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SEISMIC DESIGN OF NUCLEAR INSTALLATIONS

DRAFT SAFETY GUIDE No. DS 490

Revision of Safety Guide NS-G-1.6

FOREWORD

To be written.

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1. INTRODUCTION

BACKGROUND

1.1. This Specific Safety Guide was prepared under the IAEA's programme for safety standards for nuclear installations. It supplements and provides recommendations on meeting the Specific Safety Requirements on Safety of Nuclear Power Plants: Design [1] extending its scope to other nuclear installations than nuclear power plants. The present publication provides guidance and recommends procedures for the design of nuclear installations to cope with the effects generated by earthquakes. It supersedes the Safety Guide on Seismic Design and Qualification for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.6 (2003).

1.2. The previous versions of the IAEA Safety Guides on the evaluation of the seismic hazards and the seismic design and evaluation were ones of the most extensively used by Member States and positive feedback of their application was received from the IAEA reviews of the seismic safety of nuclear installations worldwide.

1.3. The revision of this Specific Safety Guide incorporates (i) the progress in the state-of-the-practice and research, as well as the regulatory practice in Member States, considering the lessons learned from the occurrence of recent strong earthquakes which affected nuclear installations since the publication of the previous version in 2003, (ii) recent developments and regulatory requirements on risk informed and performance based approach for assessing the safety of nuclear installations, (iii) the experience and results from seismic design conducted for new nuclear installations in Member States, and (iv) a more coordinated treatment of the design of nuclear installations against seismically induced associated geological and geotechnical hazards and concomitant events, as well as with respect to other external events affecting the nuclear installation.

1.4. This revision is also providing a clearer interface between: (i) the process for assessing the seismic hazards at a specific site, and (ii) the process for defining the related basis for design and evaluation of the nuclear installations. Thus, it bridges gaps and avoids undue overlapping on recommendations related to the two processes which correspond to and performed at different stages of the life cycle of the nuclear installation.

1.5. The process for assessing the seismic hazards at a specific site including the definition of the parameters resulting from such assessment are treated and provided by the Specific Safety Guide on Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9, [2], published in 2010 and currently under revision to ensure the clearer interface indicated in previous para 1.4 between the assessment of the seismic hazards at a site and the seismic design of the installations.

1.6. As background, an important consideration should be noticed on the difference between (i) the seismic design, and (ii) the seismic safety evaluation of nuclear installations, as indicated in the Safety Guide on Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13, [3], published in 2009. Seismic design of a new installation is distinct from the seismic safety evaluation of an existing installation in that seismic design and qualification of structures, systems and components (SSCs) is most often performed at the design stage of the installation, prior to its construction. Seismic safety evaluation is applied only after the installation has been constructed. Of course, exceptions exist, such as the seismic design of new or replacement components after construction of the installation. Conversely, the seismic safety evaluation for assessing beyond design basis earthquake of new designs prior to construction may make use of the criteria applied for seismic safety evaluation.

OBJECTIVE

1.7. The objective of this Specific Safety Guide is to provide recommendations and guidance on how to meet the safety requirements established in Ref. [1] in relation to the design aspects of a nuclear installation which are required so that an earthquake vibratory ground motion at the site, determined according to the specific site conditions and applying the guidance recommended in Ref. [2], will not jeopardize the safety of the installation. Thus, it gives guidance on a consistent application of methods and procedures currently available according to the state-of-the-practice for analysis, design, testing and qualification of structures and equipment so that they meet the safety requirements established in Ref. [1] in relation to the design, safety assessments for the design and the regulatory issues concerned with the licensing of nuclear installations.

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

1.9. This Specific Safety Guide is intended for use by regulatory bodies responsible for establishing regulatory requirements and guidelines, and for engineering organizations involved in seismic design and qualification process.

SCOPE

1.10. This Specific Safety Guide addresses an extended range of nuclear installations as defined in Ref. [4]: land based stationary nuclear power plants, research reactors, nuclear fuel fabrication facilities, enrichment facilities, reprocessing facilities and independent spent fuel storage facilities. The methodologies recommended for nuclear power plants are applicable to other nuclear installations by means of a graded approach, whereby these recommendations can be customized to suit the needs of nuclear installations of different types in accordance with the potential radiological consequences of their failure when subjected to seismic loads. The recommended direction of grading is to start with attributes relating to nuclear power plants and eventually to grade down to installations with which lesser radiological consequences are associated[1]. If no grading is justified, the recommendations relating to nuclear power plants are applicable to other types of nuclear installations.

1.11. This Specific Safety Guide is intended to be applied to the design and construction of new nuclear installations and it should not be applied to the seismic safety evaluation of existing ones. The assessment of the seismic safety of an existing nuclear installation is beyond the scope of this Specific Safety Guide; such an assessment should follow the approaches and procedures outlined in Ref. [3].

STRUCTURE

1.12. The structure of this Specific Safety Guide follows the general workflow of seismic design and qualification:

- Section 2 describes the specific safety requirements for treating external hazards and seismic actions according to the Ref [1] and provides recommendation of general nature on seismic design aspects.
- Section 3 presents the recommendations in relation to the first steps in the seismic design process, as the input required for the seismic design and qualification, including the design basis earthquake, the data obtained from the site characterization, and the seismic categorization of structures, systems and components of the installation.
- Section 4 presents specific recommendations on good practices on design related aspects for layout, structures and several component categories. For each category, this section identifies the key seismic design issues derived from earthquake experience and it gives what is currently considered as good practice in seismic design.
- Section 5 covers seismic analysis. First, the requirements for computing the site free-field response and obtaining foundation ground properties for seismic soil-structure interaction are discussed. Then, the general principles for selecting a seismically adequate plant layout are presented. The principles on how to compute the structural response of the main buildings and civil structures are introduced afterwards. Combination rules with loads other than earthquake are given and, finally, the guidelines for assessing seismic capacity (to be compared to the total demand) are given.
- Section 6 provides a unified view over seismic qualification, consistent with current state-of-the-practice. It covers qualification by test, by analysis and by earthquake experience.
- Section 7 presents the approach for assessing the seismic margin to be ensured by design. According to current practice, seismic design is performed for the design basis earthquake using criteria established in the design standards. Afterwards, the seismic margin over the design basis earthquake is assessed using best estimate methodologies, such as those described in Ref. [3].

[1] For sites at which nuclear installations of different types are collocated, particular consideration should be given to using a graded approach.

- Section 8 presents guidance on recommended seismic instrumentation and suitable monitoring procedures, and their relation to design assumptions and post-earthquake actions.
- Section 9 provides guidance on using the recommendations of this safety guide for seismic design and qualification for nuclear installations other than nuclear power plants.
- Section 10 provides the recommendations for implementation of management system, project management and peer reviews.

2. GENERAL RECOMMENDATIONS

2.1. As established in the Safety Requirements publication, Safety of Nuclear Power Plants: Design [1], the following main overarching and supporting safety requirements should be applied for design of nuclear installations to cope with the effects generated by earthquakes:

> "…
>
> ***Requirement 17: Internal and external hazards***
>
> ***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.***
>
> *5.15A. Items important to safety shall be designed and located, with due consideration of other implications for safety, to withstand the effects of hazards or to be protected, in accordance with their importance to safety, against hazards and against common cause failure mechanisms generated by hazards.*
>
> *5.15B. For multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously.*
>
> ***External hazards***
>
> *5.17. The design shall include due consideration of those natural and human induced external events[2] (i.e. events of origin external to the plant) that have been identified in the site evaluation process. Causation and likelihood shall be considered in postulating potential hazards. In the short term, the safety of the plant shall not be permitted to be dependent on the availability of off-site services such as electricity supply and firefighting services. The design shall take due account of site specific conditions to determine the maximum delay time by which off-site services need to be available.*
>
> *5.19. Features shall be provided to minimize any interactions between buildings containing items important to safety (including power cabling and control cabling) and any other plant structure as a result of external events considered in the design.*
>
> *5.21. The design of the plant shall provide for an adequate margin to protect items important to safety against levels of external hazards to be considered for design, derived from the hazard evaluation for the site, and to avoid cliff edge effects[3].*

---

[2] Requirements on site evaluation for nuclear installations are established in IAEA Safety Standard Series No. NS-R-3 (Rev 1), Site Evaluation for Nuclear Installations.

[3] 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.

5.21A. The design of the plant shall also provide for an adequate margin to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design, derived from the hazards evaluation for the site.

…….

Requirement 18: Engineering design rules

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

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

…….

Requirement 20: Design extension conditions

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

…….

Requirement 53: Heat transfer to an ultimate heat sink

The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states.

6.19B. The heat transfer function shall be fulfilled for levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site

…….

Requirement 65: Control room

A control room shall be provided at the nuclear power plant from which the plant can be safely operated in all operational states, either automatically or manually, and from which measures can be taken to maintain the plant in a safe state or to bring it back into a safe state after anticipated operational occurrences and accident conditions.

> 6.40A. The design of the control room shall provide an adequate margin against levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site
>
> ...”

2.2. The recommendations provided in following sections of this Specific Safety Guide provide detailed guidance on fulfilling the above indicated safety requirements in dealing with the protection of nuclear installations against the effects generated by earthquakes.

2.3. It should be kept in mind that the implementation of the relevant safety requirements in the design of a nuclear installation against seismic events should comply with the fundamental safety principle covering prevention of accidents. Seismic events can lead to serious challenges to the multiple layers of defence in depth, through common cause effects.

2.4. In accordance with these requirements, the seismic design of items important to safety should be performed based on the seismic hazards determined during the site evaluation stage for the specific conditions of the site applying the guidance provided in Ref. [2]. Specifically, the vibratory ground motions assessed using deterministic and /or probabilistic approaches should be available at the beginning of the process of seismic design to define the adequate earthquake design basis for the nuclear installation, as recommended in Section 3 of this Specific Safety Guide.

2.5. Seismic design should consider the influence of the layout of the plant and, consequently, of the detailed arrangements and layout of its SSCs. Specific guidance is provided in Section 4 of this Specific Safety Guide.

2.6. Specific aspects that should be considered in the seismic design of nuclear installations are related to : (i) considering protection against common cause failure of SSCs in case of earthquake occurrence affecting all units in a multi-unit site, (ii) the minimization of interaction effects, (iii) the need to provide adequate seismic margins and to avoid cliff edge effects, and (iv) to comply with proven engineering design rules as specified in relevant national and international codes and standards. All these aspects are duly considered in the recommendations and guidance provided in the corresponding sections of this Specific Safety Guide.

2.7. Special consideration should be given to para 5.21A of the Ref [1], as indicated above, regarding the need to provide in the nuclear installation design an adequate seismic margin for those SSCs ultimately required for preventing an early radioactive release or a large radioactive release in the event of an earthquake level exceeding the ones considered for design purposes, assuming that for seismic events there is not possibility to have early warnings and there is a high probability of combination with other hazards (such as fires and floods). To fulfil such requirement, in Section 3 of this Specific Safety Guide, discussions and guidance are provided to determine the beyond design basis earthquake and the categorization of the SSCs to be designed or evaluated against such event, while in other sections is discussed the applicable performance criteria in such cases.

2.8. For seismic design of research reactors, requirements from Ref [10] should be used. For fuel cycle facilities, requirements from Ref [11] should be used. Other types of nuclear installations than those or NPP should also use these requirements, as far as practicable. Application of requirements [10] and [11] should be done using the graded approach described in Section 9.

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

2.10. The design of a nuclear installation is usually a very well-structured process, conducted under the rules, procedures and conditions of a proper project management. As part of such plant design process, the seismic design process should fulfil the requirements and recommendations of a management system with adequate peer review steps. Section 10 of this Specific Safety Guide provides guidance in this regard.

3. INPUT FOR SEISMIC DESIGN

3.1. This section provides recommendations on: (i) the determination of the design basis earthquake to be used as input for calculating the seismic demand, for the design of the SSCs of the installation, (ii) the criteria for grouping the whole sets of SSCs in different categories to proceed with the seismic design applying different seismic input and performance criteria, and (iii) the definition of the applicable standards, codes and norms for seismic design purposes. The determination of the design basis earthquake includes also discussions and guidance on defining an earthquake level that exceeds the design basis to comply with the demonstration required in Ref. [1] as indicated in previous para 2.6 of the Specific Safety Guide.

GENERAL CONCEPTS OF SEISMIC DESIGN

3.2. As defined by the IAEA Safety Glossary "Terminology Used in Nuclear Safety and Radiation Protection" [4], design is the process and the result of developing a concept, detailed plans, supporting calculations and specifications for a facility and its parts. Also, qualification refers to the equipment qualification as the generation and maintenance of evidence to ensure that equipment will operate on demand, under specified service conditions to meet system performance requirements. In this sense, seismic qualification refers to a form of equipment qualification that relates to conditions that could be encountered in the event of earthquakes.

3.3. Taking those definitions as main reference, and for the purposes of this Specific Safety Guide, seismic design is the process of designing a nuclear installation to cope with the effects of the hazards generated by an earthquake event in accordance with the specified performance criteria and to comply with the prevention and mitigation requirements indicated in previous Section 2. Therefore, seismic qualification is part of the process of seismic design and refers to equipment qualification to comply with those objectives mentioned above.

3.4. Earthquakes generate several direct and indirect phenomena. From vibratory ground motions to associated geological and geotechnical hazards, as permanent ground displacement (e.g. soil liquefaction, slope instability, tectonic and non-tectonic subsidence, cavities leading to ground collapse, and settlements) to concomitant events such as seismically induced fires and floods. This Specific Safety Guide provides guidance on how to design a nuclear installation against the effects of vibratory ground motions.

3.5. Geological and geotechnical hazards that are of a nature or an intensity which cannot cope with available engineering solutions should have been excluded during the site selection and evaluation process as recommended in Ref. [2] and [5].

3.6. Thus, the seismic design process should consider the following steps:

a) Defining the earthquake levels to be used for the design, noted as design basis earthquake levels, consistent with the site specific seismic hazard, the seismic performance target of the nuclear installation, and the applicable design requirements established or adopted by the national regulatory authority.

b) Defining the seismic categorisation of the whole sets of SSCs of the nuclear installation consistent with their safety classification through a grouping of all

SSCs items of the installations in a number of categories for which different objectives should be reached in the design process.

c) Selecting the applicable standards and guidelines, consistent with the design requirements, providing the acceptable limits and conditions of the SSCs behaviour in case of an earthquake event to ensure that the intended safety functions during and after an earthquake, perform as required.

d) Evaluating the seismic demand on the SSCs due to the design basis earthquake level(s), according to relevant national or international codes, standards and proven engineering practices and as recommended or accepted by the national regulatory authority.

e) Verifying that the demand on each SSC does not exceed the seismic capacity and limits established by applicable national or international codes, standards and proven engineering practices recommended or accepted by the national regulatory authority and demonstrating that sufficient seismic margin is provided.

f) Assessing that the process above results in a design with adequate seismic margin to cope with earthquake events that exceed the design basis levels and that no cliff edge effects may be produced. This safety assessment is performed using procedures which are different from the ones used for design purposes, as utilized in the previous steps in that they emphasize the use of realistic and best estimate assessments.

DESIGN BASIS EARTHQUAKE

Required input from the site evaluation stage

3.7. The site evaluation stage conducted before the starting construction of the nuclear installation project provides detailed and specific data and information for the characterization of the site and concludes with the determination of the external hazards which may affect the nuclear installation[4]. Following such site evaluation stage in the design stage, and related to the need to cope with the effects derived from earthquake events, two important aspects treated at the site characterization or evaluation phase should be provided as input for the seismic design:

a) The determination of the specific seismic hazards at the site, particularly, the vibratory ground motion hazards, and

b) The detailed geological, geophysical and geotechnical characteristics of the site with the corresponding information on soil properties [5].

3.8. Regarding the aspect a) indicated above, the seismic hazard assessment should be available as resulting from the specific site characterization, through the application of the methods and

[4] Unless a generic design is intended for the site, in which case the site evaluation stage may occur after the reactor design. In this case the generic seismic design bases should be shown to envelope the site-specific seismic hazard challenges at the relevant hazard frequencies.

approaches recommended in Ref. [2], including the determination of the parameters (spectral representations and time histories, in horizontal and vertical directions) of the vibratory ground motions at the control point established by the user requirements, usually at the free field conditions, rock outcrop, or at the bedrock level.

3.9. If a deterministic approach was used for determining the site specific vibratory ground motion, a single value of such parameters (peak ground acceleration and spectral representation) should be selected.

3.10. If a probabilistic approach was performed for determining the site specific vibratory ground motion, hazard curves (mean and fractile curves) of the level of a relevant parameter, as the peak ground acceleration and peak spectral accelerations, and its annual frequencies of exceedance up to values compatible with the analysis needs (e.g. up to 10^{-6} to 10^{-7} per year) are the available results, including the derived uniform hazard response spectra for several annual frequencies of exceedance (e.g. 10^{-3}, 10^{-4}, 10^{-5} per year).

3.11. Regarding the aspect b) indicated in para 3.7. above, site specific static and dynamic properties of the soil parameters at the site area should be available from the geological, geophysical and geotechnical investigation campaigns, laboratory tests and engineering studies performed during the site characterization stage.

3.12. In addition to the geological, geophysical and geotechnical data and soil properties determined during the site characterization stage mentioned in para 3.7 above, , pre-construction, stage of the nuclear installation project a very detailed programme of geophysical and geotechnical investigations should be carried out to complete and refine the assessment of site characteristics to be consistent with the final layout of buildings and structures and their final location in the site area. When the final layout of the buildings, structures and support facilities is known, a differentiation should be made between structures important to safety from those structures non-important to safety in accordance with the seismic categorization. The detailed subsurface exploration and testing programme should be prepared accordingly based on such needs using either a grid boring scheme or an alternative boring scheme suited to the site and the installation under consideration. The grid spacing may vary depending on the geometry of the subsurface characteristics. The uniform grid method is especially adaptable to a site with relatively uniform soil conditions. Where dissimilarities and discontinuities are present, the usual exploration process should be supplemented with borings at spacings small enough to permit detection of the features and their proper evaluation.

3.13. As result of the geological, geophysical and geotechnical investigations conducted at the site area and at the location of the buildings and structures of the nuclear installation as described above, the following data should be basically available:

a) Static and dynamic soil properties: e.g., unit weight (γ) and/or density (δ), strength capacity in drain and/or undrained conditions, low-strain shear wave (v_s) and primary wave (v_p) velocities, variation of shear modulus (G) and damping ratio as a function of shear strain levels, with their variation in depth with indication of the types of soil and rock encountered until the bedrock level. Adequate number of soil profiles should be developed. The profile is usually defined as horizontally layers of ground, with best estimate (mean) values of layer thickness, shear wave velocity, unit weight and the shear modulus and damping ratio as function of shear strain level. The level(s) of the ground water should be also determined.

b) Variability of the thicknesses and ground layer properties to determine:

- either the Best Estimate (BE), Upper Bound (UB) and Lower Bound (LB) strain compatible soil profiles, accounting for the uncertainties in soil layer geometry and soil properties,
- or the full probability distributions of the soil parameters if the subsequent site response analysis is to be fully probabilistic.

Final site response analysis for the seismic hazard assessment

3.14. The seismic hazard assessment performed during the site evaluation stage should include a preliminary site response analysis as recommended in Ref. [2] according to the types of soil at the site area. Later, during the design stage, a final site response analysis should be performed based on the specific and detailed data and information obtained at the final location of the structures of the nuclear installation and, consequently, the final vibratory ground motions should be assessed at the control point required by the user and based on the seismic hazard assessment performed at the bedrock level.

3.15. For performing the seismic site response analyses, as defined in Ref. [5] the following site classification is used:

- Type 1 sites: Vs > 1100 m/s;
- Type 2 sites: 1100 m/s > Vs > 300 m/s;
- Type 3 sites: 300m/s > Vs;

where Vs is the best estimate shear wave velocity of the foundation medium just below the foundation level of the structure in the natural condition (i.e. before any site work), for very small strains. The site classification is valid on the assumption that the shear wave velocity does not decrease significantly with depth; other than in this case, particular analyses should be carried out according to the best practices[5].

3.16. Seismic site response analysis should be performed for soil types 2 and 3 while soil type 1 is usually considered as a hard rock site6. Soil type 1 is normally considered a rock site and a soil response analysis is not required if it can be demonstrated that negligible effect on modifying the control seismic motion. Type 3 sites (soft soil conditions) require detailed studies and site response analysis as described in Ref. [5].

3.17. As indicated in Ref. [2], basically, two approaches should properly consider the geological and geotechnical specific soil conditions at a site as part of the estimation of the seismic vibratory ground motion. The first approach is to utilize ground motion prediction equations appropriate for the specific site soil conditions, i.e. using Ground Motion Prediction Equations (GMPEs) that have

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[5] Some member states recommend not using Type 3 soft soil sites.

[6] Definition of 'rock' varies between Member States.. In some Member States a site is considered to be a hard rock site when the average shear wave velocity is larger than 2800m/s.

been developed for subsurface conditions of the type that prevails at the site. The second approach is to conduct a site response analysis compatible with the detailed and specific geotechnical and dynamic characteristics of the soil and rock layers at the site area. The decision on which approach to be used should therefore be made based on the ground motion prediction equations utilized for calculating the seismic vibratory ground motion parameters at the site.

3.18. If the first approach is utilized, the resulting vibratory ground motion parameters at the free surface of the top of the soil profile should be already the ones to be used for defining the seismic hazard design basis for the nuclear installation.

3.19. If the second approach is utilized, a step-by-step procedure should be applied as follows to determine the final seismic vibratory ground motion at the site including all parameters (spectral representations and time histories, in horizontal and vertical directions) at the specified control point location, usually the free field ground level, competent rock, or foundation level:

1) Determine the best estimate soil profile parameters based on the geophysical and geotechnical databases, for the full depth from the bedrock outcrop layer to the free surface at the site, including their uncertainties characterized either as BE, UB and LB values, or as probability distributions. That means to determine the mean values and their uncertainties for each site soil layer of the following parameters:
    a) low strain shear wave velocity (V_S),
    b) strain dependent shear modulus reduction and hysteretic damping properties,
    c) soil density, and
    d) layer thickness.
    e) For vertical component, compressional wave velocity (V_P).
2) Evaluate the correlation of soil layer properties;
3) Determine whether 1D equivalent linear analyses should be performed, or more complex approaches are needed;
4) Starting with the seismic hazard curves and associated response spectra obtained at the bedrock outcrop layer, calculate site amplification factors through convolution of the bedrock hazard curves for each spectral frequency of interest, so that they should mimic the characteristics of the principle contributors to the de-aggregated seismic hazard, including diffuse seismicity;
5) Develop the uniform hazard response spectra (UHRS) at the identified locations of interest for the nuclear installation site and for the annual frequencies of exceedance selected for defining the seismic design basis (e.g. 10^{-4} and 10^{-5} per year). Note that the final design basis ground motion should be developed with enough seismic margin beyond this level.
6) If possible verify the site response analysis results with observed instrumental records and/or microtremor surveys.
7) If subsurface structure and buried foundation are complex, soil-structure interaction analysis should be conducted.

Determination of the Design Basis Earthquake (DBE)

3.20. At the design stage of the nuclear installation, and as one of the first steps in this process, the Design Basis Earthquake (DBE) should be determined. It is used to define the level of the seismic vibratory ground motion hazards for the design of the SSCs of the nuclear installation based on the results of the site specific seismic vibratory ground motion assessment available as indicated in the section above. For such purpose, in general, two levels of seismic ground motion hazard, named SL-1 and SL-2, should be defined as the Earthquake Design Basis for each nuclear installation in order to fulfil the different needs of ensuring the safety of the installation in case of a rare earthquake (e.g. SL-2) and of ensuring the possibility of continued operation for a less severe but more probable earthquake event (SL-1). In some cases, depending on specific site conditions and country regulations (e.g., low seismically active areas), one level of seismic ground motion hazard, may be defined for design considerations.

3.21. The SL-2[7] design earthquake level should be associated with the safety requirements and is defined as the vibratory ground motion for which certain structures, systems and components of the nuclear installation should remain functional during and after the occurrence of a seismic event of such intensity and complying with applicable safety requirements.

3.22. The SL-1[8] earthquake level should be associated, mainly, to operational and licensing requirements and corresponds to a less severe, more probable earthquake with respect to SL-2 level which could reasonably be expected to occur and to affect the nuclear installation during its operating lifetime and for which those structures, systems and components necessary for continued operation should be designed to remain functional and complying with the safety objective.

3.23. The SL-2 design earthquake level is defined based on the results and parameters obtained from the seismic hazard assessment, as indicated in para 3.7 above, and according to specific criteria established by the regulatory authorities to achieve a certain target level for its annual frequency of exceedance. The SL-2 level should be characterized by both horizontal and vertical vibratory ground motion response spectra, anchored to a peak ground acceleration (i.e., at zero period of the response spectrum) and at the control point defined by the seismic hazard assessment and which usually is the free field ground surface.

3.24. If a probabilistic approach was used for the seismic hazard assessment, and according to current regulatory practice in Member States, the SL-2 level corresponds typically to a level with an annual frequency of being exceeded in the range of 1×10^{-3} to 1×10^{-5} (mean values) per reactor per year. Thus, using the seismic vibratory ground motion hazard curves and uniform hazard response spectra obtained for such level of established annual frequency of occurrence (see para

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[7] SL-2 earthquake level corresponds to an earthquake level often denoted as Safe Shutdown Earthquake (SSE) in some Member States.

[8] SL-1 earthquake level corresponds to an earthquake level often denoted as Operating Basic Earthquake (OBE) in some Member States.

3.10), the SL-2 should be calculated with due consideration of additional margins and rounding aspects[9].

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

3.26. The design basis earthquake level should include adequate design conservatism by considering the uncertainties associated with peak ground acceleration and spectral shape, based on results of the seismic hazard assessment.

3.27. The SL-1 earthquake design level corresponds typically to a level with an annual frequency of being exceeded in the range of 1 x 10^{-2}/yr to 1 x 10^{-3}/yr (mean values) per reactor per year. However, the SL-1 level is usually defined as a percentage of the SL-2 level with appropriate considerations regarding its application in the design and operation stages.

3.28. Regardless of the exposure to seismic hazard at the specific site, a new nuclear installation should be designed at least for a minimum earthquake level. In that regard, considering (i) the advances on the developments of new design of nuclear installations, (ii) the uncertainties in the seismic hazard assessment, (iii) the effectiveness in terms of cost and technical provisions of providing a high level of assurance against the seismic hazards from the conception phase of the installation, the minimum level for seismic design should correspond to a peak ground acceleration of 0.10g, and not less than values established by the national seismic codes for conventional facilities, to be considered at the free field ground surface, or foundation level. A unified, site compatible spectrum should be associated with this peak ground acceleration value and in this case, SL-1 may be assumed coincident with the SL-2 level. For plant structures, systems and components sensitive to low frequency motions (eg. SSCs on isolators), time histories/ response spectra should be examined and, if necessary, modified to take these effects into account.

BEYOND DESIGN BASIS EARTHQUAKE

3.29. In addition to the two earthquake levels defined and determined for design purposes, as indicated in the previous sub-section, an earthquake level exceeding the ones considered for design purposes should be defined as required in Refs. [1, 2, and 3]. For this earthquake level, noted as the Beyond Design Basis Earthquake (BDBE), the design should:

a) Provide adequate seismic margin for those SSCs ultimately required for preventing core damage and mitigating an early radioactive release or a large radioactive release:

b) Be consistent with mitigation measures for SSCs supporting Level 4 of the defence in depth concept and;

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[9] In some Member States, using a performance -based approach for defining specific site hazards and design, the earthquake design level is calculating scaling the site specific uniform hazard response spectrum by a design factor, (usually > than 1).

c) Demonstrate that cliff edge effects are avoided within the uncertainty of the determined DBE values.

3.30. Therefore, during the seismic design of a new nuclear installation, two different sets of earthquake levels should be determined: (i) one set, noted as DBE and constituted by the SL-2 and SL-1 levels, as defined in paras 3.20 to 3.28 above, for which adequate seismic margin should be provided by the design to avoid cliff edge effects, and (ii) a second set, noted as BDBE which aims to verify that adequate margins exist to comply with the safety requirements indicated in paragraph above.

3.31. A new nuclear installation should, first, be designed against a DBE level in accordance with specific design performance criteria and, second, should be verified that in case of occurrence of a BDBE earthquake level, specific evaluation performance criterion would also be fulfilled.

3.32. The determination of the BDBE and the associated loading conditions can be done by:

a) Defining the BDBE earthquake level by a factor times the SL-2 earthquake level[10].

b) Defining the BDBE earthquake level based on considerations derived from the probabilistic seismic hazard assessment[11].

3.33. The BDBE level should be characterized by both horizontal and vertical vibratory ground motion response spectra, anchored to a peak ground acceleration (i.e., at zero period of the response spectrum) and at the control point defined by the seismic hazard assessment and which usually is the free field ground surface.

SEISMIC CATEGORIZATION FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.34. Seismic categorization is the process by which an item of the nuclear installation is assigned to a seismic category in accordance with its required performance during and after the occurrence of an earthquake event, in addition to other classifications such as safety, quality and maintenance classifications. The relevant acceptance criterion associated with the item is part of the categorization.

3.35. Based on the current state-of-the-practice and regulatory requirements in Member States, this Specific Safety Guide recommends the grouping of the whole set of SSCs of the nuclear installation in three seismic categories:

a) Seismic Category 1

b) Seismic Category 2

c) Seismic Category 3

3.36. The Seismic Category 1 is the group constituted by the items required to remain functional during and/or after the occurrence of the SL-2 design basis earthquake. An item in this category

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

[11] An annual frequency of exceedance lower than the one used for defining the SL-2, e.g. in some Member States the mean values obtained for annual frequency of exceedance in the range of 1 to 5 x 10-5 is used.

should maintain its functionality and/or structural integrity (depending on functional requirements) and adequate seismic margin should be provided to ensure that no cliff edge effects may be produced.

3.37. Seismic Category 1 should include the following SSCs:

a) SSCs whose failure could directly or indirectly cause accident conditions;

b) SSCs required for shutting down the reactor and maintaining the reactor in a safe shutdown condition, including the removal of decay heat;

c) SSCs that are required to prevent or mitigate non-permissible radioactive releases (with limits established by the national regulatory body), including the spent fuel storage pool structure and fuel racks;

d) SSCs required to mitigate the consequences of design extension conditions, and whose failure would result in consequences of 'high' severity as defined in Ref. [6].

e) SSCs of support, monitoring and actuating systems that are needed for fulfilling the functions indicated in b), c) and d) above.

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

3.39. . For any item in Seismic Category 1, appropriate acceptance criteria[12] should be established through the acceptable values of design parameters indicating, for example, functionality, leak tightness, maximum distortion and/or deformation, maximum stress level, etc.

3.40. The Seismic Category 2 is the group constituted by those SSCs whose failure to perform the intended functions will impede or affect any of the safety functions required to be performed by Seismic Category 1 items.

3.41. Seismic Category 2 should include the following SSCs:

a) Items that may have spatial interactions (e.g. due to collapse, falling or dislodgement) or any other interactions (e.g. via the release of hazardous substances, fire or flooding, or earthquake induced interactions) with items in Seismic Category 1 including effects on any safety related operator action.

b) Items not included in Seismic Category 1 (particularly items under (b) and (c) in para. 3.37) that are required to prevent or mitigate plant accident conditions (originated by postulated initiating events other than earthquakes) for a period long

[12] Acceptance criteria are specified bounds on the value of a functional or condition indicator used to assess the ability of a structure, system or component to perform its design function. Acceptance criteria as used here means specified bounds on the value of a functional or condition indicator for a structure, system or component in a defined postulated initiating event (e.g. an indicator relating to functionality, leak-tightness or non-interaction).

enough that there is a reasonable likelihood that an SL-2 earthquake may occur during that period.

c) Items related to infrastructure needed for the implementation of the emergency evacuation plan.

3.42. The items of nuclear installations included in Seismic Category 2 should be designed to withstand the effects of a SL-2 earthquake level.

3.43. Seismic Category 3 should be the group constituted all items that are not in Seismic Categories 1 and 2.

3.44. The items of nuclear installations included in Seismic Category 3 should be designed as a minimum in accordance with national practice for seismic design of non-nuclear applications and, therefore, for facilities at conventional risk. However, for some items in Seismic Category 3 which are important to the operation of the installation, it may be reasonable to select a more severe seismic loading and more stringent acceptance criteria than the ones for conventional facilities in national practice, based only on operational needs. Such approach will minimize the need for plant shutdown, inspection and restart, thus allowing the installation to continue to operate after an earthquake occurrence.

3.45. Example of correspondence of seismic categories with the safety classes defined in Ref. [6] is given in Table 1. The inclusion of an item in a seismic category should be based on a clear understanding of the functional requirements that should be ensured for safety considerations during or after an earthquake. According to their different functions and their functional safety categories, parts of the same system may belong to different seismic categories. Tightness, degree of damage (e.g., fatigue, wear and tear), mechanical or electrical functional capability, maximum displacement, degree of permanent distortion and preservation of geometrical dimensions are examples of aspects that should be considered and determined as input for the seismic designers to allow them to establish the limiting acceptable conditions.

TABLE I. CORRESPONDENCE OF SEISMIC CATEGORIES WITH SAFETY CLASSES

Safety Classes [6]	**Seismic Category**	**Remarks**
1	1	Seismic categories 1 and 2 applies for either structural integrity, or leak tightness or functionality, or their combinations, as applicable. Both SL-1 and/or SL-2 should be used as prescribed by applicable regulations and nuclear codes.
2	1 or 2	
3	1 or 2	
Non-classified	3	SSCs that are not safety classified, and their seismic failures should not have any interactions with safety classified SSCs. National practice for seismic design of non-nuclear installations apply.

3.46. As part of the design process, and as one of its first tasks, a detailed list of all installation items should be produced with indication of their safety class and seismic categories and the applicable associated acceptance criteria.

SELECTION OF SEISMIC DESIGN AND QUALIFICATION STANDARDS

3.47. Once the seismic categories of the SSCs have been established, corresponding engineering design rules should be specified. Engineering design rules are constituted by the relevant national or international codes, standards and proven engineering practices that should be applied to the seismic design of the SSCs to meet the applicable requirements (Ref. [10-11]).

3.48. Experience from the design and construction of nuclear installations in Member States indicates that often codes, norms and standards of different origin, either by country and/or practice, are utilized. Even in a same country codes or standards for the different disciplines (mechanical, civil and electrical) are not always based on compatible performance or behaviour criteria to achieve a consistent level of safety. Therefore, in principle, this situation should be avoided, and a consistent set of codes, norms and standards should be selected for using in the seismic design. If this is unavoidable, this consistency should be attained for SSCs of same material (e.g. for reinforced concrete and steel structures) and/or same type of item (e.g., piping, mechanical and electrical components). In any case, it is recommended to perform at the beginning of the project an analysis and evaluation of the codes, norms and standards to be applied for the design, fabrication and construction of the different types and materials of the SSCs to ensure consistency and compatibility with the applicable safety requirements for the nuclear installation project.

3.49. The results of such analysis of the applicable codes, norms and standards should be well documented in the project guidelines as part of the management system.

4. SEISMIC DESIGN OF STRUCTURES SYSTEMS AND COMPONENTS

4.1. This section provides specific recommendations on good practices, that should be observed during seismic design of SSCs including installation layout, as they are recognized by the international earthquake engineering community.

4.2. These recommendations are derived from the past experience and observed performance of similar items, mainly in industrial conventional facilities, when affected by past earthquakes. These recommendations should be duly considered at the initial stages of the plant basic engineering design when adequate decisions may avoid significant problems in future stages saving time and resources.

PLANT LAYOUT

4.3. The layout of the installation should be established in the early stage of the installation design, aimed to achieve the most suitable solution for the seismic design. All procedures for seismic design should be based on a good understanding of the consequences of past destructive earthquakes, and this knowledge should be adopted and realistically applied.

4.4. In the preliminary design stages, seismic effects (in terms of forces and undesired torsional or rocking effects) should be minimized by the appropriate selection of a structural arrangement applying some general criteria, such as follows:

a) Locating the mass centre of all structures as low as practicable;

b) Locating the centre of rigidity at the various elevations as close as practicable to the mass centre to minimize torsional effects;

c) Selecting for each building plan and elevation layouts that are as simple and regular as practicable, with direct and clear paths for the transmission of seismic forces to the foundation;

d) Avoiding different embedment depths of adjacent buildings as far as practicable;

e) Avoiding buildings with large aspect ratios in plan. Plan aspect ratios should be as close to 1 as practicable and large aspect ratios should be avoided;

f) Avoiding protruding sections (i.e. lack of symmetry) as far as practicable;

g) Avoiding rigid connections between different building structures or between equipment of different categories and dynamic behaviour[13].

4.5. Adequate gap dimensions and seismic margin should be ensured in designing the structural joints between adjacent structural parts or between adjacent buildings to avoid pounding and hammering.

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[13] An example is the containment vessel and the surrounding internal concrete structures: if they are connected, they could interact during the earthquake. Since the interaction of such structures is complex and difficult to assess. the structures should preferably be decoupled above the foundation level.

BUILDINGS AND CIVIL STRUCTURES

4.6. Structural systems for buildings of nuclear installations should possess adequate strength and ductility and they should provide confinement as it is required by the intended safety functions. The following structural systems should be considered acceptable for structures of any seismic category:

a) Structures made of reinforced concrete shear walls providing the lateral force resisting system;
b) Steel or reinforced concrete moment-resisting frames, specially detailed to provide ductile behaviour;
c) Reinforced concrete slab/wall moment frames.

4.7. The following structural systems should be avoided in structures corresponding to Seismic Categories 1 and 2

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

4.8. It is recommended that detailing of structures should favour ductile failure modes in opposition to brittle failure modes. In this regard, the following should be considered:

a) In reinforced concrete structures, brittle failure in shear and/or bond of rebars or in the compressive zones of concrete should be prevented.
b) For reinforcement, an appropriate minimum ratio of the ultimate tensile stress to the yield tensile strength should be defined, to ensure a minimum ductility.
c) The lengths for reinforcing bar anchorage should generally be larger than the lengths for structures under static or non-reversing loads.
d) In steel structures, local instability should be avoided before the development of ductile failure modes based on material plasticity.
e) Structural joints, particularly in reinforced concrete structures, should be designed to accommodate ductile displacements and rotations; this provision should be consistent with the acceptance criteria specified in the seismic categorization, but is intended also to consider an adequate seismic behaviour concurrent to design extension conditions.
f) Wide enough seismic gaps between structures above ground level should be provided to avoid interaction (pounding) during seismic motion. Utilities crossing the gaps should be able to accommodate differential seismic displacements. Otherwise, structural integrity should be confirmed where interaction between structures could occur.

4.9. Structures in Seismic Category 1 can be designed to exhibit nonlinear behaviour, provided that their acceptance criteria (as expressed in terms of the value of a design parameter such as maximum crack opening, absence of buckling or maximum inter story drift) are met with a seismic margin consistent with the seismic categorization.

4.10. Structures in Seismic Category 2 can also be designed to exhibit nonlinear behaviour. Detailing of structural members, particularly joints and connections, should be consistent with the ductility level required to comply with the acceptance criteria.

4.11. Non-structural elements of the buildings, such as partition walls, ceilings, roofing, etc. should be designed so that they do not collapse and fall onto Seismic Category 1 components.

4.12. The possibility of lateral sliding during the earthquake of structures set on waterproofing material (especially if wet) should be assessed.

4.13. Masonry walls whose collapse could affect Seismic Category 1 components should be designed as Seismic Category 2 components, to avoid potential harmful interactions.

4.14. Massive mat foundations associated with nuclear buildings are generally seismically rugged and are preferred to seperate foundations for individual buildings.

EARTH STRUCTURES

4.15. The seismic design of engineered earth structures and buried structures should be consistent with the seismic design category and guidance provided in Ref. [6].

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

a) Ultimate heat sinks: dams, dykes and embankments;
b) Site protection: dams, dykes, breakwaters, sea walls, revetments;
c) Site contour: retaining walls, natural slopes, cuts and fills.

4.17. The seismic design of earth structures should account for the following seismic related effects:

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

SEISMICALLY ISOLATED STRUCTURES

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

4.19. Isolators should be seismically qualified using full scale testing of prototypes as well as during the fabrication stage. The prototypes should be tested and subjected, at least, to the maximum displacements considered in the design or for beyond design basis earthquake. The test should provide the properties to be used in the structural analysis:

a) Initial stiffness, as a function of frequency;
b) Post-yield stiffness, as a function of frequency;
c) Damping, as a function of frequency and/or maximum displacement and number of cycles expected during beyond design conditions of the isolation device.

4.20. Regarding the superstructure, the main difference between an isolated and a fixed-base structure is that the former needs a structural diaphragm above the plane of isolation (upper basemat). This diaphragm should be stiff enough to redistribute lateral loads from the superstructure into the isolation system.

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

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

(a) Uniformity of load and displacement is important. Ideally, all isolators should be of the same model, should be under the same gravity load and they should sustain the same horizontal displacement during an earthquake;

(b) Avoiding, or at least minimizing, uplift;

(c) Avoiding exceedance of ultimate deformations in the isolators during earthquakes more severe than the design basis earthquake;

(d) Allowance for in-service inspection and replacement of each individual isolator during the operational stage;

(e) Qualification conditions of isolators should be consistent with the anticipated operating temperature;

(f) The environment conditions should not present hazards e.g. fire at the level where isolators are located;

(g) Avoidance of detrimental effects to co-located SSCs protecting against other external hazards.

4.23. The substructure, the isolator pedestals (plinths) and the common footing (lower basemat), should be designed to resist not only gravity and seismic loads, but also the moments induced by the lateral displacements of the isolator system, including P-Δ effects.

4.24. A clearance space (seismic gap) should be provided around the perimeter of the upper basemat to allow for large lateral movements of the isolated structure. Generally, the isolation system is set below grade and the seismic gap takes the form of a moat. The width of such a moat should correspond to the ultimate allowed lateral displacement of the isolation system and correlated with the maximum expected displacement induced by the beyond design basis earthquake.

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

MECHANICAL EQUIPMENT ITEMS

4.26. Seismic experience about the effects of earthquakes on industrial facilities shows that most of the reported failures of mechanical equipment correspond to lack of anchorage or insufficient capacity at the anchorage. The positive anchorage of mechanical equipment to the main structure of the building should be considered the key aspect for a good seismic performance.

4.27. Design of the anchorage should take into account the following points:

a) The full load path from the base of the equipment item to the main structure should be considered;

b) The load path should have enough capacity and, enough stiffness so that the natural frequencies of the as-installed component are not significantly reduced;

c) The seismic demand at each support point should be computed from the in-structure response spectra[14], using the quasi-static method or response spectrum method with the level of damping accepted by the design standard for each particular equipment class. Simplified conservative approaches are acceptable, if justified;

d) Nozzle loads should be taken into account when computing the seismic demand;

e) Prying action at base plates should be avoided by an adequate position of fastenings (e.g. avoiding large eccentricities in the load path);

f) The portions of the load path prone to brittle failure should be oversized, in order to have ductile controlling failure modes (e.g. in cast in place bolts, the failure should take place at the bolt, not at the concrete);

g) Mixing different types of fastenings for the anchorage of the same component (e.g. welding and expansion anchors) is not acceptable unless it could be shown that the stiffness of the different fastenings is similar;

h) Flexibility of base plates can alter significantly the distribution of anchor forces with respect to the results computed with the common rigid-plate assumption. This is especially relevant when brittle failure modes are involved (e.g. pull out of expansion anchors). In those cases, the design should give consideration to the base plate flexibility.

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[14] The term in-structure response spectrum is used to mean a response spectrum computed at a point within the structure representative of the loading input point for an item of equipment. The term floor response spectrum is also often used for this purpose, but the term in-structure is preferred because not all such loading points are coincident with a floor level

i) From a seismic point of view, the preferred anchorage types are the following:

   - Cast-in-place bolts or headed studs;
   - Welding to embedded plates;
   - Undercut-type expansion anchors

j) Expansion anchors other than undercut-type should normally not be used for rotating or vibrating equipment or for sustained tension supports.

4.28. In some instances, vibration isolation devices not designed for earthquake loads have failed during earthquakes affecting industrial facilities. When a vibration isolation device is used to support a Seismic Category 1 component, the seismic capacity of the selected device should be demonstrated. In those cases, it is good practice to install limiters (bumpers) in order not to exceed maximum allowable lateral displacements.

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

STORAGE TANKS

4.30. Seismic experience shows that aboveground vertical atmospheric storage tanks are vulnerable during earthquakes, especially when they are unanchored or lightly anchored. Design of this kind of tanks should give consideration to the following points:

a) Seismic demand should be computed considering the flexibility of the tank shell and its influence in the natural frequencies of the tank;

b) A conservative freeboard should be taken to avoid damage to the roof due to sloshing of the fluid;

c) Unanchored tanks are usually not acceptable for Seismic Category 1 items. Unanchored tanks may have large uplifts and instability failures at the base of the cylindrical shell (e.g. elephant foot buckling). Large uplifts usually produce the rupture of the attached lines and the loss of the contents of the tank;

d) Capacity of the tank's foundation should be appropriately verified, especially for ring type foundations. The assessment should be consistent with the capacity assessment of the tank shell and the anchorage;

e) Global stability of the tank for overturning and sliding should be assessed;

f) Design of attached lines should allow for differential displacements between the tank and the first support consistent with the design of the anchorage (i.e. avoid placing supports very close to the tank).

PIPING

4.31. In accordance with accepted engineering practice and regulatory requirements, seismic design of piping in nuclear installations is usually done by analysis, following a national or/and international recognized piping design code. In addition, the experience from the seismic behaviour of piping to past earthquakes provides evidence of good performance of piping if it complies with the conditions below. Hence, even when the design is done by-analysis, these conditions should be followed to the extent possible:

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

4.32. Piping anchored to two different buildings, or substructures within a building, or entering a building from underground, should be sufficiently flexible to accommodate the differential motion of the attachment points at both sides.

UNDERGROUND PIPING

4.33. Underground piping is a special type of piping that is continuously supported by the ground. The design should follow the general guidelines provided in Ref. [5]. The seismic design principle of this kind of piping is to make it sufficiently flexible to follow the ground deformation during the seismic shaking.

4.34. The design should pay attention to the details at penetrations into buildings or other structures and make sure that the design has enough flexibility to allow for the expected differential displacements between the ground and the structures to which the piping is connected (Ref. [5]).

ELECTRICAL EQUIPMENT, CONTROL AND INSTRUMENTATION

4.35. Electrical equipment (cabinets, motors, transformers and similar equipment) should be seismically qualified by testing if functionality during and/or after an earthquake is required (Section 6).

4.36. Qualification tests made on equipment items do not always include the full load path of the anchorage to the main structure. Hence, the portions of the load path is not covered by the test it should be designed and assessed separately. The design should consider the following recommendations:

a) The full load path from the base of the equipment item to the main structure should be considered;

b) The load path should have enough capacity and adequate stiffness;

c) Prying action at base plates should be avoided by an adequate position of fastenings (e.g. avoiding large eccentricities in the load path);

d) The portions of the load path prone to brittle failure should be oversized, to ensure ductile failure modes (e.g. in cast-in-place bolts, the failure should take place at the bolt, not at the concrete);

e) From a seismic point of view, the preferred anchorage types are the following:

   - Cast-in-place bolts or headed studs;
   - Welding to embedded plates;
   - Undercut-type expansion anchors.

f) For motor control centres, transformers, inverters, switchgear, and control panels, use of top bracing or lateral ties to limit the differential displacements imposed on cables, conduit, bus ducts, etc. should be considered.

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

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

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

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

4.41. The seismic design aspects related to batteries and racks should ensure that the following concerns are properly addressed:

   - the lateral and transverse stiffness of the racks;
   - overturning stability;
   - anchorage to the rack supporting structure;
   - adequacy of spacers between the batteries; shims at the ends of the battery rows.

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

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

CABLE TRAYS AND CONDUITS

4.43. In accordance with accepted engineering practice, seismic design of electrical raceway distribution systems in nuclear installations is done by-analysis, following a national or an international design code. In addition, the experience from the seismic behaviour of these items to past earthquakes provides evidence of good performance of raceway systems if the design complies with the following basic rules:

a) Limited cable tray span[16];

b) Limited conduit span;

c) Tie-down of raceway members. For cantilever bracket-supported raceways, cable trays and conduits should be fastened to their supports, so that they cannot slide and fall off the supports;

d) No brittle failure at the supports. Supports should be able to withstand the earthquake without brittle failure.

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

HVAC DUCTS

4.45. In accordance with accepted engineering practice, seismic design of HVAC ducts in nuclear installations is usually done by analysis, following a national or an international design code. In addition, the experience from the seismic behaviour of HVAC to past earthquakes provides evidence of good performance of HVAC ducts if the design complies with the following basic rules:

a) Limited duct span[17];

b) Duct tie-downs. Ducts should be fastened to their supports to preclude the possibility of displacing, falling or sliding off during a seismic event. The duct should be securely attached to the last hanger support at the terminal end of the duct run. Similarly, supports

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

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

designed to limit the lateral movement of the duct system should also be attached to the duct.

c) Positive connection at joints. Ducts with slip joints without pocket locks, rivets or screws, could experience joint separation due to the differential displacement between supports.

d) Positive attachment of appurtenances. Appurtenances attached to HVAC ducts, such as dampers, turning vanes, registers, access doors, filters, and air diffusers, should be positively attached to the duct by means of screws or rivets.

e) No brittle failure of supports. Supports should be able to withstand the earthquake without brittle failure.

5. SEISMIC ANALYSIS

5.1. Once the layout of buildings and civil structures has been defined and proportioning of structural members has been carried out, seismic analysis of these structures should be performed. The purpose of seismic analysis is two-fold. On the one hand, it provides the parameters of the structural response required to verify capacity against the design basis earthquake, or to assess the seismic margin corresponding to a Beyond Design Basis Earthquake (e.g. stresses, internal forces and moments, displacements). On the other hand, seismic analysis of building and civil structures provides the seismic demand (e.g. in-structure response spectra and in-structure acceleration or displacement time histories) for seismic qualification of structures, systems and components housed by these structures.

SITE RESPONSE ANALYSIS

5.2. For soil and soft rock sites, as opposed to hard rock sites, ground (free field) response analysis should be performed with the purpose of obtaining the strain compatible soil profiles to be used in seismic soil-structure interaction analyses, and their associated uncertainties. Site response analysis is described in Section 3 of this safety guide.

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

STRUCTURAL RESPONSE

5.4. Structural response should be calculated using linear equivalent static analysis, linear dynamic analysis, complex frequency response analyses or non-linear analysis. The method depends on the relevance and complexity of the particular structure and on the national/international practice. Regardless of the method:

a) The seismic input should be defined by either design response spectra or by response spectra compatible acceleration time histories;

b) The analysis model should adequately represent the behaviour of the structure under the seismic action, considering the inertial stiffness and damping distribution of the structure;

c) Soil-structure interaction should be considered for soil and soft rock sites[18], taking into account uncertainties in ground properties;

d) Structural response should be obtained for the three orthogonal components of seismic motion (one vertical and two horizontal);

e) Potential second-order effects, if relevant, should be considered for all vertical load path

[18] Not considering soil-structure interaction in hard rock sites will eliminate radiation damping and, consequently, it could lead to conservative estimates of motion when computing in-structure response spectra.

elements (P-Δ effects[19]). Particularly, all vertical load path elements should be designed for the lateral displacements induced by seismic loads.

f) Hydrodynamic effects should be considered for SSCs containing large volumes of water, for instance fuel-pools and service pools.

5.5. It is common practice to apply two the horizontal and vertical components of the seismic input simultaneously. In this case, the components should be statistically independent. When the input components are applied individually, the corresponding structural responses should be suitably combined to account for the statistical independence of the components of the input.

5.6. Modelling of stiffness for seismic analysis should follow the national/international practice for nuclear applications. For example, in the first step the gross area of reinforced concrete sections is used to compute stiffness using linear elastic analysis. Based on the stress level identified in step 1, stiffness reduction factors are evaluated for each structural element. The corrected stiffness is then used in a second iteration, if necessary.

5.7. In many cases, when soil-structure interaction is considered, the variation of soil properties accounting for uncertainties envelops the variation in structural stiffness due to cracking. Since the two phenomena are independent, the analyst should avoid introducing artificially large uncertainties in the analysis by considering simultaneous occurrence of extremes when bounding the design space.

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

5.9. The model used for computing the seismic response should include the mass of the structure, the mass of permanent equipment and the mass of the expected live load concurrent with seismic loads.

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

5.11. For complex structures, the analyst should consider separation of the seismic model computational model into main structures and substructures. In this case, major structures that are considered in conjunction with foundation media to form a soil–structure interaction model constitute the main systems. The systems and components attached to the main systems constitute the subsystems.

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

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

5.13. Coupled analysis of a primary structure and a secondary structure, system or component should be performed when the effects of dynamic response interaction are significant.

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

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

5.16. In order to be used as design seismic input for the structures, systems and components housed by the main structure, the calculated in-structure response spectra should be peak-broadened to account for possible uncertainties in the evaluation of the vibration characteristics of the building's components[20].

DYNAMIC SOIL-STRUCTURE INTERACTION[21]

5.17. When consideration of soil-structure interaction (SSI) effects is required (see clause 5.3), the analyst should first identify acceptable models and analysis procedures based on the assessment of the following aspects:

a) Purposes of SSI analysis and use of the results (e.g. seismic response of the structure for design or assessment, input for seismic qualification of systems and components housed by the structure, basemat response for basemat design, structure-soil-structure analysis);
b) Relevant phenomena that need to be simulated (e.g. seismic wave fields, linear/equivalent linear/non-linear soil behaviour, linear/non-linear simulation of soil-foundation contact, wave incoherence);
c) Methodology/Software to be used, based on the two previous items;
d) For structures containing pools of water large enough to impact the SSI, the SSI model should incorporate the fluid-structure interaction (FSI) effect.

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

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

[21] Heavy, stiff structures founded on soft ground may experience significant differences in their seismic response with respect to the same structures founded on stiff rock. Differences may be important even for a ground with an intermediate stiffness. This effect is the result of phenomena that are jointly designated as 'soil-structure interaction' (SSI).

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

5.20. The current state of technology development provides the analyst with two main categories of acceptable methods for analysis of soil-structure interaction: direct methods and sub-structuring methods. Direct methods analyse the soil-structure system in a single step. Direct methods are applicable to (equivalent) linear idealizations and they are the only alternative for nonlinear idealizations of the soil-structure system. Sub-structuring methods divide the soil-structure interaction problem into a series of simpler problems, solve each problem independently, and superpose the results. Sub-structuring methods should be limited to (equivalent) linear idealizations, since they rely on superposition.

5.21. Uncertainties in the SSI analyses should be considered, either by the use of probabilistic techniques or by bounding deterministic analyses which cover the expected range of variation of analysis parameters affecting response, including, including soil properties. In any case, the variation of soil properties considered in SSI analyses should be consistent with the properties used for developing the design input motion (Section 3).

Direct methods

5.22. SSI analyses by direct methods should include the following activities:

a) Development of the soil-foundation-structure model, normally using a finite element modelling method;

b) Locate the bottom and lateral boundaries of the model and assign appropriate boundary conditions;

c) Define the input motion to be applied at the boundaries, compatible with the site response analysis (Section 3);

d) Perform the analyses and obtain the required response parameters.

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

5.24. Lateral boundaries should also be located at sufficient distance so that the structural response is not significantly affected by these boundaries. Minimum distances to the soil-foundation interface depend on the type of boundary being selected (elementary[22], viscous, transmitting or domain reduction method conditions).

5.25. Soil discretization should be fine enough to produce an accurate representation of all frequencies of interest in the structural response. In addition, at the soil-foundation interface, the

[22] ‘Elementary’ boundaries refer to simple boundary conditions such as setting vertical model boundaries free and establishing a kinematic connection between displacements at opposite faces of the soil model.

level of discretization should be able to accurately model the stress distribution and, if required, uplift phenomena.

Sub-structuring methods

5.26. SSI analyses by sub-structuring methods should include the following activities:

a) Site response analysis (Section 3);

b) Develop the model for the structure, normally using finite elements;

c) For rigid boundary methods[23], obtain the foundation input motion (kinematic interaction or "scattering problem");

d) Obtain the foundation impedances, using continuum mechanics methods, finite element methods or impedance handbooks;

e) Analysis of the coupled soil-structural system and obtain the required response parameters.

5.27. Implementation details vary from one type of sub-structuring method to the other (e.g. rigid boundary methods, flexible boundary methods, flexible volume methods and substructure subtraction methods).

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

Structure-soil-structure interaction[24]

5.29. The designer should assess the potential relevance of structure-soil-structure effects based on the following considerations:

a) Plant layout, separation between independent structures;
b) Soil stiffness and damping;
c) Differences in footprint and total mass among adjacent buildings, 'small' buildings located close to 'large' heavy buildings are of particular concern.

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

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[23] 'Rigid boundary' refers to the interface between the foundation and the soil being rigid.

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

5.31. Since both the foundation soil and the structures exhibit three dimensional dynamic characteristics, the structure–soil–structure interaction problem is a three-dimensional phenomenon. To represent adequately the characteristics of both the foundation soil and the structures of the nuclear installation, a three-dimensional analysis should therefore be performed.

COMBINATION OF EARTHQUAKE LOADS WITH OTHER LOADS

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

- L1: Loads during normal operation;
- L2: Additional loads during anticipated operational occurrences;
- L3: Additional loads during accident conditions.

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

a) For items in Seismic Category 1, L1 loads should be combined with the demand from design basis earthquake;

b) For items in Seismic Category 1, L1 and L2 or L3 loads should be combined with the demand from design basis earthquake if the L2 or L3 loads are caused by the earthquake and/or have a high probability of being coincident with the earthquake loads (which may be the case, for example, for L2 loads that occur sufficiently frequently, independently of an earthquake);

c) For items in Seismic Category 2 which have been identified to interact with items in Seismic Category 1, the same combinations of Seismic Category 1 should be applied, possibly associated with different acceptance criteria;

d) For items in Seismic Category 3, combinations according to national practice should be applied to the relevant design basis loads.

e) Mass of snow should be considered too for sites where design snow load is relevant (e.g. larger than 1.5 kN/m^2).

SEISMIC CAPACITY

5.34. The capacity[25] of a structure, system or component depends on the limiting acceptable condition for its intended functions. The limiting condition should be defined in terms of stress, strain, displacement, duration of electrical disturbances, etc. Seismic capacity should be derived from this limiting condition using the appropriate design code. The capacity should be larger than the demand on the structure, system or component (acceptance criterion).

[25] Seismic capacity is the highest seismic level for which required adequacy has been verified, expressed in terms of the input or response parameter at which the structure or the component is verified to perform its required safety function with high confidence of low probability of failure.

5.35. For Seismic Category 1 and 2 components, acceptance criteria for load combinations, should be derived from the applicable nuclear codes.

5.36. The acceptance criteria for Seismic Category 3 should not be less stringent than the one established by the applicable national standards and codes for conventional risk facilities.

5.37. For seismic capacity calculations, material properties should be selected according to characteristic values (e.g. 95% non-exceedance probability), supported by appropriate quality assurance procedures.

5.38. Appropriate ageing considerations should guarantee the long term safe performance of structures, systems and components (Ref. [1], para. 5.51) from seismic category 1 and 2. Main ageing mechanisms such as radiation embrittlement, fatigue, corrosion, creep, and pre-stress losses should be taking into account.

5.39. Capacities associated with foundation soil failures, such as liquefaction or seismically induced settlement, should be determined following the guidance provided in Ref. [5].

6. SEISMIC QUALIFICATION

6.1. Seismic qualification is the process of verification, through testing, analysis, or other methods, of the ability of a structure, system, or component to perform its intended function during and/or following the designated earthquake. Seismic qualification should be carried out for Seismic Category 1 and 2 components.

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

QUALIFICATION METHODS

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

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

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

6.5. For equipment, a systematic evaluation of the possible modes of failure related to earthquakes should be carried out with reference to the acceptance criteria assigned by the seismic categorization. The qualification programme should address the relevant failure modes.

6.6. Qualification by analysis should be considered acceptable for items without a functional safety requirement (i.e. passive components) and when an item is of a size or scale that precludes their qualification by testing. Structures, tanks, distribution systems and large items of equipment are usually qualified by analytical methods.

6.7. Seismic qualification of active components should include the qualification of structural integrity26 as well as the qualification for functionality. Seismic qualification should be performed (a) directly on actual or prototype component; (b) indirectly on a reduced scale model, a reduced scale prototype or a simplified component27; or (c) by means of similarity where this can be established between a candidate component and a reference component and direct qualification has been performed on the latter. Regardless the method selected, it should accurately represent the actual performance of the component when it is subjected to the prescribed effects. It should be noted that testing is limited by the ability of the test rig, or other test conditions to properly re-

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

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

create the actual in-service conditions that a component will see. When using test results to qualify components, the extent that the test process is applicable should be made clear

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

6.9. The continuing increase in analytical capabilities has allowed the use of highly sophisticated numerical models to simulate behaviour of active components during earthquake. However, as all analytical techniques have limits of applicability, an appropriate validation phase of methods and software verification should be carried out by either an independent analysis or a test.

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

QUALIFICATION BY ANALYSIS

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

6.12. The seismic demand on SSCs may be computed using linear equivalent static analysis, linear dynamic analysis, complex frequency response methods or non-linear analysis, depending on the relevance of the particular component and on the national practice. Regardless of the method:

a) The input to the SSC should be defined by either design spectra or by in-structure time histories or by response spectra compatible acceleration time histories; If design spectra (or related time histories) are used, these must be shown to envelope or be conservative to the in-structure loading conditions at the location of the SSC.

b) The computational model should conservatively represent the behaviour of the candidate item under the seismic action (mass distribution and stiffness characteristics);

c) The important natural frequencies of the SSC should be estimated, or the peak of the design response spectrum multiplied by an appropriate factor should be considered as input. Multimode effects should be considered too;

d) A load path evaluation for seismic induced inertial forces should be performed. A continuous load path, with adequate strength and stiffness, should be provided to transfer all inertial forces from the point of application to the main structure housing the item. Seismic demand for all the links of this path should be computed;

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

f) Energy dissipation should be accounted for and can be modelled for SSCs in a number of ways. If a modal analysis is being performed, modal damping values can be and are available for common types of components and materials from nuclear design codes.

6.13. For mechanical equipment the isolation devices against vibrations, the size, location and number of support gaps, the connection type (e.g. flanged), the frequency of response, and the use of yielding or energy absorbing support devices may all have an effect on the damping which should be considered in the design of the components. This effect should be carefully checked and adequately modelled.

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

6.15. Simplified analytical or design procedures could be used28. All such simplified techniques should be fully validated to show their degree of conservatism in comparison with more refined modelling techniques or test results and they should be suitably documented.

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

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

6.18. In addition to inertial effects, careful consideration should be given to the effects of differential seismic motions between supports, since experience of earthquakes has demonstrated that this phenomenon can be a major contributor to the seismically induced failure of piping systems.

QUALIFICATION BY TESTING

Types of tests and typical application fields

6.19. When the integrity or functional capability of an item cannot be demonstrated with a reasonable degree of confidence by means of analysis, a testing program should be carried out to

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

prove its seismic capability or to assist directly or indirectly in qualifying the item. Types of tests include:

- Acceptance test (proof test);
- Low Impedance test (dynamic characterization test);

6.20. The Acceptance (proof) test should be used for active electrical and mechanical components to demonstrate their seismic adequacy for the Design Basis Earthquake. It is typically performed by manufacturers to demonstrate compliance with procurement specifications. Such testing is typically carried out by using a shaking table.

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

Planning

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

6.23. Qualification by testing should account for ageing effects which may cause deterioration or alter the dynamic characteristics of the item during its service life.

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

- Applicable seismic test standards
- Acceptance criteria
- Input motions;
- Functional requirements
- Boundary (support) conditions;
- Number of repetitions of testing or cycles of loading per test
- Environmental conditions (e.g. pressure and temperature);
- Operational conditions, if functional capability has to be assessed.

6.25. Qualification tests should include:

a) Functional tests intended to verify the performance of the required safety function of the component;

b) Integrity tests aimed at proving the mechanical strength of the component;

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

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

Conduct of tests

QUALIFICATION BY COMBINATION OF ANALYSIS AND TESTING

6.27. When qualification by analysis or testing alone is not practical (this may be the case of large and complex active equipment such as motors, generators or multi-bay consoles), a combination of analysis and testing, in which an analytical procedure is fed or validated by the results of benchmark tests, should be used for qualification purposes.

6.28. Modal testing of a prototype should be considered as an aid to verification of analytical models used for qualification by analysis of large and complex items.

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

a) Justify extrapolation of qualification by testing to more complex assemblies (e.g. multi-cabinet assemblies).

b) Help define the testing program, by obtaining a better understanding of the dynamic behaviour of complex systems.

c) Investigate and explain unexpected behaviour during a test.

d) Obtain a first estimate of response before performing tests on complex systems.

e) Development of an analytical model with modal frequencies, damping, etc., verified by testing of a typical component that enables the effects of component configuration variations to be analytically simulated.

QUALIFICATION BY INDIRECT METHODS

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

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

7. MARGIN TO BE ACHIEVED BY THE DESIGN

CONCEPT OF SEISMIC MARGIN

7.1. Seismic robustness is expressed by seismic margin capacity which defines the capability of a nuclear installation to achieve certain performance for seismic loading exceeding those corresponding to SL-2. Seismic margin should be provided by conservatism associated to definition of SL-2, application of the nuclear safety requirements and applicable nuclear design codes.

7.2. If seismic failure of a main safety function occurred for the hazard severity close to the seismic design capacity and seismic performance goal is not achieved such conditions correspond to seismic induced cliff edge effect. The design should provide adequate seismic margin to (i) protect items important to safety and to avoid cliff edge effects; (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.

7.3. Seismic margin is measured by the High Confidence Low Probability of Failure[29] (HCLPF) which provides the link with the seismic fragility at the installation level. Moreover, seismic hazard severity corresponding to the initiating of seismic failure of the main safety functions can be estimated based on the mean installation level fragility.

7.4. There is a correlation between hazard level used to define SL-2, seismic margin capacity (HCLPF) and seismic performance goal (e.g. Seismic Core Damage Frequency (S-CDF), Large Release Frequency (S-LRF) or Large Early Release Frequency (S-LERF) as applicable). In this context, the minimum required seismic margin capacity of the nuclear installation should be prescribed to ensure that seismic performance goal is achieved and cliff edge effect will not occur.

ADEQUATE SEISMIC MARGIN

7.5. For nuclear installations such as NPPs and research reactors, both seismic margins capacity (expressed in HCLPF) should be calculated: a) first corresponding to prevention of core/fuel damage and b) corresponding to early or large releases. For other nuclear installations seismic margins should be consistent with the risk metrics associated to the installation accident conditions.

7.6. An adequate seismic margin expressed as minimum facility level HCLPF should be established[30]. For prevention of the core damage, the minimum facility level seismic margin HCLPF should be consistent with the required seismic performance goal (e.g. S-CDF $< 1.0^{-5}$). For

[29] HCLPF represents the peak ground acceleration (PGA), as the hazard parameter, corresponding to 5% conditional probability of failure on the 95% confidence fragility curve or alternatively can be defined as PGA on the mean fragility curve corresponding to 1% conditional probability of failure.

[30] When Seismic Margin Assessment is used for sites with low/medium seismicity the adequate seismic margin (at facility level) is typically defined by HCLPF > 1.5x SL-2.

prevention of early or large releases the minimum facility level seismic margin HCLPF should be consistent with the required seismic performance goal (e.g. LERF < 1.0^{-6}).

PROCEDURES TO ASSESS MARGIN

7.7. Procedures for quantification of seismic margins for existing nuclear installations are given in Ref. [3]. Those procedures use the as-built and as-operating conditions of the SSCs and for this reason seismic walkdowns is a key element. Same procedures for assessing the seismic margin of existing nuclear installations should be used at the design stage with the following caveats: seismic capacity of selected SSCs is not negatively affected by a) seismic interactions and b) by any design changes.

7.8. Seismic Margin Assessment is typically performed for low/moderate seismicity and Seismic Probabilistic Safety Assessment S-PSA is recommended for sites with high seismicity. S-PSA will provide, in addition to facility seismic margin, more insights about seismic robustness of the design, seismic performance expressed in S-CDF and S-LRF or S-LERF, and the significant contributors to seismic risk that may include human errors associated with recovery actions

7.9. In the probabilistic approach, the median, mean plant state fragility and seismic performance goal expressed in mean seismic CDF or other relevant risk metrics should be calculated. The plant HCLPF should be obtained from the mean plant state fragility. The plant level HCLPF can also be determined using sequence based (PRA based) seismic margin analysis.

7.10. In the deterministic approach (SMA method) the two success paths for bringing the plant in a safe shutdown mode should be identified and the HCLPF capacity is evaluated for SSCs belonging to these success paths. In this way the plant HCLPF and the SSCs that are limiting the plant HCLPF are evaluated.

7.11. The facility level seismic margin (HCLPF) should be compared with the adequate seismic margin defined in paragraph 7.6 or established by the national regulatory body.

8. SEISMIC INSTRUMENTATION AND RESPONSE TO AN EARTHQUAKE EVENT

SEISMIC INSTRUMENTATION

8.1. Seismic instrumentation — an array of strong motion accelerographs installed at the plant site and in-structure plays a key role in collecting site specific seismic instrumental data during the life cycle of the nuclear power plant. Seismic instrumentation should be installed at nuclear installations for the following reasons:

a) In some member states, to provide triggering mechanisms for the automatic shutdown of the nuclear installation in case that the earthquake exceeds a defined threshold;

b) To provide alarms for alerting operators of the earthquake occurrence, and to provide information for decision making process defined according to operation procedures;

c) To collect data on the dynamic behaviour of SSCs of the nuclear installation in case of the occurrence of an earthquake, to obtain realistic data on the structural response and to assess the degree of validity of the analytical methods used in the seismic design and qualification of the buildings and equipment;

8.2. During the site evaluation stage, as recommended in Ref. [2] a local network of seismographs (of both short period and broadband period types) should be installed and operated near the site, i.e. the zone within about 25–40 km around the plant site, to acquire detailed information on potential seismogenic sources for seismotectonic interpretation. This local network is usually connected to the regional and national seismological networks and its use refers mainly to seismological purposes.

8.3. Seismic categorization and safety classification of seismic instrumentation should be decided based on the relevance of the postulated seismic initiating event . In addition, the need for the seismic instrumentation in the emergency procedures for the nuclear installation should be taken into account.

8.4. Seismic automatic scram systems, where installed, should be properly safety classified according to Ref. [6] and adequate redundancy should be provided. All requirements for reliability, redundancy and independence of failure of any component or signal used in common with the reactor protection system should be considered.

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

8.6. Processing, interpretation and use of the data obtained from the seismic instrumentation, should be part of the operational procedures (including emergency operating procedures)of the installation and managed according to the established management system.

8.7. A suggested minimum amount of seismic instrumentation should be installed at any nuclear installation site as follows:

a) One triaxial strong motion recorder installed to register the free field vibratory ground motion;

b) Three triaxial strong motion recorder installed to register the vibratory motion of the basemat of the reactor building in a nuclear power plant,

c) Two triaxial strong motion recorder installed on the most representative floors of the reactor building in a nuclear power plant, or in the basemat of the building of structure with the biggest amount of radioactive material in other than nuclear plant installations.

In addition to the minimum seismic instrumentation described in paragraph 8.7 additional instrumentation should be considered for sites having an SL-2 free field acceleration equal to or greater than 0.2g.

8.8. The seismic instrumentation installed at the nuclear installation should be able to provide estimate of the cumulative damage parameters based on the integration of the acceleration record, thus providing a more representative parameter of earthquake induced damage in the safety related equipment and as important tool and data for assessing the installation response in case of an earthquake occurrence.

8.9. Such damage indicators should be compared with values of the same quantities derived from the free field design basis earthquake and with data from earthquake experience. Analogous comparisons should be made in other plant locations since they could provide good support for the post-earthquake walkdown and therefore for the decision on the restarting of plant operation.

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

RESPONSE TO AN EARTHQUAKE EVENT

8.11. Post-earthquake actions should be planned for a nuclear installation at design stage as part of a dedicated programme of operational response to the occurrence of such external event. This Post-Earthquake Action Programme should include a combination of pre-earthquake planning and short and long-term actions. At the seismic design stage of the nuclear installation, in accordance with the specific characteristics of the installation design and operation, the principles and general specifications of the Post-Earthquake Action Programme should be formulated and prepared.

8.12. The Post-Earthquake Action Programme should be based on:

a) A rational, experience based, approach for determining the real damage potential of felt and significant earthquakes;

b) A systematic methodology for assessing the need for plant shutdown and the plant readiness for restart, based on physical inspections and tests (if the plant has been shut down);

c) Criteria for assuring the long-term integrity of the plant.

8.13. In addition, the Post-Earthquake Action Programme should be comprehensive enough to minimize the likelihood of prolonged plant shutdowns following seismic ground motions that do not damage SSCs important to safety. For earthquakes below the design basis levels (SL1 and/or

SL2), primary emphasis is on the physical and functional conditions of the installation, as opposed to analytical evaluations. In some cases, confirmatory analytical evaluations may be performed while the installation is in operation after restart.

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

8.15. However, the intent is that the initiation of the recommended actions as part of such a programme be limited to only those earthquakes that, having been felt at the nuclear power plant, are considered to be ‘significant earthquakes’. A significant earthquake is a felt earthquake having free field surface ground motion characteristics approaching the threshold of damage or malfunction of non-seismically designed SSCs. Some typical definitions of a significant earthquake are earthquakes with a free field surface ground motion greater than 0.05g (where g is the acceleration due to gravity) or a standardized cumulative absolute velocity (CAV) greater than 0.16g·s or an earthquake with spectral accelerations in the 2–10 Hz range greater than 0.2g (5% damping) or an earthquake with spectral velocities in the 1–2 Hz range greater than 15.2 cm/s.

8.16. The designation of a significant earthquake is a function of the site and the seismic design basis of the nuclear power plant, since it may determine the actions to be taken by the licensee and the regulatory body. The definition of the significant earthquake is the responsibility of the licensee and may require agreement or approval by the regulatory body.

8.17. Given the background described above and the need for dealing with earthquakes that are felt and significant at existing nuclear power plants, a comprehensive post-earthquake action programme should be established and implemented with the objectives of providing guidance and specific and detailed procedures to the operating organization at the plant site and at headquarters, covering the complete range of seismic ground motions ranging from values lower than those corresponding to the SL-1 earthquake level to values higher than those corresponding to SL-2 earthquake level.

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

a. The post-earthquake actions to be taken will facilitate timely decision making concerning the present or future state of the nuclear power plant, for example, to shut down, to continue in operating mode or to restart;

b. Communication to all stakeholders will be timely and transparent with regard to plant status, actions taken and actions to be taken;

c. A tiered approach is to be employed starting with overall evaluations and proceeding to very detailed evaluations only when required by the situation.

d. Conforming to these principles, the two basic stages of the programme are:

(i) Planning: Pre-earthquake activities with a view to preparing an appropriate response, these activities include all tasks to be performed in advance, many of them during the design phase and before an earthquake occurs;

(ii) Response: Post-earthquake action plans defined as a function of the earthquake felt or ground motion recorded at the site and the observed consequences to the plant, after an earthquake occurs and as part of the operational response.

Specific and detailed guidance is provided in IAEA Safety Report Series 66, Earthquake Preparedness and Response for Nuclear Power Plants, 2011 [7].

9. NUCLEAR INSTALLATIONS OTHER THAN POWER PLANTS

9.1. The graded approach should ensure the seismic design criteria are commensurate with the relative importance to safety, magnitude of the seismic hazard, radiological and chemical hazards and other relevant factors.

9.2. Simplified methods for seismic hazard assessment based on more restrictive data set associated with lower return period applicable for medium and low hazard facilities should be allowed. The level of effort, complexity of analysis, and the thoroughness of documentation are commensurate with the magnitude of the facility hazard, the complexity of the facility and life-cycle phase.

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

a) The amount, type and status of radioactive inventory (e.g. solid, fluid, processed or only stored);

b) The intrinsic hazard associated with the physical processes (e.g. criticality) and chemical processes that take place at the installation;

c) The thermal power of the nuclear installation, if applicable;

d) The configuration of the installation for activities of different kinds;

e) The distribution of radioactive sources within the installation (e.g. in research reactors, most of the radioactive inventory will be in the reactor core and fuel storage pool, while in processing and storage plants it may be distributed throughout the plant);

f) The changing nature of the configuration and layout of installations designed for experiments;

g) The need for active safety systems and/or operator actions to cope with mitigation of postulated accidents; characteristics of engineered safety features for preventing accidents and for mitigating the consequences of accidents;

h) The characteristics of the structures of the nuclear installations and the means of confinement of radioactive material.

i) The characteristics of the process or of the engineering features that might show a cliff edge effect in the event of an accident;

j) The potential for on-site and off-site radiological contamination.

9.4. The nuclear installations should be categorized based on the intended design objective of the installation (i.e. the performance goal) and the consequent risk associated with it in the event of a failure of a structure(s), system(s) or component(s) relevant to the safety of installation. Based on these criteria, the nuclear installations should be placed in one of the following four categories:

- Seismic Design Category 1 (SDC1), *high hazard nuclear installations;*

- Seismic Design Category 2 (SDC2), *moderate hazard nuclear installations,*
- Seismic Design Category 3 (SDC3), *low hazard nuclear installations;* and
- Seismic Design Category 4 (SDC4), *conventional installations.*

Table 9.1 shows relation between seismic design categories and unmitigated consequences of seismic induced failure of the nuclear installation.

Table 9.1 Seismic Design Category (SDC) based on Hazard Analysis of the Nuclear Installation Failure

Nuclear installation SDC	Consequences on the site	Consequences off the site	Engineering and Safety Analysis
SDC1 High Hazard	Radiological/toxicological exposure that may cause loss of life of workers in the facility.	The hazard analysis shows the potential for significant radiological or radiological / chemical off- site consequences.	Similar rules used for NPPs apply. Engineering and safety analysis is needed to determine the preventive and mitigating features, to determine if safety objectives are met.
SDC2 Medium Hazard	The hazard analysis shows the potential for significant on-site consequences. Unmitigated release would require an emergency plan for onsite evacuation.	Small potential for off-site radiological or radiological / chemical off- site consequences	Engineering and safety analysis is needed to determine if safety objectives are met.
SDC3 Low Hazard	The hazard analysis shows the potential for only localized consequences (within 30 to 100 m from the source of releases).	No off-site radiological or radiological / chemical off- site consequences.	Limited engineering safety analysis is needed to determine if safety objectives are met.
SDC4 Conventional installations	No radiological or chemical release consequence but failure of the SSC may place facility workers at risk of physical injury.	No off-site radiological or chemical off- site consequences	Conventional Design Codes.

9.5. Structures, systems and components (SSCs) should be seismically designed to account for:

a) The seismic design category of the nuclear installations where they are to perform should a SL-2 occur;
b) The appropriate state limit should a SL-2 occur (specify the analysis methodology, design procedures, and acceptance criteria)[31].

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

c) b)c) SSCs whose seismic failures do not have any interactions with safety function should correspond Seismic Category 3. National practice for seismic design of non-nuclear installations apply.

9.6. Structures, systems and components (SSCs) should be seismically designed and qualified according to the SDC, and target seismic performance Goal as presented in Table 9.2.

Table 9.2 Relation between Seismic Design Category, Seismic Hazard Level and Design Codes for Achieving the Target Performance Goal.

Seismic Design Category	Design Codes and Standards	Seismic Hazard Level	Target Seismic Performance Goal
SDC-1 High Hazard Facility	Nuclear	SL-2 / 1.0E-4	< 1.0E-5
SDC-2 Medium Hazard Facility	Nuclear	SL-2 / 1.0E-3	< 1.0E-4
SDC-3 Low Hazard Facility	Conventional	1.5x National Seismic Code	< 5.0E-4
SDC-4 Convectional Facility[32]	Conventional	National Seismic Code	< 1.0E-3

9.7. Table 9.2 provides relation between seismic design category, performance goal, design codes and severity of seismic hazard considered in the design. The values from Table 2 are based on [12].

[32] Some high risk non-nuclear industrial facilities may be seismically designed similar with SDC3 Low Risk Nuclear Facilities.

10. APPLICATION OF MANAGEMENT SYSTEM

10.1. An integrated management system should be established covering the organization, planning, work control, personnel qualification and training, verification and documentation for activities to ensure that the required quality of the work is achieved [8].

10.2. The management system should ensure the quality and the control of the activities performed at each stage of the design.

10.3. As part of the management system, the design process or processes for the development of the concept, detailed plans, supporting calculations and specifications for a nuclear installation and its parts, should be established and conducted following the recommendations and guidance provided in Ref. [9].

10.4. Design inputs, processes, requirements, outputs, changes and records should be established and controlled. The design outputs include specifications, drawings, procedures and instructions, including any information necessary to implement or install the designed SSCs or protective measures.

10.5. Design inputs, processes, outputs and changes should be verified. Individuals or groups performing design verification should be qualified to perform the original design. Those carrying out verification should not have participated in the development of the original design (but they may be from the same organization). The extent of verification should be based on the complexity, the associated hazards and the uniqueness of the design. Some typical design verification methods include design review, carrying out calculations by an alternative method and qualification testing. Previously proven designs do not need to be subject to verification unless they are intended for different applications or the performance criteria are different. Design records, including the final design, calculations, analyses and computer programs, and sources of design input that support design output, are normally used as supporting evidence that the design has been properly accomplished [9].

10.6. The design process should include the following activities; recommendations and guidance on these activities are provided in Ref. [9, Paras 5.87–5.140]:

a. Design initiation, specification of scope and planning;

b. Specification of design requirements;

c. Selection of the principal designer;

d. Work control and planning of design activities;

e. Specification and control of design inputs;

f. Review of design concepts and selection;

g. Selection of design tools and computer software;

h. Conducting conceptual analysis;

i. Conducting detailed design and production of design documentation;

j. Conducting detailed safety analyses;

k. Defining any limiting conditions for safe operation (sometimes referred to as the safe operating envelope);

l. Carrying out design verification and validation;

m. Configuration management;

n. Management of the design and control of design changes.

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

10.8. Interfaces among all organizations involved in the design should be identified, coordinated and controlled. Control of interfaces includes the assignment of responsibilities among, and the establishment of procedures for use by, participating internal and external organizations [9].

REFERENCES

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

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, “Seismic Hazards in Site Evaluation for Nuclear Installations,” IAEA Specific Safety Guide SSG-9, Vienna, 2010.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, “Evaluation of Seismic Safety for Existing Nuclear Installations,” Safety Guide NS-G-2.13, IAEA, Vienna, 2009.

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2016 Edition, IAEA, Vienna (2016).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, “Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants,” IAEA Safety Guide NS-G-3.6, Vienna, 2004.

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, “Safety Classification of Structures, Systems and Components in Nuclear Power Plants,” IAEA Specific Safety Guide SSG-30, Vienna, 2014.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, “Earthquake Preparedness and Response for Nuclear Power Plants”, Safety Reports Series No. 66, Vienna, 2011.

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, “Leadership and Management for Safety”, General Safety Requirements GSR Part 2, Vienna 2016.

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, “The Management System for Nuclear Installations”, Safety Guide GS-G-3.5, Vienna, 2009.

[10] [9]INTERNATIONAL ATOMIC ENERGY AGENCY, “Safety of Research Reactors”, SSR-3 Specific Safety Requirements, Vienna, 2016.

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, “Safety of Nuclear Fuel Cycle Facilities”, SSR-4, Specific Safety Requirements, Vienna, 2017.

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, “Approaches to safety evaluation of new and existing research reactor facilities in relation to external events” Safety Report Series no xx (in publication).

LIST OF ABBREVIATIONS

CAV	Cumulative Absolute Velocity
HCLPF	High Confidence Low Probability of Failure
CDF	Core Damage Frequency
LERF	Large Early Release Frequency
HVAC	Heating, Ventilation and Air Conditioning
OBE	Operating Basis Earthquake
PSHA	Probabilistic Seismic Hazard Assessment
SAR	Safety Analysis Report
SL-1, SL-2	Seismic Level 1, Seismic Level 2
SMA	Seismic Margin Assessment
SPSA	Seismic Probabilistic Safety Assessment
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake

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