are one of the most significant components of the seismic safety evaluation in the SMA and SPSA methodologies. They are often referred to as 'seismic capability walkdowns' in the context of SMA approaches and 'seismic fragility walkdowns' in the SPSA approach. For existing nuclear installations, they should be performed after completion of the selected SSCs list. For new installation designs that have not been constructed, walkdowns should be performed after construction is completed to verify consistency between the as-built conditions and the as-designed conditions that were used in the safety assessment based on virtual review (see para. 5.21) and to observe any installation or site specific features. It is important that all design features used for the seismic assessment be verified in the as-built installation or any deviations addressed in order for the safety assessment to be valid. The final safety analysis report should incorporate any resulting updates to the safety assessment in accordance with regulatory requirements [12].

5.24. Each walkdown team should include qualified seismic engineers, at least one systems engineer, at least one installation operator, and support personnel as necessary (maintenance, operations, systems, and engineering). The seismic engineers should have sufficient experience in the seismic analysis, design and qualification of SSCs for resisting earthquakes and other loads arising from normal operations, accidents, and external events. One team member should be familiar with the design and operation of the SSC being walked down.

5.25. The walkdown scope should be defined to cover the requirements of the selected safety evaluation approach within the assessment conditions defined in para. 5.17. The purpose of the seismic evaluation walkdown typically includes the following:

(a) To collect information that can be used in refining the selected SSCs list;

(b) To observe and record the current as-built condition of SSCs included on the list;

(c) To verify the screening of SSCs based on very low or very high seismic capacities;

[31] It is recommended that seismically rugged SSCs be retained in the plant response logic model and assigned nominally high capacities, rather than removing them from the logic model altogether.

(d) To identify conditions in these SSCs, anchorage or their configuration (e.g. known or suspected seismically vulnerable details) for consideration in their seismic capacity evaluation;

(e) To identify the realistic failure modes of each SSC that may prevent achieving an acceptable end state;

(f) To collect key data such as dimensions that may be required in capacity evaluations;

(g) To identify SSCs that may result in seismic spatial interactions (paras. 5.20(c), 5.20(d), and 5.20(e)) not previously identified, and collect the necessary information to identify their relevant failure modes, the failure consequences, and the affected SSCs;

(h) To identify and report 'seismic housekeeping' items that can be easily addressed by the installation to reduce obvious vulnerabilities, such as temporary or left-in-place equipment that may result in seismic interactions (e.g. scaffolding, ladders, carts), missing fasteners, unsecured light fixtures, and unrestrained stored items.

5.26. The evaluation walkdown process should include preparatory activities, a preliminary walkthrough, a walkdown plan; walkdown guidance, detailed walkdowns, post-walkdown activities and documentation.

5.27. Preparatory activities for the walkdown should be performed to serve the following purposes:

(a) Plant familiarization though review of systems diagrams, layout and other drawings, previous seismic evaluations, and available documentation from prior walkdowns;

(b) Assembling a database of selected SSCs. SSC entries should include the data available prior to the walkdown and be populated later by data collected during the walkdown;

(c) Reviewing the selected SSCs list for completeness;

(d) Reviewing the selected SSCs list for groupings of similar SSCs and their locations;

(e) Identifying SSCs and the areas that may require special access and safety requirements;

(f) Identifying selected SSCs and areas for the preliminary walkthrough (see para. 5.28);

(g) Identifying access and training requirements for the walkdown team.

5.28. The objective of the preliminary walkthrough is to gain familiarity with the key areas of the installation and the general configuration and construction quality of its SSCs in order to facilitate the development of the walkdown plan. The preliminary walkthrough should include the senior members of the walkdown team. It should focus on observing SSCs which do not need special access requirements, confirming consistency of the information obtained from preparatory review (see para. 5.27) with the as-built conditions and identifying access requirements and similarity considerations for SSCs not previously identified in the preparatory activity.

5.29. A detailed walkdown plan and schedule should be prepared and shared with the installation ahead of the walkdown. The walkdown plan should specify the following:

(a) The selected SSC list, locations on layout drawings, and classification by SSC type and general location, and a description of the typical observation activities to be conducted;

(b) Lists of similar SSCs and lead items for detailed walkdowns or for confirmatory walkbys[32];

(c) Estimated time required for walkdowns and walkbys of typical SSC classes;

(d) List of SSCs with special access requirements and the support requested from the installation personnel (e.g. de-energizing active equipment to examine internals, opening equipment enclosures to observe anchorage, authorization for access to areas with high radiation levels or contamination, escorted access to high-security areas);

(e) Identification of areas in the installation where walkdowns of distribution systems and operator travel paths will be performed;

(f) Identification of the primary members on the walkdown team and confirmation of required access and training credentials;

(g) Identification of necessary safety and protection measures for the walkdown team members.

5.30. Before executing the walkdowns, project-specific guidance should be prepared, shared with, and reviewed by the seismic engineers on the walkdown team. The objective of this guidance should be to maximize the execution consistency in multiple walkdowns and the quality of the data collected for the subsequent evaluations. This guidance should include the following:

(a) Criteria for capacity screening and ranking[33];

[32] A 'walkby' is a brief, non-detailed walkdown with less extensive documentation, for instance, to confirm that an SSC is identical to another SSC that has been walked down and is free from potential spatial interaction concerns.

[33] Capacity ranking assigns a qualitative rank to each SSC based on the walkdown evaluation to prioritize the allocation of technical effort in subsequent seismic evaluations. A typical ranking system includes five grades: Low -seismically deficient, Medium -may be governed by failures external to the SSC design (e.g. anchorage, interaction), High -likely governed by failure of the SSC design, Rugged -very high seismic capacity, and Unknown -needs additional review.

(b) Class-specific actions for typical SSC classes (e.g. verify that batteries are vertically restrained);

(c) Actions for specific SSCs, typically informed by preparatory work and walkthrough (e.g. measure the as-built distances across specific building interfaces);

(d) Actions for walkby review of similar components;

(e) Criteria for assessing spatial interaction concerns (i.e. falling[34] and impact[35] hazards) and identification of known or suspected concerns to be examined;

(f) Criteria for assessing seismic-induced fire and flood interaction concerns and for known or suspected concerns to be examined;

(g) Procedure for area-based or sampling-based walkdowns (e.g. distribution systems);

(h) Procedure for walking down operator travel paths;

(i) Procedure for resolving potential in-process refinements to the selected SSCs list and addressing SSCs that get added to or removed from the final list;

(j) Information collection for applicable geotechnical failure modes (e.g. measurements to allow evaluating the liquefaction settlement capacity of a piping run);

(k) Instructions on documentation.

The appendix provides seismic failure mode considerations specific to different types of SSCs, which should be reviewed and used to inform the walkdown review and subsequent seismic capacity evaluations.

5.31. The detailed walkdown should review all the selected SSCs to the extent feasible. The seismic engineers should assess the construction and seismic robustness of the SSC, its support structure, anchorage, the potential consequences of credible sources of spatial and other seismic interactions that may affect it, and the potential and consequences of a seismic-induced fire, flood, or spray resulting from the failure of the SSC. Review of SSCs in inaccessible or restricted access locations may use available supplemental information (see para. 5.32). For groups of similar SSCs, a detailed review may be conducted of a lead item and less detailed walkbys may be conducted on the remaining items to confirm similarity and record any differences relevant to the capacity evaluation. For SSC classes with an excessively large number of often similar items (e.g. local instruments and passive elements), the walkbys may

[34] A common example is collapse of masonry walls located next to selected SSCs.
[35] A common example is impact on electrical cabinets containing chatter-sensitive devices by adjacent SSCs or debris.

be performed on a sampling basis. For distribution systems, the walkdown may be performed on a sampling basis in areas of interest that should be identified by the systems engineer and should focus on identifying representative as-built configurations for capacity evaluations.

5.32. Post-walkdown activities should be performed to resolve any actions that could not be performed in the field. These post-walkdown activities should be identified in the walkdown documentation. Examples of such actions include the review of photographs, construction records, and other documentation in lieu of field observation of inaccessible SSCs, SSC internals, SSC anchorage, or SSC seismic load path to the structure (e.g. obscured by a raised floor). The walkdown determinations should be made based on field observations to the extent feasible.

5.33. The seismic walkdown should be properly documented as an important product of the safety evaluation. This documentation should include the following:

(a) Summary of the walkdown planning (see paras 5.29(a) – 5.29(d)) and execution activities;

(b) The final list of selected SSCs (including justification for SSCs removed or added based on the walkdown);

(c) Summary of the main walkdown findings and recommendations relevant to the seismic capacity evaluation for the selected SSCs;

(d) Seismic evaluation data collected for all SSCs. This data is typically entered in template forms for each SSC class and should be populated in the SSC database (see para. 5.27(b)).

CONSIDERATIONS ON SEISMIC CAPABILITY OF NUCLEAR INSTALLATIONS FOR DEFENCE IN DEPTH LEVEL 4

5.34. The design and as-is conditions of the installation are required to provide adequate seismic margin to (i) protect items important to safety and avoid cliff edge effects; and (ii) protect items ultimately necessary to prevent an early radioactive release, or a large radioactive release, in the case that levels of natural hazards greater than those considered for design occur: see Requirement 17 of SSR-2/1 (Rev. 1) [3], Requirement 19 of SSR-3 [5], and Requirement 16 of SSR-4 [6].

5.35. Defence in Depth Level 4 concerning seismic hazard corresponds to the mitigation of severe accidents and prevention of large releases. The list of selected SSCs to be evaluated for adequate margins should include items needed to perform mitigation functions associated with design extension conditions [3]. For instance, the list should include the items for protection of the containment system, for installations with such a system, or for protection of the last confinement barrier against large releases, for other installations.

5.36. For prevention of early and/or large releases, the minimum seismic margin should be consistent with the containment or confinement seismic performance goal (e.g. a large early release frequency of less than 10^{-6} yr^{-1}) [13].

5.37. In seismic safety evaluation of adequate margins for items performing mitigation functions associated with design extension conditions [3], uncertainty in the seismic margin estimates should be properly considered.

SEISMIC MARGIN ASSESSMENT FOR NUCLEAR INSTALLATIONS

5.38. The SMA methodology should comprise the following steps:

(1) Selection of the assessment team (see para. 5.15);

(2) Selection of the reference level earthquake (see para. 5.5);

(3) Plant familiarization and data collection (see Section 4);

(4) Selection of success path(s) (see paras 5.17(b) and 5.39) and SSCs list (see para. 5.18);

(5) Systems walkdown (see para 5.21) and seismic evaluation walkdown (see para. 5.23);

(6) Determination of the seismic responses of SSCs for input to capacity calculations;

(7) Determination of HCLPF capacities for the selected SSCs and the installation;

(8) Specific considerations for nuclear reactors;

(9) Peer review (see Section 8);

(10) Documentation (see Section 8).

5.39. Specific guidance for the selection of the success path(s) and selected SSCs in the SMA methodology should include the following:

(a) Multiple alternate success paths may be selected that include available diversity and redundancy in the front-line and support systems. In some Member States, selection of at least two success paths for some installations is required by the regulatory body.

(b) The systems engineers should formulate the candidate success path(s) to reach an acceptable end state (see para. 5.16)[36] with input from operations personnel. Alternative paths should comprise differing operational sequences and SSCs to the extent possible.

(c) If multiple success paths are selected, one should be designated primary. The primary path should be the path judged easiest to demonstrate a high seismic safety margin thereto and be consistent with plant operational procedures and training.

(d) The seismic engineers should support the determination and prioritization of success paths by qualitative assessment of ruggedness and seismic vulnerability of the selected SSCs based on knowledge of the systems walkdown and previous seismic evaluations.

(e) Non-seismic failures of SSCs and system outages (e.g. random or maintenance-related) should be reviewed. Candidate success paths should avoid relying on SSCs with high random failure rates to the extent possible.

(f) The actions required of the operations staff should be reviewed and assessed given the common cause nature of the earthquake. Candidate success paths should avoid relying on operator actions that cannot be executed with high confidence given their timing, durations, installation operational and emergency procedures and training, and potential for confusion or interference with other responsibilities.

Determination of seismic responses

5.40. The seismic responses of buildings and other structures on the selected SSCs list should be determined for use in the generation of seismic input motions for SSCs supported by each structure. These responses may also be required for the seismic capacity evaluation of the structure if its failure modes of interest (see appendix) cannot be qualitatively screened out as relatively rugged in accordance with para. 5.22. The seismic responses of systems and components should be determined for their seismic capacity evaluations.

5.41. The SSC responses to the reference level earthquake should be determined with a high confidence level (e.g. Paragraph 5.1.2.6 of Ref. [10]). Determination of seismic responses may use probabilistic or deterministic methods of structure analysis. Probabilistic methods of analysis use best estimate-centred parameter values and include explicit treatment of uncertainties. Acceptable deterministic analysis methods should include conservative provisions to account for the effect of uncertainties (e.g. due to analytical procedures and

[36] For nuclear reactors, the function "removal of heat from the reactor…" in para. 5.16 involves control of the reactor coolant pressure, control of the reactor coolant inventory, and decay heat removal.

parameter values) and the sources of randomness associated with the reference level ground motions[37] that were not included in the seismic hazard analysis.

5.42. Determination of seismic responses for buildings and other structures should consider the following recommendations:

(a) New response analysis for the reference level earthquake ground motions using current mathematical models of the structure is recommended. Scaling of previous response analysis results (e.g. design-basis analyses) based on the ratios of reference-level to design-basis earthquake ground motions may be justifiable. Scaling is most appropriate for rock sites where the design-basis models of the structures are considered linear and median centred.

(b) For vibratory ground motion input, response spectrum analysis methods may be sufficient for structures without significant soil-structure-interaction (SSI) effects. Response history methods (also called time history methods) should be used otherwise. Equivalent linear or explicitly nonlinear methods may be used.

(c) For non-vibratory ground motion input (e.g. response to liquefaction settlement or slope deformation), quasi-static analysis methods should typically be sufficient.

5.43. Determination of seismic responses for systems and components should consider the following recommendations:

(a) The seismic responses may be determined using either new analysis of the response to seismic input motions at the system or component supports resulting from the reference level earthquake ground motions or scaling of previous response analysis results based on the ratios of the seismic input motions to the component/system, or physical testing.

(b) For vibratory ground motion input, analysis of the component or system response may be performed as coupled or uncoupled with the supporting structure model. Coupled response analysis should be used if significant dynamic interaction effects are expected.

(c) For non-vibratory ground motion input, quasi-static analysis methods should typically be sufficient.

Determination of HCLPF capacities for the selected SSCs and the nuclear installation

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[37] For reference, modern PSHAs incorporate most sources of ground motion randomness. One common exception is randomness due to earthquake component-to-component variability.

5.44. The seismic capacities of the selected SSCs should be characterized using HCLPF capacities. The HCLPF capacity[38] of an SSC is expressed function of the hazard parameter (PGA or spectral acceleration) corresponding to the scale factor[39] on the reference level earthquake ground motions at which there is at least 95% confidence of a 5% probability of failure. It may alternatively be represented by an earthquake motion level at which the expected (mean) probability of failure is 1% or lower.

5.45. The determination of HCLPF capacities should be performed by the seismic engineers. More detailed seismic capacity evaluations should be performed for the SSCs with relatively low HCLPF that are required in each success path. More simplified conservative, bounding-case, or screening-based capacity evaluations may be performed for other SSCs in each success path without affecting the success path HCLPF capacity.

5.46. The HCLPF capacity of a success path should be taken equal to the HCLPF capacity for the SSC with the lowest HCLPF capacity in the path. More than one independent success paths should be considered. The installation-level HCLPF capacity may be taken equal to that of the success path with the highest HCLPF capacity.

5.47. The reference level earthquake, and the installation-level and SSC HCLPF capacities should be reported. The weak link(s) in each success path should be identified for consideration of potential improvements or other actions (see Section 7).

Considerations for nuclear power plants

5.48. Seismic margins of the containment and confinement systems for nuclear power plants should be determined. Items such as penetrations, equipment and personnel hatches, impact between structures, and containment performance under elevated temperature and pressure caused by core damage should be reviewed. Credible potential seismic weak links in the containment and confinement systems should be explicitly included in the success path HCLPF capacity determination.

5.49. A detailed walkdown inside containment to verify that all small lines in a nuclear power plant can withstand the reference level earthquake is resource-intensive and possibly impractical due to (i) the radiation exposure hazard to the walkdown team, and (ii) the challenges of an exhaustive review of potential seismic spatial interactions affecting small lines in a crowded space. As a practical alternative, the SMA may be performed by ensuring that any

[38] Determining HCLPF capacities can and is often performed using deterministic evaluation methods similar to following design code procedures (e.g., the conservative deterministic failure margin method) in lieu of explicit propagation of uncertainties in the seismic capacity evaluation.

[39] The scale factor is to be multiplied by the PGA or Spectral Acceleration of the RLE, in order to get the HCLPF

success path is capable of sustaining concurrently the loss of offsite power and a small loss of coolant accident inside the containment.

PSA-BASED SEISMIC MARGIN ASSESSMENT FOR NUCLEAR INSTALLATIONS

5.50. The PSA-based SMA methodology should comprise most of the same steps of the SMA methodology (see para. 5.38), with the following exceptions:

(a) The selection of success path(s) (Step 4) is replaced by the accident sequence event tree and fault tree analysis;

(b) The identification of the selected SSCs list (Step 4) is based on the requirements of the accident sequence analysis;

(c) Determination of HCLPF capacity for the installation (Step 7) is performed differently.

(d) Enhancements of PSA-Based SMA may include Human Errors and Non-Seismic Random Failures

5.51. Development of the accident sequence event trees and fault trees logic model should be performed following the SPSA methodology (see paras 5.56 and 5.57).

5.52. The selected SSCs list should be identified similar to the selected SSCs list for the fragility evaluation in the SPSA methodology (see para. 5.58).

5.53. Determination of the HCLPF capacities for the selected SSCs is typically performed in a similar way to the SMA method. Depending on the desired end-product of the safety assessment, the following refinements should be considered:

(a) Development of conservatively biased seismic fragility estimates for the SSCs. This can be performed by assigning a generic or estimated value of the variability to define a lognormal function anchored to the HCLPF capacity at 1% mean probability of failure[40].

(b) Development of detailed seismic fragilities (i.e. similar to the SPSA method) for SSCs that are identified to govern the installation-level HCLPF capacity.

5.54. The installation-level HCLPF capacity should be determined by incorporating all minimal cut-sets that can lead to an unacceptable end state. It may be computed following one of the following two approaches:

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[40] In this case, an estimate of the variability biased low is conservative, since the fragility curve is anchored to a low capacity value, the 1% point.

(a) The ‘min-max’ approach: Each cut-set HCLPF capacity may be taken equal to the HCLPF capacity for the SSC with the highest HCLPF capacity in the cut-set[41]. The installation-level HCLPF capacity should be the lowest cut-set HCLPF capacity.

(b) The explicit quantification approach: An estimated fragility curve may be derived for each cut-set from the seismic fragilities (and non-seismic failure probabilities) of the cut-set components using a Boolean AND gate. An estimated fragility curve for the installation may be derived from the cut-set fragilities using a Boolean OR gate. The installation-level HCLPF capacity may be computed by identifying the 1% mean probability of failure point on the latter fragility curve.

5.55. The reference level earthquake, and the installation-level and all significant cut-set HCLPF capacities should be reported. The weak-link cut-sets, the corresponding accident sequences, and the failure modes and HCLPF capacities of SSCs leading to these accident sequences should be identified for consideration of potential improvements or other actions (see Section 7). Estimated fragility curves for the installation and the weak-link cut-sets, if developed, should also be reported.

SEISMIC PROBABILISTIC SAFETY ASSESSMENT FOR NUCLEAR INSTALLATIONS

5.56. The SPSA methodology comprises most of the same steps of the SMA methodology (see para. 5.38), with the following modifications:

(a) Step 4 should be replaced by the development of the accident sequence event tree and fault tree logic model and the identification of the selected SSCs list accordingly;

(b) Human reliability analysis for operator actions in the context of a seismic event should be added;

(c) Step 7 should be replaced by seismic fragility evaluation of the SSCs and seismic risk quantification for the nuclear installation.

5.57. The accident sequence logic model should include the analysis of potential seismically induced initiating events, and installation response considering the impact of the seismic event on SSCs, and operator actions. For example, the most popular approach in the Member States is to use seismic event trees to model accident sequences and fault trees to model basic seismic events[42]. If the nuclear installation has an existing internal events PSA logic model, which is typically a regulatory requirement for nuclear power plants, the seismic accident sequence logic

[41] The min-max approach produces estimates that are more approximate than the explicit quantification approach.
[42] Ref. [10] provides a more detailed description.

model should be developed by modifying the internal events logic model to account for seismic-induced failures and initiating events that are not included in the internal events PSA. For example:

(a) The common cause nature of seismic events imposes concurrent demands on the SSCs in the installation and on surrounding infrastructure and may lead to simultaneous failures whose correlation should be considered in the logic model.

(b) The range of seismic ground motions represented by the seismic hazard curve range from moderate to very large earthquakes. The resulting probabilistic distributions of seismic demands at the plant level led to distribution of the core damage frequency, large or early release frequency or other risk metrics of interest function of the hazard parameter.

(c) Earthquakes might cause initiating events not applicable to internal events PSA.

(d) Earthquakes might cause failures of passive SSCs such as structures and distribution systems that are not included in the internal events PSA.

(e) Earthquakes might result in seismic interaction failures (e.g. seismic-induced fire).

(f) SPSA accident sequence logic should include both potential seismic and non-seismic (e.g. random) SSC failures within the time required to reach an acceptable end state.

5.58. The system logic model[43], either new or modified from an existing internal events PSA model, should include all credited systems that are relied upon to prevent the progression of accidents due to seismic-induced initiating events to an unacceptable end state [15]. Existing accident sequence models (e.g. event trees) should be modified or supplemented by new ones unique to the SPSA (e.g. failure of major structures that lead directly to unacceptable end states). System reliability models (e.g. fault trees) should be modified to include all credible seismic-induced and non-seismic failure modes and to include as applicable credited recovery actions (e.g. operator intervention and mitigation systems). Common-cause correlations between basic events should be modelled.

5.59. The selected SSCs list for the seismic evaluation walkdown should include all the SSCs whose seismic-induced failures contribute to the basic events in the accident sequence logic model. This list typically includes significantly more SSCs than are needed for the SMA methodology, which only involves including SSCs sufficient to achieve a limited number of

[43] For nuclear power plants, this system logic model is commonly referred to as 'seismic plant response model'.

success paths. The selected SSCs list for the fragility evaluation should be shortened by excluding the SSCs screened in para. 5.22 and assigning them nominally high or low fragilities.

5.60. Determination of seismic responses of SSCs should generally be consistent with the recommendations provided in paras 5.40–5.43. However, in the SPSA methodology, the probability distributions of the seismic responses should be characterized in addition to generating high-confidence conservative response estimates for HCLPF computations. This characterization should be performed by using median-centred values and associated variabilities of the input parameters (e.g. material properties) and analytical models consistent with the reference earthquake ground motion level.

5.61. Fragility curves should be developed for items on the selected SSCs list. A fragility curve should characterize the probability of failure of an SSC conditioned on an earthquake loading intensity parameter. The SSC failure mode(s) evaluated for each SSC should be causally related to the basic events in the system logic model. Earthquake intensity is typically characterized by a ground motion parameter (e.g. PGA) and may alternatively be characterized by a local parameter (e.g. in-structure acceleration). The variability represented by each fragility curve should include the effects of inherent randomness and epistemic uncertainty on the corresponding SSC conditional probability of failure.

5.62. Seismic fragility evaluations should be performed at a level of rigour appropriate for the risk significance of the SSC. The following three approaches represent an ascending level of rigour:

(1) Generic fragility curves may be used for SSCs with negligible contribution to seismic risk. These may include nominally low and nominally high generic fragilities for SSCs screened in accordance with para. 5.22, and database-based (i.e. not component- and installation-specific) fragilities for other SSCs that meet certain inclusion rules[44].

(2) HCLPF capacity-based fragilities may be developed as described in para. 5.53(a). These fragilities should be sufficiently component- and installation-specific to be used for significant risk contributors. The use of these fragilities is not recommended for dominant risk contributors.

(3) Detailed fragilities incorporating expected SSC seismic responses and capacities and explicit treatment of variability due to uncertainty and randomness may be developed and used for risk-significant SSCs. The use of these fragilities is recommended for

[44] The SSCs assigned generic fragilities should be confirmed in the final risk quantification to have no significant risk contributions, which may require refinement iterations.

dominant risk contributors.

5.63. Assessment of human failure event probabilities should be performed considering the unique challenges of earthquakes and the level of damage, confusion, concurrent genuine and spurious failure alarms, and potential loss of indicator signals on shaping human performance. More guidance on human reliability modelling can be found in DS523 [15] and Ref. [21].

5.64. Risk quantification should be performed by combining the SSC fragilities, minimal cut-set Boolean math, and seismic hazard curves over an earthquake intensity parameter range of interest. The installation-level fragility curve should be computed explicitly at each intensity level from the SSC fragilities, non-seismic failure rates, and human failure probabilities in accordance with para. 5.54(b). This fragility curve should be integrated with the earthquake severity occurrence rates according to the hazard curve to compute the annual frequency of unacceptable performance. Depending on the safety evaluation objectives and regulatory requirements, this annual probability may be determined as a point estimate of the mean value or as a probability distribution.

5.65. The following SPSA outcomes should be reported:

(a) The frequencies of unacceptable end states (e.g. core damage, large early release);

(b) Description of the major seismic-induced initiating events and safety or mitigation functions included in the system logic model;

(c) Lists of seismic fragilities and non-seismic failure rates developed for all SSCs and human error probabilities developed for operator actions;

(d) Identification of the risk-significant accident sequences, seismic-induced failures and associated SSCs, non-seismic failures, and operator actions to allow understanding the likely accident scenarios and consideration of potential improvements or other actions (see Section 7);

(e) Identification of the installation-level fragility curve, the range of earthquake intensity that contribute most significantly to seismic risk, and any potential cliff edge effects;

(f) If applicable, identification of safety-related SSCs whose contribution to seismic risk is negligible for potential consideration in risk informed design decisions (see Section 7);

(g) Assessment of the sensitivity of the results to major modelling assumptions;

(h) Uncertainty ranges of annual frequencies and identification of their major contributors.

6. EVALUATION OF SEISMIC SAFETY FOR INSTALLATIONS OTHER THAN NUCLEAR POWER PLANTS

6.1. This section provides guidance on the seismic safety evaluation of a broad range of nuclear installations (see para. 1.11) other than nuclear power plants.

6.2. Seismic safety evaluation of nuclear installations other than nuclear power plants should be based on graded approach, as recommended in the following paragraphs. The intent is that the evaluation verifies that the performance of the SSCs important to safety within the installation is acceptable.

6.3. The methodology to be followed in evaluating nuclear installations other than power plants is essentially identical to that for nuclear power plants; however, the end state will be unique for each installation. In the case of a nuclear power plant the end state most common is to prevent core damage (i.e. safely shut down the plant and remove residual heat from irradiated fuel) and to prevent a large early release. For nuclear installations other than nuclear power plants, the end state may be to prevent leakage of aerosolized contaminants, for instance, in the case of a fuel processing facility. Once the desired end state is established, the methodology for assessing the installation's ability to achieve this end state should be evaluated using the SPSA, PSA based SMA, or SMA approaches presented in Sections 3 and 5 of this Safety Guide.

HAZARD CATEGORY OF A NUCLEAR INSTALLATION

6.4. For the purpose of seismic safety evaluation, each SSC that is required to perform a seismic risk mitigating function should be assigned to a seismic design class (SDC), which is a hierarchical category that denotes its importance in mitigating seismic hazard (see Section 9 of DS490 [13]). The seismic design class assigned to the SSC is a function of the severity of adverse radiological and toxicological effects — on workers, the public, or the environment — of the hazards that might result from the seismic failure of the SSC[45]. Table A–1 in the annex to this Safety Guide provides an example of criteria for use in determining the seismic design class. A framework like the one given in the annex of this Safety Guide or in Table 2 of DS490 [13] should be used in establishing the seismic design class for the SSCs of the nuclear installation.

[45] For example, in the United States of America, nuclear installations are assigned to seismic design classes (see appendix). SSCs that perform a safety function are placed into a design category based on the unmitigated consequences that may result from the failure of the SSC by itself or in combination with other SSCs. Consideration is given to consequences to the worker, the public, or the environment.

6.5. A similar approach should be used to categorize a nuclear installation into a hazard category, as a function of the risk to the public, workers, or the environment from a potential unmitigated radioactive release from the installation (see Section 9 of DS490. [13]). An example of possible nuclear installation hazard categories (high, moderate and low) is also provided in Table A–1.

6.6. A conservative screening process should be used prior to categorizing a nuclear installation. In this process, it is assumed that the complete radioactive inventory of the installation is released by a seismically initiated accident. If this screening demonstrates that there are no unacceptable consequences for workers, the public, or the environment, and no other specific requirements are imposed by the regulatory body for such an installation, the installation may be screened out from the seismic safety evaluation. For equipment or tanks that need to be operated and/or maintained in controlled atmosphere (e.g. inert glove boxes, high level waste storage tanks), the possible consequences (fire and/or explosion) of the failure of the controlled conditions should be considered in the screening process. If, even after such screening, some level of seismic safety evaluation is needed, national seismic codes for industrial facilities may be used.

6.7. If the results of the screening process show that the consequences of the unmitigated releases are unacceptable, a seismic safety evaluation of the installation should be carried out. For this purpose, the seismic hazard at the site should be determined, in accordance with the recommendations provided in paras 2.19–2.25. The seismic input for the safety evaluations should not be less than a peak ground acceleration of 0.1 g at the foundation level.

SELECTION OF PERFORMANCE TARGETS FOR EVALUATION OF SEISMIC SAFETY FOR INSTALLATIONS OTHER THAN NUCLEAR POWER PLANTS

6.8. A 'performance target', expressed as a mean annual frequency of failure due to the earthquake hazard, should be assigned to each of the seismic design classes described in para. 6.4. The performance targets represent the acceptable calculated mean annual frequency of seismic induced failure of SSCs within a seismic design class (See Section 9 of DS490 [13]). The failure of an SSC is associated with a particular failure mode and a limit state[46]. Table A–2 in the annex to this Safety Guide provides an example of performance targets selected for different seismic design classes.

[46] A 'limit state' is the limiting acceptable condition of the SSC, so that its intended safety function is kept. For example, the failure limit state for a column that is supporting a safety class pressure vessel would be the loss of load carrying capacity through either buckling or collapse. For a mechanical pump with a safety function that requires operability, the failure limit state would be the loss of operability.

6.9. A performance target should also be defined for the nuclear installation, as the maximum mean annual frequency of unacceptable performance of the installation due to the earthquake hazard (e.g. occurrence of unacceptable radioactive releases).

6.10. The overall performance of the installation (annual frequency of failure) is the result of convolving the seismic hazard (hazard curves) with the installation-level fragility (conditional probability of unacceptable installation behaviour, for each level of earthquake severity). The installation-level fragility results from the seismic capacities of the SSCs and it can be obtained from them using simplified or more rigorous methods[47]. Therefore, appropriately defined seismic design classes and performance targets for the SSCs within the installation should lead to meeting the performance target selected for the nuclear installation as a whole.

6.11. There is a correlation between the hazard level used for design, the seismic margin achieved by the design and the installation level performance goal, as described in Section 7 of DS490 [13]. In this context, the minimum required seismic margin is related to the seismic design basis and the target performance goal of the installation. Seismic margin in this context can be regarded as a surrogate for the installation level performance goal. The basis for the graded approach is described in paras 6.12 and 6.13.

GRADED APPROACH FOR ACHIEVING SELECTED PERFORMANCE TARGETS IN THE EVALUATION OF SEISMIC SAFETY FOR NUCLEAR INSTALLATIONS

6.12. A graded approach should be used for demonstrating that nuclear installations meet the performance targets (see para. 6.9) assigned to them. The level of rigour applied in the safety evaluations should range from simple, for low hazard installations, to complex, for high hazard installations, as follows:

(a) For low hazard installations, the seismic capacity evaluation methods for the selected SSCs may be based on simplified but conservative static or equivalent static procedures, similar to those used for industrial hazardous facilities, in accordance with national practice and standards. Similarly, the seismic hazard to be used in these evaluations may be taken from national building codes and map and does not need to be taken from a site-specific PSHA. If a PSHA exists, however, the seismic hazard from that study may be used.

[47] Those methods are discussed in Section 5. In deterministic SMA, the simplest method, it is usually assumed that the installation-level fragility can be derived just from the seismic capacity of the weakest SSC required to bring the installation to a safe state and keep it in a safe state during a specified period of time.

(b) For selected SSCs of installations in the moderate hazard category, the seismic safety evaluation should typically be performed using the methodologies described in Section 5, but the corresponding performance target is set lower than for installations in the high hazard category (see annex). Either the SMA or SPSA approach may be used.

(c) For selected SSCs of installations in the higher hazard category, methodologies for seismic safety evaluation as described in Section 5 should be used (i.e. no application of a graded approach).

6.13. In a particular SSC, the performance target associated with a failure mode should be demonstrated by one of the following methods:

(a) Showing compliance with a design code that was developed with a reliability-based approach[48]. The design level earthquake should be selected based on an annual frequency of exceedance that is consistent with the performance target for the particular SSC.

(b) Showing adequate seismic margin beyond a site specific reference level earthquake. The reference level earthquake should be selected based on an annual frequency of exceedance that is consistent with the performance target for the particular SSC.

(c) Explicit computation of the annual frequency of failure, using a SPSA. In the SPSA, it is very important to use ground motion from a site specific PSHA, and that the SSCs important to safety have been properly categorized and the appropriate limit states have been defined.

[48] 'Reliability-based approach' refers to an approach in which design code requirements are intended to achieve a predefined maximum probability of failure for a given set of loadings or external actions.

7. USE OF SEISMIC SAFETY EVALUATION RESULTS FOR NUCLEAR INSTALLATIONS

POST-EARTHQUAKE ACTIONS BASED ON THE SEISMIC SAFETY EVALUATION OF NUCLEAR INSTALLATIONS

7.1. The nuclear installation post-earthquake procedures, including emergency plans, procedures for post-earthquake inspections, and plans for re-start, should consider the lessons learned in the seismic safety evaluation. As a result of the seismic safety evaluation, the facility owner and the regulatory body will have a better understanding of those SSCs that are important to seismic safety. They will also have a better understanding of any seismic weak links associated with the nuclear installation. All this information should be taken into account in the definition of post-earthquake actions.

RISK-INFORMED DECISIONS BASED ON THE SEISMIC SAFETY EVALUATION OF NUCLEAR INSTALLATIONS

7.2. The programme for seismic safety evaluation of an existing nuclear installation may result in a subset of the selected SSCs that do not meet the established acceptance criteria. If that is the case, then consideration should be given to physical upgrades or strengthening programmes. The decision about implementing this kind of programme should consider the potential seismic risk reduction versus the implementation costs, and the time-at-risk concept, considering the remaining life of the installation.

7.3. In many instances there are alternate solutions for reducing the risk to an appropriate level. These may include, for instance, the following:

(a) Reducing the material at risk to moderate or low inventory levels, such that less demanding performance targets can be met;

(b) Upgrading the facility by strengthening the SSCs that limit the installation to meet the minimum seismic margin or are significant risk contributors;

(c) Hardening the primary containment such that the material at risk for which the ‘unmitigated radioactive release’ amount was calculated is reduced.

Regardless of the option taken, sufficient diligence should be exercised to be able to quantitatively calculate the reduction in risk associated with the option. This risk reduction will come in the form of an increase in the computed margin if a seismic margins assessment method was used, or in the form of a decrease in the annual frequency of failure of the selected SSCs if a SPSA method was used.

7.4. The cost associated with each of the alternate solutions should be quantified.

7.5. The risk-informed decision should look at the alternate solutions and consider both cost and the risk reduction. Options that are easy to implement and for which there is very little cost involved should be implemented. For options that are very costly and for which there is very little risk reduction, the operating organization of the nuclear installation should work with the regulatory body to determine if the costs exceed the benefits from the small amount of risk reduction.

DESIGN OF MODIFICATIONS IN EXISTING NUCLEAR INSTALLATIONS BASED ON THE SEISMIC SAFETY EVALUATION

7.6. Modifications to nuclear installations are required to be designed in accordance with recognized codes and standards and, at a minimum, to the original design standards. Design of upgrades needs to meet the design criteria and performance targets appropriate for the hazard category of the nuclear installation. More considerations for upgrading are provided in Ref. [10].

7.7. For the design of modifications, the seismic demand and the acceptance criteria should be established in compliance with the requirements of the regulatory body. The design for seismic upgrades should consider the available space and the working environment (e.g. radiation exposure). Upgrade concepts should (i) accommodate the existing configuration, to the extent possible, and (ii) observe seismic interactions based on the field inspection.

7.8. The type of upgrading of existing structures or substructures depends on the additional seismic capacity that is needed. As a consequence, the effects of the upgrades on interconnected systems and components (e.g. distribution systems) should be evaluated. Once the design of the final upgrade is completed, the need for a dynamic analysis to generate new in-structure response spectra and displacements should be evaluated.

7.9. The type of upgrading of existing systems and components depends on the additional seismic capacity that is needed. Generally, the following types of upgrading should be considered:

(a) Upgrading of anchorage, both for equipment and for supports in distribution systems;

(b) Provision of additional lateral restraint, for distribution systems;

(c) Upgrading of electromechanical relays, to models with larger seismic capacity.

(d) Upgrading of critical components, to models with larger seismic capacity.

7.10. An important consideration is to prioritize the upgrades based on contribution to the risk reduction of the installation on a cost-benefit basis.

CHANGES IN PROCEDURES BASED ON THE SEISMIC SAFETY EVALUATION OF NUCLEAR INSTALLATIONS

7.11. Existing procedures for the inspection of SSCs important to safety should be reviewed to ensure that the seismic capacity in the critical limit state for any SSC is not jeopardized as a part of normal operations (e.g. provision of scaffolding or temporary access items that may seismically interact with items important to safety).

8. MANAGEMENT SYSTEM FOR SEISMIC SAFETY EVALUATION FOR NUCLEAR INSTALLATIONS

APPLICATION OF THE MANAGEMENT SYSTEM TO SEISMIC SAFETY EVALUATION FOR NUCLEAR INSTALLATIONS

8.1. The management systems for each of the organizations involved in the seismic safety evaluation should be established and implemented before the start of the seismic safety evaluation programme [22] [23]. The management system is required to cover all processes and activities of the seismic safety evaluation, in particular, those relating to data collection and data processing, field and laboratory investigations, and analyses and evaluations that are within the scope of this Safety Guide. It is also required to cover those processes and activities corresponding to the upgrading phase of the programme.

8.2. Owing to the variety of investigations and analyses to be performed and the need for engineering judgement by the team implementing the seismic safety evaluation, technical procedures that are specific to the project should be developed to facilitate the execution and verification of these tasks.

8.3. A peer review of the implementation of the seismic safety evaluation methodology should be performed. In particular, the peer review should assess the elements of the implementation of the SMA, SPSA or PSA-based SMA methodologies against the recommendations of this Safety Guide and current international good practices used for these evaluations.

8.4. The peer review should be conducted by experts in the areas of systems engineering, operations (including fire prevention and protection specialists), earthquake engineering and other specialists depending on the focus of the seismic evaluation. Peer review should be performed at different stages in the evaluation process, as follows:

(a) The review of systems and operations should be performed first, coinciding with the selection of the success paths for SMA or the tailoring of the internal event system models for the SPSA or the PSA-based SMA.
(b) Seismic capability peer reviews should be performed (i) during and after the walkdown, and (ii) after a majority of the HCLPF values (for SMA or PSA-based SMA) or fragility functions (for SPSA) for the SSCs have been calculated. The seismic capability peer review should include a limited plant walkdown, which may coincide with a part of the plant walkdown or may be performed separately.

The findings of the peer reviews should be documented.

DOCUMENTATION AND RECORDS FOR SEISMIC SAFETY EVALUATION FOR NUCLEAR INSTALLATIONS

8.5. An important component of the management system is the definition of the documentation and records to be developed during the execution of the programme of seismic safety evaluation, and of the final report to be produced as a result of it. Detailed documentation should be retained for review and future application.

8.6. Typical documentation of the results of the seismic safety evaluation should be a report documenting the following:

(a) Methodology and assumptions of the assessment;

(b) Selection of the reference level earthquake(s);

(c) Composition and credentials of the evaluation team;

(d) Verification of the geological stability at the site (see para. 2.19(a));

(e) Success path(s) selected, justification or reasoning for the selection, HCLPF of path and controlling components (for the SMA);

(f) Summary of system models and the modifications introduced to the internal event models for the SPSA and PSA-based SMA;

(g) A table of selected SSC items with screening (if any), failure modes, seismic demand, HCLPF values (for the SMA and PSA-based SMA) and fragility functions (for the SPSA) tabulated;

(h) For the SPSA, results of quantification of the sequence analysis, including core damage frequency, dominant core damage sequences, large early release frequency or containment failure frequency, and dominant sequences for failures of the confinement function;

(i) Summary of seismic failure functions for front-line and support systems modelled, including identification of critical components, if any, for the SPSA;

(j) Walkdown report summarizing findings and system wide observations, if any;

(k) Operator actions needed and the evaluation of their likely success;

(l) Containment and containment system HCLPFs or fragility functions (if needed);

(m) Treatment of non-seismic failures, relay chatter, dependences and seismic induced fire and flood;

(n) Peer review reports.

8.7. In addition to the above information, the following detailed information should be retained:

(a) Detailed system descriptions used in developing the success path(s), system notebooks and other data (for SMA);

(b) Detailed documentation of the development of the SPSA and PSA-based SMA models, in particular, those aspects pertaining to the modifications of the internal event PSA models to account for seismic events;

(c) Detailed documentation of all walkdowns performed, including SSC identification and characteristics, screening (if appropriate), spatial interaction observations for the seismic system, and area walkdowns usually performed for systems such as cable trays and small bore piping, and to evaluate seismic induced fire or flood issues;

(d) HCLPF (for SMA and PSA-based SMA) or fragility function (for SPSA) calculation packages for all selected SSC items;

(e) New or modified plant operating procedures for the achievement of success paths;

(f) List of records and their retention times.

CONFIGURATION MANAGEMENT FOR SEISMIC SAFETY EVALUATION FOR NUCLEAR INSTALLATIONS

8.8. The operator should implement a configuration management programme to ensure that, in the future, the design and construction of modifications to SSCs, the replacement of SSCs, maintenance programmes and procedures, and operating procedures do not invalidate the results of the seismic safety evaluation.

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APPENDIX

SEISMIC FAILURE MODE CONSIDERATIONS FOR DIFFERENT STRUCTURES, SYSTEMS AND COMPONENTS IN NUCLEAR INSTALLATIONS

A.1. The failure mode considerations identified in this appendix are typical of common classes of SSCs in nuclear installations, based on experience with previous safety evaluations. These failure modes should be reviewed and used if applicable to inform the walkdown review and seismic capacity evaluations.

SEISMIC FAILURE MODES FOR BUILDINGS AND STRUCTURES IN NUCLEAR INSTALLATIONS

A.2. There are multiple potential structural failures in buildings and complex structures. Only failure modes that might lead to accident progression to an unacceptable end state should be considered. The experience of qualified seismic engineers is essential in determining the potential failure modes of interest. This experience should be informed by the seismic walkdown and the review of structural drawings and previous evaluations. These failure modes may be broadly classified as follows:

(a) Local failures of structural members that undermine the support of SSCs important to safety;

(b) Major failures of structural components that lead to unacceptable deformations, misalignments, and other causes of damage or loss of function for supported SSCs;

(c) Major failures of structural component that lead to severe damage or collapse;

(d) Global structure instability (e.g. sliding, overturning, and foundation bearing failure).

(e) Failures of structures that are part of containment or confinement systems, which can lead to a radioactive release .

A.3. Relative movements between adjacent structures should be considered with respect to the existing separations and whether they are constructed on common or separate foundations. The associated potential failure modes may be classified as follows:

(a) Major failure of one structure due to impact with a significantly heavier structure;

(b) Local failures in the structure exteriors due to impact (e.g. punching of walls);

(c) Failures of chatter-sensitive electrical components due to impact between structures;

(d) Failures of other shock-sensitive SSCs or SSC supports in the vicinity of impact;

(e) Failures of distribution systems or their supports due to separations between structures.

A.4. Seismic capacity evaluation of structures should be based on available construction information. The review of the structures during the walkdown should focus on supplementing this information with as-built observations. Example data to focus on include the following:

(a) Potential signs of degradation or distress, such as corrosion, exposure of reinforcement, and concrete spalling;

(b) Records of structure connections that appear to be field-modified from standard connections;

(c) Measurements of interface separations between buildings and description of gap filler materials, if present;

(d) Survey of equipment that enables the estimation of temporary loading during maintenance or refuelling conditions[49];

(e) Survey of as-built versus as-designed roof equipment, storage, and roofing materials.

SEISMIC FAILURE MODES FOR MECHANICAL EQUIPMENT IN NUCLEAR INSTALLATIONS

A.5. Mechanical equipment in nuclear installations typically includes process equipment, pumps, tanks and heat exchangers, fans and air handlers, and valves. The review of their seismic capacity should include the quality of anchorage, support structure, mounting configuration, equipment construction, and the ability of the equipment to function. Some damage to the equipment is tolerable if it does not compromise its ability to perform its credited function. The functional assessment includes time considerations (e.g. whether the component is needed to operate during or after the earthquake shaking and the duration of that operation without outside support). It should also include an assessment of potential seismic interactions and the flexibility of attached distribution system lines.

A.6. The review of mechanical equipment with considerable oil content should consider potential failure modes that can result in oil leakage and subsequent fire (e.g. breakage of oil level sight glass monitors on pumps).

A.7. Mechanical equipment with substantial piping (e.g. tanks, heat exchangers, and pumps)

[49] While equipment masses may be estimated from the structure design drawings for individual floors, some areas may be designed for heavy loads that are only experienced infrequently, typically when the installation is not in operation. A typical example of this is a laydown area where a nuclear reactor head is temporarily stored during a refuelling outage.

should also be reviewed for potential nozzle loads from the inertia of the attached piping.

A.8. The review of mechanical equipment supported on vibration isolators should consider their potential failure due to shaking-induced displacement.

A.9. The mountings of valves and pump shafts independently supported from the attached piping and pumps, respectively, should be reviewed for potential differential motion failures.

SEISMIC FAILURE MODES FOR ELECTRICAL EQUIPMENT IN NUCLEAR INSTALLATIONS

A.10. Electrical equipment includes instrumentation and control panels, switchgear, transformers, inverters, generators, and batteries. The review of their seismic capacity should include the same considerations identified in paras. A.5 and A.6. Many types of electrical equipment are typically vulnerable to spray (e.g. from overhead fire protection sprinklers).

A.11. The review of electrical cabinets should consider whether the internal instruments and components are positively and securely attached inside the enclosure and whether their mountings are stiff or flexible. In particular, if the internal instruments and components are on a structure that can be pulled out from the cabinet from the viewpoint of maintenance, the amplification of seismic motion due to this structure should be considered

A.12. The review of electrical cabinets that contain chatter-sensitive components should consider whether they have adequate spacing or are bolted to the adjacent cabinets to prevent pounding.

A.13. The review of diesel generators should include the exhaust and ventilation systems.

A.14. The review of batteries should consider whether they are adequately spaced and restrained.

SEISMIC FAILURE MODES FOR INDIVIDUAL INSTRUMENTS AND DEVICES IN NUCLEAR INSTALLATIONS

A.15. Local instruments and passive elements in nuclear installations are usually seismically rugged SSCs. The review of their seismic capacity should consider the adequacy of the mounting, flexibility of the attached lines, and potential spatial interactions. It should also consider the consequences of failure on the SSC function of interest (e.g. potential breakage of the glass cover on the reporting dial of a sensor).

A.16. Chatter-sensitive devices may include electromagnetic relays, switchgear circuit breakers, motor starters, and indicator switches for temperature, pressure, level, or flow. The review of their seismic capacity should consider the seismic qualification of the device model, height and their attachment to the equipment component that hosts them, and any spatial interaction concerns that might affect the host component or the device directly. These devices are typically very sensitive to transmitted shock waves resulting from impact or pounding. Chatter of these devices may be recoverable through operator actions. If credit is taken for these actions, an evaluation of the reliability of these actions after the earthquake, the time available to successfully implement these actions and the associated travel paths should be included in the analysis.

SEISMIC FAILURE MODES FOR DISTRIBUTION SYSTEMS IN NUCLEAR INSTALLATIONS

A.17. Distribution systems include piping, sampling points, cable trays and conduits, and ducting. These systems have typically high seismic capacities due to their relatively light weight and substantial ductility, since yielding in itself does not prevent the performance of their safety function. The seismic capacity review of these systems should be performed on an area basis (e.g. in a room or corridor) and consider representative configurations identified to be potentially vulnerable during the seismic walkdown (see para. 5.31). Seismically vulnerable conditions include the following:

(a) Differential motion between supports or attachment points;

(b) Flexible supports and other details that can allow large seismic displacements;

(c) Weak or brittle connections, supports, or anchorage;

(d) Long flexible runs connected to stiff branch lines or supports;

(e) Excessively loaded supports (e.g. multiple or overfilled cable trays or long spans);

(f) Degradation and corrosion.

SEISMIC INTERACTION CONSIDERATIONS FOR FAILURE OF SSCs IN NUCLEAR INSTALLATIONS

A.18. Common sources of spatial interaction include pounding between adjacent SSCs or their support structures, masonry walls, unsecured light fixtures, unanchored objects, overhead cranes, suspended ceilings, and temporary structures left in place (e.g. scaffolding). The seismic capacity review of potential spatial interaction sources should consider both the credibility and the consequences of the interaction. For example, a falling hazard from an unsecured lightweight overhead light fixture will have no consequence on an electrical cabinet that contains no soft targets or chatter-sensitive devices and need not be explicitly evaluated.

A.19. The review of seismic-fire interactions should consider the ignition sources previously identified in the internal fire safety assessment. Only ignition sources that can be potentially initiated by seismic-induced failure modes should be considered. This review should also include: (i) potential failure modes of items on the selected SSCs list that can lead to fire ignition that spreads to adjacent SSCs, and (ii) additional SSCs identified during the area-based seismic walkdowns as potential sources (e.g. non-safety-related high-voltage electrical cabinets or transformers) in applicable proximity to any of the selected SSCs. The fire area affected by each potential ignition source should be determined by the systems engineer considering the presence of combustibles, fire protection, and possible spread due to the failure of boundaries.

A.20. The review of seismic–flood interactions should consider the flood sources previously identified in the internal flood safety assessment. Only the flood sources that can be potentially initiated by seismic-induced failure modes should be considered. This review should also include: (i) potential failure modes of items on the selected SSCs list that can lead to flood that spreads to adjacent SSCs, and (ii) additional SSCs identified during the area-based seismic walkdowns as potential sources (e.g. unanchored tanks, non-ductile piping, and non-safety-related heat exchangers) that can affect any of the selected SSCs. The flood area affected by each potential source should be determined by the systems engineer considering the volume of released fluid, flow paths within a floor plan and from higher to lower elevations within a building, potential barriers or path diversions inside the building, and the configurations of the SSCs in the flooded area(s).

A.21. The review of seismic-flood and seismic-spray interactions should consider the seismic vulnerabilities of the fire protection systems and other non-ductile piping. Experience has shown that these systems are susceptible to seismic-induced shaking. Known vulnerabilities of fire protection systems include mechanical couplings, threaded pipe connections, easy-to-damage sprinkler heads (i.e. due to impact with adjacent objects) in wet systems, and inadvertent actuation of dry systems. Seismic capacity review of these systems should be performed on an area basis as described in para. A.17, considering in particular the proximity of known seismically deficient system components to spray-sensitive SSCs.

OPERATOR TRAVEL PATHS

A.22. The seismic capacities that should be reviewed depend on the understanding of the expected movements necessary to execute operator actions credited in the safety evaluation and on considering seismic-induced failures that may impede access, travel, or egress along these paths. Common potential impediments to travel include masonry walls that may collapse and block a pathway, normally shut doors that may be distorted due to seismic damage and rendered unopenable, seismic-induced fire and flood along the travel path, and blocked access to storage locations of tools.

A.23. If outside help is credited in the safety evaluation, the seismic capacity review should also consider potential failures along additional travel paths that are needed for the arrival and deployment of this help within the necessary time. Examples include critical highway bridges and road junctures, access roads to the nuclear installation, and entry points to the buildings.

SPECIFIC CONSIDERATIONS FOR SEISMIC FAILURE MODES FOR NUCLEAR POWER PLANTS

A.24. An explicit evaluation of the seismic capacity of the primary reactor system and components should be performed. A review of design documentation and previous evaluations should be performed to identify credible seismic-induced failure modes. The candidate failure modes should be evaluated using the seismic demands of the reference level earthquake to identify the governing failure mode or modes. Several governing failure modes may be identified that lead to different consequences for the installation end state.

A.25. The seismic capacity of the primary (and secondary, if applicable) containment should be explicitly evaluated. All credible seismic-induced failure modes that can lead to loss of structural integrity in the containment pressure boundary should be included.

NON-VIBRATORY GROUND MOTION-INDUCED FAILURES IN NUCLEAR INSTALLATIONS

A.26. Potential SSC failure modes due to geotechnical failure hazards that could not be screened out (see paras. 2.19 and 5.11) should be considered in the seismic walkdown and capacity review. The corresponding seismic demands are typically permanent displacements rather than accelerations. The seismic capacity review of the affected SSCs should focus on their ability to perform their credited functions when subjected to the imposed displacements. This capacity will typically depend on the flexibility and ductility of attached distribution systems, which should, if feasible, be assessed during seismic walkdowns, as follows:

(a) Settlement of structure foundations due to liquefaction or dry sand settlement may result in failure of buried distribution systems at the interface with the structure;

(b) Relative vertical displacements between adjacent structures due to differential settlement may result in failure of interconnecting distribution systems;

(c) Differential settlements under the foundations of a structure may result in permanent distortion, internal damage to structure members, and failures of attached lines;

(d) Slope displacements may result in failure of buried and aboveground lines and SSCs below the slope;

(e) Fault rupture, subsidence, and lateral spreading displacements may result in failure of buried and aboveground lines and SSCs spanning the ground displacement zone.

A.27. Potential SSC failure modes due to concomitant phenomena that could not be screened out (see paras 2.19 and 5.11) should be considered in the seismic walkdown and capacity review, for example, as follows:

(a) The seismic capacity of an upstream dam whose breach can result in flooding of the nuclear installation site should be explicitly evaluated. This seismic capacity should be mapped to the consequences on the installation in accordance with SSG-18 [14], considering the vulnerability of individual SSCs to the flood level and the lower reliability of emergency response procedures in the combined aftermath of earthquake and flood.

(b) The assessment of the consequence of a tsunami hazard on the safety functions of nuclear installations located near coastlines should include evaluating the potential malfunctioning of equipment located at a low level, such as seawater pumps, in accordance with SSG-18 [14] and IAEA Safety Standards Series No. NS-G-1.5, External Events Excluding Earthquakes in the Design of Nuclear Power Plants [24];

(c) The seismic slope stability and displacement capacity to trigger a landslide that could affect the nuclear installation site should be explicitly evaluated. The consequences of this landslide on the safety-related functions should be assessed considering the slope discharge along the landslide path and the distance to the installation.

(d) The potential for seismic failures in adjacent nuclear and industrial facilities that might affect the nuclear installation should be identified during the walkdown and reported for further assessment.

ANNEX

EXAMPLE OF CRITERIA FOR DEFINING SEISMIC DESIGN CLASSES AND PERFORMANCE TARGETS IN NUCLEAR INSTALLATIONS

SEISMIC DESIGN CLASSES FOR SSCs IN NUCLEAR INSTALLATIONS

A–1. Table A–1 provides an example of criteria for defining seismic design classes of SSCs in a nuclear installation, taken from the practice of one Member State (United States of America) [A–1]. SSCs with a safety function are assigned into one of the five classes given in the table, based on the unmitigated consequences that may result from the failure of the SSC by itself or in combination with other SSCs.

A–2. A similar approach has been used to categorize nuclear installations into high (SDC-4, SDC-5), moderate (SDC-3) and low (SDC-1, SDC-2) hazard categories, in accordance with the risk to the public, workers, or the environment from a potential unmitigated radioactive release [A–1]. These hazard categories are also shown in Table A–1.

PERFORMANCE TARGETS FOR SSCs AND NUCLEAR INSTALLATIONS FOR SEISMIC EVALUATION PURPOSES

A–3. A 'performance target' is a selected annual frequency of failure due to the earthquake hazard. Performance targets are linked to seismic design classes for SSCs. Table A–2 shows an example of selected performance targets taken from the practice of one Member State (United States of America) [A–2].

A–4. In Table A–2, the annual frequency of failure (performance target) ranges from that assumed for normal building structures in some Member States (i.e. about $P_f = 10^{-3}$ per year) to that approaching a mean core damage frequency for seismically induced core melt, which is considered acceptable in some Member States (i.e. about $P_f = 10^{-5}$ per year). The performance targets for the intermediate seismic design classes are between these two values.

TABLE A–1. SEISMIC DESIGN CLASS BASED ON THE UNMITIGATED CONSEQUENCES OF FAILURE [A-1] (COURTESY OF THE AMERICAN NUCLEAR SOCIETY)

Seismic Design Class	Hazard Category	Unmitigated Consequences of Failure		
		Worker	Public	Environment
1[a]	Low	No radiological or toxicological release consequences but failure of SSCs may place facility workforce at risk of physical injury.	No radiological or toxicological release consequences.	No radiological or toxicological release consequences.
2[a]		Radiological/toxicological exposures to workers will have no permanent health effects, may place more facility workers at risk of physical injury, or may place emergency operations at risk.	Radiological/toxicological exposures of public areas are small enough to require no public warnings concerning health effects.	No radiological or chemical environmental consequences.
3	Moderate	Radiological/toxicological exposures that may place facility worker's long-term health in question.	Radiological/toxicological exposures of public areas would not be expected to cause health consequences but may require emergency plans to assure protections.	No long-term environmental consequences are expected, but environmental monitoring may be required for a period of time.
4	High	Radiological/toxicological exposures that may cause long-term health problems and possible loss of life for a worker in proximity of the sources of hazardous material, or place workers in nearby on-site facilities at risk.	Radiological/toxicological exposures that may cause long-term health problems to an individual at the exclusion area boundary for two hours.	Environmental monitoring required and potential temporary exclusion from selected areas for contamination removal.
5		Radiological/toxicological exposures that may cause loss of life of workers in the facility	Radiological/toxicological exposures that may possibly cause loss of life to an individual at the exclusion area boundary for an exposure of two hours.	Environmental monitoring required and potentially permanent exclusion from selected areas of contamination.

Notes:

(a) ”No radiological/toxicological releases” or “no radiological/toxicological consequences” means that material releases that cause health or environment concerns are not expected to occur from failures of SSCs assigned to seismic design classes 1 or 2.

TABLE A–2. EXAMPLE OF PERFORMANCE TARGETS [I-2] (COURTESY OF THE AMERICAN SOCIETY OF CIVIL ENGINEERS)

Seismic Design Class	Hazard Category	Performance target (yr^{-1})
1	Low	$< 1 \times 10^{-3}$
2		$< 4 \times 10^{-4}$
3	Moderate	$\sim 1 \times 10^{-4}$
4	High	$\sim 4 \times 10^{-5}$
5		$\sim 1 \times 10^{-5}$

REFERENCES TO ANNEX

[A–1] AMERICAN NUCLEAR SOCIETY, «Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design, » Standard ANSI/ANS 2.26-2004, La Grange Park, Illinois, 2017.

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