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EVALUATION OF SEISMIC ROBUSTNESS USING PLANT HCLPF CAPACITY

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INTRODUCTION

After the Fukushima accident, stress tests were performed both on national and European level in Germany. One of the subjects in these stress tests is the robustness of the plant against earthquakes with at least one intensity beyond the Design Base Earthquake's (DBE): This paper demonstrates the performed analysis to clarify the extended intensity beyond DBE, at which the vital safety functions of the plant are still assured.

First, the characteristics of the DBE for the plant site will be presented.

Next, the plant specific Seismic Probabilistic Assessment (SPSA) will be described which is used to verify the plants robustness against higher earthquakes levels. For the SPSA, the calculation of core damage frequency is a result of the convolution of the seismic hazard with the fragility functions over the event trees and fault trees where the structures, systems, and components (SSCs) are included. The end results are point estimates of the end metrics of interest.

These end results are used to develop the plant level fragility curve which depicts the conditional probability of seismic core damage for different levels of ground motion input. By entering the plant level fragility curve corresponding to 95% confidence at 0.05 conditional probability of failure, the plant HCLPF capacity can be obtained as the respective robustness criterion.

WORDING OF THE RECOMMENDATIONS

It is stated in RSC (2011) that from the point of view of the RSC, German nuclear power plants have demonstrated significant design reserves for beyond-design-basis earthquakes. To protect this assessment, the RSC recommendation of September 2012 suggested:

- For plants where results of probabilistic seismic safety analyses are available, the robustness must be assessed against the beyond-design-basis earthquake effects. The assessment should be based on the HCLPF values in order to guarantee the vital safety functions of the necessary structures and equipment.
- For plants where no results from probabilistic seismic safety analyses are available, the method can be chosen. A transferability consideration (potentially supported by a plant inspection by an expert commission) based on results in accordance with a) an assessment of the robustness against beyond-design-basis earthquake effects is to be carried out.

With regard to a definition of robustness against a beyond-design-basis earthquake, the assessment criterion using in RSC (2011) is that

- The vital functions to ensure compliance with the protection objectives are ensured.
- Effective emergency measures can also be considered.

DESIGN-BASIS EARTHQUAKE CHARACTERIZATION

A maximum free-field acceleration of 170 cm/s² (maximum potential earthquake or safe shutdown earthquake) was defined for the design of the plant approval procedure against earthquakes.

The specification was made on the basis of a deterministic seismic hazard analysis, taking into account the special requirements for nuclear power plants regarding security against earthquake effects. This lead to an earthquake on site with an intensity of I = VIII (MSK), which will subsequently take on the written form of I = 8. This intensity was assigned to the maximum ground acceleration of 170 cm/s². Dynamic calculation models that use the response spectra method were used for the earthquake design of the block. The spectral course of the ground response spectrum corresponds to the USAEC spectrum Schneider (1975).

Since the licencing procedure, the seismic hazard at the site has been repeatedly examined and assessed by independent expert groups using different methods. These inspections revealed no indications of deficits in seismic load assumptions. No change was made for the seismic load assumptions.

The design of the systems, plant parts and construction works against earthquakes was carried out according to classification I or IIa components, whereby the consistent maximum ground acceleration of 170 cm/s² was taken as a basis.

SPECIFICATION OF ROBUSTNESS REQUIREMENTS FOR EARTHQUAKE LOAD CASE

In principle, the design-basis earthquake, which is the basis for the original design, can be used as the reference base for the assessment of robustness. Furthermore, the design-basis earthquake required to comply with NSSC KTA (2012) with a maximum exceedance probability of 10^{-5}/a can also be used as a reference base. In a statement in May 2011 on the plant-specific safety reviews of German nuclear power plants RSC (2011), the RSC defined the earthquake intensity levels as the reference value for robustness in the event of an earthquake. The proof of robustness is to be obtained by increasing the earthquake intensity by one or two steps based on the intensity level, which equates to a maximum exceedance probability of 10^{-5}/a.

Figure 1 shows the exceedance probability of an earthquake on site as well as the maximum free-field acceleration based on realistic free-field response spectra depending on the intensity level. It is evident from this Figure that the exceedance probability decreases sharply as the intensity level increases. The exceedance probability of 10^{-5}/a equates to an intensity level of 7.1, which can be assigned a maximum free-field acceleration of 85 cm/s² based on the realistic free-field response spectra. Already, the increase of one intensity level leads to a decrease in the exceedance probability by almost 3 to the power of 10. The exceedance probability with regard to the intensity levels decreases to such an extent that an increase of two steps over the 10^{-5}/a intensity can no longer be presented within Figure 1. If the originally determined design-basis earthquakes (I=8) were assessed according to today's point of view, the exceedance probability would be significantly below the required maximum exceedance probability for design-basis earthquake of 10^{-5}/a according to NSSC KTA (2012).

For the reasons stated above, the exceedance probability or the associated maximum free-field acceleration will subsequently be used as an alternative reference value for the assessment of robustness against earthquake load case. The exceedance probability as an assessment scale corresponds to the general concept according to the NSSC understanding of the safety philosophy RSC (2013), according to which the categorization of events into safety levels takes place according to frequency of occurrence.

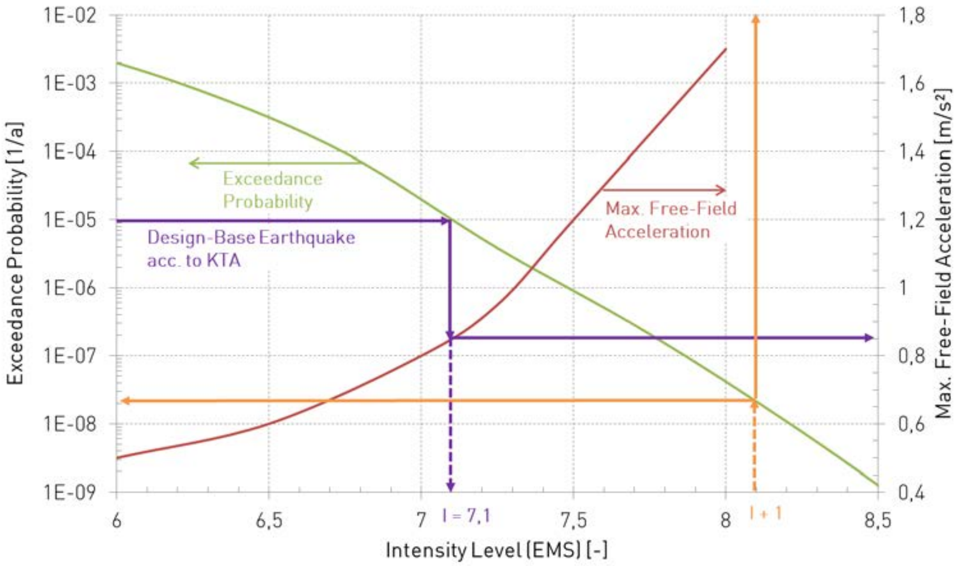

Figure 1: Exceedance Probability and maximum Free-field Acceleration depending on Intensity Level at plant site

EARTHQUAKE ACCIDENT CONTROL CONCEPT

Relevant to the following considerations are those safety features which are actually required and which could be endangered by the earthquake itself or by the associated consequential damages.

In order to ensure these safety features, the buildings in which the corresponding systems or system sections are located must be designed to be earthquake-proof. Additionally, those buildings are designed to a high standard, the failure of which could cause inadmissible consequential damage or compromise the boundary conditions of the radiological incident calculations for earthquake event. According to the protection concept for the mitigation of external influences KWU (1982) the following essential buildings are designed to withstand design-basis earthquakes:

- Switchgear building
- Emergency power generating and central water chiller building
- Cable bridges, Switchgear building/reactor building
- Cable- and pipe duct Switchgear building/emergency power generating and central water chiller building
- Reactor building interior (Containment)
- Reactor building – annulus
- Main steam – and feedwater valve compartment
- Equipment air lock enclosure
- Emergency feed building
- Pipe and cable ducts Emergency feed building/ reactor building
- Cell cooling tower structures
- Nuclear and emergency service water supply (intake structures, pumps, piping)
- Reactor auxiliary building
- Vent stack (to avoid exhaust stack debris)

The mitigation concept against comprehensive errors in actively engineered safety features essentially consists of spatial separation of redundant subsystems and corresponding structural protection. In addition to these precautions pertaining to engineered safety features, there are further measures which prevent or limit the occurrence or spreading of incidents which cause ripple effects. This primarily involves passive measures that were implemented during building design (for example, creation of fire zones and earthquake design for all important safety-related buildings). However, the design or spatial arrangement of procedural systems also serves to protect against system-wide accidents (for example, reactor coolant system earthquake design, flood-proof arrangement of systems within containment). Finally, there is particularly active equipment that can be used to avoid and mitigate system-wide accidents (for example, fire detection and firefighting equipment).

ASSESSMENT OF DESIGN RESERVES

In principle, the Seismic Probabilistic Safety Analysis (SPSA) or the deterministic Seismic Margin Assessment (SMA) are available as methods to determine reserves in the seismic design. An SPSA provides information on the earthquake-induced contribution towards the overall core damage frequency and the contributing risks associated with earthquake effects. A less complex alternative to the SPSA is the deterministic Seismic Margin Assessment. Table 1 shows a comparison of the essential features of the SMA and SPSA. A detailed description of both methods is given in IAEA (2009).

Table 1: Comparison of SMA and SPSA essential features (according to IAEA (2009))

	SMA	**SPSA**
Earthquake level to be considered	RLE determined from deterministic or probabilistic procedure RLE: review level earthquake	Probabilistic seismic hazard assessment
Plant/ System models	Success path(s)	Event trees / fault trees
Seismic response analysis	Deterministic or probabilistic best-estimate	
	For the RLE	For a further range of earthquake ground motion
Capacity or fragility assessment	HCLPF	Fragility functions — probability of failure as a function of earthquake level
Quantification or end metrics	Deterministic calculation of SSC and plant HCLPF	Probabilistic calculation of core damage frequency ➔ Basis for calculation of a „Plant Level Fragility“ (also see EPRI (2002))

In principle, both methods ultimately enable the quantification of seismic robustness via the servicing of a plant HCLPF. The HCLPF value (High Confidence of Low Probability of Failure) is defined as the maximum free-field acceleration at which the failure probability of the equipment with a confidence level of 95% is less than 0.05.

The basis for a probabilistic analysis is the failure curve (distribution function) to be determined for the respective SSC, which indicates the probability for the earthquake-induced failure of the SSC depending on the macroseismic parameter (max. free-field acceleration). A graphic representation is given with

Figure 2, the meaning of the HCLPF value is also included. A detailed explanation can be found in EPRI (2002) and BfS (2005). The HCLPF value is therefore a reliable measurement of the seismic capacity against the design basis earthquake.

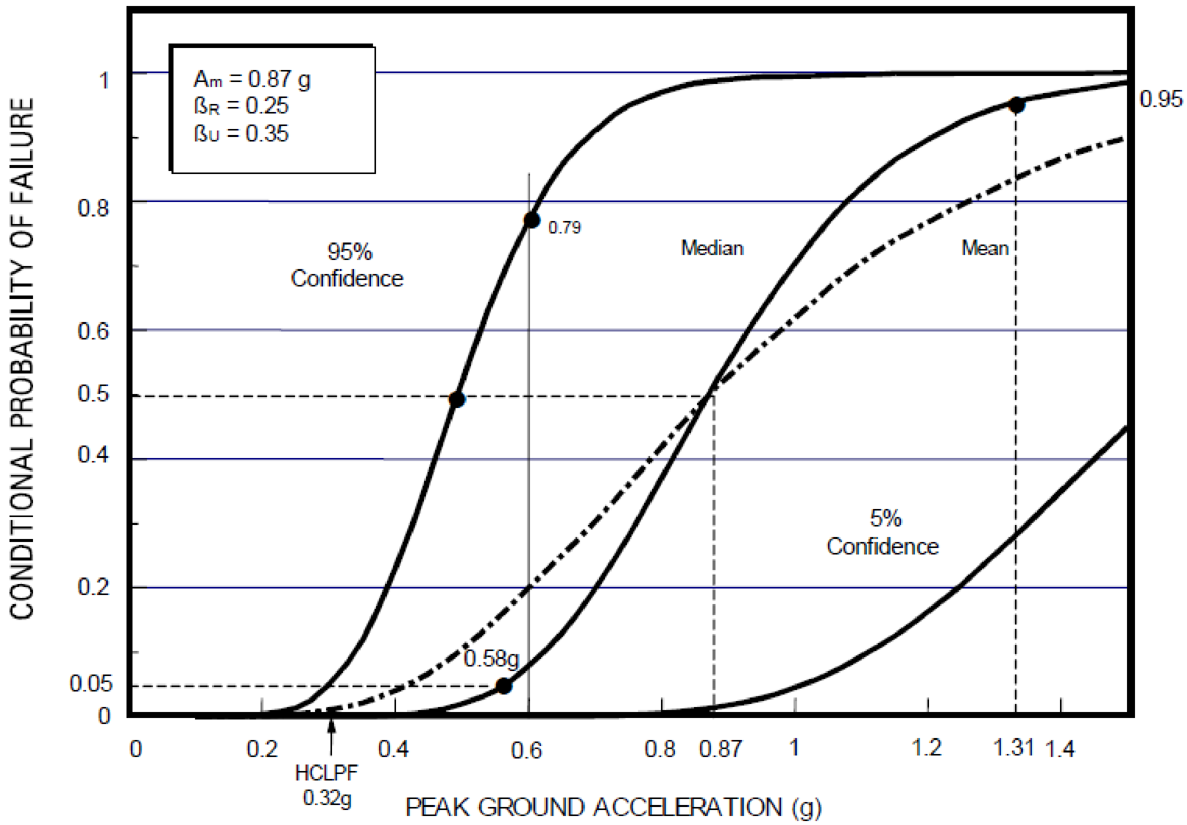

Figure 2: Failure curves for a hypothetical component EPRI (2002)

In the case of an SPSA, however, it must be taken into account that a HCLPF value determined from the so-called "Plant Level Fragility" (PLF) is based on a detailed modeling of the overall plant behavior in the event of an earthquake EPRI (2002). The PLF reflects the conditional probability of an earthquake-induced core damage condition subject to the seismic momentum used (e.g. PGA value). An equivalent HCLPF value can be derived by using the PLF for a corresponding confidence level of 95% with a conditional core damage frequency of 0.05.

In contrast, the plant HCLPF as end result of a deterministic SMA represents the weakest link within the strongest success path. If reaching a safe state is dominated by a single component, similar results are obtained using both methods. However, if only the combined failure of several components results in failure to reach a safe state, the HCLPF values from both methods can differ considerably from one another. Since the SPSA results are significantly more conclusive in the latter case, and these results are also available for a NPP of KONVOI-type, only the SPSA methodology and the resulting statements regarding seismic robustness will be outlined in the following sections.

Seismic Probabilistic Safety Analysis Methodology

The key elements of the SPSA are:

- Seismic risk analysis:
  Calculation of the annual exceedance probability of earthquake intensities and realistic site-specific free-field response spectra for different earthquake intensities.
- List of structures, systems and components to be assessed with respect to earthquakes
  Whose earthquake-induced failure can prevent safe plant shutdown with residual heat removal, can lead to core damage and which are so susceptible to earthquake-induced failure that they contribute to the frequency of hazard or core damage conditions due to earthquakes.
- Determination of the earthquake –induced failure probabilities
  For the equipment whose failure can lead to hazard or core damage conditions.
- System analysis
  With modeling of the different combinations of failures of equipment and personnel actions that can trigger an earthquake-induced course of events with hazard or core damage conditions.
- Quantification of the frequency of events with hazards or core damage conditions
  Amalgamation of earthquake risk analysis results, earthquake-induced failure probability analysis and the system analysis to determine the frequencies of hazard and core damage conditions, their result uncertainties and different degrees of importance.

The seismic capacity for all safety-related SSCs is assessed in the SPSA. The assessment is carried out either by means of generic considerations or by means of a detailed identification of the individual safety factor which are combined in the overall safety factor or in the so-called safety reserve factor of the equipment under consideration. The safety reserve factor together with uncertainty parameters β_R and β_U provide the calculation parameters for the specification of the failure curve of the respective equipment, i.e. the conditional failure probability at a specific maximum free-field acceleration. Within the course of this process, a sufficient robustness is determined for the SSC whose earthquake-induced failure can be neglected by further system analyses.

For all other SSCs, the associated failure curves are to be linked with the plant system logic (event flow diagrams and error trees of PSA models). This makes it possible to assess the design reserves of the entire system with regard to a hazard condition (HC) or a core damage condition (CDC) (after taking emergency measures into account).

Assessment of HCLPF values within the SPSA framework

The SSC to be assessed concerning earthquakes whose earthquake-induced failure prevents safe plant shutdown with residual heat removal and can lead to core damage, and that are so susceptible to earthquake-induced failure that they contribute to the frequency of hazard or core damage conditions due to earthquake are differentiated into:

- SSC whose earthquake-induced failure leads to a direct accident (DA),
- SSC whose earthquake-induced failure leads to an initiating event (IE) or
- SSC that have a safety-related function and are required for the mitigation of an accident (SR)

An initial so-called screening process is carried out with regard to the consideration of SSC. For this purpose, a corresponding HCLPF value is determined as a comparative value for a sorting level, this being 0.3g for components and non-safety-related structures or 0.5g for safety-related structures. SSCs which have a larger HCLPF value and thus have a minimum robustness can be sorted and exclude from further consideration in the PSA model.

For SSCs that could not be sorted during the screening process, corresponding failure curves have been determined within the framework of the SPSA for a NPP of KONVOI-type which result in the respective HCLPF value for the SSC under consideration.

When evaluating the overall system robustness, it must be taken into account that the earthquake-induced failure of a single component (with the exception of the SSC which lead to a direct accident) does not directly lead to core damage due to the general design philosophy. As a result, the overall plant robustness is still significantly above the HCLPF values of these individual components. The assessment of the overall plant robustness versus the design basis earthquake is given in the following section.

Assessment of seismic PSA results

In line with the second supervisory authority safety review, a Seismic Probabilistic Safety Analysis (SPSA) was submitted for a NPP of KONVOI-type.

With this SPSA the earthquake capability of the plant facilities was reviewed completely by an international team of experienced engineers in the field of earthquake stress using combined on-site inspections and documentation reviews. At the same time, the facilities were reviewed with regard to the design and capability procedures used for earthquakes and the earthquake induction of internal fire and flood events with particular consideration towards earthquake-induced interactions.

Characteristic for individual site-specific key elements of the implemented SPSA is:

The seismic hazard analysis shows, among other things, that the realistic, site specific free-field response spectra for all annual exceedance probabilities up to and including 10-7/a are covered by the design basis earthquake spectrum.

The development of the list of SSCs to be assessed with regard to earthquakes ultimately concentrates on only a small number of devices with significant earthquake-induced accident potential. In a first list, devices with potential to prevent earthquake-induced core damage are determined by means of documents (3.409). However, this list is then concentrated on a few devices by a staggered, engineer-related selection procedure, and their failure probabilities are to be determined. Thus, the first step in this selection procedure already leads to a reduced list of 646 devices with earthquake-induced accident potential due to system-wide established guidelines, which come from the plant inspection that was carried out according to international regulations. This further reduces the list to 78 devices, of which another 48 entries can be sorted due to re-examination into sever quake behavior. Therefore, only 37 devices remain, for which an analysis of the failure probabilities is made according to the internationally regulated safety reserve method.

The quantification covers the earthquake-induced frequencies of hazard and core damage conditions. Such a condition is present if the earthquake-induced hazard to the reactor core is already specified before or after the implementation of internal emergency protection measures. The quantification shows extremely low frequencies in both cases. The expected value of overall frequency of earthquake-induced hazard conditions amounts to 5.8×10^{-9}/a and that of core damage conditions is approximately 9.3×10^{-10}/a. As a result, these overall frequencies are smaller by more than two orders of magnitude compared to those of the PSA for internal events and thus contribute marginally to the overall frequencies of hazard and core damage conditions. Additionally, the uncertainty analysis shows that even the 95% confidence boundaries of the earthquake-induced frequencies of hazard and core damage conditions only contribute marginally to the overall frequencies of hazard and core damage conditions, which also applies to the very pessimistic cases treated in the sensitivity analysis.

In an additional step the results of the SPSA have been processed further to develop a so-called "Plant Level Fragility" (PLF) pursuant to EPRI (2002). The PLF reproduces the conditional probability of an earthquake-induced core damage condition (system unavailability (CDC)) depending on the seismic design values used (e.g. PGA value) and was determined for the considered earthquake magnitudes by separating the occurrence frequency of the earthquake magnitude from the frequency of core damage condition.

Figure 3 shows the PLF for a NPP of KONVOI-type including the 90% confidence interval (5%/95%) resulting from the performed uncertainty analyses. For the sake of clarity, the acceleration values were plotted against the PGA value for an earthquake with an exceedance probability of 10^{-5}/a. By means of relevant evaluations of the available results, the proportions of random or earthquake-induced failure could also be reported separately and can contribute particularly to the explanation of the curve.

For a better connection between the earthquake hazard degree to the PLF, additionally the figure shows the exceedance probability of the earthquake.

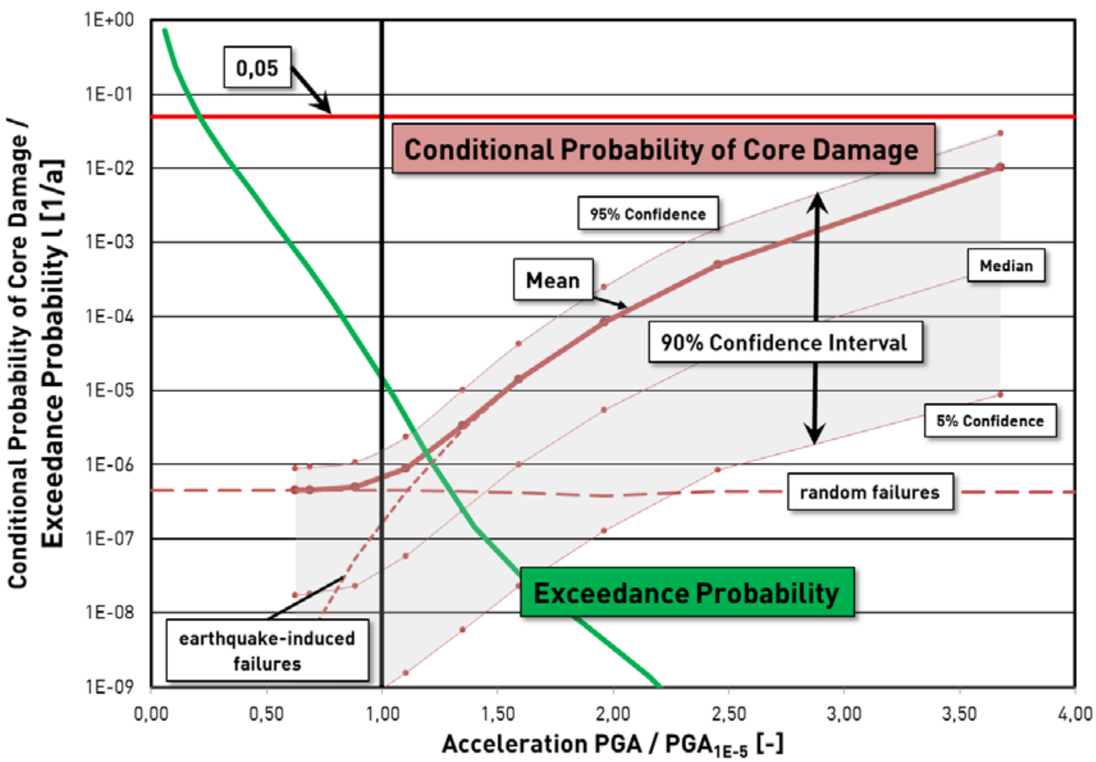

Figure 3: "Plant Level Fragility" for a NPP of KONVOI-type

Overall, the SPSA resulted in a very small earthquake-induced core damage frequency of less than 10^{-9}/a. The investigated earthquake magnitudes or accelerations were initially extended to the design range of the plant, but were extended to a PGA value of 0.3 g for the additional evaluations. The earthquake-induced plant shutdowns have the main share in the result as far as the design range of the system, without additional earthquake failure of devices, which mainly stems from random failures in the system engineering that are independent of earthquakes. The earthquake-induced failure of equipment designed specifically for earthquakes only contributes marginally to the earthquake-induced core damage frequency. The same applies to the earthquake-induced failure of devices that are not designed to withstand the design basis earthquake. It could been shown that no dominant relative contributions result from the earthquake-induced failure of structures, systems and components. Thus, even in earthquake events with very low exceedance probability, no cliff edge effects can be found.

Extrapolations on the basis of the available PSA model beyond the design range show an increasing relevance of results of earthquake-induced device failure as expected, wherein firstly earthquake-induced failures of devices that have not been designed to withstand the design basis earthquake are determined by the result.

The failure probabilities calculated by means of the existing PSA model are very small at least until the increase of earthquake strength to three orders of magnitude above the earthquake magnitude; consequently, the conditional probability of an overall plant earthquake-induced core damage condition remains small even in this area of exceedance probability, which is why cliff edge effects are not to be expected here.

To summarize, based on the SPSA for a NPP of KONVOI-type it can be shown that in terms of design, the investigated NPP has a high earthquake capacity with regard to the potential earthquake events in the plant environment.

CONCLUSION

In this paper, the robustness of a NPP of KONVOI-type for earthquake load cases was demonstrated, with regards to RSC recommendations RSC (2012), in which the following aspects were considered:

- Design basis earthquake conservatism
- Seismic probabilistic safety analysis results (SPSA)

This paper has shown that both in terms of the estimated seismic loads and in terms of the seismic capacities based on the conservative design, the plant can be considered as robust against an earthquake with an exceedance probability of 10^{-5}/a.

The robust design is firstly underlined by the results of the SPSA with a very small earthquake-induced core damage frequency of less than 10^{-9}/a. In accordance with the procedure recommended by the IAEA in IAEA (2009), a corresponding evaluation of the results of the SPSA was carried out which, with reference to the earthquake (10^{-5}/a) to be used for comparison, shows a clear robustness, which is also reflected in the determined "Plant Level Fragility".

Irrespective of the considerations presented, the plant has design reserves against the plant-specific earthquakes which are determined according to science and technology (basis: exceedance probability 10^{-5}/a) in such a way that the vital functions for observing the protection objectives are ensured even in case of an increased intensity level.

The assessment intensity of I = VIII, which is based on the original design, contains a reserve according to today's assessment criteria. This is based on the fact that, according to the current design procedure KTA (2012), only the assessment intensity of I = VII is determined for the required annual exceedance probability of 10^{-5}/a. This means that the plant was initially designed for a higher intensity (intensity VIII) than would be required by today's assessment criteria (intensity VII).

If the intensity is increased by one step to intensity VIII, the load level for which the system was originally designed is reached. An earthquake of this strength is therefore mitigated in accordance with design principles and thus there are no restrictions on the relevant infrastructure, the safety-related systems and vital functions.

REFERENCES

BfS (2005), „Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke“, Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR-37/05, August 2005

EPRI (2002), “Seismic Fragility Application Guide”, *Electric Power Research Institute*, Palo Alto, California, EPRI 1002988

IAEA (2009), "Safety Guide NS-G-2.13: Evaluation of Seismic Safety for Existing Nuclear Installations", 2009

KTA (2012), "KTA 2201, Design of nuclear power plants against seismic events, Parts 1-6", 2012

KWU (1982), “Schutzkonzept zur Beherrschung von Einwirkungen von außen”, VRS1/154c/80, Siemens KWU Offenbach 1982

German Reactor Safety Commission (RSC) (2011). “Summarising assessment and recommendations of the STATEMENT - Plant-specific safety review (RSK-SÜ) of German nuclear power plants in the light of the events in Fukushima-1 (Japan) [from the 437th RSK meeting from 11 to 14 May 2011]“.

German Reactor Safety Commission (RSC) (2012). “RECOMMENDATION - RSK’s understanding of safety philosophy (published in the Federal Gazette, Official Section, on 05.12.2013, B4) [from the 460th RSK meeting 29 August 2013]“.

German Reactor Safety Commission (RSC) (2013). “STATEMENT - Recommendations of the RSK on the robustness of the German nuclear power plants [from the 450th RSK meeting from 26 to 27 September 2012]“.

Schneider, G. (1975) „Assessment of the seismological situations for the NPP Obrigheim, Philippsburg II, Neckarwestheim II”, University Stuttgart.