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OVERVIEW OF THE SEISMIC PROBABILISTIC SAFETY ASSESSMENT APPLIED TO A NUCLEAR INSTALLATION LOCATED IN A LOW SEISMICITY ZONE

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ABSTRACT

Permanent concern on the safety of nuclear installations shall be assured in order to maintain the protection of workers, individuals from the public and the environment. Safety analysis methodologies for both approaches, deterministic and probabilistic, have been developed and updated based on operational experience, investigation of past incidents or accidents, and analysis of postulated initiating events. In general terms, the main objectives of a nuclear safety study are the identification of a comprehensive list of accident initiating events, the evaluation of their impact on the installation and the assessment of the total radiological risk resulting from accidents with off-site releases.

Among all initiating events and hazards, there are external hazards that continually challenge the safety of a nuclear facility or its nearby area. In particular, seismic events represent a major contributor to the risk of a nuclear facility. Large levels of ground motion induced by earthquakes may be experienced due to the propagation of mechanical waves on the ground, caused by the displacement of tectonic plates. In this context, a seismic hazard analysis can be carried out in order to predict local acceleration levels with the associated uncertainty distribution, allowing an adequate seismic classification of plant structures, systems and components, including installations located in sites with low seismicity.

In order to estimate the risk of a nuclear installation concerning accidents induced by seismic events, a Seismic Probabilistic Safety Assessment (Seismic PSA) shall be performed. In this article, a general description of the Seismic PSA methodology is presented, with emphasis on the supporting studies for this assessment.

Finally, this study is under the scope of a master degree project at IPEN – CNEN/SP which intends to apply the methodology described in this article to an experimental nuclear installation containing a PWR reactor designed for naval propulsion to be installed in a low seismicity zone in Brazil.

1. INTRODUCTION

The design of a nuclear installation is feasible when the radiological risk associated with its operation is as low as reasonably achievable. Radiological levels are evaluated in the design basis and are supposed to be maintained at an acceptable value over the plant lifetime [1]. The evaluation of the installation radiological risk with regard to the public health, the economy and the environment and taking into consideration abnormal or accident conditions may be performed by means of two complementary approaches, the Deterministic Safety

Analysis (DSA) and the Probabilistic Safety Assessment (PSA) [2]. In general, the DSA is the primary approach for the safety assessment of nuclear installations and is required by regulatory authorities for the operator or license applicant to demonstrate safety margins during normal operations and transient conditions expected to occur during plant lifetime. In addition, the adequacy of items designed to prevent design basis accidents and to minimize their consequences should also be demonstrated. The PSA is a comprehensive and integrated assessment of the safety of a nuclear plant that derives numerical estimates for the risk of accidents with significant damage to the radioactive sources in the plant by considering the initial plant state and the probability, progression, and consequences of multiple failures (equipment failures, human errors and internal/external hazards). A PSA can be used as a complementary tool in the regulatory decision making process, and it can be used by the designer or operator in several applications, in all lifetime phases of the plant (design, construction, commissioning, operation and decommissioning). Besides, to comprehensively assess the risk associated with accidents involving damage to radioactive sources at the plant, it is essential that all groups of internal events and internal/external hazards be considered in the safety analysis.

Internal events originating from sources located inside the reactor building or on the plant site are related to random failures of Structures, Systems and Components (SSCs) as well as to human errors in the execution of procedures. Internal hazards such as fire and floods must also be included in the scope of the safety analysis. External hazards originating outside the plant site may be natural or man-induced, or a combination thereof.

With the evolution of the design of nuclear plants regarding the use of passive safety systems, acquired experience with the operation of existing plants and the use of new equipment technologies, the risks associated with internal events have decreased, leading to an increase of the importance of studies on the impacts of internal fire, internal floods, seismic events and other external hazards.

External hazards analyses are largely site and plant specific. These events can mainly affect building structures (including containment and/or confinement), ventilation and air-conditioning systems, cooling systems and ultimate heat sink, electrical power supply, components of safety systems and the accessibility of the plant.

Among the fundamental roles of a PSA, the identification of the SSCs capacity to support the occurrence of an external event is one of the most important results. However, the estimation of risk metrics, such as Core Damage Frequency (CDF) and/or Large Early Release Frequency (LERF) or Large Release Frequency (LRF), concerning low probability and high impact external hazards is still a challenge. This is the case of seismic events, as they continually challenge the safety of a nuclear plant and large levels of earthquake-induced ground motions can be potentially experienced in the site of a nuclear plant and in its vicinity.

Furthermore, earthquake-induced initiating events represent hazards with complex characteristics. The range of ground motion levels form a continuous scale and the failure probabilities of SSCs depends on particular ground motions. In addition, the following specificities may be mentioned [3]:

- Seismic events can damage not only active components, but also structures which under normal conditions have extremely low failure probabilities, generating specific failure modes that are not reflected in the accident sequence models for other initiating events;

- Seismic events can have a large spatial impact, damaging multiple structures, redundant systems and multi-unit sites;
- Mitigating the effect of a seismic event may require more complex actions than the ones taken to mitigate other initiating events;
- Seismic PSA comprise larger uncertainties, which propagate from hazard and fragility analyses;
- Depending on the magnitude of the seismic event, ground motions in the plant may exceed the design basis criteria. Therefore, in a Seismic PSA, the failure probabilities of SSCs shall be evaluated considering beyond design basis ground motions.

Some supporting analyses are critical with respect to the development of a Seismic PSA. The probability of earthquakes of different intensities (seismic hazard), including the structural response of the plant (seismic demand) and the susceptibility of the SSCs to a given earthquake intensity (seismic fragility) are the main contents generated before the seismic risk quantification. Then, the expansion of the event tree and fault tree models due to earthquake-induced initiating events is performed to identify the impact on the plant [4]. Finally, the quantification of the CDF and/or LERF or LRF is made based on plant fragility (fragility curves) and probability of occurrence of seismic events (seismic hazard curves).

The technical elements necessary for the development of a Seismic PSA are still analytically sophisticated and require extensive engineering judgment, as shown in Fig. 1. Thus, in this article, focus is given on the existing guidance and current practices regarding the elements listed below:

- Seismic Hazard Analysis: consists of the characterization of the site in order to verify the frequency of occurrence of seismic events for a given ground motion parameter (GMP). The main results of the analysis are the Seismic Hazard Curves and the Uniform Hazard Response Spectra (UHRS);
- Seismic Demand Analysis: based on the UHRS, Soil-Structure Interaction (SSI) is evaluated to verify site conditions and SSCs response to a given GMP level. The main product of the analysis is the Floor Response Spectra (FRS); and
- Seismic Fragility Analysis: based on the FRS, SSCs capabilities to support a given GMP level are assessed. Fragility curves of the SSCs are generated. In addition, it allows the identification of the weak links of the plant systems for specific ground motion levels.

2. SEISMIC HAZARD ANALYSIS

In general, choosing the location or region in which a nuclear plant is to be installed is not a trivial task with regard to safety evaluation for seismic hazards. It is known that the characterization of the phenomenon depends on local environmental conditions and it is still difficult to predict the occurrence, location and magnitude of seismic events.

A starting point for predicting calculations is the history of these events over time. In regions of high seismicity, i.e., where earthquakes occur with greater frequency and considerable magnitudes, associated with regional geology and existing faults, important data are provided for the formulation of prediction models of the phenomenon in a specific region. On the other hand, in low seismicity regions, the quantification of parameters used in prediction models (recurrence interval, magnitude and distance epicenter-site) must incorporate, in an appropriate way, the uncertainties involved.

The Brazilian territory is an example of a region of low seismicity. It is located on South American Plate [5], as shown in Fig. 2.

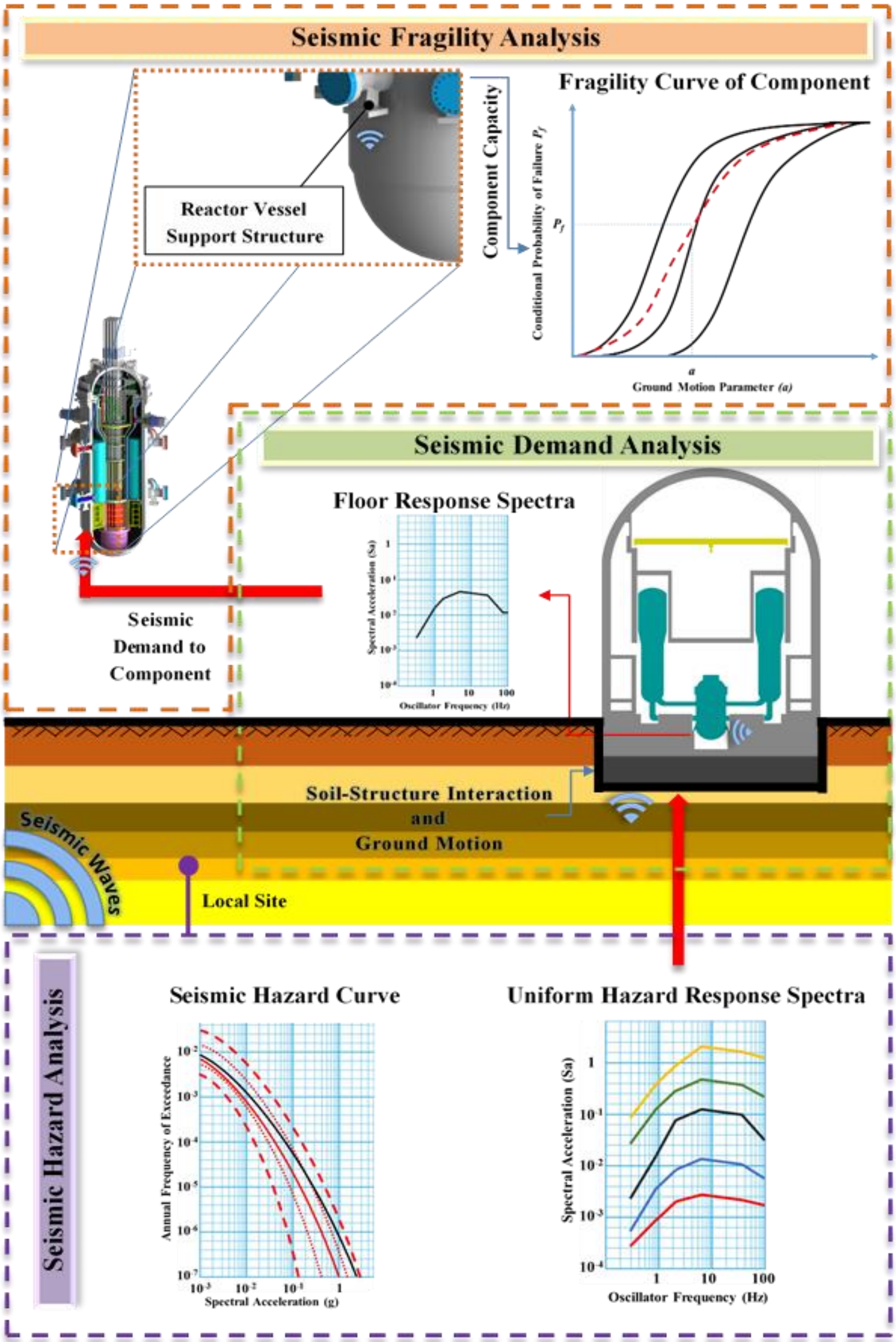

Figure 1. Technical elements necessary for the development of a Seismic PSA.

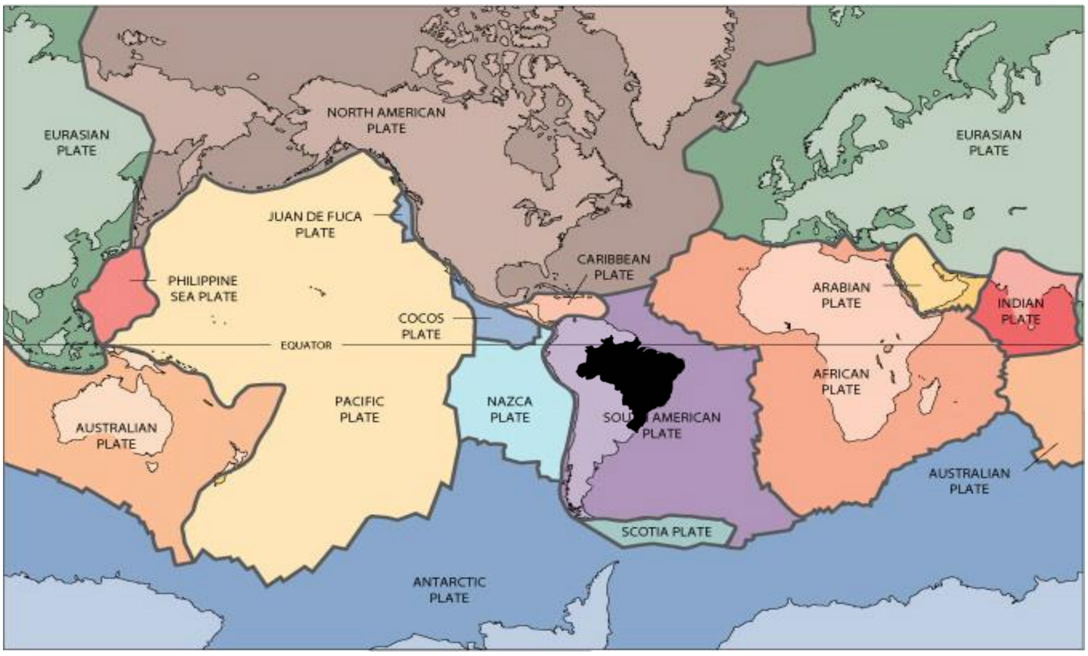

Figure 2: Division of the earth's crust into 12 tectonic plates. Brazil appears in black in an intraplate region on South American Plate.

2.1. Process overview

A seismic event is a natural phenomenon with large energy release that propagates in mechanical waves and can follow different paths, including the terrestrial mantle, as in the case of body waves. On the other hand, the surface waves can reach longer distances propagating through the terrestrial crust, where the civil structures are founded [6]. These structures can experience any type of wave at different times, but considering the main event, when the mainshock occurs, the duration of the phenomenon is generally about 30 seconds, long enough to cause great destruction. In a region of low to moderate seismicity – classified with degree VI - VII in the Modified Mercalli Intensity scale (MMI) – as the Brazilian territory, even being located in an intraplate area, reflections and refractions of waves originated from great distances can be experienced [7].

Although there have been reports of perceptions of seismic phenomena in Brazil since 1560 [8], the rate of occurrence of activity with magnitudes that produce destructive effects is low. These low activity rates generally yield a small number of events of sufficient magnitude and/or ground motion amplitude to be of engineering or seismological interest. As a result, Ground Motion Prediction Equations (GMPEs) derived for these areas will have large uncertainties [9]. And it can be difficult to properly partition the epistemic and aleatory components of uncertainty in the GMPEs. Besides, it can be much more difficult to confidently identify and characterize seismic sources [10].

A common practice to verify the behavior of seismic phenomena in zones of diffuse (low to moderate) seismicity is to compare this behavior with results of parameters of prediction models obtained for other similar intraplate regions. An example is the work done for the region of Thyspunt, a rocky stretch of coast in the Eastern Cape, province of South Africa [11].

The Probabilistic Seismic Hazard Analysis (PSHA) plays an essential role in modeling the epistemic uncertainties involved in the study of seismic event prediction, whether for high seismicity regions or for diffuse seismicity regions. The uncertainties in the determination of Safe Shutdown Earthquake (SSE) can be considered in the calculations performed in the PSHA. However, the assumptions adopted and the methodology applied in a PSHA depend on the tectonic configuration of the site and the proper characterization of the input parameters for the analysis, making it specific for an area or for a set of areas of the same seismic source [12].

In general terms, for the identification of all possible seismic events and resulting ground motions, together with the associated probabilities of occurrence, a PSHA can be performed following the steps below [13] [14]:

- <u>Treatment of input data for PSHA calculations</u>: the seismic phenomenon is analyzed based on its time history, characterization of seismic sources and local geology. Two characterizations are performed: Seismic Source and Ground Motion. In the first case, the qualitative and instrumental historical records of seismic events and failures in the region of interest are collected and treated. Initially, the magnitudes of the historical data are homogenized in a moment magnitude (M_w). Then, an algorithm is used to exclude records from beforeshock and aftershock earthquakes. The remaining main events are grouped in seismic area source zonation, in which the magnitude-frequency relations, probability distribution of magnitude and earthquake recurrence model are generated. In the second part, the physical and mechanical characteristics of regions similar to the site of interest are studied for the scaled backbone GMPEs. After assembly of the backbone, GMPEs that are compatible with the site of interest are selected through expert assessment. At the end of each characterization, uncertainties in earthquake size, location and ground motion are combined, using the total probability theorem, through the logic-tree models. It is a technique used to identify and estimate the epistemic uncertainties associated with the lack of knowledge about seismic processes, seismicity characteristics and ground motion in the specific region under study. For its construction and generation of results, weights are defined from the evaluations carried out by several specialists, starting from a specific database.

- <u>Hazard calculations:</u> based on the logic-trees developed for the characterizations carried out in the previous step, the maximum and minimum magnitudes for data integration are defined. The PSHA results are generated from the combination of the branches of the logic-trees through the use of specific codes. Some examples are presented in sub-item 2.2.

- <u>Hazard results</u>: the main results generated are the Seismic Hazard Curves, Uniform Hazard Response Spectra and Disaggregation. Vertical Response Spectra and Response Spectra for alternative damping values may also be generated.

Fig. 3 shows the flowchart of the activities above-mentioned performed in each step of the PSHA process.

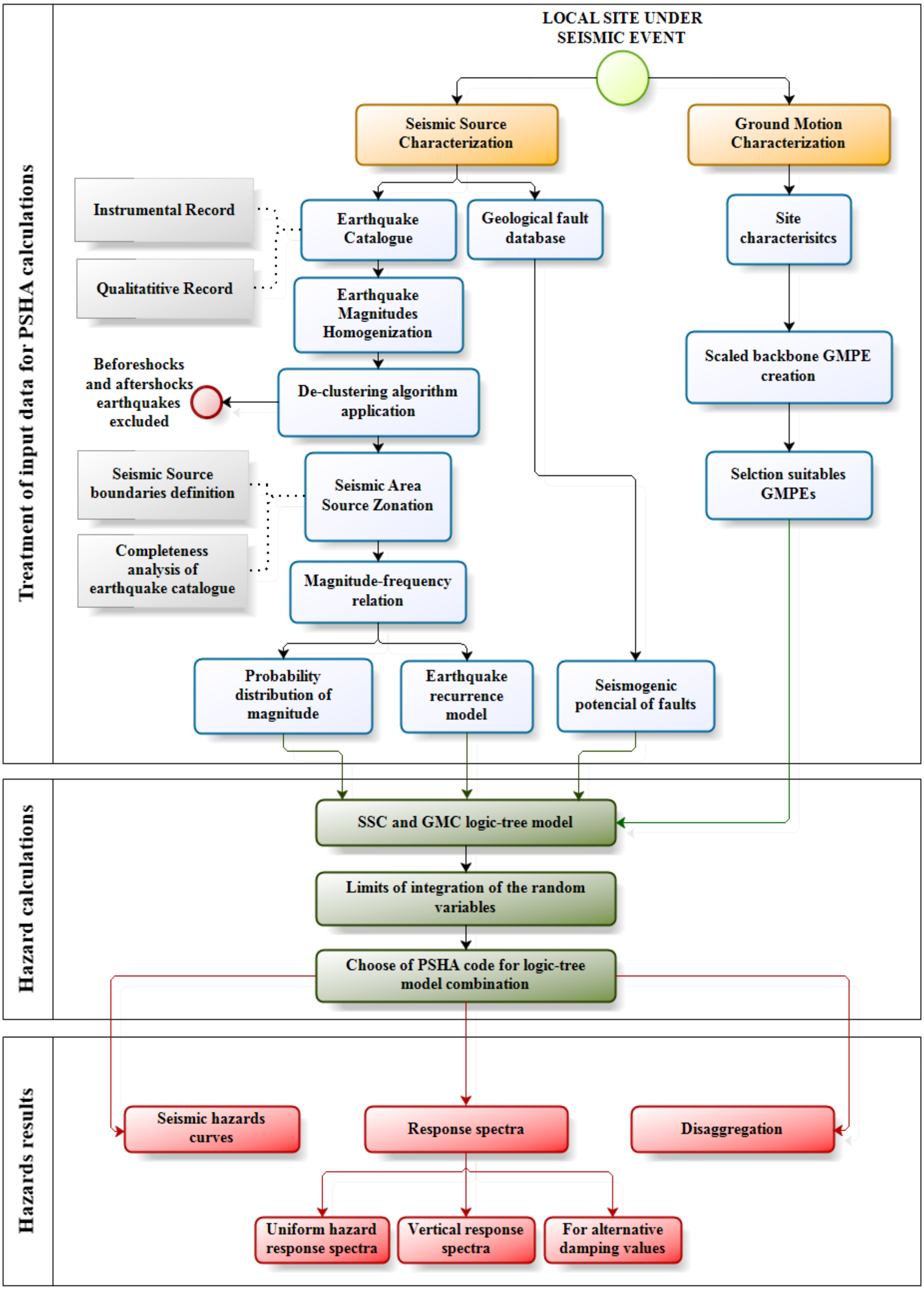

Figure 3. PSHA process calculation.

2.2. PSHA computer codes

The parameters considered in the calculations performed in a PSHA follow probability distributions of random variables and their integration, depending on the complexity of the study, may require a significant number of operations and the use of computational codes for numerical calculation. Examples of well-known and open source computational codes used in PSHA studies are *OpenQuake* (GEM Foundation) [15], *R-CRISIS* (Instituto de Ingeniería – Universidad Nacional Autónoma de México & Evaluación de Riesgos Naturales – ER) [16] and *OpenSHA* (Field, E.H., Jordan, T.H., and Cornell, C.A.) [17].

2.3. Main results

Seismic Hazard Curves

Seismic Hazard Curve is the most important and most commonly used tool in seismic hazard analysis. It consists of a graph whose abscissa represents a Ground Motion Parameter (GMP) and the ordinate represents the Annual Frequency of Exceedance (AFE) of the observed value for this GMP. A family of curves are plotted on the graph, which show the probability that a specific GMP level will be exceeded at the point of interest. A hazard curve is developed for each seismic source. The final hazard curve represents the sum of the rates at which a given GMP level is exceeded in each area of seismic source.

Some examples of seismic hazard curves for frequencies of 1, 10 and 100 Hz calculated for a specific area located in the Southeast region of Brazil are shown in Fig. 4 [7]. These curves are calculated as a function of Spectral Acceleration (Sa), which is a GMP, with their respective fractiles, mean and median. In the case of the 100Hz frequency plot, the Spectral Acceleration equals the Peak Ground Acceleration (PGA).

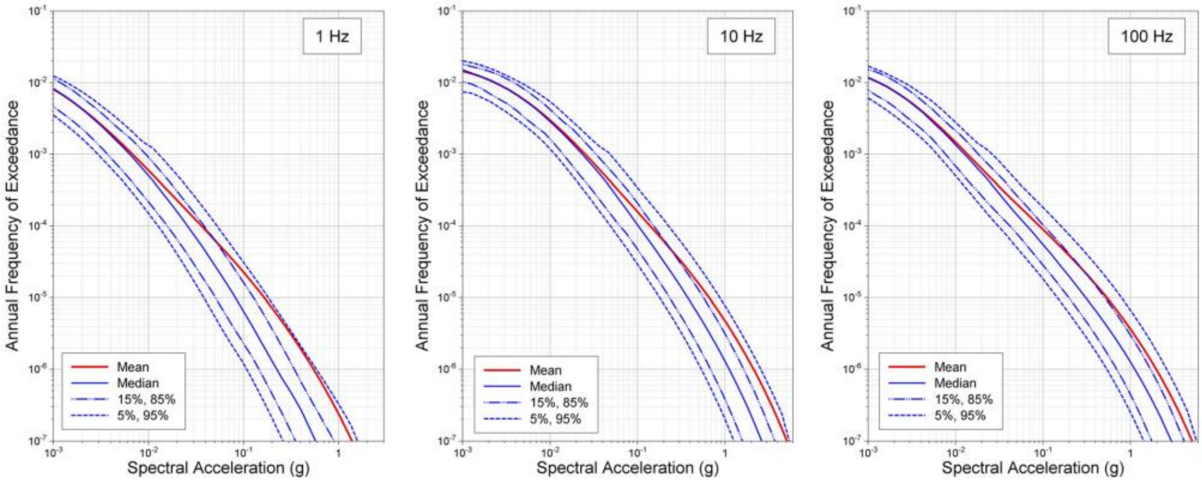

Figure 4. Seismic hazard curves for the 1, 10 and 100Hz frequencies in an area located in the Southeast region of Brazil.

Uniform Hazard Response Spectra – UHRS

The UHRS are the response spectra obtained from mean hazard curves derived from a GMP (PGA or Sa, for example) at frequencies of multiple responses, such that each acceleration value has the same AFE. The UHRS is used to obtain the Ground Motion Response Spectra (GMRS), which, in turn, is used as input for calculation of structural seismic loading.

Disaggregation

Seismic hazard curves show the combined effect of all magnitudes (M), distances (R) and standard deviations (ε) in the calculation of the probability of a GMP to exceed a certain value. Based on these curves, it is difficult to identify specific combinations, that is, event scenarios in which M, R and ε most contribute to the seismic hazard. A common practice is to decompose the total probability of exceedance into different degrees of magnitude and distance. This allows a better identification of the "seismic control events", i.e., the subsets of seismic events responsible for the level of ground motion severity at the site of interest.

3. SEISMIC DEMAND ANALYSIS

A seismic phenomenon consists of a large release of energy in wave form and movement. Generally, the GMPs used to characterize this movement are the PGA or Sa. In projects that require seismic qualification for the SSCs, the determination of the intensities of the seismic phenomena in a certain place of interest is fundamental for the structural dimensioning, directly affecting the cost of construction. For nuclear plants, it is mandatory to demonstrate that ground accelerations will not result in damage to the structures and items supported therein [18], whereas acceleration amplification may occur – e.g., an acceleration of 0.1g on a foundation can be amplified to 0.3g at a higher level of the structure, and the acceleration can reach 3g or more on internal items fixed at this level. The interaction of a civil structure under seismic action may attenuate or amplify the acceleration from the base motion. From the engineering point of view, the damage shall be characterized in terms of intensity associated with a GMP, usually the PGA.

In order to verify the effects of the ground that is under the action of a seismic event in a place of interest, one must analyze its characteristics. The geological characterization generated in the Seismic Hazard Analysis, specifically in the local PSHA, provides the information needed for this task. Acceleration amplifications in the foundation of a nuclear plant can occur due to the different materials (different module of elasticity) present in the ground. To reduce undesired effects, it is recommended that the foundation be built on a competent material (rock) [18]. Thus, it can be considered that, approximately, the ground motion (competent rock) will be the same as that of the foundation, simplifying the calculations of the structure.

Dynamic finite element models of the structures of a nuclear plant are established and the responses of the structures at the desired locations are determined by dynamic structural analysis. If the plant is founded on soil (absence of competent rock), soil-structure interaction should be considered when performing dynamic structural analysis [6].

A Site Response Analysis is performed to evaluate the effects of soil conditions on earthquake-induced motions at a specific elevation level, such as the free surface or foundation level, due to propagation of shear waves in the ground. In this way, the results generated in this analysis are considered as inputs for Seismic Demand Analysis, which is the analysis of the structure response based on the ground under study. Table 1 shows the main calculations involved in these two analyses: the Site Response Analysis and the Seismic Demand Analysis.

Table 1: Calculations performed in Site Response and Seismic Demand Analyses

Analysis	Calculations
Site Response	• Developing input ground motions at reference hard rock • Establishing geotechnical model for the site • Calculating amplification functions and strain-compatible soil properties • Determining seismic hazard curves at foundation level • Determining the GMRS and Foundation Input Response Spectra (FIRS)
Seismic Demand	• Modelling of Structures • Time-History, Direct and Scaling methods for generating Floor Response Spectra (FRS) • Generating FRS considering Soil-Structure Interaction (SSI)

Finally, the seismic demand used to dimension plant SSCs determines the Structure Capacity and FRS calculations. The latter term consists of a set of response spectra for each level of the structure, considering its inherent characteristics (mass-stiffness). The FRS is one of the main inputs to qualify internal items of the structure, such as pipes, valves, electrical panels, etc. Thus, all seismic items will have the appropriate seismic loading for their dimensioning and qualification.

4. SEISMIC FRAGILITY ANALYSIS

After the modeling of the structural seismic demand described in the previous section, it is necessary to identify the plant fragility, that is, the probability of failure of the SSCs for a given level of ground motion parameter (PGA [g] or Sa [g]).

The seismic fragility of an SSC can be given by:

$$F(A) = P[SSC\ failure|GMP = a] \tag{1}$$

with:

- A = ground acceleration capacityof SSCs*; and*
- a = *GMP level.*

Due to the lack of knowledge of the actual seismic capacity of the SSCs, the uncertainties associated with their design, analysis, qualification test and construction can be expressed only in probabilistic terms and represented by a fragility curve [19].

The basic (standard) fragility model is based on a lognormal probability distribution:

$$P_{F|a,Q} = \Phi\left(\frac{ln\left(\frac{a}{A_m}\right) + \beta_U \Phi^{-1}(Q)}{\beta_R}\right) \tag{2}$$

with the following parameters:

- a: GMP level;
- A_m: median ground acceleration capacity;
- β_R: variability due to inherent randomness of response;
- β_U: variability due to lack of knowledge of structural response and uncertainties in the capacity;
- Q is the desired confidence level associated with the uncertainty variability; and
- Φ: the standard Gaussian cumulative distribution function.

The mean fragility curve is calculated by using the equation below:

$$P(a) = \Phi\left(\frac{ln\left(\frac{a}{A_m}\right)}{\beta_C}\right) \tag{3}$$

with a composite variability calculated by:

$$\beta_C = \sqrt{{\beta_R}^2 + {\beta_U}^2} \tag{4}$$

The Seismic Fragility Analysis can be represented, in a simplified way and considering the Seismic Demand Analysis activities, by the flowchart of Fig. 5.

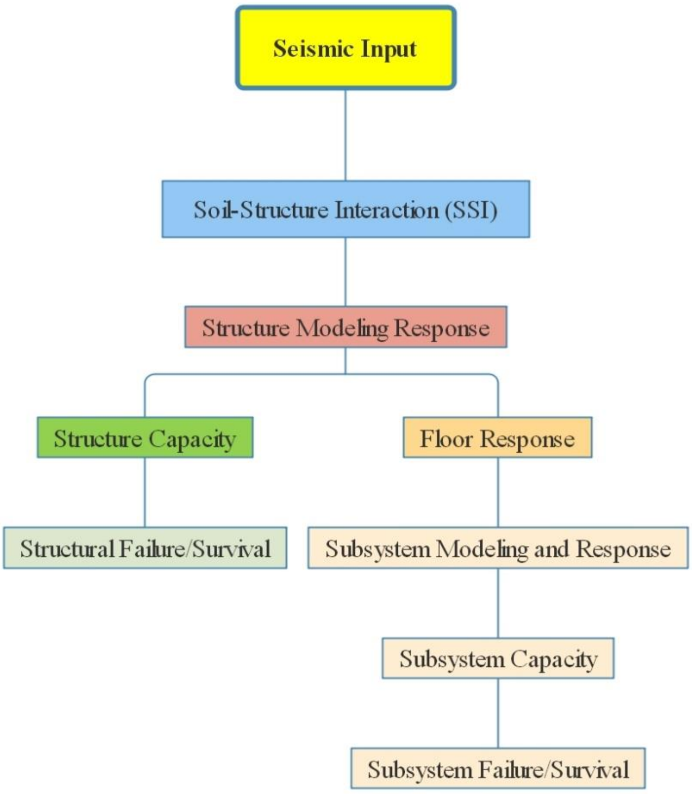

Figure 5. Simplified flowchart of Seismic Fragility Analysis.

One of the objectives of the seismic fragility analysis is to generate a family of curves expressed by fragility parameters. As shown in Fig. 6, the conditional probability of failure P_f at acceleration 0.87g that has a 95% nonexceedance subjective probability (confidence) is obtained from Eq. 2 as 0.79. The 5% to 95% probability (confidence) interval on the failure at 0.6g is 0 to 0.79. A mean fragility curve is obtained using Eq. 3 but replacing β_R with the composite variability β_C using Eq. 4. The median ground acceleration capacity A_m, and its variability estimates β_R and β_U are evaluated by taking into account the safety margins inherent in capacity predictions, response analysis, and equipment qualification [19].

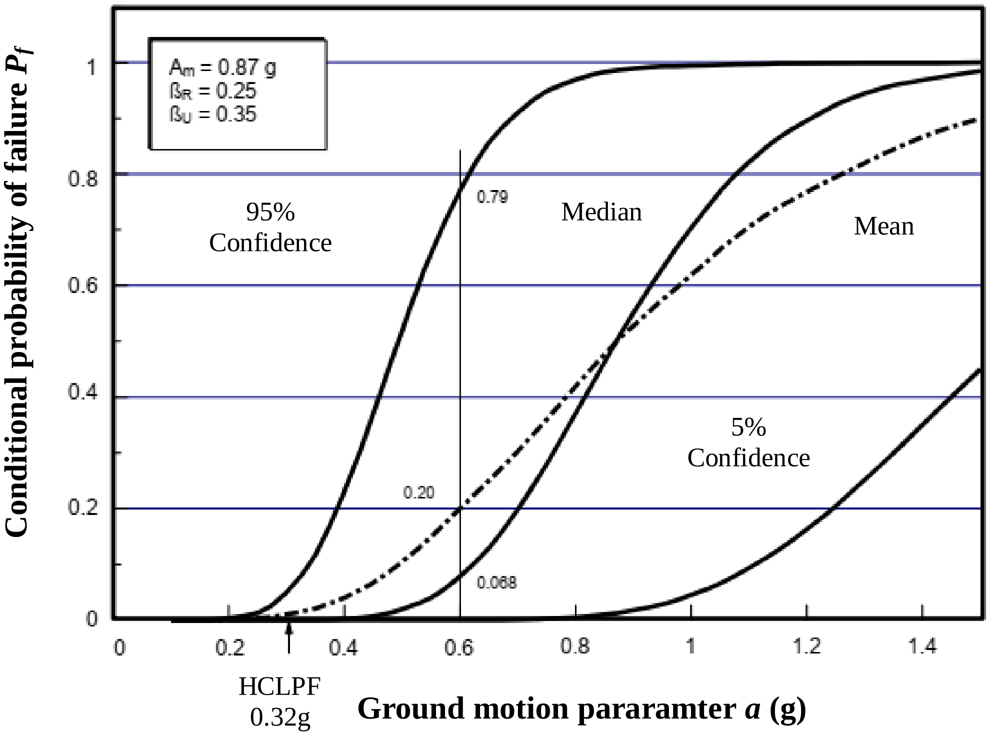

Figure 6. Fragility curves for an SSC with Am= 0.87g, β_R = 0.25, β_U = 0.35 and composite fragility curve, with β_C = 0.43.

Finally, an important parameter in a seismic fragility analysis is the High Confidence of Low Probability of Failure (HCLPF), used as a screening criterion for high seismic capacity components based on review of seismic qualification criteria and screening documents, and walkdown qualification. The HCLPF is defined as the seismic capacity of an SSC expressed in terms of a given ground motion parameter, corresponding to the α% probability value in the mean fragility curve.

5. IMPLEMENTATION OF THE RESULTS OF THE SEISMIC ANALYSES IN THE EXTENDED PLANT PSA MODEL

In general, the Seismic PSA is developed as an extension of Level 1 and/or Level 2 PSA for a nuclear installation, based on the supporting analyses described in sections 2, 3 and 4. Thus, for the extension of a Level 1 PSA, the quantification of the seismic risk associated with a nuclear installation results in a measure of the annual core damage frequency induced by seismic events.

In developing an extended Seismic Level 1 PSA, the following steps shall be followed [3] [6] [20]:

1. Selection of initiating events (IE) relevant for the Seismic PSA – i.e., review of plant safety analyses;
2. Assembly of the Seismic Equipment List (SEL), based on the relevant IE;
3. Assessment of SEL components by seismic capability engineers;
4. Extension of the existing Level 1 PSA model, to include:
    a. dedicated event trees for seismic induced IE;
    b. fragility basic events in the existing fault trees, in order to model the seismic induced failure of those SSCs which are relevant for the mitigation of the seismic induced IE – the fragility analysis of selected SEL components should take into account walkdown observations;
    c. import fragility parameters from the Seismic Fragility Analysis into the PSA model;
    d. assess the impact of seismic events on the Human Reliability Analysis performed for the Level 1 PSA model;
5. Perform the quantification of the seismic induced core damage frequency.

The risk quantification is performed by appropriate integration of the seismic hazard, fragility and the Level 1 PSA model of the plant. Once the hazard curves and fragility curves for a failure event are obtained, the two sets of curves are combined two at a time (i.e. one hazard curve and one fragility curve) to obtain the probability distribution of the unconditional CDF, P_F, as presented in Eq. 5.

$$P_F = \int_0^\infty H_a \frac{dP_{f/a}}{da} da \tag{5}$$

with:

- a: GMP level;
- H_a : the mean annual frequency of the hazard curve with respect to the ground motion variable a; and
- $dP_{f|a}$: Derivative of the conditional probability of failure (fragility curve).

6. CONCLUSIONS

The basic parts of a Seismic PSA for a nuclear installation consist of identifying seismic hazards, analysing the systems (or SSCs), evaluating the seismic fragility of the plant, and performing risk quantification. Each of these four distinct areas requires a good engineering basis and some level of specific training. Nowadays, Seismic PSA is relatively more mature as compared to the PSA for other external hazards. Also, various guidelines are available publically, providing practical methods and covering a broad spectrum of PSA tasks. On the other hand, some basic elements of the Seismic PSA are still analytically sophisticated and require extensive engineering judgement. In order to elaborate appropriate guidelines to extend the PSA for seismic hazards, it is crucial to have the support of a multi-disciplinary team of specialists (seismologists, structural integrity engineers, geologists, etc.) capable of evaluating these hazards.

In this article, the analyses that should be conducted as support for a Seismic PSA were described in a simplified way. In Seismic Hazard Analysis, emphasis was placed on the methodology for the evaluation of epistemic uncertainties, which may represent a great contribution to the structural design of a nuclear plant. It is important to note that the PSHA shall be carried out specifically for the region of interest. The epistemic uncertainties associated with the lack of sufficient historical records compromise the adequate characterization and prediction of seismic events in regions of low seismicity, especially events with higher degrees of intensity. This is the case of the Brazilian territory, which has not experienced seismic events with intensities greater than VII on the MMI scale yet. The Seismic Demand Analysis, which deals with the response of the structure under a load originated from the ground motion, depends on the local geological characteristics. It is recommended that the SSI be reduced and the foundation installed on a competent rock. Next, the FRS of the plant structure is considered the main input data for the design and qualification of the SSCs that will be fixed at each level. This design is directly linked to the Seismic Fragility Analysis of the SSCs, by means of which the probabilities of failure of these SSCs are calculated for a given acceleration of the ground. Fragility parameters are calculated and incorporated into the event tree and fault tree models developed in the Level 1 PSA for the plant. Based on these results, the identification of the SSCs weak links, especially those that perform safety functions, may help in decision making processes regarding plant safety.

It is important to mention that the methodology presented in this article will be used to evaluate the seismic induced CDF of an experimental nuclear installation containing a PWR reactor designed for naval propulsion to be installed in the southeast region of Brazil. In this case, it is important to highlight that a specific Level 1 PSA for internal events and operation at-power has been performed for the plant and it will be taken as the base PSA for the development of the Seismic PSA. Finally, for future research, it would be helpful to develop computational codes that could integrate all seismic support analyses required for the extension of a Level 1 PSA to a Seismic PSA.

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REFERENCES

1. Internacional Agency Energy Atomic – IAEA, "Safety Fundamentals No SF-1 - Fundamental Safety Principles," https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1273_web.pdf (2006).
2. The American Society of Mechanical Engineers – ASME. ASME-ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frenquecy Probabilistic Risk Assessment for Nuclear Power Plant Applications. (2013).
3. J. Prochaska, P. Halada, M. Pellissetti, M. Kumar, "Report 1: Guidance document on practices to model and implement SEISMIC hazards in extended PSA - **Volume 2** (implementation in Level 1 PSA)," http://asampsa.eu/wp-

content/uploads/2014/10/ASAMPSA_E-D50.15-REPORT1-EARTHQUAKEPSAvol-2.pdf (2017).

4. M. Rausand, A. Hoyland, “System Reliability Theory Models and Statistical Methods,” https://notendur.hi.is/hthb1/%C3%81rei%C3%B0anleiki%20kerfa/System%20Reliability%20Theory%20Models%20and%20Statistical%20Methods.pdf (2004).
5. U.S.Geological Survey, “The layer of the Earth we live on is broken into a dozen or so rigid slabs (called tectonic plates by geologists) that are moving relative to one another,” https://pubs.usgs.gov/gip/dynamic/slabs.html, (2011).
6. Wei-Chau Xie et al. *Seismic risk analysis of nuclear power plants*, Cambridge University Press, New York & U.S.A. (2019).
7. A. A. D. Almeida, M. Assumpção. J. J. Bommer, S. Drouet, C. Riccomini, C. L. M. Prates, “Probabilistic seismic hazard analysis for a nuclear power plant site in southeast Brazil”, *Journal of Seismology*, **Volume 23**, Issue 1, pp 1–23 (2019).
8. J. Berrocal et al, Sismicidade do Brasil, Editora Esperança – IAG/USP, São Paulo & Brazil (1984).
9. J. Douglas, “Ground motion prediction equations 1964-2018,” http://www.gmpe.org.uk/gmpereport2014.pdf (2019).
10. J. Ake, “Uncertainties in Probabilistic Seismic Hazard Analyses for Regions of Low-to-Moderate Seismic Potential: The Need for a Structured Approach,” https://www.nrc.gov/docs/ML0808/ML080870337.pdf (2012).
11. Julian J. Bommer et al, “A SSHAC Level 3 Probabilistic Seismic Hazard Analysis for a New-Build Nuclear Site in South Africa”, *Earthquake Spectra*, **Volume 31**, No. 2, pages 661–698, (2015).
12. U.S. Nuclear Regulatory Comission, “Regulatory Guide 1.208 - Performance-based approach to define the site-specific earthquake ground motion,” https://www.nrc.gov/docs/ML0703/ML070310619.pdf (2007).
13. R. J. Budnitz et al, Recommmendations for Probabilistic Seismic Hazard Analyses: Guindance on Uncertainty and Use of Experts,” file:///C:/Users/ELLISON%20AMARO/Downloads/SSHACNUREG-CR-6372-orig.pdf (1997).
14. Jack W. Baker, “Introduction to Probabilistic Seismic Hazard Analysis,” https://web.stanford.edu/~bakerjw/Publications/Baker_(2013)_Intro_to_PSHA_v2.pdf (2013).
15. Global Earthquake Model Foundation, “OpenQuake (OQ),” https://docs.openquake.org/manuals/OpenQuake%20Manual%20%28latest%29.pdf (2019).
16. Ordaz M. and Salgado-Gálvez M.A., “R-CRISIS Validation and Verification Document,” http://www.r-crisis.com/Content/files/R-CRISIS%20V_AND_V%20Document_V1%20Full%20document.pdf (2019).
17. Field, E.H., T.H. Jordan, and C.A. Cornell “OpenSHA: A Developing Community-Modeling Environment for Seismic Hazard Analysis”, Seismological Research Letters, 74, no. 4, p. 406-419. (2003).
18. U.S. Nuclear Regulatory Comission, “10 Code of Federal Regulation - Appendix S to Part 50—Earthquake Engineering Criteria for Nuclear Power Plants,” (2007).
19. Electric Power Research Institute – EPRI. Seismic Fragility Application Guide. Palo Alto: 2002.
20. 17. Electric Power Research Institute – EPRI. Seismic Probabilistic Risk Assessment Implementation Guide. Palo Alto: 2013.