Seismic Re-evaluation of the Unit 1&2 of the Mochovce NPP

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Abstract: The Slovak Nuclear Regulatory Authority (UJD) requires the licensee (Slovenske Elektrarne) to keep the seismic capacity of the plant in operation on a level generally accepted by the international community. The re-evaluation of the seismic capacity of an existing plant is generally due to the following reasons: 1) evidence of a seismic hazard at the site that is greater than the design basis earthquake, arising from new or additional data, 2) existing facilities were designed and constructed to withstand low seismic loads only and 3) regulatory requirements stricter than those valid at the time of design and construction and take into account the state of knowledge and the actual condition of the plant. The original design basis earthquake of the plant with VVER440 type reactors was 0.06 g. The first project of seismic re-evaluation in 1997 increased the RLE (Review Level Earthquake) to 0.10 g. Significant number of plant upgrading measures were implemented. As a consequence of a new hazard evaluation carried out at the site, a new project was launched by the licensee, which further upgrades the seismic capacity of the plant to the RLE = 0.15 g (SL-2 value). The seismic re-evaluation is being performed on the basis of the IAEA guideline "Evaluation of Seismic Safety for Existing Nuclear Installation (NS-G-2.13)". The paper summarizes the basic assumptions and methods being applied for the re-evaluation, which follows the generic methodology of the Seismic Margin Assessment (SMA) combined with a Seismic Probabilistic Safety Assessment (SPSA).

1. INTRODUCTION

The Mochovce NPP is located in the southwest region of Slovakia. The site is about 17 km from the town of Levice (population: 40 000), 30 km from the town of Nitra (population: 90 000), and 120 km from Bratislava, the capital of Slovakia. The supply of technical water for the plant is provided by the pumping station at the river Hron near water reservoir (3.1 million cubic meters), located about 5 km from the site.

The plant is made up of two twin units, i.e., four units equipped with VVER440/V213 type reactors. Unit 1 and 2 were commissioned in 1998 and 1999, respectively. Unit 3 and 4 are under construction. Unit 3 will be given into operation in 2022.

UJD requested Slovenske Elektrarne to upgrade the seismic capacity to a peak ground acceleration value (PGA) of 0,15 g also in coincidence with the approach used for Units 3 and 4. The mean PGA corresponding to SL-2 level (frequency of $1x10^{-4}$/year) is equal to 0.143 g, conservatively the mean SL-2 PGA was selected 0.15 g.

The seismic updating concept follows the generic methodology of the Seismic Margin Assessment (SMA) combined with a Seismic Probabilistic Safety Assessment (SPSA).

The main steps of the re-assessment process are the following:

- Data collection for the seismic re-assessment
- Independent assessment of seismic hazard for the Mochovce site

- Identification of the safety requirements to be applied for the re-evaluation of seismic resistance of structures, systems and components (SSCs)
- Selection of criteria for the inclusion of any item into the SSC list to be re-assessed by a deterministic process called Seismic Margin Assessment (SMA)
- Re-assessment of seismic capacity of all SSCs by means of the conservative deterministic failure margin (CDFM)
- Re-assessment of the seismic risk by means of the seismic probabilistic safety assessment (SPSA)
- Identification of plant upgrading measures to achieve the required seismic capacity of the plant

The paper is prepared on the basis of the report [1]. The safety requirements are discussed in chapter 2 of the paper. Chapter 3 is focused on engineering standards, chapter 4 on SMA and chapter 5 on SPSA. Synthesis of the SMA and SPSA results is in chapter 6. Conclusion is described in chapter 7.

2. SAFETY REQUIREMENTS

Safety requirements are in general derived from the basic safety documents aiming at the minimization of the risks associated with nuclear power plants operation. In this context the General Nuclear Safety Objectives are met by identifying means of effective defence against radiological hazards as necessary means to protect individuals, society and the environment from harm. Two complementary safety objectives support the general one, dealing respectively with radiation protection and technical aspects.

The radiation protection objective states to ensure that in all operational states radiation exposure within the plant or due to any planned release of radioactive material from the plant is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.

The technical safety objective refers to take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the plant, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

Meeting these three safety objectives requires that, in the design re-evaluation of an existing plant, all actual and potential sources of radiation exposure are identified and properly considered, and provision be made to ensure that sources are kept under strict technical and administrative control.

The general approach to meet these objectives is the application of the defense in depth concept. It aims at ensuring that all safety activities, whether organizational, behavioral or design related, are subjected to overlapping provisions, so that if a failure occurs, it will be detected and then mitigated by appropriate measures.

A set of requirements as well as technical recommendations have been developed by the international community according to the defense in depth approach, to be applied in areas such as siting, design and operation, especially for the seismic scenarios [4].

For an existing plant it is neither necessary nor feasible to comply with all current safety standards in force for new plants [4]. When the safety assessment is applied to an existing plant, reference has to be made to high level safety requirements and special rules have to be developed for their practical implementation in the day-to-day work. The practical implementation must be in accordance with the operating conditions. For example reconstruct works can only be performed during refueling outages.

The guideline of IAEA [4] is developed for the seismic re-evaluation of existing plants. For existing plants, the seismic re-evaluation of their safety is based on a special set of assumptions which represent a partial relaxation of the general safety objectives, according to the best international practice.

For analyses full power operation is the starting operating mode of the plant to be brought into safe shutdown. The SSC list will contain the components needed for shutdown, containment and RCS heat removal for the operating modes 1 to 6, as defined by the emergency operating procedures. This means considering the plant having the reactor coolant system at or near operating pressure and temperature. The range of time to be considered is 72 hours after the safe shutdown earthquake occurrence. During this time the plant is able to reach and maintain the safe shutdown. Offsite power is considered not available at least for 72 hours after the earthquake. In addition, the occurrence of small break LOCA is considered.

The SSCs involved in the seismic re-evaluation program are traditionally selected as the seismically classified items of the plant. This is the so called SSC list.

UJD Decree No. 430/2011 [7] classifies the SSCs according to the following categories:

- Seismic category 1
    - 1a - active SSCs that must keep their functionally at or after earthquake.
    - 1b - SSCs that must have seismic robustness related to their strength and ability to stay hermetical and to keep integrity.
- Seismic category 2
    - 2a - SSCs that due to seismic interactions directly or by means of side effects may cause loss of functionality, strength, integrity or stability of seismic category 1 SSCs or cause disconnection of necessary access routes,
    - 2b - other SSCs.

The safety analysis will identify a special subset of items able to guarantee the following three basic safety functions under the above specified conditions, namely:

1. Plant cold shutdown
2. Residual heat removal (for 72 hours)
3. Containment integrity

SSCs in seismic category l are analyzed and the main success path and alternative success path are selected to perform the safety functions. SSCs in category 2a are placed in proximity of the selected category 1 SSCs. They are also involved in a special section of the SSC list and are covered by dedicated methodologies during category 1 SSCs assessment. In general, the seismic interaction effects are considered as:

- proximity
- structural failure and falling
- flexibility of attached pipelines and cables

Within seismic interaction analysis also the potential seismically induced internal fire and internal flooding events are analyzed.

A number of existing nuclear power plants throughout the world have been and are being subjected to review of their seismic safety. Rational feasible criteria for resolving the main issues were developed, particularly in the USA [2]. These criteria in some instances have been adapted for the specific conditions in Western and Eastern European countries by the relevant utilities and approved by the competent safety authorities.

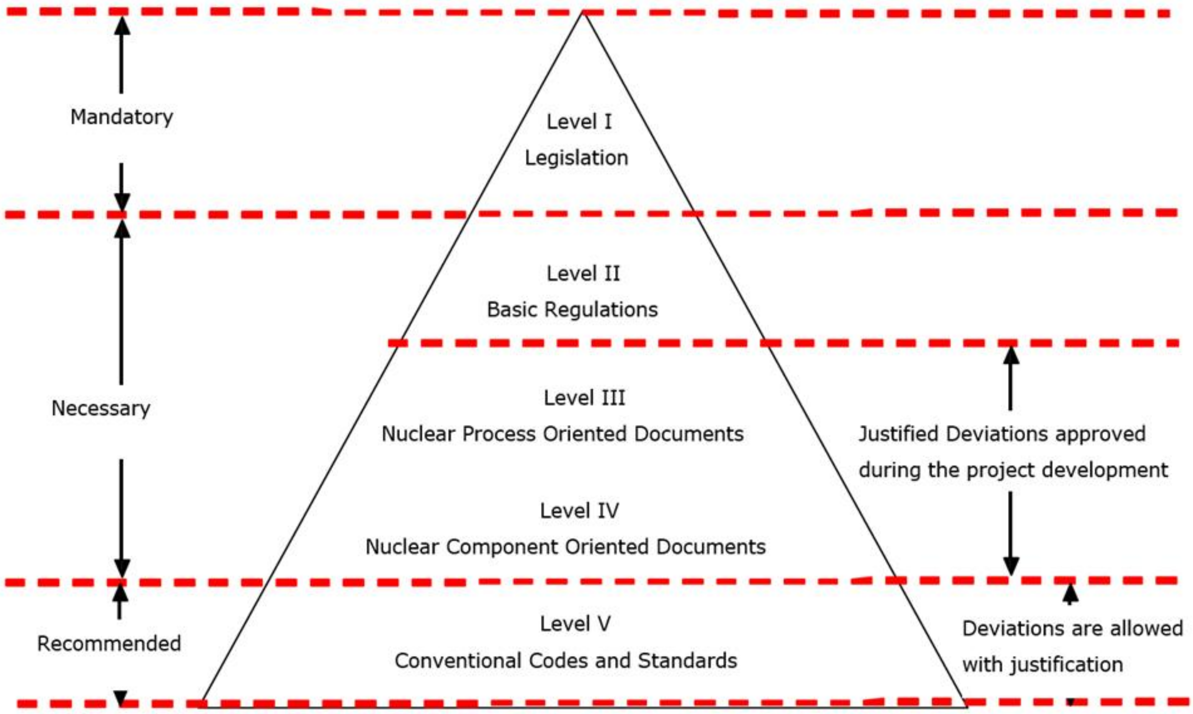
However, due to their development process, such re-evaluation programs at existing operating plants are unique and therefore plant-specific or regulatory specific. A state-of-the-art approach following the generic methodology of the SMA combined with a SPSA is tailored to the specificity of the Mochovce plant.

3. ENGINEERING STANDARDS

3.1. Design Standards

The codes and standards are set in such a way that the assessment and design of upgrading measures is compliant with the Slovak legislative requirements, international codes, standards and practices. The selection of codes and standards follows a hierarchic structure developed according to the European Utility Requirements for LWR plants (See Figure 1). A long list of codes and standards is provided in [1].

Figure 1: A hierarchy structure of legislations and standards



This hierarchy corresponds to the importance of the levels, with increasing level of the obligatory force from bottom to top. Characteristics for all levels are described below.

<u>Level I (mandatory) – Slovak and EU Legislation</u>

This level includes the generally binding rules of Slovakia (acts, laws, governmental decrees). All international legislations and agreements ratified by Slovakia also belong to this level, too. Special attention is paid to the relevant EU legislation. The project meets this level of legislation.

<u>Level II (necessary) – Basic Regulations</u>

This group contains regulations, which reflect commonly accepted criteria and principles of nuclear safety. In particular, the design and construction of the plant, including conditions for safe operation, are in coincidence with relevant IAEA Safety Requirements. International standards on quality management systems are also included into this level. The project meets this level of regulations.

Level III (necessary) – Nuclear Process Oriented Documents

These documents relate to the most important safety aspects of nuclear installations and determine adequate requirements on processes and systems. Currently, mainly relevant IAEA Safety Guides belong to this level and are applied to the project.

Level IV (necessary) – Nuclear Components Oriented Documents

This group of documents consists of specific rules and standards for nuclear installations in the fields of mechanical and electrical engineering, I&C, manufacturing, construction and tests, etc.

Level V (recommended) – Conventional Codes and Standards

This level includes technical rules and standards applicable to SSCs of Slovak conventional industry.

3.2. Quality Assurance

A Quality Assurance Program (QAP) is established in coincidence with the UJD requirements for the implementation of the seismic re-evaluation and upgrading program.

The QAP requirements were defined at the beginning of the seismic re-evaluation and involve the following issues:

a) project management
b) training and qualification of personnel
c) non-conformance control and corrective actions
d) document control and records, including configuration control
e) design control
f) procurement control of items and services
g) inspection and testing for acceptance
h) management self-review
i) independent review

A specific QA topic is related to the review of the computer codes to be used in the numerical analyses. The QA reviews the suitability of such tools for the project and provide evidence of the required certification.

3.3. Revision System

The review system of the re-evaluation program is part of the QAP but it has also a crucial role in the project.

The review relies on a three-stage approach:

1. the design reviews internal to the design organization
2. the independent verification carried out by an external organization on the selected areas
3. the UJD review of the plant upgrading proposal

3.4. Safety Analyses Report

The Safety Analysis Report (SAR) of the plant is being updated and submitted for review and approval to UJD, according to the results of assessments and upgrades performed. All seismic failures modes (scenarios) with the potential for serious radiological consequences to the plant staff, the public and the environment are identified and mitigation measures are proposed and documented.

4. SEISMIC MARGIN ASSESSMENT

SMA is a deterministic approach to evaluate the seismic capacity of the plant. Two success paths (the main success path and the alternative success path) are identified. The seismic capacity of the SSCs involved in the success paths is assessed. The seismic capacity of the weakest SSC of the stronger success path (path with higher seismic capacity) determines the seismic capacity of the plant (nuclear unit).

4.1. Finite Elements Models

The objective is to assess the capability of selected SSCs to withstand the seismic load, combined with other applicable loads, remaining functional and within material properties limits, as well as to compute the floor response spectra needed for the subsequent equipment assessment. Detection of those SSCs is needed which do not meet the seismic requirements. In structure response spectra (ISRS) is computed for the different floors of each building. In case that the need of reinforcement or upgrading of the structures (buildings) arises, that affects their structural behavior and dynamic response. The ISRS will be computed for the final upgraded condition of those buildings.

Finite element model (FEM) is used to represent SSCs and calculate their seismic capacity, except for simpler SSCs, that can be assessed by simpler calculations. These models are carefully developed taking into account the complexity of the considered items, in order to represent with a sufficient level of detail their behavior.

One combined three dimensional FEM is set up for each complex object under assessment, in order to compute its seismic response. Particularly, the FEM is arranged in such a way to be able to represent with sufficient approximation the stiffness as well as the mass distribution of the real SSC.

The reactor coolant system (RCS) is analyzed by developing a dedicated FEM that takes into account the complete primary loop with primary coolant piping, pumps, steam generators, valves, pressurizer, etc., chosen as representative of all the loops. The boundary conditions in terms of motion to the restraints and supports will be assumed consistently with the outcomes from the analysis of the whole reactor building.

The representation of the SSCs by the FEMs is accomplished by an appropriate selection of finite elements, in terms of shape, type and dimension, in order to represent the major characteristics of the structural system under investigation and their dynamic response in the range of frequency of interest, and can be tuned differently to represent different portion of a complex structure, in order to ensure a reasonable balance between accuracy and computational effort/feasibility. The mesh size is assessed to ensure a proper and sound response of the model. The finite element selection also takes into account the mathematical analytical theory of the elements, number of nodes, as well as their aspect ratio to ensure accuracy in the response of the model.

Nearly all SSCs exhibit at least some ductility, i.e., the ability to strain beyond the elastic limit, before failure. Taking into account the oscillatory nature of earthquake ground motion, this energy absorption is beneficial in increasing the seismic margin against failure of SSCs. Ignoring this effect would usually lead to unrealistically low estimations of the seismic safety margins. This inelastic energy absorption capacity is accounted for in the evaluation approach by specifying a so-called inelastic energy absorption factor F_{μ}, also called ductility factor.

The seismic capacity evaluation of the SSCs consists in a deterministic approach based on the methodology described below.

4.2 Seismic Capacity Evaluation

The seismic capacity of SSCs is defined in the form of HCLPF (High Confidence of Low Probability of Failure). The Conservative Deterministic Failure Margin (CDFM) approach is used, under the hypotheses:

$$\frac{SeismicDemand}{Capacity} \leq F_\mu \qquad F_\mu \ inelastic\ energy\ absorption\ factor \tag{1}$$

Therefore, under above hypotheses and input parameters calculated using FEM, the HCLPF is defined as follows:

$$HCLPF = \frac{C_{CDFM} - D_{NS}}{D_{CDFM} + \Delta C_{CDFM}} F_\mu \cdot RLE \tag{2}$$

where:
C_{CDFM} deterministic seismic capacity
D_{CDFM} deterministic elastic seismic demand computed at RLE =0.15 g
D_{NS} non-seismic demand for all the non-seismic loads in the load combination
ΔC_{CDFM} reduction in the capacity due to concurrent seismic loads
F_μ deterministic inelastic energy absorption factor, or ductility factor.

For complex SSCs of a success path the HCLPF will be assumed as the lowest HCLPF computed for the single structural members belonging to the structural system/building.

In addition to the analyses, the seismic capacity of SSCs can be determined using:

- assessment by similarity
- assessment by test based upon the existing test results
- mixed approach which is applied for the assessment of complex equipment, e.g., for the evaluation of pumps, the main equipment evaluation could be performed by similarity and the large nozzles and anchors ruggedness are assessed by analysis.

For specific equipment, such as tanks and heat exchangers, distribution systems (including cold and small piping and HVAC ducts) and cable and conduit raceway, suitable simplified approach is used. This is the GIP-VVER approach developed for VVER plants, compliant with the US-GIP approach. The two approaches are formally different, but they rely on the same technical basis and only minor differences affect the qualification procedures. The IAEA TECDOC-1333 summarizes the differences between the standard US-GIP data base and the GIP-VVER [5].

4.3 Corrective Measures

Taking into account the outcomes of the HCLPF calculation, those SSCs that exhibit an HCLPF lower than RLE in terms of PGA are subjected to corrective measures in order to ensure that everywhere HCLPF > RLE.

These corrective measures are determined from case to case, according to the specific SSC affected, tailored on the considered object, and according to the applicable standards.

In the final step, results from SMA process are compared with those from SPSA process in order to identify the final corrective measures. From these two processes a final list of SSCs to be upgraded is optimized. This list undergoes a specific analysis to assess the compatibility with plant operation.

The final list then undergoes a basic design and a time/cost analysis, in order to decrease the contribution of the seismic PSA to the overall CDF and LERF.

5. SEISMIC PSA

In addition to deterministic seismic analyses also probabilistic seismic analyses are required by the regulatory authority. The seismic PSA (SPSA) is carried out to evaluate the risk associated with occurrence of a seismic event [6]. In probabilistic studies (Level 1 and 2 SPSA), the risk of core damage is generally referred as CDF and LERF and represents a bounding estimation of the annual frequency of occurrence of an accidental event leading to core damage which can possibly result into radiation releases.

The objective of the SPSA is to calculate CDF and LERF for all operating modes associated with fuel damage connected with fuel assemblies:

- in reactor pressure vessel
- in spent fuel pool.

The goal of the SPSA is:

- to understand the overall seismic risk
- to identify the dominant contributors to the seismic risk
- to determine the acceleration ranges (earthquake intensities) which dominate the plant risk
- to compare seismic risk to risk originating from other events
- to propose safety measures to improve the plant safety given seismic event.

The main differences of SPSA with respect to SMA are the following [4]:

- the seismic hazard in SPSA is represented by family of curves, which associates to each seismic magnitudes (evaluated in terms of peak ground acceleration) a probability of occurrence and considers a spectrum of intensities beyond the RLE
- in SPSA, the fragility of SSCs is estimated in a probabilistic manner, hence there is a finite probability that a SSC fails below its HCLPF
- in SPSA, both seismically qualified and non-qualified components are credited for reaching the safe shutdown condition
- multiple initiating events due to the simultaneous failure of several SSCs are considered in SPSA
- the correlation between failure of SSCs with similar fragility characteristics or with spatial interactions is included in SPSA analysis
- in SPSA, the effect of the seismic event on the human performance (e.g., due to additional level of stress) is accounted for in the calculation of CDF and LERF

The use of probabilistic methodologies allows also a quantitative evaluation of the additional safety margin which can be associated to the corrective measures to be implemented in the framework of seismic re-qualification process.

5.1. Input Data

The input data necessary for SPSA are listed below:

- The seismic hazard curves for the site
- The design documentation of the plant and seismic safety technological concept
- The fragility curves of SSCs
- The report assessing the results of observations during plant walkdown
- The existing internal event Level 1 and 2 PSA model for internal events

5.2. Methodology

The basic idea of SPSA is to evaluate the risk of fuel damage connected to a wide range of seismic intensities. In general, the probability of occurrence of a seismic event whose intensity exceeds the threshold decreases with the seismic intensity itself. By contrast, the fragility of SSCs increases with the seismic magnitude. Large number of components is involved, each of them is characterized by its own fragility characteristic, the solution is generally evaluated using a PSA tool (e.g., Risk Spectrum) and may necessitate the definition of a limited number of acceleration intervals for the computation of the overall seismic risk.

In the following, some general indications concerning the main steps to be carried out in SPSA analysis are detailed.

Seismic Hazard Analyses
A family of hazard curves is constructed using the probabilistic seismic hazard analyses (PSHA). For the convenience of calculation, the seismic hazard curves are decomposed into a number of discrete acceleration ranges. The calculations are performed for these discrete ranges, characterized by the mean frequency of the median acceleration for each range. The seismic hazard curves are considered as basic input for the SPSA.

Initiating event identification and event tree development
The first step in the SPSA analysis consists in the identification of initiating events (IE). The purpose of this activity is to identify an exhaustive set of IEs that can be originated by a seismic event. The activity begins with the evaluation of the list of IEs retained from internal event PSA and the identification of those IEs that can be induced by a seismic event. The list is then completed with those IEs that are specific to the seismic hazard.

While in internal event PSA the simultaneous occurrence of multiple IEs is screened out based on the associated low frequency of occurrence, a seismic event can trigger the simultaneous failure of several SSCs. For the probabilistic viewpoint, this implies that there exists a correlation between the different failures which can generate an initiating event condition and the frequency of simultaneous occurrence becomes thus non-negligible a priori. The connection of seismic event with internal fires and with floods is also assessed within the scope of SPSA.

In particular, the major steps are:

- Identification of the multiple initiating events occurring at the same time due to earthquake as a common cause initiator leading to complex plant transients
- Construction of the event trees to identify accident sequences leading to core damage
- Definition of the initiating events and construction of the event trees for each seismic acceleration range

Characterization of the accidental scenario and construction of the event trees
The corresponding accidental scenarios are characterized and the associated success criteria are defined based on the internal event PSA. In the definition of the accidental scenario, conservative assumptions are applied if the thermal-hydraulic calculation corresponding to exact accidental conditions is not available from internal event PSA calculations. The dependencies between systems and components can be explicitly included in the event trees or hidden in the fault trees. The event trees are implemented in the SPSA model with the appropriate boundary conditions.

Fragility analyses
The seismic fragility of SSCs is defined as the conditional probability of failure at a given value of peak ground acceleration from the seismic hazard curves. The development of fragilities is based upon the use of log normal models of variables that contribute to structural response, equipment response and structural and component capacity. The distributions on response and capacity are

determined individually for each important variable and then combined in accordance with the mathematical procedures applicable to a log normal model. Several studies have been conducted that verify the separation of variables approach and the use of a log normal model, thus this methodology is well proven. Based on this approach, the fragility of an element corresponding to a particular failure mode can be expressed as

$$f(a) = \Phi\left(\frac{ln\left(\frac{a}{A_m}\right) + \beta_u \Phi^{-1}(Q)}{\beta_r}\right) \tag{3}$$

where:

a – seismic acceleration (typically expressed in peak ground acceleration)
A_m – median ground acceleration capacity
β_r – logarithmic standard deviation expressing uncertainty due to randomness
β_u – logarithmic standard deviation expressing knowledge uncertainty
Q – subjective probability (confidence) that the conditional probability of failure is less than f for peak ground acceleration a.

System Analyses

The impact of the seismic event on system performance is accounted for in the system reliability analysis with dedicated basic events, which express the fragility of the SSC to a seismic event. The probability of this seismically-induced failure depends on the seismic level and can be updated using exchange events to reduce the manipulation required in the calculation phase.

In particular, the analysis takes into account:

- Seismic equipment failures dependent on earthquake intensity
- Additional component failure modes screened out from an internal event PSA due to negligible probability of occurrence
- Seismic failure of structures
- Correlation between seismic failures
- Fault tress are constructed for each acceleration range

Human Reliability Analysis

The effect of the seismic event on human performance is evaluated, taking into account:

- Effects of seismic motion on performance condition and human reliability and calculation of human error probability including the effects of additional stress induced by the seismic event
- Additional operator actions required to mitigate combinations of seismic and non-seismic failures.

Seismically induced fires and floods

The seismic event is the initial hazard. The subsequent internal fires and internal floods are considered to be directly caused by the seismic event at about the same time of its occurrence. The approach addresses seismically induced internal fires and internal floods which are not already captured in the internal event PSA. The potentially risk significant scenarios are considered. Those scenarios are screened out that are not expected to contribute significantly to the plant risk, or are subsumed by other damage states, e.g., failures of the buildings.

Quantification

The quantification of the CDF and LERF associated with a seismic event is performed based on the hazard curves characterizing the site and the resistance of SSCs. As the fragility curves corresponding to the SSCs are available, the probability of failure curve can be directly computed as a convolution of hazard curves and fragility curves using the HazardLite code which is a seismic module of the RiskSpectrum code. In the general case, each SSC credited in SPSA model has its own fragility and is subjected to a different ISRS depending on its location. For this reason, the seismic risk is evaluated using a SPSA tool, discretized hazard and fragility curves and repeating the calculation in each interval.

In the following, the steps of the quantification phase are described.

The first step in the quantification is related to the evaluation of the risk connected to the different levels of seismic event considered. For each interval, the following data are updated in the model:

- Initiating event frequencies
- Failure probabilities of SSCs
- Calculation of CDF and LERF.
- The evaluation of the importance and sensitivity factors.

CDF and LERF are computed considering that all the basic events induced by an earthquake are independent. This assumption is generally accepted due to the low probability of occurrence of major earthquakes, despite of the fact that the earthquake represents a common mode of failures for the plant. The interrelations generally considered in SPSA are:

- correlation of SSCs characterized by similar seismic resistance: multiple failures of components is likely if they have similar seismic fragilities and are subjected to comparable seismic floor spectra;
- correlation of SSCs due to spatial interactions: the failures of several components located in the same room can be associated with the collapse of the civil structure or the rupture of a component jeopardizing the functionality of several SCCs in the same room.

The identification of interrelation between components may be supported by walkdown on the plant site. The evaluation of the impact of correlation on results can be carried out after the identification of minimal cut sets (MCSs) either with a new calculation with SPSA model or with postprocessing of the MCSs. Different options can be implemented to account for the correlation aspect. A realistic evaluation requires the calculation of the joint probability of statistically-correlated events, which generally involves multivariate integrations. A simpler option which ensures conservative results is to use the same basic event for modelling the seismic failure of components with similar characteristics, considering thus the failures as fully correlated.

The robustness of the SPSA results will be consolidated performing uncertainty analysis. The uncertainties on the input data are propagated through the SPSA model using Monte Carlo technique and the distribution of the CDF and LERF by providing upper and lower estimates of the seismic risk (typically corresponding to 95% and 5% of the distribution). Uncertainty analysis is performed by taking into account the following sources of uncertainty:

- uncertainty in the seismic hazard frequency over the range of seismic accelerations modelled in SPSA
- uncertainty in fragilities of essential SSCs
- uncertainty in human error rates
- uncertainty in the likelihood of random, non-seismic component failures.

Sensitivity studies are based on the importance factor analysis. The most critical assumptions are highlighted and the effect of the modelling choice on SPSA results can be evaluated with dedicated sensitivity analysis.

The list of modifications identified using SMA and SPSA analysis are implemented in the model to verify the CDF and LERF reduction due to the improvements. The quantification steps are repeated and the differences with respect to the reference case (i.e., without corrective measures) is highlighted.

6. SYNTHESIS OF THE SMA AND SPSA RESULTS

The SMA process provides a set of SSCs to be upgraded, with reference at the main success path and alternative success path selected during the SSC list development (loss of off-site power and small break LOCA are considered) and to the initial plant design (already upgraded to 0.15 g).

The SPSA provides the list of SSCs which represent the major contributors to the risk for the "as-is" plant configuration and proposes safety measures to improve the plant safety given a seismic event.

The results obtained from the two processes described above are analyzed and merged by means of the following rules:

1. All SSCs resulted as to be upgraded in the SMA process will be considered for the upgrade.
2. The SSCs recognized as major contributors to the risk in the SPSA process may or may not be present in the list of point 1 because the SPSA takes into account a wider range of scenarios:

- if the major contributors to the risk are already in the list of SSCs to be upgraded resulted from the SMA process, they will be considered for the upgrade;
- if some major contributors will not be in the list of components to be upgraded resulted from the SMA process, a deeper sensitivity analysis will be performed, in order to clarify if the upgrades are minor (e.g., additional anchorage, lateral supports and tying the cabinets together) or if the upgrades are more substantial (e.g., replacement of major components or adding of shear walls). In the first case, the SCCs will be considered as to be upgraded while in the latter case, detailed analysis and refinement of fragilities will be attempted before considering the upgrading.

From these two processes a final list of SCCs to be upgraded is determined. This list undergoes a specific analysis to assess the compatibility with plant operation. The final list undergoes a basic design and a time/cost analysis.

7. CONCLUSION

This paper summarizes the ongoing process of the seismic re-evaluation of the Unit 1&2 of the Mochovce NPP. The objectives of this project is to increase the design basis earthquake from 0.10 g to 0.15 g. The paper described the safety requirements of the project, used engineering standards, application of SMA and SPSA within the project and synthesis of the SMA and SPSA results.

The project contributes to the state of knowledge and deep understanding of the plant seismic risk. It is important due to growing recognition of the safety significance of the seismic hazard. In addition, further areas may be identified for improvement the existing methods and models.

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