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Definition and classification scheme of SSCs for specific and generic seismic fragility evaluation

Authors : Mr. Oleksandr SEVBO (Energorisk), Oleksandr Sevbo

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Project officer: Katerina PTACKOVA

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Summary

The present deliverable is focused on definition and classification of systems, structures and components of nuclear power plant, in order to perform generic fragility analysis or detailed specific fragility analysis. It also deals with the screening criteria for selection of systems, structures and components for fragility analysis. Fragilities of systems, structures and components are needed as input to seismic probabilistic safety assessment logic models. The intended use of the described below approaches is for performing seismic probabilistic safety assessment (SPSA) for an operating plant reflecting as-designed, as-built, and as-operated conditions. The screening guidance will be quite different for an advanced reactor SPSA that is to be used for the design purposes. The document is based on a literature review of several reference publications. The main results of the efforts are: ? Definition of process for identification of systems, structures and components for fragility analysis; ? Qualitative and quantitative criteria to screen out systems, structures and components from further consideration; ? Quantitative criteria to decide which fragility analysis, detailed plant specific or generic, should be performed for systems, structures and components; ? List of systems, structures and components for the METIS study case for detailed plant-specific fragility assessment.

Approval

Date	By
2021-08-31 12:48:10	Mr. Konstantin GOLDSCHMIDT (TUK)
2021-08-31 14:54:16	Dr. Irmela ZENTNER (EDF)

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Definition and classification scheme of SSCs for specific and generic seismic fragility evaluation

Deliverable D6.1

Version N°1

Authors: Oleksandr SEVBO (Energorisk),

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	N°1	O. SEVBO	N. CHOKSHI I. ZENTNER K. GOLDSCHMIDT	

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Abbreviations and Acronyms

Acronym	Description
AC	Alternate Current
BI	Birnbaum Importance
CCF	Common Cause Failures
CDF	Core Damage Frequency
DC	Direct Current
ECCS	Emergency Core Cooling System
FDF	Fuel Damage Frequency
FV	Fussell-Vesely Importance
HCLPF	High Confidence of Low Probability Failure
UHS	Uniform Hazard Spectra
LRF, LERF	Large Release Frequency, Large Early Release Frequency
PGA	Peak Ground Acceleration
PSA	Probabilistic Safety Assessment
RAW	Risk Achievement Worth
RRW	Risk Reduction Worth
SEL	Seismic Equipment List
SFP	Spent Fuel Pool
SPSA	Seismic Probabilistic Safety Assessment
SSC	Systems, Structures and Components
VVER	Water-Water Energetic Reactor
WP	Work Package

Summary

The present deliverable is focused on definition and classification of systems, structures and components of nuclear power plant, in order to perform generic fragility analysis or detailed specific fragility analysis. It also deals with the screening criteria for selection of systems, structures and components for fragility analysis. Fragilities of systems, structures and components are needed as input to seismic probabilistic safety assessment logic models. The intended use of the described below approaches is for performing seismic probabilistic safety assessment (SPSA) for an operating plant reflecting as-designed, as-built, and as-operated conditions. The screening guidance will be quite different for an advanced reactor SPSA that is to be used for the design purposes.

The document is based on a literature review of several reference publications. The main results of the efforts are:

- Definition of process for identification of systems, structures and components for fragility analysis;
- Qualitative and quantitative criteria to screen out systems, structures and components from further consideration;
- Quantitative criteria to decide which fragility analysis, detailed plant specific or generic, should be performed for systems, structures and components;
- List of systems, structures and components for the METIS study case for detailed plant-specific fragility assessment.

Keywords

Fragility; systems, structures and components; seismic equipment list; importance measures

Introduction

This report presents results of Task 6.1 "Definition and classification scheme of SSCs for specific and fragility evaluation". The purpose of this deliverable D6.1 is to provide methods and detailed information on definition and classification scheme of systems, structures and components (SSCs) for specific and generic seismic fragility evaluation. This technical report includes description of acceptable approach for selection of SSCs; classification of SSCs according to their relative importance into Tier1/Tier2 for the METIS case study. The place of the Task 6.1 within METIS WP 6 workflow is shown on Figure 1.

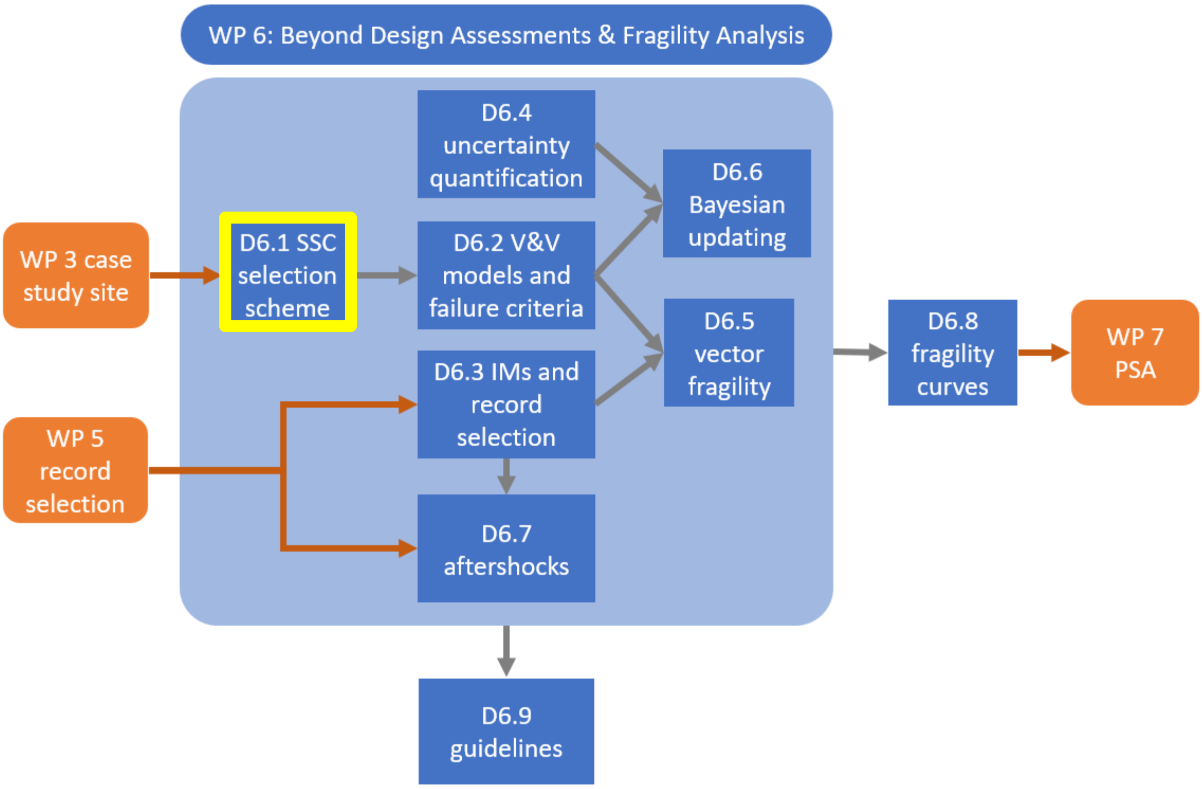

Figure 1: Flow chart of METIS work package 6

Section 1 deals with development of seismic equipment list – list of systems, structures and components that are necessary to ensure fundamental safety functions for further fragility analysis.

Section 2 presents an approach for classification into two Tiers systems, structures and components included in the seismic equipment list that have survived initial screening.

Short description of Zaporizhzhia nuclear power plant (ZNPP) unit 1 SPSA is presented in Section 3. This section presents also results of application of the Section 2 approach for classification of ZNPP Unit 1 SSCs into two Tiers.

Section 4 contains the results, recommendations and conclusions.

1. Definition of SSCs for fragility analysis

1.1. General

Seismic PSA is a comprehensive, structured approach to identifying failure scenarios due to seismic events, constituting conceptual and mathematical tool for deriving numerical estimates of risk. The objective of SPSA include the following /IAEA 2020a/:

- to develop an appreciation of accident behaviour (i.e. consequences and role of operator);
- to gain understanding of the overall likelihood of core damage induced by earthquakes;
- to identify the dominant seismic risk contributors associated with earthquakes;
- to identify the range of peak ground acceleration that contributes significantly to the plant risk;
- to compare seismic risk with risks from other events and establish priorities for addressing identified vulnerabilities

The main technical elements of SPSA are /IAEA 2020a/:

1. Probabilistic seismic hazard assessment;
2. Development of seismic equipment list;
3. Seismic fragility analysis;
4. Seismic plant response analysis;
5. Seismic risk quantification and interpretation of results.

This deliverable deals with the second technical element of SPSA – development of seismic equipment list, it adjustment and selection of SSC for further analysis. The result of this technical element is the list of SSCs for which fragility parameters have to be determined. The seismic equipment list is to be developed as combined effort of the system analyst and the seismic fragility analyst. The process for identification and definition of SSCs for fragility analysis consist several iterative steps, which are presented in Figure 2. The following general steps of the definition process should be performed:

1. Development of seismic equipment list (SEL);
2. Screening of SSCs from further consideration in fragility analysis and/or SPSA probabilistic model. It should be noted that there is distinction between screening for fragility analysis and screening from SPSA model. The screening from SPSA model implies that seismic failures of SSCs are not included in the probabilistic model, while their random failures are retained in the model;
3. Selection of SSCs for detailed and generic fragility analysis.

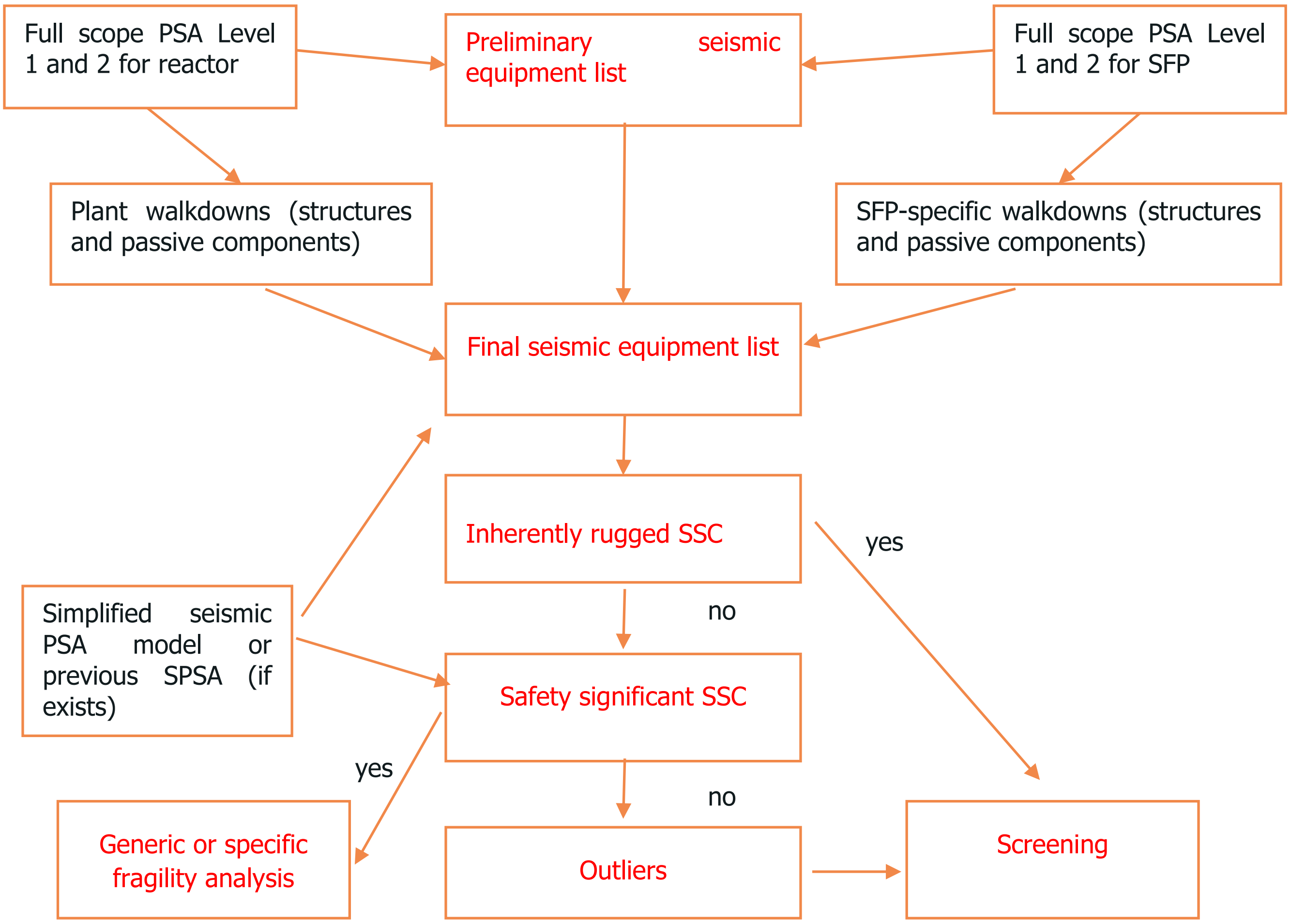

Figure 2: Definition of SSCs for fragility analysis

The starting point for definition of SSCs for fragility analysis is development of seismic equipment list. Seismic equipment list must contain all SSCs that are needed to prevent the progression of the seismic initiating event to core damage and/or to other undesirable consequences. Further such SSCs are modelled in Seismic PSA models to evaluate such risk metrics as core damage frequency, fuel damage frequency and large early release frequency. The seismic equipment list is significant for demonstrating completeness of the SPSA. The candidate seismic equipment list is supported by initial plant walkdowns that focus on the identification of potential systems interactions and reviewed for completeness (by the plant operators and system analysts). This review is performed consistent with the development of seismic initiating events and the development of seismic event tress (which is part of SPSA technical element on seismic plant response analysis). The seismic equipment list should include all relevant SSCs involved in the analysis of seismic initiating events and the development of seismic event tress.

An important aspect is that the seismic equipment list is usually very large. According to /IAEA 2010/, /IAEA 2010a/, /ASME 2013/ the list of structures and components for seismic fragility analysis should include all structures and components that are included in the PSA model for seismic hazards. The initial list of systems and components should be based on the list of components from Level 1 PSA. Further the list should be expanded to include all structures and components (active and passive), and their combinations that, if failed, could contribute to seismically-induced core damage frequency, fuel damage frequency or large release frequencies.

Depending on scope of internal events Level 1 PSA and SPSA (number of plant operational states, number of radiation sources at the plant, completeness of initiating events), it may contain from several hundred SSCs up to the more than thousand components.

This set should be appropriately reduced by screening the SSCs. According to /IAEA 2009/ screening of components may be performed on the basis of their high seismic capacity, the lack of seismically induced failures due to system interactions (verified in the plant walkdown), the level of seismic demand to which they are subjected at high levels of earthquake ground motion. The SSCs screened out using this approach should be replaced in the system models by a surrogate element of high capacity (or low fragility). The screening level and associated value of the fragility surrogate element should be established such that the surrogate element is not a dominant contributor to the end metrics. The end result is a list of selected SSCs for which further evaluation should be performed. /EPRI 2013/ also recommends that some equipment may be initially screened from the list if the conservative assumption is made that no credit will be taken for it performing its function. Equipment may also be removed from the list if a bounding analysis can demonstrate that the seismic core damage frequency (CDF) and LERF are not sensitive to its seismically induced failure probability.

So, for the following SSCs seismically induced failures can be screened out from consideration in SPSA model:

- Inherently rugged or robust SSCs;
- SSCs which seismically induced failures have negligible (or insignificant) impact on risk metrics under consideration.
- SSCs with high seismic capacity (or, in other words, with low fragility);

Therefore, the identification of SSCs for fragility analysis is an iterative process that consists of sequences of screenings and additions involving interaction between PSA analysts and structural or fragility analysts. PSA analyst should ensure that screening does not ignore the important risk contributors. However, the systems analysis may benefit in future risk-informed applications by keeping the high-capacity components in the model rather than screening them.

An important aspect for SPSA is that the seismic equipment list is further used not only for fragility analysis and probabilistic modelling, but also for identification of human actions to be considered in seismic PSA model:

- some human actions previously included in the internal events PSA might be eliminated or modified due to such reasons, as: the condition for human action has been screened out during the development of SEL; it was determined during development of SEL that particular SSC assumed to be failed due to seismic.
- new human actions that were not modelled in the internal events PSA might be considered, e.g. seismic related control room actions and /or recovery actions (e.g., recovery of relay chatter); undesired operator response to false alarm and indications (triggered by relay chatter).

For all structures and components that appear in dominant accident sequences, it should be ensured that the associated site-specific fragility parameters are derived on the basis of plant specific information. This is essential to avoid distortion of the contribution of seismic hazards in the results of, and insights from, the Level 1 PSA.

1.2. Development of seismic equipment list

The aim of this step is to develop a list of SSCs that are necessary to ensure fundamental safety function (safe shutdown list, /IAEA 2009/), mitigation functions, as well as SSCs needed to address seismically induced events (like internal fires and floods, loss of coolant accidents, loss of offsite power, reactor pressure vessel rupture and externally induced events). This action covers the followings:

- compiling basic SSC list for SPSA considering adverse effect of collapse of non-safety SSCs on safety SSCs performance.
- compiling list of important structures and passive components that was not considered / screened out from the PSA for internal events due to their high reliability;
- compiling SSC list related to the internal fires and floods;
- assembling list of pipes that can induce seismic LOCA;
- assembling list of relevant civil structures and facilities affected by seismic events. This shall take into account natural formations that collapse or change due to seismic event (including secondary effects, e.g. potential liquefaction scenarios and slope-stability, as well impacts between the buildings) and - can disturb normal operational conditions which can influence fundamental safety functions of the analysed plant, and industrial facilities, product lines (oil, gas etc.) that collapse due to seismic event and - can disturb normal operational conditions which can influence fundamental safety functions of the plant.

Actions on compiling basic SSC list for SPSA are presented in a number of international guidelines such as /IAEA 1993/, /EPRI 2013/, etc. Methods and steps for development of the list are well defined used for a long time. Depending of scope of available plant-specific PSAs, steps are slightly differing.

In case if documentation, models and results only for internal events (full-power, low power and shutdown) PSA Level 1 and level 2 are available, the SEL definition process should start from **(A)** identification of SSCs that are important to safe shutdown. Definition of safe shutdown SSCs is covered by /IAEA 2009/. Further analyst should **(B)** identify structures and passive components that are important to seismic response. The purpose of this step is to identify such passive components (tanks, pipes, cabinets, cables trays, interface lines, ventilation duct, etc,) whose seismically induced failure could affect the safety functions modeled in the seismic PSA. Passive components that were screened from the internal events PSA (due to their low probability of failure) should be included in consideration. It is recommended that scope of analysis should include not all passive items, but only those passive components that support functions modelled in the SPRA must be considered. As well such structures located at the plant, as reactor/containment building, turbine building, electrical/control buildings, auxiliary buildings, diesel-generator buildings, intake structures, should be incorporated in the SEL. **(C)** To adjust the basic SEL (include additional SSCs or eliminate SSCs), a plant walkdowns are used. Plant walkdown may be performed as a separate walkdown specifically for the SEL development or as part of the activities on seismic fragility assessment. **(D)** It is recommended by /EPRI 1991/ to perform comparison of the basic SEL against the system piping / instrumentation / electrical diagrams, as part of walkdown preparation. This will help ensure that SSCs screened from explicit inclusion in the initial PSA study are reconsidered for the seismic PSA. This would add to the robustness of the SEL. Purposes of plant walkdowns could be as follows:

- to define and verify the location / elevation of SSCs identified at previous steps (A), (B). This is important since some of the information presented in plant documentation may not reflect the current as-built, as-operated plant (e.g., SSC removed or abandoned in place). For multi-unit sites, this task could also include verification or observation of any noted asymmetries between the NPP units at the site;
- to identify additional SSC, not considered at previous steps (A) and (B). Depending on scope of initial PSA study, examples of such SSCs may be: - previously omitted SSCs, such as interfaces, relay, cables, cabinets, etc.; – SSCs added to enhance plant core damage or fuel damage prevention in response to security issues or the Fukushima event that are not currently credited in the initial PSA study (such as diverse and flexible mitigation capability, or equipment for diverse and flexible coping strategies (FLEX); – SSC (structures, pipes) that may adversely impact other SEL equipment or impede pathways in support of operator

response to the seismic event. – SSCs that may contribute to significant fire or internal flooding scenarios identified in the PSA. This task will support evaluation of seismically induced fire and seismically induced flooding scenarios;

- to combine multiple individual items and components into group, that further would be treated as one SSC. Multiple components on a common skid (such as DG components and support equipment) may be grouped together when judged appropriate. Another example may be grouping several instruments according to their common instrumentation rack and adding the rack to the SEL;
- to exclude SSCs from the SEL, if the SSCs are not presented/located as documented in the existing plant documentation. As such, these SSCs may be re-evaluated as outside the scope of the model. In such instances, the SSCs could be removed from the SEL and documented correctly.

Procedures and recommendations for the seismic walkdowns can be found at several sources, e.g. /EPRI 1991/, /EPRI 2013/.

In case if documentation, models and results for seismic (full-power, low power and shutdown) PSA Level 1 and level 2 are available, the SEL definition process should start from SEL established for initial SPSA study. This list contains SSCs identified for the plant at the past periods, and attention should be paid on review of the plant documentation and the SEL to account safety-important and seismic-important modifications and modernizations implemented at the plant after completion of the initial SPSA. This would include performing steps (C) and (D).

Output of this step is developed list containing plant specific relevant SSCs both for reactor facility and spent fuel pool facility, as well list of relevant inside and outside building/structures. It is recommended that the SEL, for each item, should contain at least: identification, brief description, location/elevation, assumed failure modes including description of failure impacts. Optional information can be formed by SSC categorization, e.g. /APSA 2017a/:

- basic internal SSC ensuring fulfilment of fundamental safety functions including (internal) seismic events; plus, a list of relays that chattering can evoke functional failures of SSCs (see Section 2.2);
- threatening internal SSC which collapse can affect performance of basic internal SSC;
- flood internal SSC that failure can lead to internal floods;
- fire internal SSC acting as potential ignition sources;
- external SSC capable evoking induced events;
- special internal SSC that involve in-site effects like multi-unit effects, impact of seismic event on nuclear facilities located in-site area.

1.3. Screening

The seismic equipment list is typically a long list of components and structures of many different classes and wide variety of seismic ruggedness. Since it is impractical to develop detailed fragility analysis of all potentially significant SSCs, screening analysis is typically applied. Screening analysis is a process to eliminate SSCs from further consideration based on their negligible contribution to the probability of a significant accident or its consequences. Any screening approach adopted should ensure that the final seismic core damage frequency (CDF) and large early release frequency (LERF) would not change appreciably, if any of the screened components were instead to be included.

Screening criteria are established in several guidelines and SPSA studies. Multi-steps initial screening process can be applied during selection of SSCs for fragility assessment:

- Identify and screen out inherently rugged or seismically insensitive SSCs from further consideration.
- Use previous SPSA model (if exists) or create simplified SPSA screening model to screen out low risk-significant SSC.

An additional option to reduce SEL is to identify and screen out low capacity SSCs that will be assumed to fail in case of seismic. It may be reasonable to remove from further fragility analysis those systems and components modelled in the PSA that have very low seismic capacity or provide a minimum mitigation potential in the SPSA. Usually it relates to such balance of plant systems that are not seismically designed / manufactured (e.g. component cooling systems, instrument air, etc.). If such SSCs are removed from further fragility analysis, they are assumed to fail in SPSA model. However, even such SSC may have some inherent capacity to survive seismic event. Care should be taken for low seismic hazard sites (or for low peak ground accelerations, PGA), since assuming failure for low capacity SSC could result in significant overestimation of core /fuel damage at low seismic levels.

Due to large scope of detailed walkdowns and fragility analysis for high seismic hazard sites, this screening option may be advantageous for that sites. Early screening of low capacity SSCs (that will be postulating fail) and that will not contribute to prevention of core damage or large early release may save resources on fragility analysis /IAEA 2020a/.

Also, should be noted that when structures and components of a low fragility are to be screened out on the basis of generic data, it should be proven that the generic data are used in a conservative manner and that no relevant plant and site-specific features are neglected. After completion of probabilistic models and preliminary quantification of SPSA results, the correctness of the screening assumptions and results should be numerically checked to verify that the screening process has not incorrectly excluded important SSCs. Sensitivity analysis is recommended for that checking.

1.3.1. Inherently rugged components

Inherently seismically rugged components are components able to withstand a strong seismic impact without significant loss of function, i.e., have a very low probability of failure due to seismic event. Inherently rugged is understood to require a significant beyond-design-basis g level to fail the equipment /EPRI 2013/.

Knowledge on the inherent capacity of components can obtained from past earthquake experience and past SPSAs. Several methods and guidance on identification of inherently seismically rugged SSCs are available, e.g., in /EPRI 1991/, /EPRI 1995/. Examples of SSC that are generally agreed as inherently seismically rugged are shown in Table 1, /IAEA 1993/, /EPRI 1995/, /EPRI 2013/, /NRC 1985/, /IAEA 2020a/.

Since seismic failures of inherently rugged components have no impact (or negligible impact) on risk metrics, as a rule, they are not included in the PSA models. Depending on plant response to different seismic events only random failures of inherently rugged components may be included in the probabilistic model. However, those rugged components whose seismic-induced failures are considered as directly leading to core/fuel damage or are significant to the risk results should not be screened out of the SPSA models. It should be also noted that seismic event generates challenging situation since the whole nuclear power plant is affected. Seismically-induced spatial effects and events can lead to cliff-edge effects having deep impact on potential radioactive releases even if contribution to the CDF is low. This implies that only high capacity SSCs not threatened by others SSCs can be screened out from further consideration. Such screening should be based on the review of seismic qualification criteria and qualification documents of relevant SSCs and verified by walkdown, if appropriate. Specifically, the walkdowns are performed in order to exercise engineering judgement (e.g., to verify anchorage of pumps), to confirm the seismic capacity of typically rugged components and to enlarge list of screened SSCs by items that are also considered inherently rugged by a trained and experienced walkdown team of experts.

SSC	Comment
Motor operated valves	That do not need actuation (i.e. to change state) to perform the intended functions. Active valves that change state are included for seismic evaluation. Motor operated valves are also included in the relay chatter evaluation for possible spurious operation due to relay chatter.
Manual valves	Mechanically (versus electromechanically) actuated devices are inherently rugged devices and are considered not susceptible to contact chatter
Air operated valves	That do not need actuation (i.e. to change state) to perform the intended functions. Walkdown should still be performed if valves are on small lines (less than one inch) to confirm that the valve / valve operator support is adequate for large operators. Active valves that change state are included for seismic evaluation. Air operated valves are also included in the relay chatter evaluation for possible spurious operation due to relay chatter.
Check valves	That do not need actuation (i.e. to change state) to perform the intended functions
Dumpers	That do not need actuation (i.e. to change state) to perform the intended functions
Pumps	Motor driven pumps judged as inherently rugged in /EPRI 1991a/ Because of the vibrations and stresses which occur for normal operations, pumps have inherent capacity to resist earthquakes. IAEA recommends to consider pumps as inherently rugged as long as they are properly anchored /IAEA 1993/. High capacity of pumps anchorage or mounting should be validated by an experienced seismic walkdown team.

	For most sites, pumps are treated generically or screened out. High seismic sites of NPP may require more specific evaluation.
Motor-generator sets	Due to the normal operating vibration inherent in rotating machinery, a motor has sufficiently rugged construction which should preclude any concerns about the ability of a motor to operate after a seismic event, given that anchorage adequacy is validated /EPRI 1991a/
Specific piping	Experience from past earthquakes in industrial facilities indicates that piping is rugged and can resist earthquakes of at least 0.5g pga, which is the limit of the experience data /NRC 1985/. Welded and bolted piping is considered to be inherently rugged, but cast-iron fire mains are not judged to be inherently rugged. Inherently rugged is understood to require a significant beyond-design-basis g level to fail the equipment. Piping may be defined as rugged or not rugged by the PRA analyst with support from the fragility experts /EPRI 2013/
Solid state relays with no mechanically moved parts, small safety and relief valves	These types of valves are considered to be of sufficiently high seismic capacity and can be screened out.
Manually operated control switches, Limit and torque switches found on motor operated valve actuators, Position switches found on circuit breakers	Mechanically (versus electromechanically) actuated devices are inherently rugged devices and are considered not susceptible to contact chatter
Batteries	Batteries mounted in braced racks designed for seismic loads or qualified by dynamic testing do not require evaluation. Rigid spacers between batteries and end restraints are required. Batteries should be tightly supported by side rails, /WEST 1991a/.
Sensors (temperature elements, level switches and transmitters, pressure switches and transmitters)	Seismic inertial loads for pipe-mounted temperature elements may be inconsequential. Flow transmitters. Level and pressure switches and transmitters are considered rugged, inadvertent actuation is the most probable seismic failure mode. However, inadvertent actuation of safety systems results in safety success and, therefore, these components are screened out /IAEA 2020a/

Filters, strainers, heat exchangers	Only failure modes related to blockage of filters, strainers, etc. are screened out
Control circuits	Control circuits (solid state) are assumed to be seismically rugged. For relay chatters separate assessment is needed

Table 1: Inherently seismically rugged SSC

1.3.2. Low risk significant SSC

All structures, systems and components modelled in the Level 1 and Level 2 PSA for internal initiating events and those structures, systems and components for which seismically induced damage can have an effect on accident sequences should be incorporated into the SPSA model /IAEA 2009/. Failures of many SSCs have lesser impact than others on the plant risk profile. Some of them have such small impact on risk of core / fuel damage and/or large release of radioactivity, that they can be neglected. Non-inclusion of such failures to probabilistic models would not significantly change not core damage frequency (CDF) nor large early release frequency (LERF), as well would not jeopardise the plant risk profile

Definition of low-significant SSC usually is based on the following:

- ranking of components/failure modes by importance measures; or
- impact of components/failure modes on risk metrics, CDF or LERF (screening by impact).

The importance measures from the previous SPSA results can be used to rank the SSC seismic fragilities. Typically, Fussell-Vesely (FV), risk achievement worth (RAW), risk reduction worth (RRW) and Birnbaum importance measures are used.

The FV importance measure is the fractional contribution of a given basic event to the probability of the undesired consequence when the basic event probability is changed from its base value to zero (i.e. the basic event never occurs) or equivalently the (conditional) probability that at least one "minimal cut set" containing the basic event occurs (given that the undesired consequence is occurred), /APSA 2017/. Referring to an individual basic event, the Fussell-Vesely Importance measure is defined as:

$$FV_i = \frac{f - f(P_i = 0)}{f} \sim \frac{f\left(MC_{including\ i}\right)}{f} = \frac{a \cdot P_i}{a \cdot P_i + b}$$

where $f(Pi = 0)$ is the probability of the undesired consequence when the basic event probability is zero.

The higher the value for FV risk importance means the larger fractional contribution to risk of the cutsets containing the event. FV values are always less than one. Typical SSCs with high FV importance include structural failures of buildings, including failures of electrical structures (e.g. off-site switchyards) leading to loss of power.

The Risk Achievement Worth (RAW) measures the "worth" of a given basic event in achieving the present risk level (probability of the undesired consequence in the following), by considering its maximum that is when the basic event always occurred. It indicates the importance of maintaining the current level of reliability for the basic event i. Referring to an individual basic event, the Risk Achievement Worth is defined as /APSA 2017/:

$$RAW_i = \frac{f(x_i = 1)}{f} = \frac{f(P_i = 1)}{f} = \frac{a + b}{a \cdot P_i + b}$$

where $f(xi = 1)$ is the probability of the undesired consequence when $xi = 1$ (i.e. the basic event always occurs).

The RAW value should always be greater than or equal to one (that is, failure of a component should always result in a higher or equal CDF value). Typical SSCs with high RAW importance include structural failures of buildings and multi-train, common-cause failure events. RAW values are typically lower for SPSA results than they would be for other PSA hazard assessments. This is primarily due to two reasons: 1) SPSA has high failure probabilities in the failure of many SSCs (1E-01 or higher). This means that the factor increase in risk if the component was considered completely unavailable for the SSC is lower in a seismic PRA than might be seen in and internal events PRA, and therefore, the value for the RAW of the SSC is also lower. 2) Most of the SPSA risk value is concentrated in the failure of major structures and systems. This means that there are relatively few cut sets with the failure of SSCs. This causes the RAW value to be lower for those SSCs, as well (the failure of major structures or systems can be considered to be masking the failure of components). However, at very low seismic accelerations, the RAW importance measure could be misleading because it assumes that the SSC has no seismic capacity. It may be more useful to use RAW above some fragility curve truncation level /EPRI 2013/.

The Risk Reduction Worth (RRW) measures the "worth" of a given basic event in reducing the risk level (probability of the undesired consequence in the following), by considering its maximum decrease that is when the basic event never occurs. It indicates the importance of reducing the current level of unreliability for the basic event i. Referring to an individual basic event, the RAW is defined as /APSA 2017/:

$$RRW_i = \frac{f}{f(x_i = 0)} = \frac{f}{f(P_i = 0)} = \frac{a \cdot P_i + b}{b} = \frac{1}{1 - FV_i}.$$

The Birnbaum Importance measure is the rate of change in the risk (probability of the undesired consequence in the following) as result of the change in the probability of a given basic event, or equivalently the difference in the probability of the undesired consequence when the basic events always occurs and never occurs, or equivalently the probability to be in a "critical" status for the particular basic event (i.e. the undesired consequence occurs only if the basic event occurs).

Referring to an individual basic event, the Birnbaum Importance is defined as /APSA 2017/:

$Bi = (xi = 1) - (xi = 0) = (Pi = 1) - \varphi(Pi = 0) = \partial\varphi/\partial Pi = a = RAWi + RRWi.$

The following technical aspects should be considered while using importance measures:

- Truncation value. The truncation value applied for initial PSA /SPSA should be low enough so that the truncated set of minimal cut sets contains all the significant contributors and is low enough to cover at least 95% of the core/fuel damage frequency. Depending on the scope and level of detail of the PSA (modelling at component level vs subsystem/train level), the truncation value may vary from 1E-12 to 1E-8 per reactor year, /NRC 2002/.
- Completeness of PSA model - the initial PSA /SPSA model should be sufficiently complete to address all important modes of operation for the SSCs being analyzed. SSCs contributing to fulfillment of fundamental safety functions both for reactor and spent fuel pool for all plant operational states (nominal [power, low power and shutdown modes) should be considered in PSA, /NRC 2002/.

- Risk profile – the risk profile for NPPs may be unbalanced in the sense that the failure of a single SSC, or of a very small number of SSCs, contributes disproportionally and dominates the risk profile. Such an unbalanced risk profile means that the NPP does not have high level of defence in depth. NPP with strong defence-in-depth attributes would not have a risk profile dominated by a single failure or a very small number of failures, and the risk is balanced among a variety of different contributors to overall risk. Since there are no disproportionally dominated SSCs failures for balanced risk profile, importance measures (especially FV) for many SSCs will be low and very low.

- Uncertainties. It should be satisfactory level of confidence that SSCs ranking is not affected by data uncertainties. The sensitivity study may be performed in order to show sensitivity SSCs ranking to uncertainties in the reliability parameters, /NRC 2002/.

- Common cause failures (CCF). SSCs screening should take into account combined effect of random and common cause failures for particular failure mode of SSC in question. Common cause failure probabilities can affect PSA results by enhancing or obscuring the importance of components. SSC may be considered as risk significant mainly because of its contribution to CCFs, or SSC may be erroneously treated as a low risk significant mainly because it has negligible or no contribution to CCFs, /NRC 2002/.

- Recovery actions. Recovery actions are modelled for dominant accident sequences/cut sets. Quantification of recovery actions typically depends on the time available for diagnosis and for performing the action, as well as the training, procedures, and knowledge of operators. There is a certain degree of subjectivity involved in estimating the success probability for the recovery actions. The concerns in this case stem from situations in which very high success probabilities are assigned to a sequence, resulting in related components being ranked as low risk contributors. Furthermore, it is not desirable for the risk evaluation of SSCs to be affected by recovery actions that sometimes are only modelled for the dominant scenarios. Sensitivity analyses can be used to show how the SSC importance measures would change if all recovery actions were removed, /NRC 2002/.

By /ASME 2007/, significant basic event is a basic event that has a Fussell-Vesely importance greater than 5E-03, or a risk-achievement worth greater than 2. SSCs represented in SPSA model by basic events with importance measures (FV and RAW) below these values can be considered as non-risk significant. Having described above technical aspects, it is recommended for screening of SSC to use the following numbers to judge SSC as low risk significant:

$$\text{FV} \leq 1\text{E-04 and RAW} < 1.5.$$

It should be noted that care should be taken while screening out SSCs for NPPs with well-balanced risk profile. The SEL should contain SSCs contributing at least 95% to risk metric (CDF, FDF, LERF). Depending on the risk picture, in order to fulfil this rule, it may be appropriate to retain in the SEL all items with FV more than 1E-05.

As regard for screening by impact, typically a CDF screening threshold is established by the system analyst whereby the components which are not modelled in detail, can be screened out, or else surrogate elements can replace groups of elements that are screened (at a high capacity level). In simple terms, this approach consists in setting a bounding (limit) fragility for the SSCs that replaces real seismic fragility of SSCs. Then convolution of this bounding fragility curve with the hazard curve results in a (bounding) failure frequency of these SSCs. If the bounding fragility is suitably chosen, it can be demonstrated that those SSCs for which the bounding fragility is applicable, have very small contribution to risk and such low significant SSCs can be screened out. Alternatively, so called surrogate elements can be used. Such elements represent whole groups of seismic components, with the objective to retain the risk contribution of those SSCs whose individual risk contribution is negligible. In the case of seismic PSA, the correct implementation of screening by risk impact forms a time-consuming process (which can require similar amount of resources as normal analysis). Care must be taken to ensure an exact counting of potential failure modes of seismic components and an adequate treatment of the correlation of component seismic failures. In addition, this approach should also consider impacts on Level 2 PSA results what introduces further complexity. Another drawback consists in difficulties to set some reasonable screening threshold for contribution to the CDF similarly as for event screening by frequency. Based on the above introduced reasoning, in particular the work intensity required for a well performed screening (e.g. correct implementation should also evaluate impact on Level 2 PSA) this method is not recommended by /APSA 2017a/, unless it is used in combination with the screening method based on seismic capacity.

2. SSC Classification

2.1. Tiers

In order to facilitate the SPSA development process, all SSCs selected for fragility analysis (see Section 1.2) are distributed between two Tiers:

- Tier 1 SSCs are unique/critical items that require detailed specific fragility analysis;
- Tier 2 SSCs can be dealt by more generic fragilities.

According to /IAEA 2009/, /IAEA 2020a/ for all structures and components that appear in dominant accident sequences, it should be ensured that the associated site-specific fragility parameters are derived on the basis of plant specific information. By /ASME 2013/ one of the objectives of seismic fragility analysis is to provide fragility realistic and plant-specific seismic fragilities for the significant contributors to seismic CDF and/or seismic LERF. This is essential to avoid distortion of the contribution of seismic hazards to the results of, and insights from, the Level 1 PSA.

Based on that, the following rules for inclusion of SSCs to Tier 1 are used in this report:

- SSCs with FV > 1E-03 and RAW ≥ 2, or RAW > 100, or FV>1E-01 (see Figure 2 for illustration); or
- Dominant SSCs ranked by Fussel-Vesely importance; or
- Dominant SSCs ranked by Birnbaum importance.

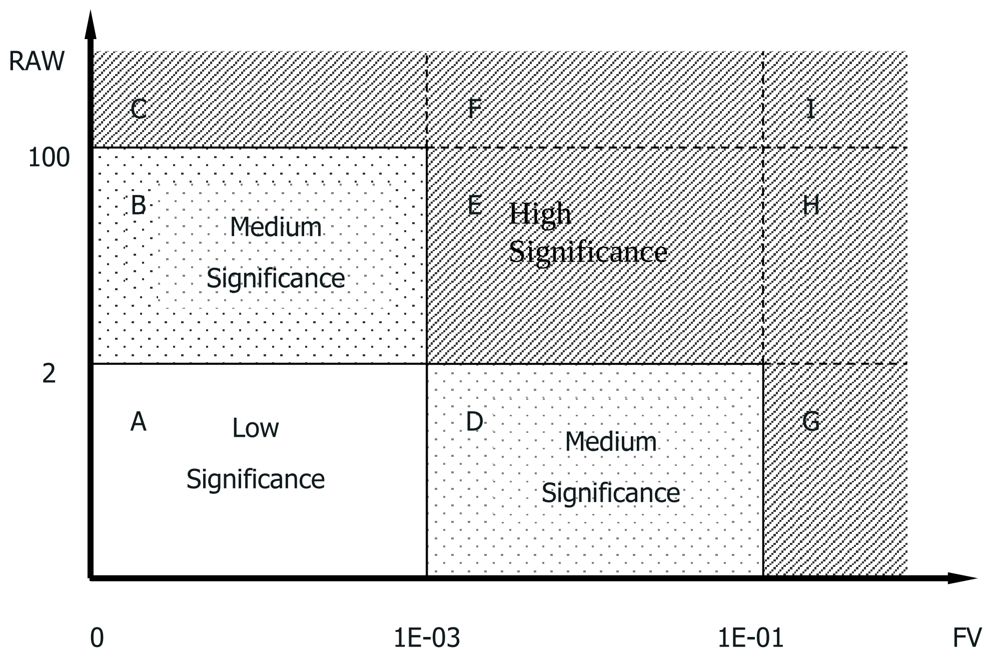

Figure 2: SSCs categorization by importance measures

A Possible option could be combined ranking of SSCs using Fussel-Vesely and Birnbaum importance measures. This will give more refined view on the selection and prioritization of structures and components for detailed fragility analysis.

Detailed fragility analyses for large number of systems, structures and components modeled in SPSA is time consuming and resource intensive. Amount of SSCs for detailed plant-specific fragility analysis varied from study to study, from several components to several dozens components. For example, for Loviisa NPP (Finland) with VVER 440 reactors, seismic fragilities were generated for the 30 components exhibiting the lowest seismic capacity, with generic fragilities being assigned to other generic classes of components. Those fragilities were used together with plant specific seismic hazard curves to quantify the core melt frequencies for the various accident sequences, /WEST 1991/. For Paks NPP (Hungary) more than 60 groups of SSCs failures with associated fragilities were defined for modelling and quantification in PSA: 27 groups of mechanical equipment, grouping based on equipment type and/or location; 9 groups of electrical and I&C cabinets, grouping based on cabinet location; 20 groups of electrical and I&C relays (contact devices), grouping based on relay type; 11 structural failures; 2 degrees of liquefaction, based on the differences in consequences, /OECD 2007/.

2.2. Selection

Two stages for selection of SSC for detailed fragility analysis (screening) are used:

- Screening of seismically rugged SSCs and preliminary fragilities are developed using generic data and design information (capacity screening);
- Resulting from initial seismic risk quantification (or previous SPSA) perform detailed fragility analysis for risk-significant contributors.

To reduce number of required detailed fragility functions, screening by high seismic capacity is used. Generic seismic high confidence of low probability failure data (defined as a level of earthquake ground motion at which there is a 95% confidence of an at most 5% probability of failure, or, equivalently, 1% mean probability of failure) is used for SSCs failures in PSA model to assess contribution of SSCs to seismic risk, and to screen out non-significant items. Generic high confidence of low probability failure (HCLPF) for different types of SSC are presented in /EPRI 1991/, guidance on calculating a screening HCLPF capacity value are documented in several document, e.g. /EPRI 2003/, /EPRI 2013/. For selecting a representative value for screening of SSCs, data provided in Table H-1 of /EPRI 2013/ could be useful. If components have wide variability in median capacities (e.g. electrical components, cables, relays), it is reasonable to use fragilities on the lower end of the ranges in order to avoid screening SSC on the basis of a high generic parameters. If structures and buildings have wide variability in median capacities, it is reasonable to use fragilities on the higher end of the ranges in order to not understate structural capacities, as this can disguise important SSCs within the structure.

As per recommendation of /EPRI 2012/, the screening HCLPF value of SSCs for a site should be calculated by convolving the fragility of a single SSC with the site-specific hazard curve such that the seismic CDF is at most about 5E-7 per year. There is also alternative criterion, that the screening level HCLPF be about 2.5 times the ground motion response spectra. Because each site will have a different hazard curve, the screening HCLPF value for each seismic PSA needs to be separately derived.

Another approach is to use In other cases, HCLPF levels to develop simplified fragility estimates for use in the PSA models, by adding the screening level fragility as surrogate element to each accident sequence to account for SSCs that were not modeled. Simply say this approach consists in setting a target (limit failure probability) for the surrogate elements that replace real components and that have very small contribution to risk. In practice correct implementation of surrogating approach forms time consuming process and care must be taken to exact counting of potential failure of seismic components and careful treatment of the correlation of components represented by the surrogate elements in order to avoid under-estimation of results, /EPRI 2002/. In addition, this approach should also consider impacts on L2 PSA results what introduces further complexity. Based on this reasoning as well as work intensity to perform well performed screening (e.g. correct implementation should also evaluate impact on L2 PSA) this way is not recommended by /APSA 2017a/.

As input data for screening, at least site-specific uniform hazard response spectrum (UHS) is needed. The results of the simplified seismic PSA model or previous seismic PSA (if exist) should be reviewed to determine whether or not an SSC modeled at the screening level could be included in Tier 1. For such SSCs detailed fragility calculations should be performed, and CDF /LERF should be quantified with the new fragility data. Then the importance parameters can be reviewed again to re-evaluate the distribution of SSCs into Tier 1 and Tier 2.

The distribution of SSCs between Tier 1 and Tier 2 may also account for other factors such as the following /EPRI 2013/:

- The initial, representative fragility values assigned to individual SSCs could be too low. If the initial fragility value were potentially too low, the importance measures could be overestimated.
- The initial representative fragility values assigned to individual SSCs could be too high. If the initial fragility value were too high, the importance measures would be underestimated. See, also discussion in Section 3.3.
- The initial SPSA model assumptions may skew the importance results. For example, if the SPSA model does not credit offsite alternate current (AC) power recovery or long-term direct current (DC) battery capacity during an extended station blackout scenario, the importance measures of mitigation systems powered by DC have very low values. It is understood that such mitigation systems are important for delaying potential core damage events. However, if offsite AC power recovery and long-term DC capability are not credited during station blackout events, core damage is likely assured regardless of whether such mitigation systems are available.
- Use of Level 2 PSA importance measures may provide a different risk ranking of SSC fragility events. For example, failures of containment venting systems or mobile pumps may result in a LERF end states.
- When eliminating SSCs from consideration as risk-important using representative fragilities, care should be taken to understand why the SSC is less important. This should include an understanding of whether the contribution of the SSC will be controlled by the failure of other SSCs, or if it is just the assumed representative fragility that is the basis. If it is just the assumed fragility that makes the SSC less important, this insight should be discussed with fragility experts.
- SSCs that have a significant uncertainty in the initial general fragility data and are satisfy Tier 1 criteria should be priority categorized for detailed fragility analysis. If there is significant uncertainty in the general fragility data, consider performing plant-specific fragilities for the SSCs that have the highest potential to impact the CDF/LERF.

In addition to SSCs directly included in seismic equipment list, there is another aspect, related to relay chatter, that need to be considered for inclusion in the scope of fragility analysis. During seismic event, relay chatter can affect the functionality of components required to bring the reactor to a safe shutdown state. Due to seismically induced chatter relay may send spurious signals to other electrical and control devices such as circuit breakers, motor starters or other relays. The consequence of these spurious signals would be unintended equipment shutdowns or actuations. Besides direct impact on availability of SSCs required to shutdown and maintain the reactor in a safe shutdown state, this can result in such effects, as: operator confusion due to unusual equipment operating configurations as well as inconsistent and erroneous indications on control panels; occurrence of initiating events, e.g. interfacing LOCA. Relays whose chatter during an earthquake could result in adverse effects on safety should be identified and further evaluated. It should be noted that impact of relay chatter on seismic risk may be dominant failure modes for some designs of NPP, e.g. relay chattering of core cooling and service water pumps have 8.1% change in CDF for Krsko NPP, /NEA 2020/. However, for some studies was assumed that relay chatter issues are generally issues for older plants; or have negligible effect on CDF for some reactors.

To identify relay for further consideration, the seismic equipment list is used as the basis. Often, rather than further modelling the response of the systems to relay chatter, a deterministic screening is conducted to identify relays with high and low capacity and to determine if relay chatter is detrimental, /EPRI 2003/. Relays can be screened out by demonstrating that they do not participate in any important safety functions; or by reference to the test data base that demonstrates that they are very rugged; or by a detailed circuit analysis to show that their chatter is benign. Typically, original relay list that may involve hundreds of relays is usually reduced to a very few (typically less than ten, sometimes even none) of concern, /NEA 1998/. As stated in /NEA 1998/, after the analysis has identified any relays whose chatter can be troublesome to important safety functions, the next step is to remedy the situation by either (i) replacing the particular relay, e.g. low ruggedness relays that can cause adverse effects /EPRI 2003/; or (ii) changing the circuit to eliminate seal-in or reset problems; or (iii) instructing the operators to be alert to post-earthquake relay-chatter problems; or some combination. Some relays with intermediate capacities may be modelled depending on their impact on the plant. Relay chatter that can lead to the spurious actuation of valves resulting in a bypass of containment are especially of concern.

2.3. Generic fragility classes

Typical scheme for definition of SSC general classes included in Tier 2 is illustrated in Figure 3.

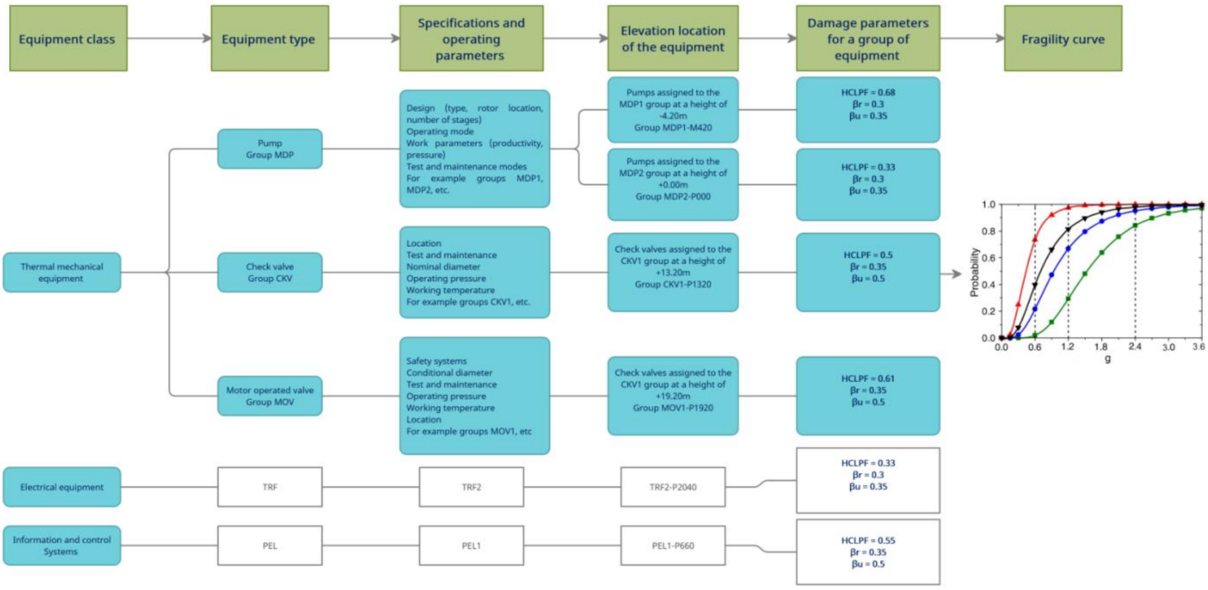

Figure 3: SSC classification process, /ZNPP 2019/

All components identified in SEL should be grouped into SSC types and for each type a specific type code and failure probability is to be assigned. As per figure 2, SSCs classification (or grouping) for consideration in probabilistic models is based on the following:

- SSCs should be of the same type;

- SSCs should have the same boundaries;
- SSCs should have the same operational mode;
- SSCs should have the same failure mode;
- SSCs should have the similar operational parameters/modes;
- SSCs should have the similar surveillance requirements, test and maintenance practice;
- Seismic fragility for the same SSCs depend on elevation (floor) at which this SSC is located.

3. METIS case study

3.1. General Information

The Zaporizhzhya NPP site is situated in the Kamenka-Dniprovska district of the Zaporizhzhya region on the left bank of the Kakhovka water reservoir (Dnipro river). Zaporizhzhya NPP The district center, the town of Kamenka-Dniprovska, is situated at a distance of 12 km from the Zaporizhzhya NPP site, 52 km from the regional center, the city of Zaporizhzhya, and at a distance of 5 km from the satellite town of Energodar. The local relief of the Zaporizhzhya NPP site is flat, with alternating sand hummocks and hollows. The site leveling elevation is taken as 22.0 m. There are six power units operated at the Zaporizhzhya NPP site with WWER-1000/320 reactors with the total electric power of 6000 MW.

General layout of the ZNPP site showed on Figure 4.

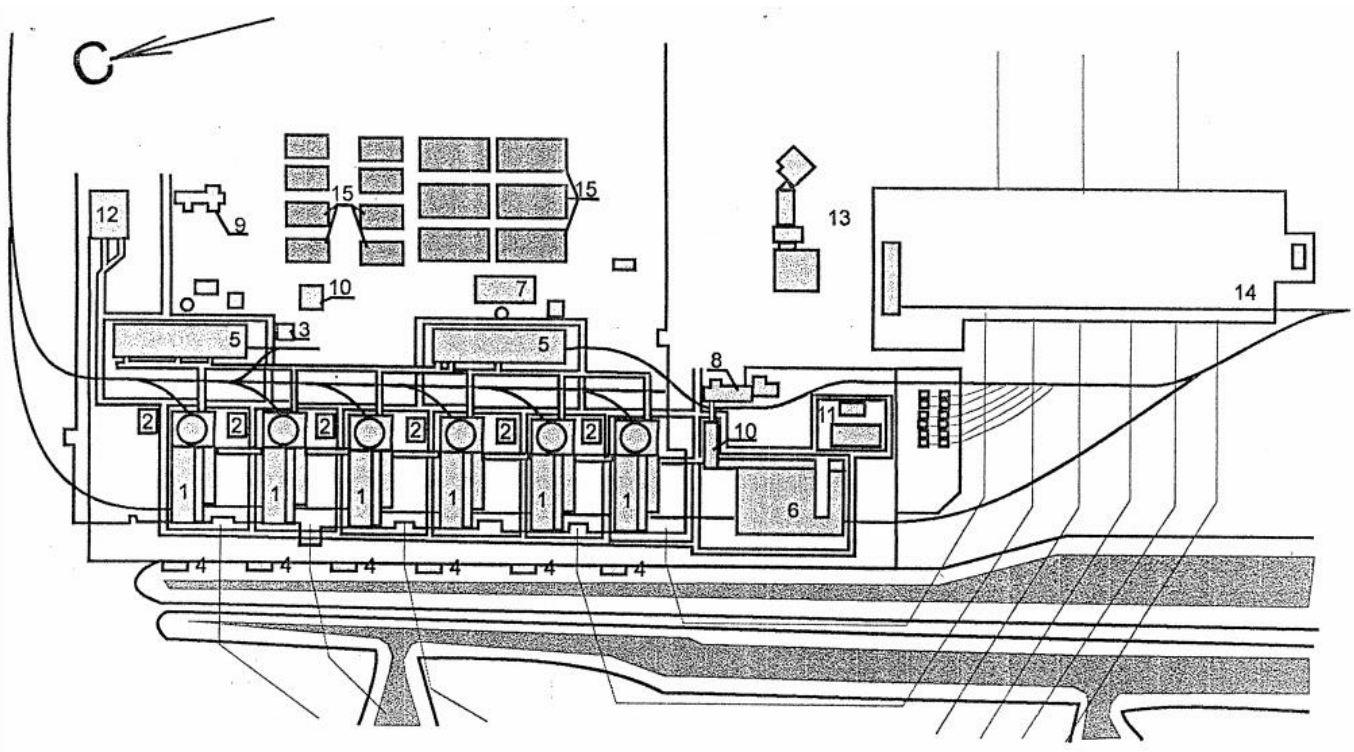

Figure 4: ZNPP site layout.

Where: 1- Main building; 2- Diesel-generators; 3 -Common-unit diesel-generators (ZNPP-5,6); 4- Cooling pump station; 5- Special building; 6- Common auxiliary building; 7-Radioactive waste storage; 8- Administrative building; 9- Checkpoint; 10-Labs; 11 – Oil/ DG fuel building; 12 – Dry spent fuel storage facility; 13 – Training center; 14- Off-site switchyard 750 kV; 15 - Essential service water ponds.

Each reactor facility (RF) is equipped with a water-cooled water-moderated pressurized power reactor WWER-1000/ 320 series. Layout diagram of main equipment of WWER-1000/320 is shown of Figure 5.

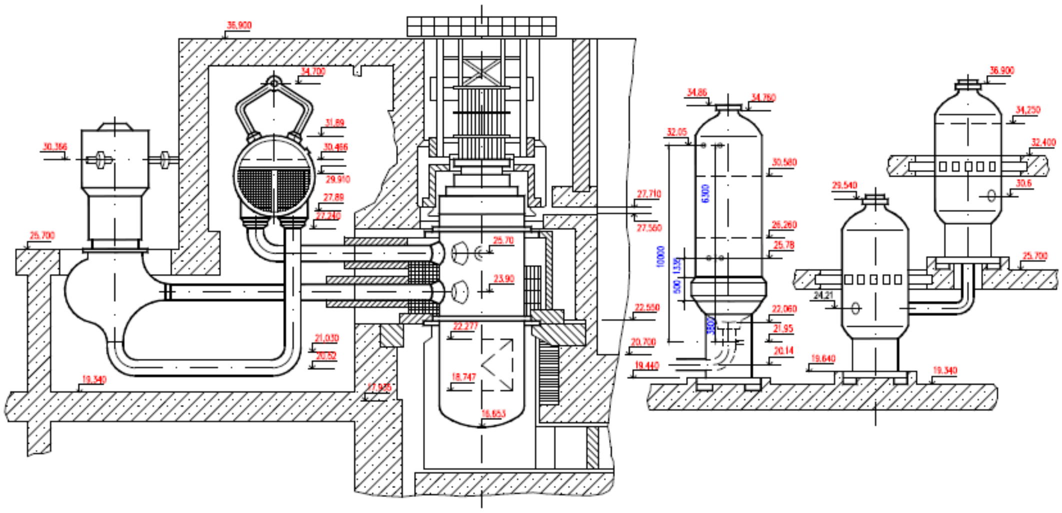

Figure 5: Diagram of main equipment of WWER-1000/320.

RF equipment is housed in the pre-stressed leak-tight reinforced-concrete containment having the shape of a hollow cylinder with a spherical dome and flat bottom. The reinforced concrete wall of the containment is 1.2 m thick in its cylinder-shaped section and 1.1 m thick in its dome section. There is leak-tight 8-mm metal lining on the internal side of the containment.

The spent fuel assemblies unloaded from the core are stored in racks in the spent fuel pools. Before placing for storage, the fuel assemblies are subjected to fuel cladding leak testing. Based on the testing results, a spent fuel assembly is placed either in the rack slots or in a sealed canister. The spent fuel storage system is housed in the reactor compartment provided with all necessary rooms and equipment to receive and store the spent fuel assemblies. The SFP is housed inside the containment and consists of three compartments designed for storage of spent fuel assemblies, and a well which is an area for loading of transport casks with spent fuel assemblies and unloading of fresh fuel casing. Dividing the SFP into three compartments allows for maintenance in one of them while spent fuel assemblies are placed into the remaining two. The well stands separately from the fuel storage area, which permits installation of a fresh fuel casing into the dry well. The SFP is adjacent to the reactor and is connected with upper part of the reactor cavity by a refuelling channel for transport of fuel assemblies. The pool is equipped with spent fuel storage racks.

3.2. Seismic PSA study

3.2.1. Overview

Seismic PSA study for ZNPP Unit 1 was completed (including resolution of regulatory review comments) and approved by State Nuclear Regulatory inspectorate of Ukraine in 2019. Scope of the SPSA include development of Level 1 and Level 2 PSA for two sources of radioactivity – reactor and spent fuel pool. All operational states (POS) are considered in the study: nominal power (POS0), low power (POS1,2,3,4,5,6, 13, 14,15) and shutdown modes (POS7-12), as well as refueling and long-term storage states for SFP. Five earthquake levels (PGA) are modeled:

- Q1 - 0.085 g
- Q2 - 0.17 g
- Q3 - 0.2 g
- Q4 - 0.3 g
- Q5 – 1.45 g.

The SPSA encompasses the following main subtasks:

- Determination of earthquake reoccurrence parameters for source, calculation of earthquake frequencies for specified ground accelerations;
- Plant familiarization and data collection (identification of SEL, analysis of equipment qualification for seismic events, seismic capability walkdown);
- Determination of seismic response of SSCs for input to fragility calculations, fragility calculations for SSCs;
- Analysis of scenarios for selected levels of seismic events (identification of seismically induced initiating events and hazards (internal floods, fires), systems/accident sequence analysis leading to event trees/fault trees modelling);
- Accident sequence quantification and sensitivity analysis (development of PSA models; component and human reliability data re-assessment; quantification of CDF, FDF, LRF from reactor, LRF from spent fuel pool).

3.2.2. Systems and safety functions

The set of VVER-1000/320 safety functions required to prevent core or fuel damage and front-line systems which can perform each function are listed in Table below.

Safety function	System	Plant designation
Reactivity control		
Reactor scram	Reactor scram	AZ
Boron injection	Chemical and volume control system	TK + TB10
Boron injection	High pressure emergency core cooling system	TQ 13/33

Boron injection	High pressure injection system	TQ 14/34
Boron injection	Emergency core flooding system	YT
Primary coolant inventory control		
	Chemical volume and control systems	TK + TB10
	High pressure injection system	TQ 13/33
	Emergency core flooding system (hydroaccumulators)	YT
	Low pressure injection system operation via containment sump	TQ12/32
Secondary heat removal		
SG feeding	Auxiliary feedwater system (AFW)	RL
SG feeding	Emergency feedwater system (EFW)	TX10/30
Secondary pressure maintenance	Steam dump valve to atmosphere (BRU-A)	TX
Secondary pressure maintenance	Steam dump valve to condenser (BRU-K)	RC
Secondary pressure maintenance	SG safety relief valve (SG SRV)	TX
Secondary cooldown	BRU-A	TX
Secondary cooldown	BRU-K	RC
Primary heat removal		
Primary cooling down & decay heat removal	LPIS (taking into account planned cooldown line)	TQ12/32
Primary cooling down & decay heat removal	LPIS from sump	TQ12/32
Primary cooling down & decay heat removal	HPIS from sump	TQ13/33
Primary pressure control		
Primary pressure control	Primary pressure control system (spray into pressurizer by main coolant pump)	YP
Primary pressure control	Primary pressure control system (spray into pressurizer by CVCS)	TK
Primary pressure control	Emergency gas evacuation system	YR
Primary overpressure prevention	Subsystem of steam dumping from PRZ into bubble condenser	YP
Steam generators isolation		
SG steam side isolation	Fast acting steam isolation valve	TX

SG feedwater side isolation	Isolation valves of feed water systems	RL + TX
Power supply		
Power supply	Essential power supply system	DG, BV, BW, BX
SFP cooling		
SFP water level control	Containment spray system	TQ11
SFP cooling	SFP cooling system	TG
SFP water level control	SFP feeding system	TM50
Boron injection	Boron concentrate system	TB30

Table 2: ZNPP safety functions and systems

3.2.3. Seismic equipment list

Development of lists of ZNPP-1 components to determine boundary seismic resistance (SEL) has been performed in /ZNPP 2019/ by the following steps:

- Step 1. Refinement of existent list of components, pipelines, buildings and structures of ZNPP Units 1, 2 for which it is necessary to perform justification of seismic resistance. This list was developed using approach illustrated on Figure 6. The refinement was performed to account for actual conditions of ZNPP and to ensure completeness of the list. The following criteria were applied to include SSC in the list:
  - Seismic failure of SSC can lead to occurrence of initiating event;
  - Seismic failure of SSC lead to degradation of the safety functions, necessary for the NPP safe shutdown and for maintaining it in a stable state;
  - SSC can lead to the occurrence of internal hazard and, as a consequence, to the occurrence of IE or degradation of the safety function necessary to ensure the NPP safe shutdown of the unit;
  - Electrical equipment and instrumentation, which performs supporting functions in relation to the heat and mass transfer equipment involved in the safe shutdown of the unit;
  - Buildings, structures and interfaces that houses elements of components, including instrumentation and control, electrical equipment necessary for safe shutdown of the unit and maintaining it in a safe final state.

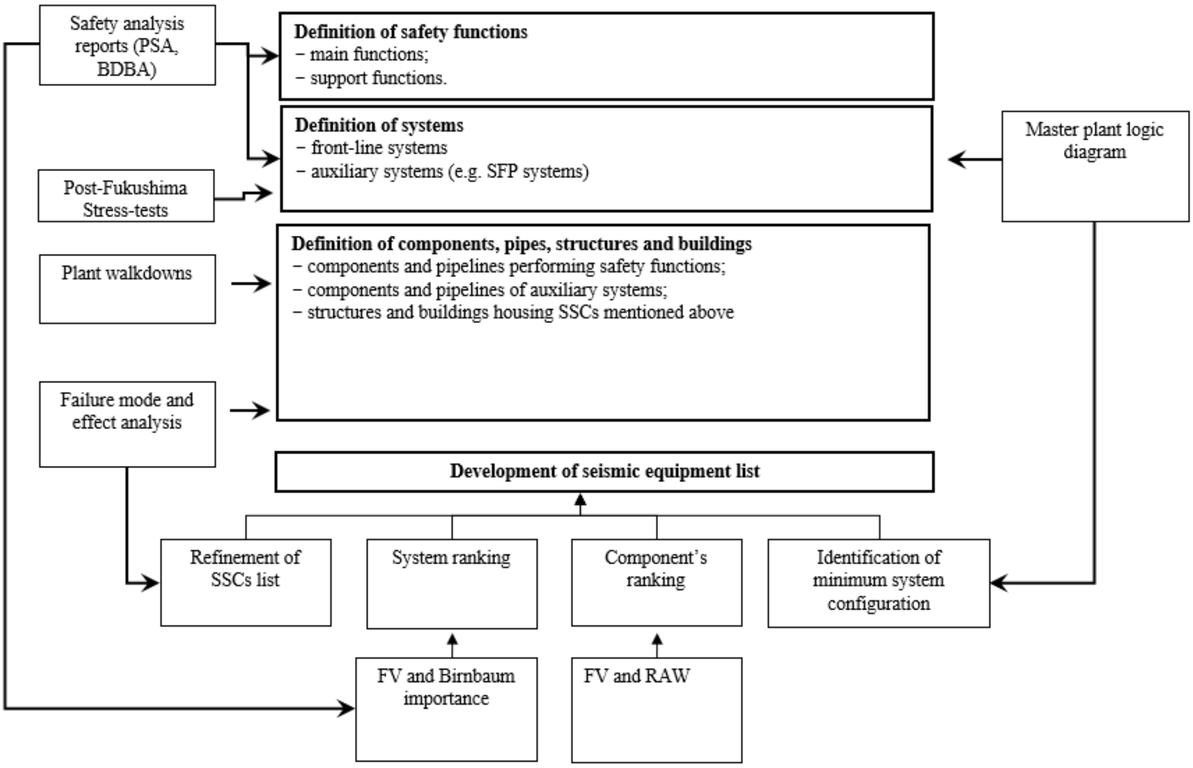

Figure 6: Development of VVER-1000 SSCs list for justification of seismic resistance, based on /NAEK 2012/

- Step 2. Evaluation of the SEL developed at step 1, taking into account components boundaries and assumptions defined during development of full-scope PSA. During evaluation:
  - components (elements) which do not change their position during IE mitigation process, are included in the boundaries of the corresponding pipeline systems, tanks, containers, heat exchangers;
  - components (elements) that are not part of the systems that ensure fulfillment of safety functions, but the failure of which, due to an earthquake, can lead to IE, are included in the corresponding boundaries of pipeline systems, tanks, tanks, heat exchangers;
  - mapping of SSCs with basic events (BE) from full-scope PSA model is performed. BE representing more than one component from the SEL are defined. SSCs from SEL that are not modelled in full-scope PSA, but seismic failure of which can be important for mitigation of accident sequences (e.g., some panels from main control room, sensors, etc.) are identified;
  - re-evaluation of SSCs that were previously screened out from the PSA study.
- Step 3. Compiling of final list of SSCs for SPSA. Quantification of importance measures taking into account such seismic effect as loss of off-site power.

As the result of performing steps 1 through 3, list of ZNPP systems to be modeled in SPSA includes the following systems: emergency core flooding system; low pressure injection system, high pressure injection system, high pressure emergency core cooling system, primary pressure control system, emergency gas evacuation system; ventilation systems; emergency feedwater system; steam dump valves; essential power supply system; essential service water system; containment spray system;

SFP cooling system; instrumentation and control, mobile pump system. Civil structures and facilities affected by seismic events are also included in the SEL.

ZNPP Unit 1 final seismic equipment list contains more than 1.5 thousand items /ZNPP 2019/:

- Heat and mass transfer equipment – 653 items;
- Electric equipment – 111 items;
- Instrumentation and control – 231 items;
- Diesel-generator electrical equipment – 152 items;
- Components not modelled in internal events PSA – 180 items;
- Pipelines – 498 items;
- Civil structures and facilities – 10 items.

It should be noted that the approach used to develop original ZNPP SPSA seismic equipment list is consistent with the approach presented in Section 1.

3.3. Importance measures

ZNPP Unit 1 seismic PSA probabilistic model in SAPHIRE 8 code has been used to re-calculate importance measures for all SSCs, using 1E-12 as truncation value to calculate CDF. Distribution of SSCs by FV and RAW importance measures is illustrated on Figure 7. According to Figure 7, majority of ZNPP Unit 1 SSCs have low significance. About 20 SSCs are ranked as high and very high significance, and about 50 can be considered as medium significance.

Dominant contributors to total seismic CDF ranked by FV, as well by other criteria from Section 2.1 are shown in Table 3.

To check influence of seismic levels on SSCs failures risk distribution, importance measures for individual earthquake levels were also calculated. As example, dominant contributors to seismic CDF and seismic FDF for PGA 0.085 g are shown on Table 4 and

Table 5, respectively.

It should be noted that, as PGA levels increase, importance measures become less reliable due to the nature of seismic PSA cut sets. Presence of high and very high probabilities of SSC fragilities in case of higher PGAs leads to large numbers of cut sets created at higher PGA levels with very similar values. Importance measures for higher PGAs being weighted in favour of the more severe initiating events, such as direct core damage. For this reason, the FV metric in some extent may over-represents the real risk decrease that would be seen by improving the fragility event for the given components for higher ground motions.

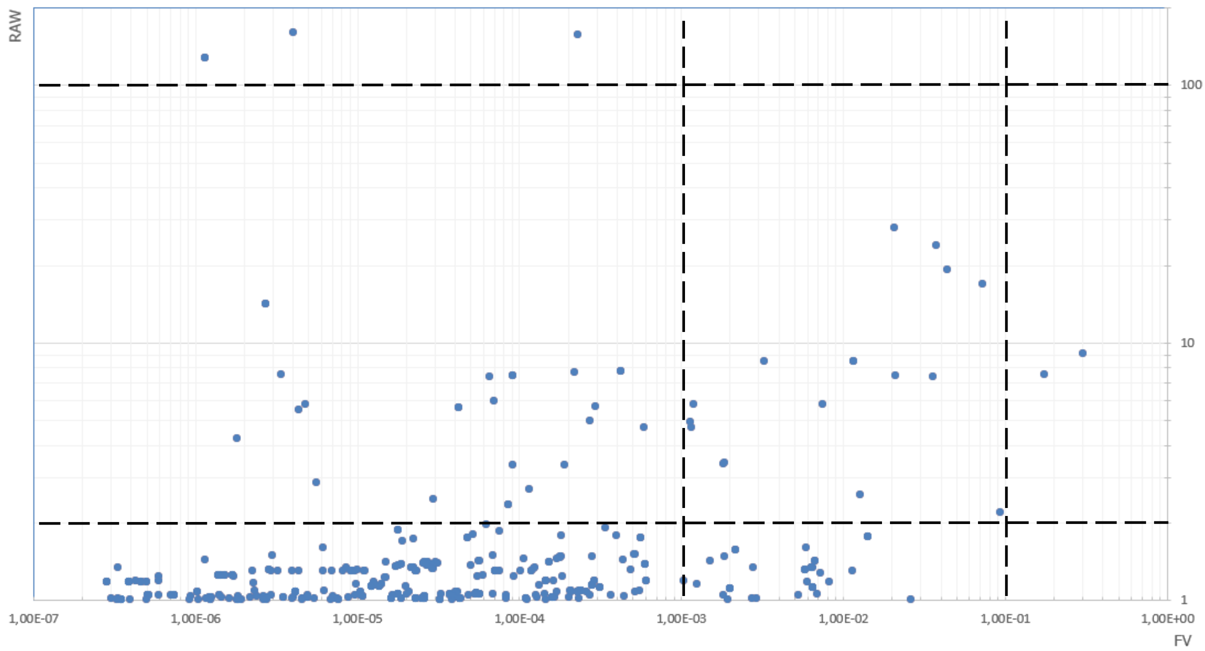

Figure 7: ZNPP Unit 1 SSCs categorization by importance measures

SSC failures	FV	RAW	BI	Rank		
				FV/RAW	FV	BI
Reactor internals cavity (damage at 0.3 g)	3,02E-01	9,08E+00	3,18E-05	High (H)	1	11
Reactor building (damage at 0.3 g)	1,77E-01	7,49E+00	2,67E-05	High (H)	2	18
Diesel generator DG-1	9,41E-02	2,19E+00	4,77E-06	High (E)	3	48
Reactor internals cavity (damage at 0.2 g)	7,25E-02	1,69E+01	5,94E-05	High (E)	4	8
Reactor building (damage at 0.3 g)	4,44E-02	1,91E+01	6,75E-05	High (E)	5	7
Reactor internals cavity (damage at 0.2 g)	3,80E-02	2,37E+01	8,44E-05	High (E)	6	6
Essential power supply transformers 0.4 kV (located at elevation 20.4, damage at 0.3 g)	3,59E-02	7,38E+00	2,38E-05	High (E)	7	23
Reactor building (damage at 1.45 g)	2,63E-02	1,00E+00	9,86E-08	Medium (D)	8	400
Essential power supply busbars 0.4 kV, CV, CW, CX (located at elevation 20.4, damage at 0.3 g)	2,11E-02	7,40E+00	2,38E-05	High (E)	9	22
Reactor building (damage at 0.17 g)	2,07E-02	2,78E+01	9,97E-05	High (E)	10	5
Essential service water discharge pipelines, section 212 (damage at	1,44E-02	1,76E+00	2,87E-06	Medium (D)	11	57

SSC failures	FV	RAW	BI	Rank		
				FV/RAW	FV	BI
0.3 g)						
Essential service water discharge pipelines, section 21541 (damage at 0.3 g)	1,44E-02	1,76E+00	2,87E-06	Medium (D)	12	58
Essential service water drain pipelines, section 224 (damage at 0.3 g)	1,44E-02	1,76E+00	2,87E-06	Medium (D)	13	59
DG building, trains 2 and 3 (damage at 0.3 g)	1,28E-02	2,56E+00	5,83E-06	High (E)	14	44
Microprocessing control units HV, HW, HX (located at elevation 13.2, damage at 0.3 g)	1,17E-02	8,43E+00	2,76E-05	High (E)	15	12
Control and monitoring cabinets (located at elevation 13.2, damage at 0.3 g)	1,17E-02	8,43E+00	2,76E-05	High (E)	16	13
ECCS train 1 pipe, section TQ11-168 (damage at 0.3 g)	1,14E-02	1,30E+00	1,16E-06	Medium (D)	17	133
ECCS train 2 pipe, section TQ21-169 (damage at 0.3 g)	8,22E-03	1,18E+00	7,11E-07	Medium (D)	18	189
Essential power supply cabinets HG10,11, 14,20 etc. located at elevation 13.2 (damage at 0.3 g)	7,49E-03	5,74E+00	1,76E-05	High (E)	19	27
Reactor building (cracks at 0.3 g)	7,25E-03	1,27E+00	1,03E-06	Medium (D)	20	158
Mobile pump for reactor (damage at 0.3 g)	6,90E-03	1,05E+00	2,20E-07	Medium (D)	21	274
DG3 pipelines (damage at 0.3 g)	6,76E-03	1,42E+00	1,58E-06	Medium (D)	22	84
DG2 pipelines (damage at 0.3 g)	6,70E-03	1,42E+00	1,57E-06	Medium (D)	23	85
DG1 pipelines (damage at 0.3 g)	6,52E-03	1,34E+00	1,30E-06	Medium (D)	24	111
ECCS pipe, section YT11-005 (damage at 0.3 g)	6,49E-03	1,12E+00	4,71E-07	Medium (D)	25	227
ECCS pipe, section YT12-006 (damage at 0.3 g)	6,49E-03	1,12E+00	4,71E-07	Medium (D)	26	228
ECCS pipe, section YT13-007 (damage at 0.3 g)	6,49E-03	1,12E+00	4,71E-07	Medium (D)	27	229
ECCS pipe, section YT14-008 (damage at 0.3 g)	6,49E-03	1,12E+00	4,71E-07	Medium (D)	28	230

SSC failures	FV	RAW	BI	Rank		
				FV/RAW	FV	BI
Essential service water pipe, section VF30-213 (damage at 0.3 g)	6,45E-03	1,34E+00	1,29E-06	Medium (D)	29	112
Essential service water pipe, section VF30-216 (damage at 0.3 g)	6,45E-03	1,34E+00	1,29E-06	Medium (D)	30	113
Essential service water pipe, section VF30-225 (damage at 0.3 g)	6,45E-03	1,34E+00	1,29E-06	Medium (D)	31	114
Essential service water spray pools (damage at 0.3 g)	6,04E-03	1,18E+00	6,93E-07	Medium (D)	32	190
Seismic failure of DG-1, DG-3 (damage at 0.3 g)	5,95E-03	1,60E+00	2,24E-06	Medium (D)	33	66
Essential service water pipe, section VF10-211 (damage at 0.3 g)	5,84E-03	1,31E+00	1,16E-06	Medium (D)	34	126
Essential service water pipe, section VF10-214 (damage at 0.3 g)	5,84E-03	1,31E+00	1,16E-06	Medium (D)	35	127
Essential service water pipe, section VF10-223 (damage at 0.3 g)	5,84E-03	1,31E+00	1,16E-06	Medium (D)	36	128
ECCS pipe, section YA10Z01C	5,34E-03	1,04E+00	1,50E-07	Medium (D)	37	296
Sensors (primary pressure, containment pressure) located at elevation 6.6 (damage at 0.3 g)	3,29E-03	8,43E+00	2,76E-05	High (E)	38	14
Test/maintenance unavailability of safety train 2	2,92E-03	1,01E+00	3,26E-08	Medium (D)	39	360
DG trains 1 and 3 building (damage at 0.3 g)	2,82E-03	1,34E+00	1,29E-06	Medium (D)	40	115
Test/maintenance unavailability of safety train 3	2,76E-03	1,01E+00	3,07E-08	Medium (D)	41	361
DG-2 failure to run	2,18E-03	1,56E+00	2,10E-06	Medium (D)	42	68
DG-3 failure to run	2,17E-03	1,56E+00	2,09E-06	Medium (D)	43	69
High pressure injection pipe, section TQ23-062 (damage at 0.3 g)	2,03E-03	1,11E+00	4,05E-07	Medium (D)	44	234
High pressure injection pipe, section TQ23-065 (damage at 0.3 g)	2,03E-03	1,11E+00	4,05E-07	Medium (D)	45	235
High pressure injection pipe, section TQ23-068 (damage at 0.3 g)	2,03E-03	1,11E+00	4,05E-07	Medium (D)	46	236
High pressure injection pipe, section TQ23-080	2,03E-03	1,11E+00	4,05E-07	Medium (D)	47	237

SSC failures	FV	RAW	BI	Rank		
				FV/RAW	FV	BI
(damage at 0.3 g)						
Test/maintenance unavailability of safety train 1	1,95E-03	1,00E+00	2,17E-08	Medium (D)	48	401
ECCS hydroaccumulators check valves located at elevation 23.00 (damage at 0.3g)	1,86E-03	3,41E+00	8,96E-06	High (E)	49	38
DG-1 failure to run	1,85E-03	1,48E+00	1,78E-06	Medium (D)	50	74
ECCS hydroaccumulators check valves located at elevation 19.20 (damage at 0.3g)	1,84E-03	3,38E+00	8,83E-06	High (E)	51	39
ECCS pipe, section YA10Z01H	1,84E-03	1,04E+00	1,42E-07	Medium (D)	52	297
Seismic failure of DG-1, DG-3 (damage at 0.17 g)	1,51E-03	1,41E+00	1,54E-06	Medium (D)	53	87
High pressure injection pump TQ23D01, failure to run	1,26E-03	1,15E+00	5,75E-07	Medium (D)	54	207
Essential power supply transformers located at elevation 0.0 (damage at 0.3g)	1,20E-03	5,73E+00	1,76E-05	High (E)	55	28
Essential power supply transformers 0.4 kV (located at elevation 20.4, damage at 0.2 g)	1,16E-03	4,66E+00	1,36E-05	High (E)	56	35
Essential power supply cabinets HG70-80, located at elevation 20.40 (damage at 0.3 g)	1,15E-03	4,93E+00	1,46E-05	High (E)	57	34
Essential power supply batteries, located at elevation 13.20 (damage at 0.3 g)	1,04E-03	1,19E+00	6,93E-07	Medium (D)	58	178
Essential power supply busbars 6 kV BV, BW, BX, located at elevation 20.40 (damage at 0.3 g)	6,14E-04	1,19E+00	6,93E-07	Low	59	179
Essential power supply direct current buses EE01,02,03, located at elevation 20.40 (damage at 0.3 g)	6,14E-04	1,19E+00	6,93E-07	Low	60	180
Reactor building (damage at 0.085 g)	2,33E-04	1,57E+02	5,77E-04	High (C)	98	2
Control rods failure	4,03E-06	1,60E+02	5,88E-04	High (C)	332	1
Relays 1_HSP_2-RYL-F	1,16E-06	1,27E+02	4,66E-04	High (C)	386	3

SSC failures	FV	RAW	BI	Rank		
				FV/RAW	FV	BI
Relays 2_HSP_2-RYL-F	1,16E-06	1,27E+02	4,66E-04	High (C)	387	4
Common cause failure (clogging) of ECCS heat exchangers	2,76E-06	1,41E+01	4,85E-05	Medium (B)	346	9
Common cause failure (rupture) of ECCS heat exchangers	2,76E-06	1,41E+01	4,85E-05	Medium (B)	347	10

Table 3: ZNPP Unit 1 dominant SSCs

SSC	FV	RAW	BI	Rank		
				FV/RAW	FV	BI
Reactor building (damage at 0.085 g)	9,74E-02	6,49E+04	5,77E-04	High (F)	1	1
Reactor building (damage due to soil liquefaction at 0.085 g)	1,91E-02	6,39E+04	5,68E-04	High (F)	2	2
DG-2 failure to run	6,01E-03	2,55E+00	1,39E-08	High (E)	3	23
DG-3 failure to run	5,62E-03	2,45E+00	1,30E-08	High (E)	4	24
DG-1 failure to run	5,20E-03	2,35E+00	1,20E-08	High (E)	5	25
High pressure injection pump TQ23D01, failure to run	2,50E-03	1,31E+00	2,74E-09	Medium (D)	6	28
Control rods failure	1,33E-03	5,24E+04	4,66E-04	High (F)	7	3
Seismic failure of DG-1, DG-3 (damage at 0.085 g)	9,59E-04	6,92E+01	6,06E-07	Medium (B)	8	9
Common cause failure (clogging) of ECCS heat exchangers	8,65E-04	4,10E+03	3,64E-05	High (C)	9	6
Common cause failure (rupture) of ECCS heat exchangers	8,65E-04	4,10E+03	3,64E-05	High (C)	10	7
High pressure injection pump TQ13D01, failure to run	7,68E-04	1,09E+00	8,42E-10	Low	11	31
High pressure injection pump TQ33D01, failure to run	7,68E-04	1,09E+00	8,42E-10	Low	12	32
Valve control electrical cabinet RTZO DU01	6,86E-04	1,15E+01	9,34E-08	Medium (B)	13	12
Valve control electrical cabinet RTZO DU03	6,86E-04	1,15E+01	9,34E-08	Medium (B)	14	13
Essential power supply cabinet HG10	6,86E-04	1,15E+01	9,34E-08	Medium (B)	15	14
Essential power supply cabinet HG11	6,86E-04	1,15E+01	9,34E-08	Medium (B)	16	15
Essential power supply cabinet HG21	6,86E-04	1,15E+01	9,34E-08	Medium (B)	17	16

Essential power supply cabinet HG30	6,86E-04	1,15E+01	9,34E-08	Medium (B)	18	17
Essential power supply cabinet HG31	6,86E-04	1,15E+01	9,34E-08	Medium (B)	19	18
Essential power supply busbar LV	6,86E-04	1,15E+01	9,34E-08	Medium (B)	20	19
Essential power supply busbar LX	6,86E-04	1,15E+01	9,34E-08	Medium (B)	21	20
Emergency feedwater pump TX10, 20,30, CCF to run	4,88E-04	6,19E+00	4,61E-08	Medium (B)	22	22
Relays 1_HSP_2-RYL-F	4,85E-04	5,24E+04	4,66E-04	High (C)	23	4
Relays 2_HSP_2-RYL-F	4,85E-04	5,24E+04	4,66E-04	High (C)	24	5
Emergency feedwater pump TX10, failure to run	4,37E-04	1,04E+00	3,55E-10	Low	25	33
DG-1 failure to run during 5 hours	3,99E-04	1,39E+00	3,49E-09	Low	26	26
DG-3 failure to run during 5 hours	3,99E-04	1,39E+00	3,49E-09	Low	27	27
ECCS pipe, section TQ11-168 (damage at 0.085 g)	3,94E-04	1,15E+01	9,34E-08	Medium (B)	28	21
DG-2 failure to run during 5 hours	2,65E-04	1,26E+00	2,32E-09	Low	29	29
Control rod drives control circuit YS16-SUP	1,39E-04	1,40E+02	1,24E-06	High (C)	30	8
CCF to open of ECCS hydro accumulators check valves YT11 (12, 13, 14) S03	1,25E-04	1,44E+01	1,19E-07	Medium (B)	31	10
CCF to open of ECCS hydro accumulators check valves YT11 (12, 13, 14)S04	1,25E-04	1,44E+01	1,19E-07	Medium (B)	32	11

Table 4: ZNPP Unit 1 dominant failures of SSCs for reactor (PGA 0.085g)

SSC	FV	RAW	BI	Rank		
				FV/RAW	FV	BI
CCF of DG to run	3,55E-01	4,34E+03	5,86E-04	High (I)	1	2
CCF of essential power supply section breakers BV(WX)02A C-BN02A-CBA-E-ABC	1,68E-01	4,34E+03	5,86E-04	High (I)	2	3
CCF of essential service water check valves to open C-QFN1S0N-CKV-O-ABC	1,65E-01	4,34E+03	5,86E-04	High (I)	3	4
CCF of DG to run during 5 hours	9,31E-02	4,34E+03	5,86E-04	High (F)	4	5
CCF of essential service water pumps to start	6,60E-02	4,34E+03	5,86E-04	High (F)	5	6

C-QFN1D0N-MDP-S-ABC						
CCF of essential power supply DC buses C-EE0N-DCP-F-ABC	6,04E-02	4,34E+03	5,86E-04	High (F)	6	7
CCF of essential power supply circuit breakers C-BN0NA-CBA-O-ABC	1,40E-02	4,34E+03	5,86E-04	High (F)	7	8
CCF of essential power supply transformers C-BNF0N-TRF-F-ABC	9,99E-03	4,34E+03	5,86E-04	High (F)	8	9
DG-1 failure to run GV01-DGN-R	9,98E-03	3,58E+00	3,50E-07	High (E)	9	20
DG-2 failure to run GW01-DGN-R	9,96E-03	3,58E+00	3,49E-07	High (E)	10	21
Essential service water spray ponds (damage at 0.085g)	9,95E-03	4,34E+03	5,86E-04	High (F)	11	10
DG-3 failure to run GX01-DGN-R	9,88E-03	3,56E+00	3,46E-07	High (E)	12	22
CCF of diesel-generators to start C-GN01-DGN-S	5,50E-03	4,34E+03	5,85E-04	High (F)	13	11
CCF of EPS AC buses C-CN0N-ACB-F-ABC	4,16E-03	7,88E+03	1,06E-03	High (F)	14	1
Seismic failure of DG-1, 3 (damage at 0.085 g) EQ-1-DGN13-P0	3,20E-03	2,28E+02	3,07E-05	High (F)	15	18
DG-1 failure to run during 5 hours GW01-5-DGN-R	2,51E-03	3,47E+00	3,34E-07	High (E)	16	23
DG-2 failure to run during 5 hours GV01-5-DGN-R	2,48E-03	3,45E+00	3,30E-07	High (E)	17	24
DG-3 failure to run during 5 hours GX01-5-DGN-R	2,46E-03	3,42E+00	3,28E-07	High (E)	18	25
CCF of EPS batteries C-EA0N-BAT-F-ABC	1,08E-03	4,27E+03	5,77E-04	High (F)	19	12
EPS circuit breaker BW02A-CBA-E	9,00E-04	3,32E+00	3,14E-07	Medium (B)	20	26
CCF of essential service water filters C-QFN1N01-BST-Q-ABC	8,86E-04	4,27E+03	5,77E-04	High (C)	21	13
Failure to open of essential service water valve VF20S05-CKV-O	8,86E-04	3,32E+00	3,14E-07	Medium (B)	22	27
Failure of essential power supply circuit breaker BV02A-CBA-E	8,81E-04	3,28E+00	3,07E-07	Medium (B)	23	28
Failure of essential power supply circuit breaker	8,81E-04	3,28E+00	3,07E-07	Medium (B)	24	29

BX02A-CBA-E						
Failure to open of essential service water valve VF10S05-CKV-O	8,67E-04	3,28E+00	3,07E-07	Medium (B)	25	30
Failure to open of essential service water valve VF30S05-CKV-O	8,67E-04	3,28E+00	3,07E-07	Medium (B)	26	31
CCF of EPS circuit breakers C-BN10A-CBA-K-ABC	7,30E-04	2,39E+02	3,21E-05	High (C)	27	17
CCF of essential service water pump to run C-QFN1D0N-MDP-R-ABC	4,23E-04	4,07E+03	5,50E-04	High (C)	28	14
Failure of DC bus EE01-DCP-F	2,47E-04	2,77E+00	2,39E-07	Medium (B)	29	32
Failure of DC bus EE02-DCP-F	2,47E-04	2,77E+00	2,39E-07	Medium (B)	30	33
Failure of DC bus EE03-DCP-F	2,47E-04	2,77E+00	2,39E-07	Medium (B)	31	34
DG-1 failure to start GV01-DGN-S	1,21E-04	2,59E+00	2,14E-07	Medium (B)	32	35
DG-2 failure to start GW01-DGN-S	1,21E-04	2,59E+00	2,14E-07	Medium (B)	33	36
DG-3 failure to run GX01-DGN-S	1,21E-04	2,59E+00	2,14E-07	Medium (B)	34	37
Reactor building (damage at 0.085 g)	9,29E-05	6,29E+01	8,49E-06	Medium (B)	35	19

Table 5: ZNPP Unit 1 dominant failures of SSCs for spent fuel pool (PGA 0.085g)

Based on importance measures for dominant failures regarding CDF, the following SSCs ranked as high significance can be recommended for Tier 1:

- Reactor internals cavity (combined FV 4.1E-01, max RAW 23.7);
- Reactor building (combined FV 2.7E-01, max RAW 157);
- Diesel-generators 1, 2, 3 (combined FV 1.1E-01, max RAW 2.19);
- Essential power supply transformers 6/0.4 kV (plant designation BU05, 06,07, 26, 27, 28) that provide power to essential power supply busbars CV, CW, CX (combined FV 3.7E-02, max RAW 7.38);
- Essential power supply busbars 0.4 kV (plant designation CV, CW, CX), (combined FV 2.2E-02, max RAW 7.4);
- Diesel-generators buildings (combined FV 1.6E-02, max RAW 2.56);
- Micro processing control units HV, HW, HX (combined FV 1.2E-02, max RAW 8.43);
- Control and monitoring cabinets (plant designation HV063,064,065, 089,090,091), 10,11,14,20, 21, 24, 30, 31,34) located at elevation 13.2 (combined FV 1.2E-02, max RAW 8.43);

- Essential power supply cabinets (plant designation HG10,11,14,20, 21, 24, 30, 31,34) located at elevation 13.2 (combined FV 8.0E-03, max RAW 5.74);
- Sensors (primary pressure, containment pressure) located at elevation 6.6 (combined FV 3.6E-03, max RAW 8.43)
- ECCS hydro accumulators check valves (plant designation YT11S03, YT11S04, YT12S03, YT12S04, YT13S03, YT13S04, YT14S03, YT14S04 (combined FV 4.5E-03, max RAW 3.41)
- Essential power supply transformers (plant designation BU23,24,25) located at elevation 0.0 (combined FV 1.2E-03, max RAW 5.73);
- Essential power supply cabinets HG70-80, located at elevation 20.40 (combined FV 1.2E-03, max RAW 4.93);
- Control rods drives and control circuits (combined FV 4.03E-06, max RAW 160);
- Relays of reactor protection control and Instrumentation system (combined FV 2.2E-06, max RAW 127); plus
- ECCS heat exchangers (plant designation TQ10W01, TQ20W01, TQ30W01), that were ranked as high significant for initial PGA levels.

Regarding SSC required to prevent fuel damage at spent fuel pool, the following SSCs ranked as high significance can be recommended for Tier 1:

- Diesel-generators 1, 2, 3;
- Essential power supply components
  - busbars 6 kV (plant designation BV, BW, BX) and associated section breakers;
  - DC buses (plant designation EE01,02,03) and batteries;
  - busbars 0.4 kV (plant designation CV, CW, CX);
  - transformers (plant designation BVF01 02; BWF01 02; BXF01 02).
- Essential service water components
  - filters,
  - pumps (plant designation QF)
  - check valves on QF pumps discharge;
- Essential service water spray ponds.

Other SSCs should be considered as part of Tier 2 group.

4. Conclusion

The report provides a description of approaches used for identification of seismic equipment list to support development of seismic PSA. It is based on survey of guidelines available in different literature documents, as well on authors experience in developing PSA studies and defining risk significance of SSCs for different PSA applications. It presents screening guidelines to select systems, structures and components for further detailed plant specific and/or generic fragility analysis.

Risk-informed approach is proposed to screen out risk insignificant SSCs, and to select SSCs for detailed fragility evaluations. Qualitative and quantitative screening criteria are stated, as well as important factors that should be accounted for during development and adjustment of seismic equipment list.

Evaluation of importance measures for Zaporizhzhia NPP Unit 1, which is chosen as the METIS case study, was performed. 16 SSCs groups important to prevent core damage at reactor facility and 9 SSCs groups important to prevent fuel damage at spent fuel pool are preliminary proposed for inclusion into Tier 1. These lists will be used as basis for further selection of SSCs for detailed fragility evaluation under METIS project, depending on availability and completeness of plant-specific documentation and data needed for fragility analysis.

5. Bibliography

/APSA 2017/ A. Wielenberg, C. Hasnaoui, L. Burgazzi, E. Cazzoli, P. Jan, S. La Rovere, H. Löffler, T. Siklóssy, J. Vitazkova, E. Raimond. Risk Metrics and Measures for an Extended PSA. Technical report ASAMPSA_E / WP30 / D30.7 / 2017-31 volume 3 (2017)

/APSA 2017a/ 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). Technical report ASAMPSA_E ASAMPSA_E/ WP22/ D50.15/ 2017-33/volume 2 (2017)

/ASME 2007/ American National Standard External-Events PRA Methodology. ANSI/ANS-58.21-2007. The American Society of Mechanical Engineers (2007).

/ASME 2013/ Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. Case for ASME/ANS RA-Sb–2013. The American Society of Mechanical Engineers, Two Park Avenue, New York (2013).

/EPRI 1991/ Reed, J W, Kennedy, R P, Buttemer, D R, Idriss, I M, Moore, D P, Barr, T, Wooten, K D, & Smith, J E. A methodology for assessment of nuclear power plant seismic margin. EPRI NP-6041-SL (1991).

/EPRI 1991a/ Generic Seismic Ruggedness of Power Plant Equipment (Revision 1). EPRI NP-5223-SL (1991).

/EPRI 1995/ Generic Seismic Technical Evaluations of Replacement Items for Nuclear Power Plants. EPRI Technical report 104871 (1995)

/EPRI 2002/ Seismic Fragility Application Guide, EPRI Report 1002988, Palo Alto, CA USA, (December 2002)

/EPRI 2003/ Seismic Probabilistic Risk Assessment Implementation Guide, EPRI Technical Report 1002989, Palo Alto, CA (2003).

/EPRI 2012/ Seismic Evaluation Guidance. Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, EPRI Report 1025287, Palo Alto, CA (2012).

/EPRI 2013/ Seismic Probabilistic Risk Assessment Implementation Guide, EPRI Technical Report 3002000709, Palo Alto, CA (2013).

/IAEA 1993/ International Atomic Energy Agency, Probabilistic safety assessment for seismic events, IAEA TECDOC-724, IAEA, Vienna (1993).

/IAEA 2009/ International Atomic Energy Agency, Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13, IAEA, Vienna (2009).

/IAEA 2010/ International Atomic Energy Agency, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3, IAEA, Vienna (2010).

/IAEA 2010a/ International Atomic Energy Agency, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4, IAEA, Vienna (2010).

/IAEA 2020/ International Atomic Energy Agency, IAEA, "Methodologies For Seismic Safety Evaluation of Existing Nuclear Installations", IAEA Safety Report Series No. 103, IAEA, Vienna (2020).

/IAEA 2020a/ International Atomic Energy Agency, IAEA, "Probabilistic Safety Assessment for Seismic Events", IAEA TECDOC 1937 , IAEA, Vienna (2020).

/DOE 1997/ Seismic Evaluation Procedure for Equipment in U.S. Department of Energy Facilities DOE/EH-0545

/SQUG 2001/ Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment, Revision 3A. Seismic Qualification Utility Group (SQUG): December 2001. ADAMS No. ML040560263

/NAEK 2012/ List of components, pipelines, buildings and structures for which it is necessary to perform justification of resistance on seismic impact of seismic resistance on

NPPs with reactor facility V-320. Final report (in Russian). NAEK Energoatom (2012).

/NEA 1998/ State-Of-The-Art Report On The Current Status Of Methodologies For Seismic PSA. Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, NEA/CSNI/R(97)22 (1998).

/NEA 2020/ Use and Development of Probabilistic Safety Assessments at Nuclear Facilities. Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, NEA/CSNI/R(2019)10 (2020).

/NRC 1985/ R. J. Budnitz, P. J. Amico, C. A. Corneli, W. J. Hall, R. P. Kennedy, J. W. Reed, M. Shinozuka. An Approach to the Quantification of Seismic Margins in Nuclear Power Plants, NUREG/CR 4334, US Nuclear Regulatory Commission (1985)

/NRC 2002/ An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis, RG-1.174, US Nuclear Regulatory Commission (2002)

/NRC 2012/ R. J. Budnitz, P. J. Amico, C. A. Corneli, W. J. Hall, R. P. Kennedy, J. W. Reed, M. Shinozuka. Criterion for Capacity-Based Selection of Structures, Systems and Components for Fragility Analyses in a Seismic Risk-based Evaluation, US Nuclear Regulatory Commission (2012)

/OECD 2007/ Specialist Meeting On The Seismic Probabilistic Safety Assessment Of Nuclear Facilities, SEOGWIPO ICC, Jeju Island, Korea, 6-8 November 2006, NEA/CSNI/R(2007)14 (2007)

/ZNPP 2019/ Zaporizhzhie NPP. Probabilistic safety assessment of seismic impact on power unit 1. Final report (in Russian). AT75/208-13.1000.ОД.4 (2019)

/WEST 1991/ Ravindra, M.K. et. al., "Seismic Fragilities of Selected Components in Loviisa Nuclear Power Plant", Draft report prepared for Imatran Voima Oy by EQE Engineering, Inc. and Westinghouse Energy Systems International (1991)

/WEST 1991a/ Monette, P., Baltus, R., Yanev, P., Campbell, R., Stigter, H., 1991, "Seismic Ruggedness Evaluation of Kozloduy WER Units 1 to 4, Rev.1, Report prepared by Westinghouse Energy Systems International and EQE Engineering for the IAEA.