PSA-based seismic margin assessment of a NPP with advanced passive safety features

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SUMMARY:
The ability of the boiling water reactor KERENA to safely handle beyond-design seismic ground motions is analyzed within a seismic margin assessment (SMA) supported by PSA (probabilistic safety analysis) modeling. The performed analysis covers the two major domains of PSA-based SMA, namely i.) fragility analysis of safety-relevant structures and equipment, and ii.) system and accident sequence analysis using the fragility-augmented PSA model (fault trees and event trees). The focus of the fragility analysis is on the civil structures and on the advanced passive safety equipment of the KERENA design, which are designed against standard design spectra (EUR-spectra) with a peak-ground acceleration (PGA) of 0.25g.
The results of the fragility analysis indicate a high robustness of the analyzed structures and equipment to design-exceeding seismic ground motions. The PSA based system analysis also manifests the benefits of the passive safety features, which remain functional irrespectively of the availability of electrical power supply.

Keywords: NPP, Seismic Margin Assessment, SMA, Probabilistic Safety Analysis, PSA, Seismic PSA, Fragility Analysis

1. INTRODUCTION

1.1. Background and motivation

The design of new nuclear power plants (NPPs) for external events includes design provisions for adequate performance in case of seismic events. Adequate performance means that in case of a seismic event the systems, structures and components (SSCs) that are required to fulfill the fundamental safety functions (reactivity control, residual heat removal and activity confinement) must maintain stability, integrity or operability, depending on their role in the safety concept of the plant.
The definition of the seismic accelerations used in the design, depends on the seismic hazard at the site and is provided in the form of design ground response spectra. With reference to the purpose of the seismic design (i.e. to fulfill the fundamental safety functions) these design ground response spectra are referred to as Safe Shutdown Earthquake (SSE).
As in other industries, there are both commercial and technical incentives to apply a standardized design unless counter-indications prevail. This includes the definition of standard design ground response spectra which are supposed to cover a reasonable range of site-characteristic ground response spectra. These standard spectra are referred to as Design Basis Earthquake (DBE) in the present paper. It should be noted that varying definitions of SSE and DBE have been used in the past; the above definitions are consistent with the European Utility Requirements (EUR) for LWR Nuclear Power Plants (2001).
In order to minimize the risk of seismic-induced severe accidents, plant owners are required to supplement the seismic design (based on the SSE definition) by a *seismic risk analysis*. Essentially, seismic risk analysis aims at showing that in case of seismic events that *exceed* the SSE, a disproportionate increase of the radiological consequences is *not to be expected*. This formulation ("*not to be expected*") is intentional and indicates that the demonstration is *explicitly* based on

probabilistic concepts.
There exist two well-established methodologies for providing this demonstration, namely Seismic Margin Assessment (SMA) and Seismic PSA.
In both methods the conservatisms inherent in the seismic design are analyzed in order to determine the maximum level of seismic accelerations, for which the considered SSC (system, structure or component) is still behaving as intended. And in both methods the response of the plant in case of seismic-induced events (i.e. deviations from normal operating conditions, possibly leading to accidents) is analyzed in detail, by considering the interaction of the involved safety systems.
The main difference between SMA and Seismic PSA is the way in which the statement "*risk of core damage due to design-exceeding seismic events is low*" is supported.
In SMA this is done by defining a *specific* design-exceeding seismic event (e.g. an earthquake leading to ground motions that are 40% higher than the SSE). It is then explicitly shown that the risk of a core damage in case of this specific seismic event is *low* and that the confidence of this risk-related statement is *high* (despite unavoidable uncertainties).
In Seismic PSA the statement "*risk is low*" is supported by an explicit evaluation of the seismic-induced core damage frequency (CDF); this risk-indicator can then be compared with the risk resulting from other event types (e.g. random component failures, internal fires etc.) or - more generally - with risk-targets, cf. INSAG-12 by the IAEA (1999). The seismic risk - in terms of the CDF - is not the result of a specific design-exceeding seismic event, as in SMA; instead, the entire range of possible seismic events contributes to the CDF. The underlying analysis requires *probabilistic* site-specific seismic hazard data, notably the hazard curve.
Due to the earthquake-induced accident at the Fukushima-Daiichi plant in 2011, as well as other design-exceeding seismic events (e.g. at Kashiwazaki-Kariwa, Japan, in 2007 and at North-Anna, USA, in 2011), seismic risk assessment is currently attracting increased interest. In Europe, numerous additional safety evaluations („stress tests") have been performed or initiated after the Fukushima accident. One major goal of these stress tests is to evaluate the plant's behaviour in case of design-exceeding seismic events.
The present paper deals with the seismic risk analysis for the KERENA NPP design. KERENA is a boiling water reactor for which the basic design has been completed. In the wake of the Fukushima accident a seismic risk assessment of the KERENA design has been performed within the so-called post-basic-design phase which was dedicated to analysing the basic design with regard to lessons learned from the Fukushima event.
The seismic design of KERENA is a standardized one (see section 1.3) and, consequently, not specifically geared to a particular site. This implies that the analysis must be performed using representative assumptions regarding the seismic hazard instead of site-specific data.
Under these circumstances an SMA is more suitable to assess the seismic risk and has therefore been selected in the case of KERENA. Indeed, Seismic PSA is more demanding in terms of site-specific data, as it requires a hazard curve (see above). Should a site-specific seismic risk analysis of KERENA be required at a later stage, it can be performed on the basis of the generic SMA presented in this paper.

1.2. KERENA boiling water reactor

KERENA (formerly known as SWR 1000) is a medium capacity Generation III+ boiling water reactor (BWR). The evolutionary design of KERENA includes advanced passive safety features. These ensure the fulfillment of the fundamental safety functions independently of active safety systems, thus providing diversity to the active systems for residual heat removal, safety injection and control rod insertion.
Due to the large water inventories available in the core flooding pools and in the shielding/storage pool, the passive safety systems are capable of managing accidents - and in fact also severe accidents - autonomously for 72 hours following event onset, even in total absence of power supply and without any operator action.
The reactor building of KERENA is shown in Fig. 1 and in Fig. 2 the key safety functions of the KERENA BWR are indicated schematically.

Figure 1. Reactor building of the KERENA BWR

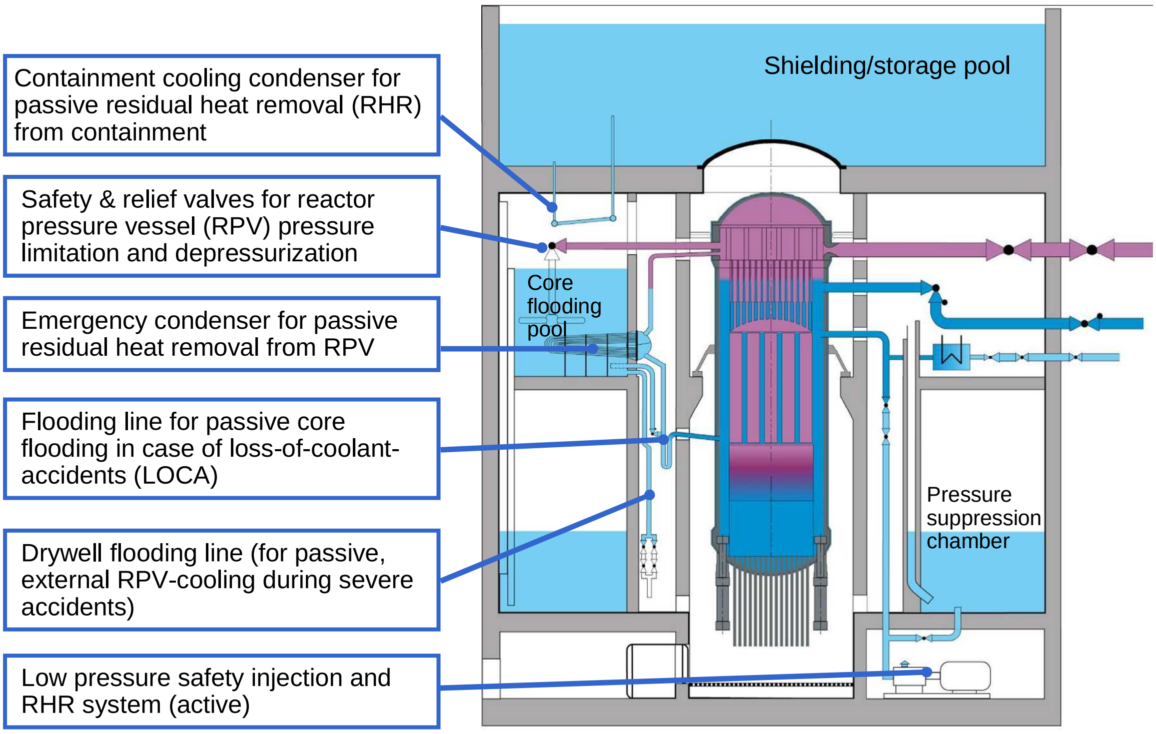

Figure 2. Schematic drawing of the KERENA BWR indicating key safety functions

1.3. Seismic design basis

The design basis earthquake (DBE) applied in the KERENA design is based on a set of standard design ground response spectra issued by European Utilities (2001). These spectra are referred to as EUR-spectra (EUR is the acronym for European Utility Requirements) and are intended to envelope the site-specific ground motions at a majority of potential Western European sites. The EUR-spectra are anchored to a horizontal peak ground acceleration (PGA) of 0.25g. For additional information on the EUR-spectra it is referred to Bommer et al. (2011).
In the dynamic analysis performed for structural and equipment design, each of the EUR-spectra for different soil conditions (soft, medium and hard) is combined with three sets of soil parameters. In the generation of the floor response spectra – for use in equipment design against DBE – a conservative enveloping process has been applied, as indicated in Fig. 3.

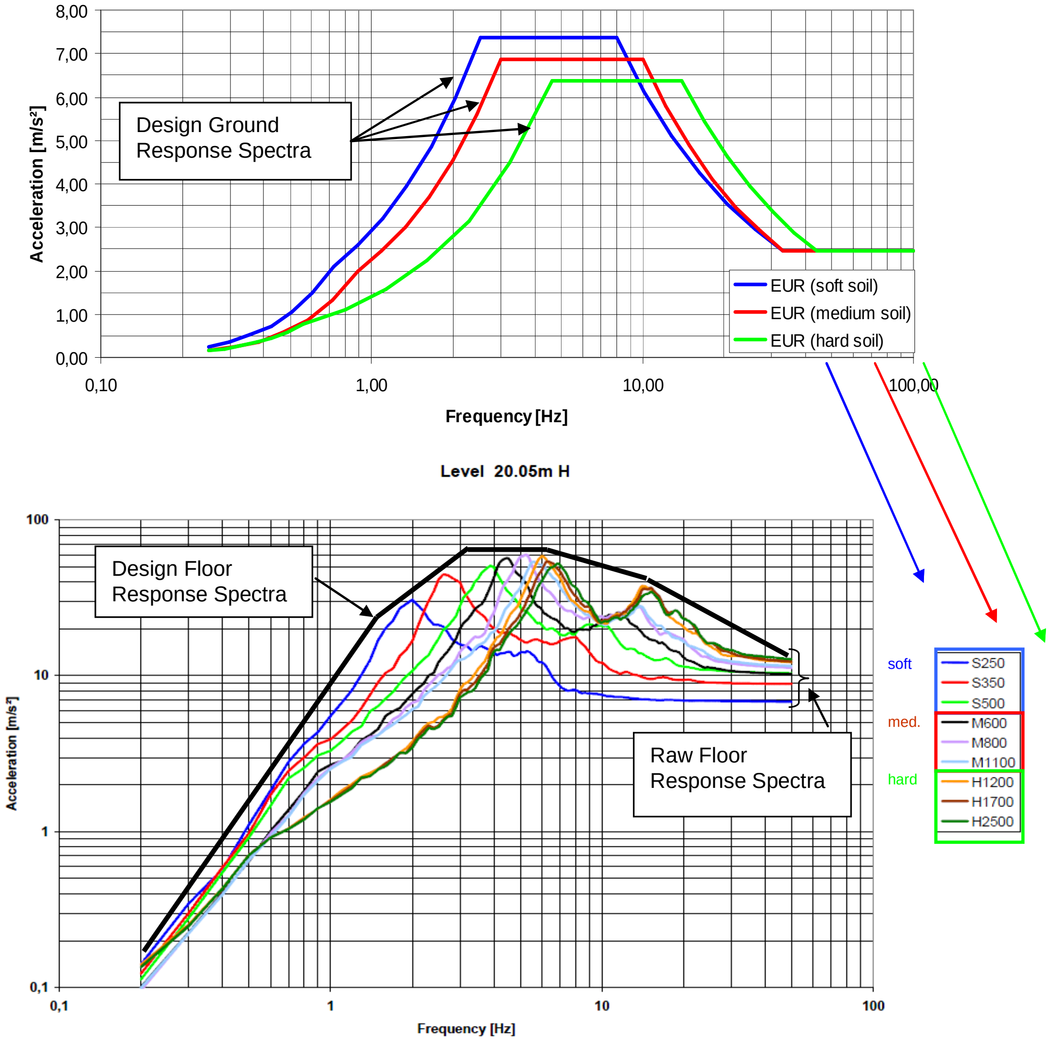

Figure 3. Generation of design floor response spectra in the KERENA seismic design

2. METHODOLOGY

The key elements of a SMA include the following:

- Seismic Hazard Analysis (definition of a site-specific spectral shape and ground conditions)
- Seismic Fragility Evaluation
- Systems/Accident Sequence Analysis
- Seismic Margin Quantification

For the latter two steps (system analysis and seismic margin quantification) there exist two different approaches, which are frequently referred to as EPRI-style SMA and NRC-style SMA.
In the EPRI-style SMA - see the referenced guideline document EPRI NP-6041 (1991) – the system analysis focuses in identifying success paths for which the plant is brought in a safe condition and maintains that condition for at least 72 hours. For each success path, the seismic margin is identified by the seismic capacity of the weakest component required.
In the NRC-style SMA – see the referenced standard ASME/ANS RA-S-2008 for details – the seismic capacity is evaluated on the basis of the PSA-model, by identifying the combinations of seismic and non-seismic failure events leading to core damage.
The discussion of the relative merits of the two approaches is beyond the scope of the present paper. A major reason for adopting the latter approach ("NRC-style SMA") in the present analysis is the availability of a well-maintained and up-to-date PSA model for the KERENA design, cf. Abusharkh and Schmaltz (2012).

2.1. Seismic hazard

One of the first steps in an SMA is the definition of the so-called review-level-earthquake (RLE), i.e. the level of design-exceeding seismic ground motions for which analytic evidence is produced that the risk of core damage is low. The EUR specify a PGA of 40% above the SSE; since for KERENA no site-specific SSE can be defined, a bounding approach consist in applying the 40%-margin to the PGA of the DBE. This implies that the PGA of the RLE is 0.35g in the present analysis.
The other relevant feature of the RLE spectra is the *spectral shape*, resulting from the normalization of the ground response spectrum to the reference PGA. As the seismic design of KERENA is a standardized one, the fragility analysis must be performed using representative assumptions regarding the seismic hazard instead of site-specific data. The approach applied in the Generic Design Assessment (GDA) of the UK EPR^{TM}, i.e. the use of the Uniform Hazard Spectra (UHS) of two candidate sites with enveloping spectra for medium-soil and hard-soil site, respectively (see Thiry et al. (2011)), could not be adopted in the present case.
In an effort to derive a representative assumption for the spectral shape, a literature review of median uniform hazard spectra has been conducted. The pertinent data for the following NPP sites are reported in the following Fig. 4:

- Kori (Korea), cf. Choi et.al. (2008)
- Krško (Slovenia), cf. Uršič et.al. (2005).
- Leibstadt (Switzerland), cf. HSK (2007)
- Loviisa (Finland), cf. Varpasuo (2008)

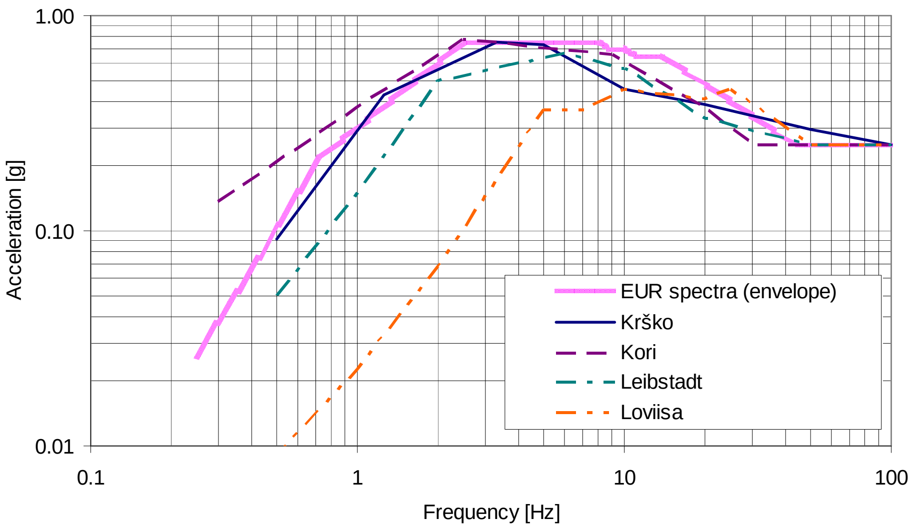

Figure 4. Spectral shapes of median uniform hazard spectra in the literature

Based on the above figure, the spectral shapes of the above sites are enveloped by the EUR-spectra used in the KERENA design with the exception of Loviisa, but not by a very significant amount in the most relevant frequency range for the structural response (i.e. 5 to 10 Hz).
Considering this observation and the objective to quantify the seismic margins of KERENA conservatively, in the present SMA the margin between the spectral shapes of the DBE and the spectral shape of the RLE has *not* been credited. However, the approach of defining two different RLE spectra for the two classes of sites “medium-soil site” and “hard-soil site” has been adopted from the UK EPR™ GDA. The respective spectral shape is then represented by the corresponding EUR spectra (see Figure 3).

2.2. Fragility analysis

In the present paper the log-normal fragility model according to Kennedy and Ravindra (1984) is used,

$$F(a) = \Phi\left(\frac{\ln\left(\frac{a}{\breve{A}}\right) + \beta_U \Phi^{-1}(Q)}{\beta_R}\right) \tag{2.1}$$

In the above equation, a is the peak ground acceleration (PGA), $\breve{A}$ is the median seismic capacity (in terms of PGA), β_U and β_R are the logarithmic standard deviations due to random variability and uncertainty, respectively, Q is the confidence level and Φ is the cumulative distribution function of the standard normal distribution. The fragility F is then the conditional failure probability, in case of a seismic event leading to a PGA equal to a. A key quantity associated with the fragility model is the HCLPF capacity, defined as the value of the maximum ground acceleration for which there is a high confidence Q (95%) that the probability of failure F does not exceed 5% (the acronym HCLPF stands for “High Confidence of Low Probability of Failure”).
In SMA the seismic capacity is expressed exclusively in terms of the HCLPF value. In other words, it is the HCLPF that is compared with the PGA of the RLE (see section 2.1).
The main advantage of the log-normal fragility model is that it facilitates a divide-and-conquer approach. The fragility parameters $\breve{A}$, β_U and β_R are thereby estimated in terms of the conservative bias and the variability introduced in the various steps of the analysis, starting from the surrounding soil, proceeding through the building, finally considering the design of the individual component (“separation-of-variables”).

2.3. System analysis

The system/accident sequence analysis models the combinations of structures, systems and component failures that could initiate a seismic induced accident sequence and lead to core damage. The seismic margin is then calculated by combining the plant accident sequence logic with the component fragilities. The seismic accident sequence model is developed as summarized below:

- Seismic initiating events are determined from the internal events PSA, cf. Abusharkh and Schmaltz (2012). Structures and other passive components that are typically not included in the internal events PSA must also be considered, particularly those that could lead directly to core damage or radioactivity release.
- Seismic event sequence models are developed for each initiating event as appropriate using the internal events PSA models. Some simplifications are applied, considering that seismic events have some particular charateristics (e.g., offsite power recovery is not modeled).
- System analysis fault trees from the internal events PSA are the framework within which the seismic-induced failures are integrated. In particular, passive components – which are typically not considered in non-seismic PSA due to their low non-seismic failure rate - must

be considered in the seismic analysis. The seismic induced failures are included in the fault trees as basic events; these “fragility basic events” will then appear in the minimum cutsets and thus indicate which combinations of seismic induced failures would lead to core damage. Utilization of the internal events PSA fault tree models also ensures that random non-seismic equipment failure probabilities are included in the analysis.

- Human actions in the model are also reviewed and evaluated relative to potential seismic impact on the human reliability. In the SMA evaluation, the operator actions are evaluated qualitatively.

The accident sequence cutsets resulting from the model evaluation are the inputs for the assessment of the seismic robustness of the plant’s safety functions (see section 4).

The SMA evaluation is performed not only for full power operation but also for representative shutdown states, i.e. the various Refueling Outage States (ROS):

- ROS0: Reactor pressurized, RPV closed
- ROS1: Reactor vessel depressurized, RPV closed
- ROS2: RPV opened, flooding of shielding pool not completed
- ROS4: RPV opened, flooding of shielding pool completed

3. FRAGILITY OF INDIVIDUAL STRUCTURES AND EQUIPMENT

3.1. Structures

The fragility analysis was limited to those civil structures of the nuclear island which are safety relevant in the context of the PSA model. Detailed finite element models were used for the evaluation of the seismic margins, which are summarized in the following table:

Table 1.1. Seismic robustness of KERENA buildings (HCLPF values)

	HCLPF capacity	
	Medium soil site	Hard soil site
Containment (UJA) and Reactor Building (UJB)	1.93 g	0.94 g
Unit Control Room Building (UCA)	0.80 g	**0.62 g**
Emergency Diesel and Service Water Pump Building (UBP/UQB)	1.51 g	1.79 g

These values indicate that for the structural capacity the margins - with respect to the target capacity of 0.35g (see section 2.1) - are very significant, especially for the reactor building and the containment.

3.2. Equipment

Numerous equipment items of the KERENA NPP have been analysed for their seismic fragility. However, the focus of the present paper lies on the emergency condensers and the containment cooling condensers, since these components represent key elements of the safety concept of KERENA, based on passive residual heat removal. In addition, the seismic fragility of the reactor pressure vessel (RPV) is discussed.

3.2.1. Emergency Condenser

The emergency condenser is a key component in the passive residual heat removal (RHR) concept of KERENA. It transfers the heat from the RPV to the core flooding pools inside the Containment. The finite element model used for the fragility analysis of the component is shown in the following Fig. 5.

The emergency condenser design indicates a particular high seismic ruggedness. The HCLPF capacity is 6.4g for a medium-soil site and 5.9g for a hard-soil site.

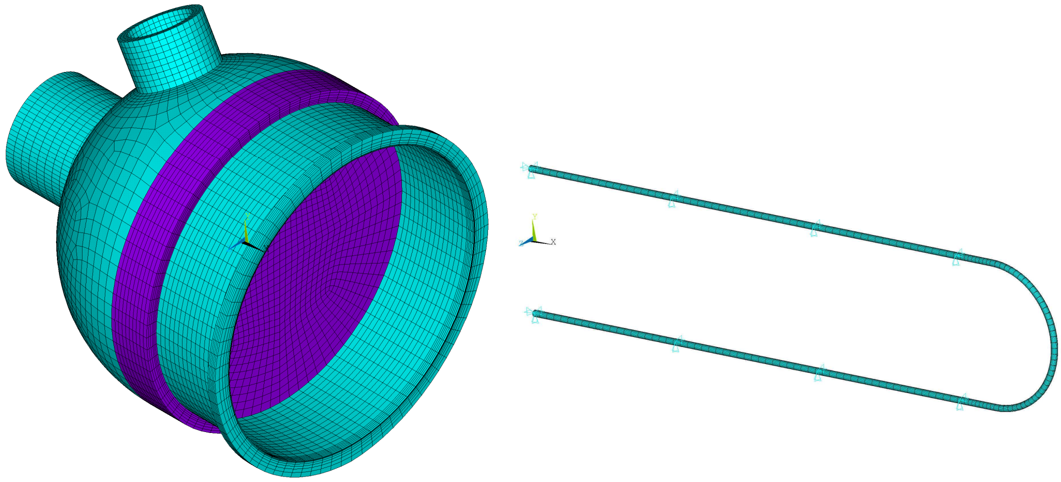

Figure 5. Finite element model of emergency condenser wall (left) and tube (right)

3.2.2. Containment Cooling Condenser

The containment cooling condenser ensures the transfer of the heat from the containment to the shielding/storage pool. The pertinent finite element model is shown in the following figure.

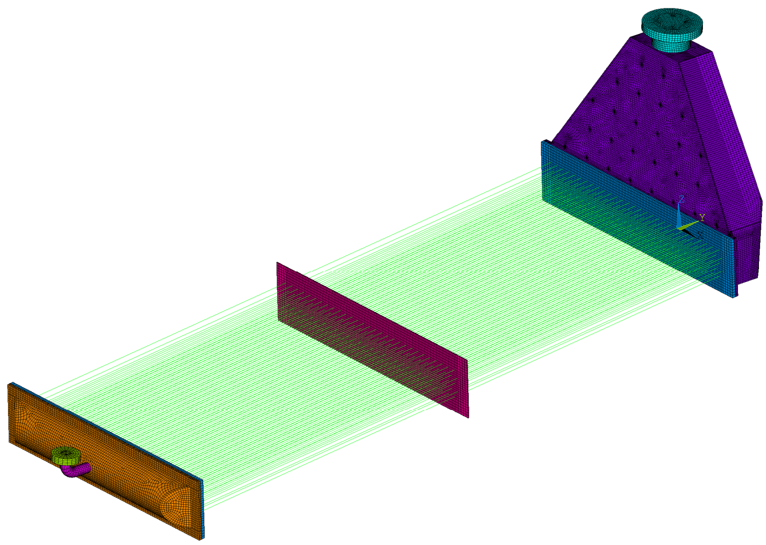

Figure 6. Finite element model of the containment cooling condenser

The containment cooling condenser design has a HCLPF capacity of 0.55g for a medium-soil site and 0.48g for a hard-soil site.

3.2.3. Reactor Pressure Vessel

The RPV design has a HCLPF capacity of 0.61g for a medium-soil site and 0.59g for a hard-soil site. These values are governed by the capacity of the RPV support skirt. The seismic capacity of the internals is higher than that and exceeds 0.95g for a medium-soil site and 0.88g for a hard-soil site.

3.2.4. Piping of the Reactor Coolant Pressure Boundary

The analysis encompassed the piping of the feedwater system, the main steam system, the emergency condenser system, the steam relief system, the scram system and the fast boron injection system. For these systems detailed stress analyses are available from the basic design phase of the KERENA

development, which could be used for the estimation of the seismic margins.
The results indicate that the seismic capacity of the most important KERENA piping systems is very high (≥0.71g).

4. PLANT ROBUSTNESS IN CASE OF SEISMICALLY-INDUCED ACCIDENTS

In the previous section, the fragility of individual structures and equipment was discussed. Clearly, the seismic induced failure of a single component is generally not going to lead to core damage. Using the event tree models of the potential accident sequences and fault tree models for modeling the failure of plant's safety functions (e.g. safety injection), it is possible to identify the combinations of component failures which lead to a core damage. These combinations are called minimum cutsets (MCS).

4.1. Power states

The system analysis has been conducted for the following seismic-induced initiating events during power operation:

- Loss of Offsite Power (LOOP)
- Loss of Feedwater (LFW) and Loss of Main Heat Sink (LMHS)
- Anticipated Transient Without SCRAM (ATWS)
- Small Loss of Cooling Accident (LOCA)

The following compilation summarizes the most important combinations of (seismic-induced) safety function failure that would lead to core damage. This compilation is based on the Minimum-Cutset analysis of the above listed event trees.

- Event Tree: LOOP (same combinations lead to core damage in the event tree LFW+LMHS)
    - Failure of Active RHR + Failure of S/R Valves (uncontrolled depressurization)
    - Failure of Active RHR + Failure of Containment Cooling Condensers
- Event Tree: ATWS (Anticipated Transient Without SCRAM)
    - Failure of SCRAM tanks + Failure of Motor Insertion of Control Rods
    - Failure of SCRAM + Failure of 4/8 S/R-Valves
- Event Tree: Small LOCA (Loss of Cooling Accident)
    - Failure of Passive Core Flooding + Failure of Active Injection
    - Failure of Drywell Flooding (inventory lost) + Failure of Active Injection
    - Total Failure of Depressurization (e.g. all S/R-Valves)

The above compilation shows that in all considered event scenarios the seismic induced loss of an active safety function only leads to core damage, if its passive – and hence diverse – counterpart also fails.

4.2. Shutdown states

Based on the system analysis for low-power (shutdown) operation, the most critical operational state is the one in which the RPV is open without complete flooding of the shielding/storage pool (ROS2). In this case there are the following single-fragility cutsets:

- Unit control room building (UCA)
- Cable trays
- Safety chilled water system

Regarding cable trays it is pointed out that the modeling of their seismic failure within the fragility-augmented PSA model is quite simplified, also in view of obvious limitations on the availability of information on cable routing. More specifically, it is assumed that all cable trays fail at once.
This assumption of full correlation within distribution systems (such as piping, ducting and cable routing) is frequently adopted in order to err on the safe side. The conservatism associated with this assumption has to be analyzed case-by-case, if necessary. In recent case studies it has been found by Pellissetti and Klapp (2011) that even a moderate relaxation of this assumption results in significantly

reduced risk of seismic induced failure of the associated support function (e.g. power supply, in the case of cable routing).
Considering that it is the purpose of SMA to assess the seismic risk, it is important to note that the duration of the shutdown state ROS2 is actually very short. Hence the risk that a design-exceeding seismic event occurs during this ROS is orders of magnitude smaller than the risk that a design-exceeding event occurs at all. While in the internal event PSA – and in the seismic PSA – the duration of the various states is taken <u>explicitly</u> into account through the frequency of the initiating event, this is not the case in SMA.

5. CONCLUSIONS

Based on the fragility analysis of selected SSCs and the system analysis based on an up-to-date PSA model, the following conclusions can be reached regarding the response of the KERENA BWR to beyond-design seismic events:

- The seismic margin in the analyzed civil structures is large, compared to the EUR-based target capacity of 0.35g; this holds especially for the reactor building and the containment (HCLPF ≥ 0.94 g).
- For the analyzed passive mechanical equipment including those which are characteristic of the KERENA design (emergency condensers and containment cooling condensers) abundant seismic margins were found by means of numerical calculations (HCLPF ≥ 0.55g for medium-soil sites, HCLPF ≥ 0.48g for hard-soil sites).
- The PSA-based cutset analysis indicated good seismic robustness also from the system perspective. This is reflected by the fact that there are no single-fragility cutsets leading to core damage during power operation.
- The only single-fragility cutsets are associated with a shutdown state (open RPV without complete flooding of the shielding/storage pool). However, the contribution of this event scenario to the total risk of seismic-induced core damage is judged to be low, because of the very short duration of this outage state.

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