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Continued on next page

REVISION HISTORY (Cont'd)

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06	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes	17-05-2012

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SUB-CHAPTER 15.6 – SEISMIC MARGIN ASSESSMENT (SMA)

1. OBJECTIVE OF THE SEISMIC MARGIN ASSESSMENT (SMA) FOR THE UK EPR

Sub-chapter 13.1 of the PCSR describes the seismic design principles for the UK EPR. The EPR design objective is that, following an earthquake, the safety functions needed to return the plant to a safe shutdown state should not be unacceptably affected. Design of new reactors to withstand seismic events is necessary to comply with HSE Safety Assessment Principles (SAPs) [Ref-1] and Technical Assessment Guides (TAGs) [Ref-2] [Ref-3] and the EUR requirements [Ref-4].

The EPR Nuclear Island buildings and equipment are designed to withstand a Design Basis Earthquake (DBE), which is defined using a set of standard ground motion spectra (EUR 0.25g ground spectrum defined for six different ground conditions). Using these standard spectra, seismic analyses are carried out to calculate spectra for the design and qualification of safety related Structures, Systems and Components (SSC), within the Nuclear Island. For site specific structures outside the Nuclear Island, which are safety classified, such as the Pumping Station, seismic motion spectra that are specific to the particular plant are defined. For EPRs sited in the UK, the DBE will be shown to bound seismic events with a recurrence frequency of one in 10,000 years, to comply with requirements of HSE Safety Assessment Principles.

HSE Safety Assessment Principles and EUR design principles require an additional demonstration that the reactor design is robust against events more severe than that assumed for the plant design, so that no ‘cliff edges’ exist beyond the design basis. The purpose of the current Seismic Margin Assessment (SMA) is to demonstrate that this requirement is achieved so that safe shutdown can be achieved in seismic events that exceed the DBE by a certain amount.

The seismic margin of the UK EPR is assessed by a PSA-based SMA, following a methodology developed by the US NRC [Ref-5]. This approach uses the PSA model to identify combinations of seismic equipment failures which could result in core damage, as well as combinations of seismic failures, random failures and human errors which contribute significantly to seismic risk. By identifying which equipment items and structures are of critical importance in seismic events, the analysis approach ensures that vulnerabilities in the design are identified allowing them to be corrected if necessary, thus helping ensure that the seismic risk is ALARP.

EUR principles require that the plant should be able to withstand an earthquake with a horizontal Peak Ground Acceleration[1] (PGA) which is 40% above the DBE level (i.e. 0.35g PGA = 0.25g PGA DBE multiplied by 1.4). For the current SMA carried out for the UK EPR a more conservative PGA target of 1.6 times DBE (=0.4g PGA) is adopted for the Seismic Margin Earthquake (SME).

The detailed PSA-based SMA is performed for at-power states. A simplified approach is used for shutdown states. Also, at this first stage, internal hazards that might be caused by a seismic event, such as fire or flooding, are not analysed in detail and are not included in the PSA model supporting the SMA.

[1] Peak Ground Acceleration refers to Zero Period Acceleration

2. GENERAL METHODOLOGY

This section summarises the methodology of the PSA-based SMA. The methodology involves the four steps described in sections 2.1 to 2.4 below:

2.1. SEISMIC HAZARD ANALYSIS

The purpose of the SMA is to show that the SSCs critical to achieving a safe shutdown state following an earthquake are designed with large safety margins so that they have a low probability of failure in the Seismic Margin Earthquake (SME), which has a PGA of 1.6 times that assumed for the DBE. The first step in this process is to define the ground motion spectrum for calculating the seismic capacities (fragilities) of the SSCs. The ground motion spectrum is a characteristic of the EPR site in question.

For the purposes of the GDA, the free field ground motion spectra used as input data for the estimation of seismic capacities of equipment and structures, are a bounding ‘hard’ and ‘soft’ site spectra derived by enveloping Uniform Risk Spectra (URS) for prospective UK new build sites. Derivation of the bounding spectra is described in section 3 of this sub-chapter.

2.2. IDENTIFICATION OF THE SEISMIC EQUIPMENT LIST AND SEISMIC FRAGILITY EVALUATION

To perform the SMA, it is necessary to produce a Seismic Equipment List (SEL) containing the SSCs whose seismic capacities need to be evaluated for the SMA. The seismic fragility analysis for these SSCs is then performed.

The SEL for the UK EPR SMA is developed using expert judgement in combination with the Level 1 PSA model. The use of the PSA model to identify critical combinations of component failures serves to confirm the completeness of the SEL. Derivation of the SEL is presented in section 4.2 of this sub-chapter.

The fragility assessment of the SSC items in the SEL evaluates the PGA at which their response will exceed a threshold of acceptability for the characteristic motion spectrum adopted. As noted above, the motion spectrum depends on the ground conditions.

The fragility assessment of the SSCs considers the capacity to withstand ground motion of each component and its associated uncertainties. The capacity is defined as the free field PGA value for which the seismic response at the component location exceeds the component resistance capacity, resulting in the probability of failure of the SSC, i.e. the probability that the response exceeds a defined threshold. The PGA capacity of the SSCs is estimated using information on the plant design and ground parameters, test data from SSC qualification and fragility tests, data from generic seismic tests, earthquake experience results, material property data, etc. Where identical components occur in different redundant trains, the seismic capacity of all the components is conservatively set to that of the most vulnerable component, taking no benefit for the redundancy of the system. Similarly the seismic capacity of electrical cables is assumed to be the capacity of the cable tray anchorages, which is conservatively set to the capacity at the most seismically vulnerable anchorage location.

Because of the large number of variables involved in the estimation of the seismic capacity, the fragility is represented by a family of fragility curves with a probability assigned to each curve to reflect uncertainties. The slope of the curves represents the inherent randomness in seismic loading and the resulting component failure probability.

The family of fragility curves is described by three parameters:

- A_m = median PGA capacity
- β_R = logarithmic standard deviation for randomness
- β_U = logarithmic standard deviation for uncertainty

For the SMA, a HCLPF (High Confidence of Low Probability of Failure) capacity is derived for each significant SSC considered [Ref-3]. The HCLPF capacity is the maximum PGA value, for the assumed ground motion spectrum, at which there would be a 95% confidence that the probability of failure would be less than 5%. The HCLPF capacity is computed from the fragility parameters as follows:

$$HCLPF = A_m \exp[-1.65(\beta_R + \beta_U)]$$

For the purpose of GDA, fragility data for the UK EPR are obtained from the following sources:

- Design and qualification data from FA3 or OL3 EPR studies when applicable,
- Design criteria and qualification procedures applicable to the UK EPR,
- Results from generic databases and the literature,
- Expert judgement.

Section 5 of this sub-chapter presents the HCLPF capacity values used in the PSA based SMA and calculated in the documents which present the seismic fragilities of structures and equipment [Ref-1] and of primary equipment [Ref-2].

Note that for an existing plant, a walkdown would be carried out to identify and screen out components with high seismic capacities, in order to reduce the scope of the detailed fragility analysis required. The walkdown would confirm the quality of seismic anchorages (i.e. confirm that there are no inadequate anchorages) and also confirm the absence of spatial interactions (i.e. exclude seismic failures of non-classified SSCs that could impact on SSCs on the SEL). As the current SMA is performed before construction, the SEL is based on design information from FA3 EPR and typical data for plant items identified as important in the PSA analysis. A confirmatory plant walkdown will be performed once the plant is built and equipment fully installed to confirm the fragility assumptions for plant items on the SEL are correct (see section 5.2).

2.3. SYSTEM/ACCIDENT SEQUENCE ANALYSIS

After the seismic capacities of the SSCs on the SEL have been evaluated, event tree analysis is used to determine the failure combinations that are likely to make a dominant contribution to the risk of core damage due to seismic events. Derivation of the seismic event sequences considered in the SMA is described in detail in section 4 of this sub-chapter and is summarised below.

Seismic initiating events are determined from the experience of past seismic PSAs and a review of the internal events of the level 1 PSA. The objective of this review is to determine, through the analysis of the seismic capacities of the systems involved in the initiating events, which ones need to be included and analysed in the seismic PSA model. Structures and other passive components that are typically not included in the internal events PSA must also be considered, particularly those that could lead directly to core damage or activity release.

In the SMA, the initiating event is not evaluated considering the seismic hazard curve related to a specific site, but the accident sequence is studied on the basis that the seismic induced initiating event occurs: the frequency of occurrence is set to 1.0 in the model.

Four event trees types are identified and analysed in the SMA:

- Event trees where the initiator occurrence leads directly to core damage. These initiators correspond to seismically induced failure of a critical SSC whose failure could lead directly to core melt. An example of such an SSC is the polar crane in the reactor building, failure of which could lead to RPV failure by direct impact. Another example is reactor building structure, the HCLPF capacity of which is expected to be high enough such as that if it would fail, it can be considered that initiating event would occur and mitigating systems would also fail due to the seismic event.

- Seismically induced LOOP event tree. It is assumed that loss of off-site power occurs with unit probability following the SME event.

- Seismically induced small LOCA event tree, caused by failure of small pipework or the Reactor Coolant Pump seals.

- Event tree for ATWS caused by the failure of Control Rods to insert following the seismically induced LOOP, due to rod blockage or I&C failure.

The first event tree type comprises only the seismic initiator and the failure of the critical SSC.

In the other three event tree types, the core damage frequency is composed of multiple event combinations, consisting of the seismic initiator, seismically induced equipment failures, human action failures and random equipment failures.

The Level 1 PSA model described in PCSR Sub-chapter 15.1 is used as a basis to analyse these ‘seismic event trees’. To perform the analysis, the Level 1 PSA fault trees for individual protection systems (such as the Emergency Feedwater System (ASG [EFWS]), Low Head Safety Injection (LHSI), etc…) are modified to include a seismic failure mode (system failure or operator action failure). This is done by introducing a seismic failure basic event in the fault tree, which has an associated seismic HCLPF capacity for the system. Each system is then assigned an HCLPF capacity based on the lowest capacity determined for components in the system train (MIN). The seismic failure is inserted into the system fault tree as a basic event with a label showing the system HCLPF capacity (for instance in the fault tree “EFWS01001A FCD” modelling partial or total loss of ASG [EFWS] train one, the basic event “SEIS EFWS - 0.63g” is introduced, see section 6 of this sub-chapter).

In addition to identifying the SSCs which are likely to dominate the seismic risk, it is also desirable to identify critical operator actions which are relied upon to mitigate the event. This is done by assigning all operator actions in the event tree a probability of failure of 1.0 per demand, to ensure that they appear at the top of the list of cutsets when the event tree is evaluated.

The Level 1 PSA seismic event trees are reviewed and modified, where necessary, to take into account reduced capability of the mitigation systems to achieve a safe and stable state in the event of an earthquake. This is done on the basis of the evaluation of systems and safety functions. Special consideration is given to passive equipment to ensure that the Seismic Equipment List (SEL) is complete. Pessimistic assumptions are made concerning seismic impact modelling (e.g. offsite power recovery is not modelled).

A detailed SMA is performed for at-power states and a simplified approach is taken for shutdown states. With the exception of critical dropped load hazards such as failure of the polar crane, internal hazards that could be induced by the seismic event, such as fire or flooding, are not included in the SMA model. This is because the detailed design of the UK EPR against internal hazard has not been fully developed within GDA, and it is anyway considered unlikely that a seismically induced fire, flood or dropped load, would be able to cause widespread failures over all four trains of the safeguard systems (see section 4.5).

The final step in the SMA analysis is to evaluate the individual event trees to determine the seismic capacity of the entire plant.

2.4. SEISMIC MARGIN EVALUATION FOR PLANT

For each seismic event tree the frequency of the seismic initiating event is set to 1.0 /yr. The seismic event trees are then evaluated using the RiskSpectrum code to express the core damage frequency as a sum of cutsets involving component unavailabilites (due to seismic and random failures and human action failures).

The cutset list derived for the event trees generally contains single element cutsets (consisting of a single seismic failure only), cutsets that contain multiple seismic failures and cutsets that contain combinations of seismic failures, random failures and/or human action failures.

The HCLPF capacity for a system is evaluated using the MIN-MAX method, which applies the following rules:

- The equivalent HCLPF capacity for components operating under OR logic in the fault tree is taken as the lowest HCLPF capacity corresponding to the weakest link.
- The equivalent HCLPF capacity for the simultaneous failure of components operating under AND logic in the fault tree is taken as the highest HCLPF capacity value.
- This seismic failure basic event is assigned to all system trains as a common cause failure since the equipment is identical and located in similar structures and locations. It is generally accepted that this simplifying assumption is strongly conservative, compared to more realistic modelling of the dependencies between seismic failures [Ref-1].

All the seismic failure basic events (system failure or operator action failure) are assigned an arbitrary high failure probability, so that they appear at the top of the list of cutsets when the event tree is evaluated.

To determine the HCLPF capacity for the plant, the MIN-MAX approach is applied to the cutset list. For each cutset containing only seismic failures, the highest HCLPF capacity of any SSC in the cutset is assigned to the cutset as a whole (MAX). The lowest HCLPF capacity is then selected for all the cutsets in the summation (MIN). For mixed cutsets (seismic and random failures or operator failures), the cutset HCLPF capacity is actually higher than the seismic failure part of the cutset. These cutsets are qualitatively evaluated to identify plant vulnerabilities but are not used to determine the seismic margin of the plant. Similarly cutsets containing no seismic failures (i.e. containing only equipment random failures and operator failures) are also evaluated qualitatively, but are not used to determine the seismic margin of the plant.

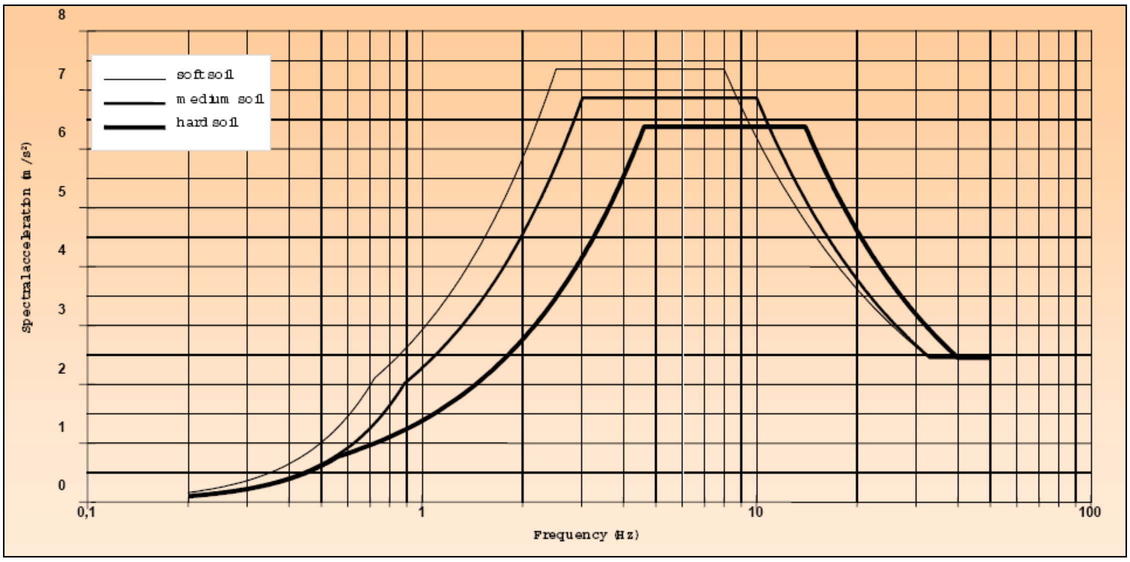
The HCLPF capacity of the plant is the lowest HCLPF capacity of all the cutsets generated by the event tree analysis. The seismic margin assessment consists of comparing the HCLPF capacity with the PGA for the SME. As the plant HCLPF capacity exceeds the SME PGA by a substantial margin, it is concluded that core damage will be avoided at earthquakes beyond the DBE, meeting EUR requirements for an SMA.

The Seismic Margin Evaluation is described in section 6 of this sub-chapter.

3. SEISMIC HAZARD ANALYSIS

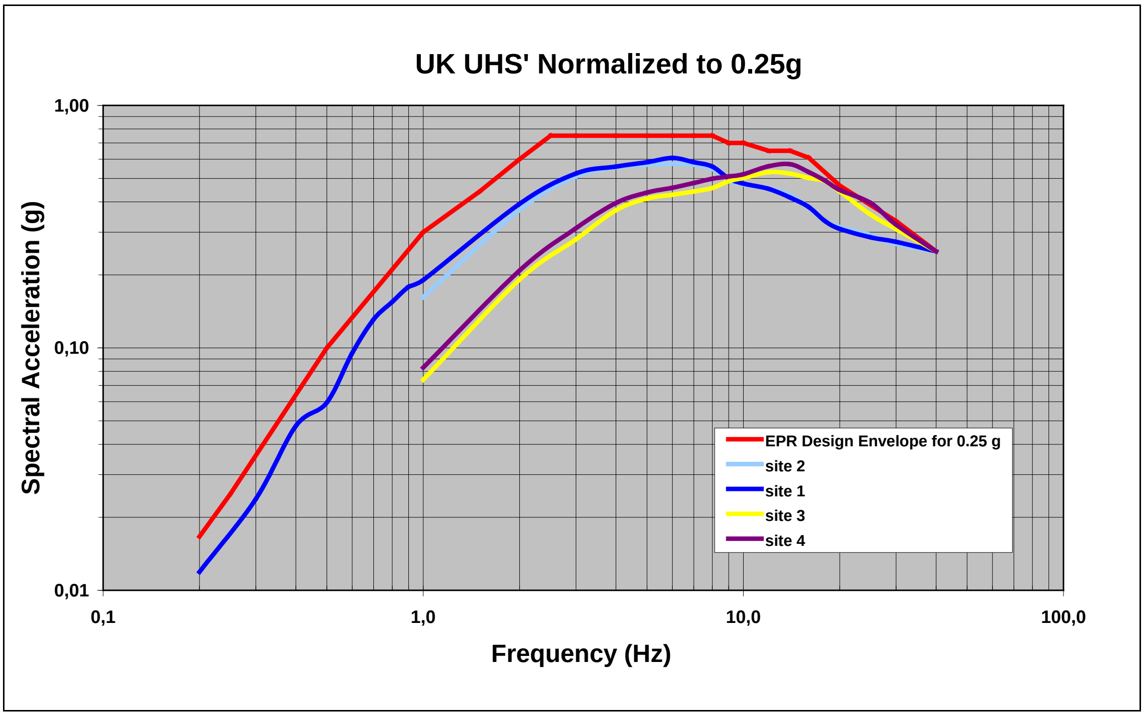
Design Basis Earthquake

Based on the Seismic Design Motions defined in Chapter 13, the UK EPR is designed for the enveloping response from the 5 soil/rock cases (SA, MA, MB, MC, and HA) and a specific Flamanville case (HF). The input ground response spectrum is represented by one of three standard spectra (i.e. the design basis horizontal ground motion spectra EURH, EURM and EURS shown on the figure below) normalised to 0.25g. Depending on the soil type, one of the three standard spectra is used to define the applicable Design Basis Earthquake (DBE) for a particular site. The vertical motion spectra are defined as being 2/3 of the horizontal motion spectra.



Median Ground Response Spectrum

A median spectrum shape for evaluating the seismic fragilities must be selected. For the UK EPR, at the GDA stage, ground response spectra provided by HSE for four potential UK sites were considered. These spectra were Uniform Risk Spectra (URS) for the four sites, as shown in the figure below (based on HSE report [Ref-1], normalised to 0.25g PGA): they correspond to a seismic event with an annual frequency of exceedance of 1.0E-4/yr. A review of these spectra led to the choice of Site 2 (Soil Site) and Site 4 (Rock Site) as the most representative of future UK EPR sites. The seismic fragilities were therefore calculated on the basis of these two spectra.



Reference Ground Motion Parameter

The reference ground motion parameter used in this study is the peak ground acceleration (PGA) associated with the appropriate median spectral shape. The HCLPF capacity of buildings and equipment is thus calculated in terms of PGA based on both the Site 2 and Site 4 spectra.

4. SEISMIC SYSTEMS/ACCIDENT SEQUENCE ANALYSIS

This section describes the seismic PSA model and the derivation of the seismic equipment list (SEL) (equipment whose survival is important to achieving success state following a seismic event). The internal events PSA model for power operation provides the basis for development of the seismic PSA model. The seismic model thus includes random failures and human errors modelled in the internal events PSA. Consideration of radiological releases and low power and shutdown states in developing the SEL is described in section 4.2 of this sub-chapter.

4.1. SEISMIC PSA MODEL DEVELOPMENT

This section describes how the at-power level 1 PSA model for internal events is used to develop the seismic PSA model and the initial SEL (see section 4.2 of this sub-chapter). The containment response and the effect of shutdown states are considered in sections 4.3 and 4.4 of this sub-chapter. The effect of seismically induced internal hazards on the SMA is described in section 4.5 of this sub-chapter.

4.1.1. Seismic Initiating Events

The review summarised below was performed to select the representative initiating events to be studied in the seismic PSA model:

- Transients

  The most likely seismic failures are those associated with non-safety classified equipment (non-seismic category). Previous PSAs have shown that loss of the offsite grid is the most important equipment failure in this category; this failure has the greatest impact since it results in unavailability of all non-safety equipment powered by normal AC supplies. It also results in a challenge to the emergency diesels supplying the safeguard trains. As with previous seismic PSAs, Loss of Offsite Power (LOOP) is therefore included as a key initiating event, and is included in the SEL (see section 4.2 of this sub-chapter). The HCLPF capacity of the offsite grid is expected to be lower than the SME. Although its value is not required for the SMA (since the seismic risk is not being quantified at this stage) it is included for completeness.

  It is noted that lower level earthquakes that do not cause failure of offsite power could lead to a plant trip or shutdown in which non-safety equipment remains available. As the frequencies of such events are lower than those of the reactor trip and turbine trip initiating events already modelled in the PSA, the risk from these events is considered as insignificant.

- Supporting systems

  Apart from LOOP described above, failures of other supporting systems do not need to be modelled as initiating events. These systems and their failure modes are included in the seismic PSA model and in the SEL because such systems are required to mitigate the LOOP initiating event. As a result, the fragilities of these systems will need to be high enough to withstand the SME. Since the earthquake itself can be considered as the initiating event (rather than the system failures), and the supporting systems are accounted for in the model, there is no need to treat these failures as explicit initiating events.

- LOCA

  Major equipment such as reactor vessel supports, reactor coolant pump supports, steam generator supports, pressuriser, and reactor coolant system piping are in the SEL because of their potential to cause a beyond design basis LOCA involving multiple major passive failures. The seismic capacity of these components is expected to be so high that if they were to fail, there would be a high conditional probability that mitigating equipment would also fail. Therefore in the current study it is assumed conservatively that a failure of these components would lead directly to core damage.

To ensure completeness, and address uncertainties regarding the possibility of RCP [RCS] leaks in very large earthquakes (due perhaps to multiple failure of small diameter connected pipes), a seismic small LOCA (S SLOCA) initiating event is included in the seismic PSA model. This ensures that the SEL includes equipment needed to mitigate LOCAs that may not be needed for transients. It is assumed that the seismic capacity of the RCP [RCS] with respect to medium and large LOCAs is higher than the SME. However even if such failures were to occur, no additional safety equipment would be required to mitigate these events, that is not already included in the S LOCA and LOOP event tree models.

- <u>Steam Generator Tube Rupture (SGTR)</u>

  Steam generator tubes are expected to have a seismic capacity greater than the SME, and a seismically induced SGTR is judged not to be credible. Thus, the SG tubes are not included on the SEL. Even if small leaks were to occur, the systems required for mitigation would be those already identified in the LOOP and SLOCA event trees, with the possible exception of the steam generator isolation valves and the “high level SG” and “high activity” signals. SG isolation valves such as feedwater lines and steam line isolation valves are included in the SEL to give additional confidence in the mitigation capability. Also, the above “signal” components are included.

- <u>Secondary Breaks</u>

  Secondary coolant components such as steam generators and steam and feedwater pipework are included in the SEL. As described above for LOCA, the capacity of these components is expected to be high enough so that if they were to fail, there would be a high conditional probability that mitigation equipment will also fail. Therefore these components are presently modelled assuming that their failure would lead directly to core damage. Leaks on the secondary side are judged likely to have only a minor impact on mitigation equipment. The Feedwater Isolation Valves and Main Steam Isolation Valves (VIV [MSIV]) are included in the SEL to provide additional confidence in the isolation capability of piping in the turbine building. The plant arrangement is such that leaks in the secondary piping areas will not impact mitigating equipment (e.g. Emergency Feedwater System (ASG [EFWS]) and Safety Injection System (RIS [SIS]) and the mitigation systems are the same as those in the LOOP and SLOCA event trees, except for the feedwater lines and steam line isolation valves which are included in the SEL. Also, steam generator pressure signals are included in the SEL.

- <u>Structures</u>

  A failure of key structures would be likely to cause an initiating event and could result in enough plant damage to cause core damage. The seismic capacity of structures is expected to be very high so that if they did fail, there would be a high conditional probability that mitigation equipment would also fail. Thus, failures of the key structures are identified as initiating events leading directly to core damage.

Based on this review, the seismic PSA model used for the SMA is reduced to event trees for the following initiating events only:

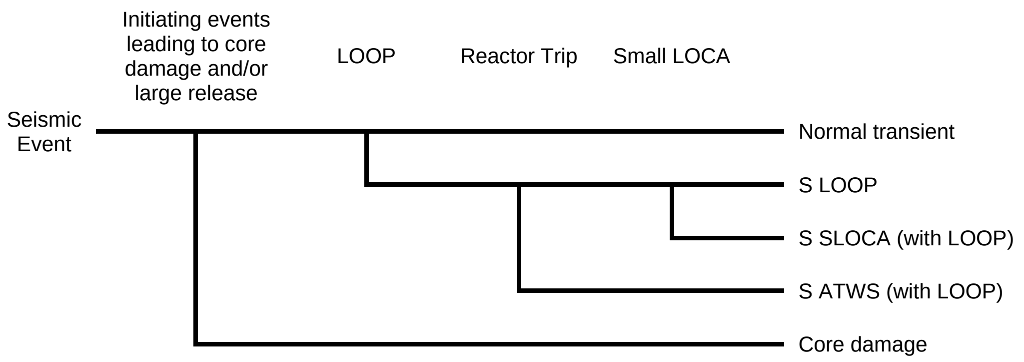
<u>Initiating events leading directly to core damage:</u> Failure of a number of structures, systems and components, which have a high seismic capacity, is assumed to lead to core damage (or a high conditional probability of core damage given their severe consequences).

Seismic Loss of Offsite Power (S LOOP): Seismically induced loss of off site power in power states is the most likely consequential event other than reactor or turbine trip. It is modelled in the event tree analysis to identify the functions, systems and components required to mitigate the event.

Seismic Small LOCA (S SLOCA): Seismically induced small LOCA is included for completeness and to address uncertainty with regard to a large earthquake potentially causing Reactor Coolant System (RCP) [RCS] leakage. This ensures that the seismic model considers such an event. Any equipment required to mitigate the S LOCA not included in the S LOOP model is added to the SEL.

Seismically Induced Anticipated Transient Without Scram (S ATWS): Failure of the reactor trip following a LOOP initiating event would lead to an ATWS scenario. An ATWS following a seismic LOOP is modelled in the event tree analysis, since it requires specific systems and components for mitigation. Equipment required to mitigate ATWS sequences in the level 1 PSA is added to the SEL.

These initiating events can be represented on the following generalised event tree for seismic failure scenarios:



The event trees representing the plant response to these initiating events are discussed in sections 4.1.3 to 4.1.5 of this sub-chapter.

4.1.2. Plant Response and Review of Mitigation Systems

The accident response analysis and the level 1 PSA model for internal events developed for the UK EPR in the framework of the GDA were used to develop the seismic PSA model for SMA and the SEL.

The LOOP scenario was first adapted for the seismic model. Since this model includes the possibility of Reactor Coolant Pumps seal LOCA, the LOOP accident response model also includes the systems and models required for mitigation of small LOCAs.

As described in section 4.1.1 of this sub-chapter, LOOP is the most likely initiating event to be induced by a seismic event and is thus a key transient. Recovery of offsite power after the seismic event is not considered. As a result, a major simplifying assumption can be made to limit the scope of the SEL. All systems that depend on normal AC power, such as the Main Feed Water System (ARE [MFWS]), the Start-up and Shutdown System (AAD [SSS]) pump, the main condenser, and their supporting systems can be excluded because the fragility of these systems can be taken as that of the offsite power supply. Also, systems that are non-Seismic Category 1 (SC1) are generally excluded unless it is determined that their seismic capacity is potentially important.

The Chemical and Volume Control System (RCV [CVCS]) is only partly SC1 classified. The RCV [CVCS] pumps are not qualified to be operable following a seismic event. As there are different backup systems available to perform RCP [RCS] coolant makeup and boration, the RCV [CVCS] is conservatively considered unavailable and not included in the SEL. Additionally, the auxiliary pressuriser spray is not included in the SEL as it is not required to mitigate a seismic event. However, the containment isolation valves of the RCV [CVCS] are classified SC1 and they are included in the SEL to ensure containment isolation performance (see section 4.3) and that the risk due to potential loss of inventory to outside the containment is considered.

The LOOP event tree of the level 1 PSA model for internal events was used as a starting point for developing the seismic LOOP event sequence model and event tree. The following summarises the evaluation of the safety functions:

Reactivity Control (reactor trip): The following equipment which is required to remain operational to support the trip function is included in the SEL:

- Reactor internals (damage must not prevent rod drop),
- Fuel assemblies (damage must not prevent rod drop),
- Control rods (must drop in the core),
- Reactor Protection System (RPR [PS]) with instrumentation, input signal, logic and cabinets, and power supply to I&C,
- Reactor trip breakers and actuators.

In the UK EPR, batteries are used as backup to the AC power source for the Control Rod Drive Mechanisms (RGL [CRDM]s), which are kept withdrawn in case of failure of normal AC power. Although they are located in Conventional Building, the backup batteries are conservatively assumed not to fail because of AC power failure. In case of total loss of external power, the RPR [PS] and supporting trip signals and equipment are required to open reactor trip breakers and perform Reactor Trip (RT). They are therefore included in the SEL.

The ATWS event is modelled in the PSA to represent the case of failure of RT. Although the seismic capacity of equipment whose failure would result in ATWS is expected to be greater than the SME, the ATWS event tree is modelled in the seismic PSA model used for the SMA. The following systems are required for mitigation of ATWS:

- Pressuriser Safety Relief Valves (PSRVs): Opening of at least 1 out of 3 PSRVs is required for overpressure protection. Also, the closure of any previously opened PSRV is necessary to prevent occurrence of a small break LOCA. The PSRVs and their pilots are therefore included in the SEL. The function of overpressure protection ensured with 3 out of 3 PSRVs for some transients would have the same seismic capacity.

- Extra Boration System (RBS [EBS]): this system is necessary to mitigate ATWS, as it enables the boration of the primary coolant for control of reactivity on the medium term. The RBS [EBS] is SC1 classified and fed by emergency power. The RBS [EBS] is therefore included in the SEL and is automatically actuated by the ATWS signal produced by the RPR [PS] and its electrical supporting systems, or by the operator from the control room.

- The remaining systems necessary for heat removal in the ATWS sequence are addressed further with the associated safety functions.

Reactor Coolant System (RCP [RCS]) Integrity: The failure of the Reactor Coolant Pump Seal leads to a small LOCA but does not lead directly to core damage as the mitigation of a small LOCA is considered in the seismic small LOCA sequence analysis (see section 4.1.4).

The seismic capacity of equipment preventing a seal LOCA is expected to be high and greater than the SME. Seal injection via the RCV [CVCS] is assumed to be lost following the seismic event. Following loss of RCV [CVCS], the integrity of the reactor coolant pumps seals is ensured by the thermal barrier cooled by the Component Cooling Water System (RRI [CCWS]). In case of loss of the thermal barrier cooling, as the Reactor Coolant Pumps stop following the loss of normal AC power supply, the Stand Still Seal System (DEA [SSSS]) is still available to ensure the RCP [RCS] integrity with a high reliability. However in SBO conditions, the RCP [RCS] pressure and temperature must be slightly reduced after a few hours, in order to ensure leak-tightness of the pump seals and of the DEA [SSSS]. This is ensured by the secondary side residual heat removal addressed below. The following are included in the seismic model and SEL:

- The reactor coolant pump thermal barrier cooling by the RRI [CCWS] and its supporting systems (emergency electrical AC power and Essential Service Water System (SEC [ESWS]) pumps),

- The Stand Still Seal System, with necessary supporting systems,

- The three seal leak-off line isolation valves.

The seal LOCA prevention function has a high reliability and the associated equipment is expected to have a seismic capacity larger than the SME. Even so, the mitigation of a small LOCA is considered in the accident response addressed below.

Reactor Coolant System (RCP [RCS]) Inventory control: This function is provided by the Safety Injection System (RIS [SIS]). Safety Injection is performed by:

- four Medium Head Safety Injection (MHSI) trains and their supporting systems,

- four Accumulators and associated Motor Operated Valves (MOVs), normally open in power states,

- four Low Head Safety Injection System (LHSI) trains and their supporting systems,

- safety injection signal - I&C.

The RIS [SIS] components are SC1 classified and their seismic capacity is expected to be higher than the SME.

Secondary Side Residual Heat Removal: This function is ensured by ASG [EFWS], Main Steam Relief Trains (VDA [MSRT]) and Main Steam Safety Valves (VVP [MSSV]). As indicated in the initiating event analysis, steam generators and connected piping systems, including isolation valves, are included in the SEL. The AAD [SSS], Main Feedwater System (ARE [MFWS]), Condenser, Main Steam Bypass are assumed to be unavailable as they depend on offsite power and are non-SC1 classified. The following other SC1 equipment is included in the PSA model and the SEL:

- Four ASG [EFWS] trains and their supporting systems including signal for system control,
- Four VDAs [MSRTs] and their supporting systems including signal for system control,
- Eight VVP [MSSV]s.

According to the at power level 1 PSA model for internal events, the above systems are sufficient to achieve a success state. In the case of a seismic event, it is considered that the makeup pumps used for ASG [EFWS] tank filling are unavailable. However, the capacity of water storage of the ASG [EFWS] tanks is sufficient to allow conditions for RHR connection to be reached. The long term residual heat removal function is carried out by the Residual Heat Removal System, which is not modelled in the seismic PSA model. However, the four trains of LHSI equipment and their supporting systems, which are addressed below, are included in the PSA model and added to the SEL.

The ASG [EFWS] tanks are also included in the SEL (see section 4.2).

Feed and Bleed (F&B): The secondary cooling success paths are expected to have a capacity well above the SME and the F&B function has significant redundancy. Yet, this function is included in the seismic PSA model similar to the internal events PSA. The necessary equipment is included on the SEL to ensure a comprehensive seismic PSA model. Moreover, most of the same equipment is already required to support other functions and the SLOCA initiating event, thus the equipment added to the SEL is minimal, if any. The following are included to satisfy this function:

- Opening of 3 out of 3 PSRVs on demand with associated solenoid pilots and their supporting systems (Bleed function),
- Opening of 1 out of 2 Severe Accident Dedicated Relief Valves (SADVs); SADVs and its support functions are SC1 classified (Bleed function),
- Safety injection signal - I&C (Feed function),
- Four MHSI trains and their supporting systems (Feed function),
- Four Accumulators and associated motor operated valves (MOVs) (Feed function),
- Four LHSI trains and their supporting systems (Feed function),
- In-Containment Refuelling Water Storage Tank (IRWST) cooling is ensured by LHSI trains or one Containment Heat Removal System (EVU [CHRS]) train. Heat exchangers of the LHSI are cooled by RRI [CCWS] and heat exchangers of the EVU [CHRS] are cooled by EVU [CHRS] dedicated cooling chains. These dedicated cooling chains are cooled by the Ultimate Cooling System (SRU [UCWS]).

The following operator action is required:

- Manual Initiation of Feed & Bleed: This consists of opening the PSRVs or SADVs. Although Safety Injection occurs automatically when these valves are opened, it is assumed the operators follow the procedure and start the pumps before opening the valves.

SLOCA Considerations: to mitigate a small LOCA, there is little difference in required systems and equipment compared to what has already been identified above for the transient response model. The success criteria are slightly different, but since F&B was included in the transient accident response model, most of the equipment required for SLOCA has been identified. The following additional equipment has been identified for inclusion in the seismic PSA model and SEL:

- Partial cooldown (PCD): PCD actuates and opens the VDAs [MSRTs] to reach a lower pressure and allow MHSI makeup to the RCP [RCS]. The operator action that consists of manual initiation of fast cooldown is needed in case of failure of MHSI, to reach LHSI injection pressure.

The failure of a PSRV to close on demand during a plant transient could lead to a small LOCA. However, the equipment used to mitigate a stuck open PSRV is the same as for a small LOCA, and thus already included in the SEL.

4.1.3. Seismic LOOP Event Sequence Model

This section presents the event sequence description of the seismic LOOP. The seismic event is assumed to be severe enough to induce a loss of offsite power. In the case of a low level earthquake with offsite power available, the sequence is assumed to be bounded by normal plant transients. The accident sequences are developed within a level 1 seismic PSA model for at-power states (S LOOP): the event tree is shown in Sub-chapter 15.6 - Figure 1.

A Success sequence (Sequence #1) requires correct operation of the following functions, all of which are expected to have a high seismic capacity (i.e. seismic capacity above the SME):

- Reactor Trip (CRDM06): The actuation of the reactor trip is considered to be highly reliable due to redundancy, and does not require any operator action. However, as the RGL [CRDM] are kept withdrawn by SC1 batteries that backup the normal AC power source in the switchgear building of the conventional island, a reactor trip signal is needed to trigger reactor trip. A failure of reactor trip could thus result from a failure of the I&C providing the reactor trip signal in the PS or SAS from sticking of control rods, or from a mechanical blockage due to damage to the fuel and/or reactor internals. The sequence following the failure of the reactor trip is analysed further in the S ATWS event tree model. The seismic ATWS sequence model and required mitigating functions are described in section 4.1.5 below.

- I&C Power Supply for 2 hours (I&C_L2H): Total loss of I&C electrical power supplies would lead to core damage, due to the unavailability of active system functions in such a case. The different power sources available to the I&C are normal AC (considered lost due to the initiating event), the Emergency Diesel Generators (EDGs, one per division), the Station Black-Out (SBO) Diesel generators (LJP for division one / LJS for division four), and the batteries.

- Ultimate Power Supply (SBO): In the case of failure of all emergency diesels, the RRI [CCWS] would no longer be available. However, ASG [EFWS] pumps and LHSI trains 1 and 4 could still be fed by SBO diesel generators in divisions 1 and 4. The seismic capacity of the SBO diesels is expected to be higher than the SME. So, the loss of emergency power (EDG) would not lead directly to core damage. An operator action is required to start the SBO diesels manually from the Main Control Room (MCR) or locally in case of battery failure.

- Prevention of a reactor coolant pump seal LOCA (RCP_07): This function is also highly reliable and does not depend on operator action. In the internal event PSA model for the LOOP scenario, the reactor coolant pump is assumed to coast down following the initiating event. Either one of the following functions can prevent a seal LOCA:

  - Thermal barrier cooling with RRI [CCWS] is able to protect the Reactor Coolant Pump seals and significantly reduces the likelihood of a Reactor Coolant Pump seal LOCA. There are four RRI [CCWS] trains. Two redundant trains supply one common header that cools two of the four Reactor Coolant Pump seals. Two other redundant trains supply a second common header that cools the other two Reactor Coolant Pump seals. The RRI [CCWS] pumps are supplied by emergency power.

  - The DEA [SSSS] seal system and the three seal leak-off lines are isolated in the event of loss of Reactor Coolant Pump seal cooling and seal injection, in order to prevent the occurrence of a seal LOCA. This function is an automatic action and does not need operator action. The electrical supporting systems required are the emergency busbars, which are seismically classified at SC1.

- Secondary side Residual Heat Removal (SCD_11): Opening of at least 1 out of 4 MSRVs or one out of eight MSSVs is required to ensure residual heat removal via secondary steam release. SG feed is provided by 1 out of 4 ASG [EFWS] trains. The secondary side Residual Heat Removal (RHR) function is automatically actuated but a manual back-up in case of failure of ASG [EFWS] automatic regulation is available. The RHR function is highly redundant and reliable.

  Although this function is highly reliable, and expected to have a high seismic capacity, the failure of this function is modelled in the S LOOP event tree. Backup cooling using Primary Feed & Bleed is modelled consistently with level 1 PSA for internal events.

In summary, there is high confidence in achieving success path #1 due to reliability of the functions available to mitigate the seismically induced LOOP transient. The high reliability is ensured by the redundancy of safety functions and their high seismic capacities (>SME).

In success sequences #2 and #7 in the seismic LOOP event tree model, the Feed & Bleed function is used following failure of secondary cooling, assuming sufficient EDG power supply and either one MSRV or one MSSV remain available. This function requires operator actions to initiate F&B within 2 hours, primary feed with RIS [SIS], primary bleed, and IRWST long term cooling, as discussed below:

- Manual initiation of F&B before 2 hours (OPE_07): It is assumed that if secondary side RHR fails at initiation, the operators have about 2 hours to initiate F&B to prevent core damage. The failure of these actions is assumed to lead to core damage.

- During this period, overpressure protection is achieved by opening of at least 1 of the 3 PSRV (PZR_03) that are operating by cycling open and shut.

- Primary Bleed (PBL_02): At least 1 out of 2 bleed lines or 3 out of 3 Pressuriser Safety Relief Valves must open to give a sufficient reduction in the RCP [RCS] pressure to ensure safety injection with MHSI. Bleeding with the SADV or PSRV lines is credited in the model as they are SC1 classified. This equipment is reliable and is expected to have a high seismic capacity (>SME). The failure of this action is assumed to lead to core damage.

- Feed Function availability: The success criterion for RCP [RCS] feed is the operation of 2 out of 4 MHSI trains (SISM16A), or the operation of 1 out of 4 MHSI (SISM16B) and 4 out of 4 Accumulators (SISA01D). The following function events are thus modelled in the event tree:

  - MHSI (SISM16A or SISM16B): 2 out of 4 MHSI, or 1 out of 4 MHSI with 4 out of 4 accumulators enable the success criteria for primary feed to be achieved, preventing core damage. The MHSI has redundancy and is expected to have a high seismic capacity (>SME). Failure of this function is assumed to lead to core damage.

  - Accumulators (SISA01D): 4 out of 4 accumulators are assumed necessary for the success of this function. The only active component within the function is the check valve in each injection path that is required to open. The function is expected to be highly reliable. Failure of this function is assumed to lead to core damage.

  - IRWST Cooling (SIS_06): IRWST cooling requires at least 1 out of 4 LHSI trains in minimum flow operation, or 1 out of 2 Containment Heat Removal System (EVU [CHRS]) trains to ensure the operability of the RIS pumps. The function has high redundancy and the EVU [CHRS] and LHSI are expected to have a high seismic capacity (>SME). Failure of this function is assumed to lead to core damage.

In summary, there is high confidence in achieving success sequences #2 and #7 due to high redundancy and reliability and the expected high seismic capacity (>SME). As operator action is required within 2 hours under conditions where the level of stress on the operator is still high in the immediate post-seismic period, manual initiation of F&B by the operator is assumed to be less reliable than the equipment.

Although it is considered highly reliable, failure of the Reactor Coolant Pump sealing function leading to a small LOCA is considered in the seismic PSA model for completeness. Success sequences #15 and #17 contain events following the Reactor Coolant Pump seal leakage. In this scenario, partial cooldown is automatically initiated to reach MHSI pressure injection. The function events required to mitigate Reactor Coolant Pump seal LOCA are similar to those required for mitigation of a seismically induced small break LOCA in an at-power state. The mitigation of a RCP [RCS] leak is modelled as a seismic small LOCA event sequence. The necessary function events are modelled in the event tree shown in Sub-chapter 15.6 - Figure 2 and mitigating systems are described in section 4.1.4.

Sequence #24 models the response of the EPR to a station blackout in the case of a seismic event. The SBO diesels are assumed to be available in the seismic PSA model as they are SC1 classified and diverse from the EDGs. As emergency power has high redundancy, and the capacity of EDG is high (>SME), the risk of station blackout is low. Even so, in the event of a station blackout, the Standstill Seal System (DEA [SSSS]) would be automatically actuated. Following this event, 2 different possibilities are foreseen with respect to the failure of the Reactor Coolant Pump seal system:

- If the Reactor Coolant Pump seals (without considering the DEA [SSSS]) are unable to withstand the extreme temperature and pressure (RCP_07), RCP [RCS] cooling is required using the 4 Steam Generators to ensure that the pressure and temperature remain within the design envelope for Stand Still Seal System (DEA [SSSS]) protection and thus to guarantee the primary system integrity. The cooling to 280°C must be symmetrical for the 4 loops. It requires at least 1 ASG [EFWS] pump, supplied by the SBO diesels, and manual opening of the ASG [EFWS] header to allow cooling of the 4 SGs and, after 2 hours, local opening of the Main Steam Relief Trains in the steam valve room (basic event OP_EFW/MSRT_2H LOCAL). Failure of this cooling leads to a LOCA and subsequently to core damage due to an assumed complete dependency between this cooling and the initiation of a subsequent fast secondary cooldown (OPE_66)

  If a seal LOCA occurs (not caused by the failure of the symmetrical cooling), it is managed by initiation of fast secondary cooldown (OPE-66 - OP_FSCD_30MN) requiring 2 SGs (FSCD02) which allows safety injection with LHSI (SISA01A and SISL40). IRWST cooling (SIS_06B) is then required.

- If the Reactor Coolant Pump seals (without the DEA [SSSS]) are successful in withstanding the extreme conditions, cooling can be performed with only one steam generator (SCD_11A).

The systems involved in these sequences have been previously described.

In summary, there is a high confidence in achieving a success path in a station blackout situation as the mitigation systems have high redundancy and reliability and are expected to have a high seismic capacity (>SME).

4.1.4. Seismic Small LOCA Event Sequence Model

The seismic small LOCA model presented in Sub-chapter 15.6 - Figure 2 is used for the seismically induced RCP [RCS] leakage. As described above in the initiating event evaluation, LOCAs due to seismically induced RCP [RCS] leakage are considered unlikely (seismic capacities > SME). However, the ability to mitigate such events is considered to address uncertainties in plant response, particularly with regard to the integrity of small diameter pipework connected to the RCP [RCS]. The event tree developed for a S SLOCA is based on the level 1 PSA event tree for SLOCA adapted for the seismic event: in particular occurrence of a seismically induced LOOP is assumed.

The mitigation systems modelled in the S SLOCA event tree are similar to those considered in modelling Reactor Coolant Pump seal LOCA. For the primary success path #1, the main differences, particularly with regard to success criteria and I&C signal occurrence, are summarised below:

- Partial cooldown (PCD01): Partial cooldown is initiated automatically on receipt of an SI signal and can be performed using at least 1 out of 4 steam generators with 1 out of 4 ASG [EFWS] trains and 1 out of 4 MSRV trains. This function thus has high redundancy and is expected to have a high seismic capacity (>SME). Failure of this action is conservatively assumed to lead to core damage in the present study, because the success criteria for F&B require the availability of one RCV [CVCS] train (RCV [CVCS] is not SC1 classified).

  Feeding of the steam generator can also be performed by the AAD [SSS] and steam blowdown can also be performed using the GCT [MSB]. However, these systems are reliant on normal AC power and are not SC1 classified: consequently they are not credited in the model.

Operator action to start PCD within 15 minutes in case of PCD signal failure is modelled in the level 1 PSA. This operator recovery action is conservatively ignored in the seismic PSA model due the high operator stress following a seismic event.

- RCP [RCS] inventory (SISM04): Makeup is required to compensate for the RCP [RCS] leakage in the case of a small LOCA. 1 out of 3 MHSI injection trains is necessary. The RIS [SIS] is assumed to start automatically on a low pressuriser pressure signal and makeup to start when RCP [RCS] pressure is sufficiently reduced by the PCD. There is high redundancy for this function and the seismic capacity is expected to be high (>SME).

- IRWST cooling (SIS_04): because of the small LOCA, IRWST cooling is required to achieve a primary success path, whereas it is only required in case of F&B for seismic induced transients.

In summary, success path #1 in S SLOCA event tree has a high probability of success similar to that for a Reactor Coolant Pump seal LOCA (although with slightly different success criteria): partial cooldown is required to reach the MHSI pressure injection to ensure RCP [RCS] makeup and IRWST cooling is necessary to ensure containment residual heat removal. The redundancy and seismic capacity of the success path is expected to be high (>SME).

In the unlikely event of the failure of MHSI to provide RCP [RCS] makeup, fast secondary side cooldown (FSCD01) can be actuated manually at 30 minutes (OPE_24A) to reduce RCP [RCS] pressure to the accumulator and LHSI injection pressure (success path #3). The success path for a small break LOCA without MHSI is similar that in Reactor Coolant Pump seal LOCA without MHSI. As for PCD, no credit is taken for the AAD [SSS] for achieving fast cooldown.

As in the event trees for transients, all the required functions are automatic, except for manual initiation of fast cooldown and primary F&B.

In the unlikely event of failure of the partial cooldown, the event sequence is conservatively assumed to lead to core damage, because the RCV [CVCS] is not SC1 classified; according to the success criteria one RCV [CVCS] train is needed for successful F&B.

4.1.5. Seismic ATWS Event Sequence Model

The seismic ATWS model which is shown in Sub-chapter 15.6 - Figure 3 is included for completeness to address the unlikely situation where the reactor trip function fails due to the seismic event. The failure of the reactor trip function could be due to failure of the I&C system to provide a reactor trip signal, failure to open of the RT Breakers or Contactors or mechanical blockage of control rods (due to stuck rods or deformation of fuel assemblies or reactor internals).

The model used is the same as the model of ATWS following a LOOP initiating event used in the level 1 PSA for internal events. It should be noted that Reactor Coolant Pump trip is guaranteed to succeed given that offsite power to the pumps is lost in the seismic event. The S ATWS event tree involves several new safety functions that are not included in the previous event trees for seismic transients and small LOCAs:

- Boration for ATWS (EBS_02): For the ATWS transient modelled in the PSA, 1 out of 2 trains are required, with the PSVs cycling open and shut, for long term reactivity control. The RBS [EBS] piston pumps provide the required injection flow rate with high reliability. In the very unlikely event of failure of the ATWS signal to initiate boration (I&C_80), operator action to start the system from the control room would be required within 1 hour (OPE_60).

- Overpressure protection (PZR_03): 1 out of 3 PSRVs are required to open on demand to protect the RCP [RCS] from overpressure given the Reactor Coolant Pump trip due to the LOOP.

  NB. The function of overpressure protection with the opening of more than 1 out of 3 PSRVs would present the same seismic fragility, as a common cause seismic failure is considered for the three PSRVs.

- Pressuriser safety valves re-close (PZR_02): If 1 out of 3 of the PSRVs does not re-close, a LOCA condition is assumed in the seismic PSA model. However given the high reliability of the reactor trip function and the reliability of the PSRVs, this sequence is not studied further.

4.1.6. Dependencies on Supporting Systems

The availability of mitigating systems described in the previous sections depends on the availability of supporting systems. The following supporting systems dependencies modelled in the level 1 PSA are considered in the seismic PSA model:

- Each of the four trains of ASG [EFWS] is supplied by a separate train of emergency AC, I&C and DC. The cooling of the ASG [EFWS] pumps (motor and bearings) is provided by water pumped from its first stage that is discharged to the ASG [EFWS] tank. The pump cooling does not depend on the RRI [CCWS]/SEC [ESWS] systems.

- Each of the four trains of MHSI is supplied by a separate train of emergency AC, I&C, and DC. Each train depends on its respective train of RRI [CCWS]/SEC [ESWS] for pump cooling.

- Each of the four trains of LHSI is supplied by a separate train of emergency AC, I&C, and DC. Trains 2 and 3 also depend on their respective train of RRI [CCWS]/SEC [ESWS] for pump motor cooling. Pump motor cooling for Trains 1 and 4 is supplied alternatively by the respective trains of RRI [CCWS]/SEC [ESWS] or DEL [SCWS] air cooled chilled water systems. The motors of the DEL pumps are powered by SBO-emergency switchgears. The heat removal function of each of the 4 LHSI trains, via its heat exchanger, depends on that respective RRI [CCWS]/SEC [ESWS] train.

- The two trains of RBS [EBS] located in the fuel building are powered by emergency AC trains 1 and 4. The RBS [EBS] does not depend on the RRI [CCWS] or SEC [ESWS]. The RBS [EBS] is started on an ATWS signal generated by the PS, but can also be started by operator action.

- The four emergency diesel trains are air cooled and do not depend on the SEC [ESWS]. However, the diesels depend on the DC batteries power to start.

- Automatic operation of the above systems and their supporting systems is dependent on the I&C systems and control power (DC), which is supplied by emergency AC and backed up with battery power.

- The Reactor Trip function depends on I&C control power (DC), which is supplied by emergency AC and backed up with battery power. Although backup batteries are located in conventional buildings, it is conservative to assume the survival of this equipment in the seismic PSA model.

- Reactor Coolant Pump Seal LOCA prevention depends on uninterruptible AC power and I&C control power (DC), which is supplied by seismic AC and backed up by batteries.

- PSRV and MSRV operation depends on uninterruptible AC power and I&C control power (DC), which is powered by emergency AC and backed up with battery power. SADV operation also depends on uninterruptible AC power and I&C control power (DC).

- Two station blackout diesels (SBO) provide backup to the emergency diesels. The SBO diesels are air cooled. They can be started from the MCR using 2 hour batteries or locally by an operator. The SBO diesels are SC1 classified and are in SC1 located classified buildings, thus they are included in the seismic PSA model.

4.2. SEISMIC EQUIPMENT LIST (SEL)

The seismic systems and accident sequence analysis enables the construction of a so-called Seismic Equipment List (SEL). This list is presented in this section. The SEL includes:

- The active or passive systems and components identified in the previous step using the level 1 PSA for internal events in at-power states, credit for which is taken for reaching a safe state following the seismically induced initiating event;

- Other active or passive components that have a key role in the level 1 PSA for shutdown states and in the level 2 PSA;

- Other equipment which is not modelled in the PSA because of its assumed high reliability, notably structures.

In addition to the PSA model review performed in 4.1, the completeness of the SEL is ensured by the review of flow diagrams and system descriptions from applicable System Design Manual, general arrangement drawings, and single line electrical diagrams to make sure that the Seismic Equipment List is exhaustive. Also a review with plant layout personnel to identify the room location of the different equipment items, based on information currently available within GDA.

The SEL for the UK EPR is summarised in the following tables. The impact of the seismic failure of the components and structures is indicated in the tables. Structures, systems and components whose seismic failure is assumed to lead directly to core damage (see section 4.1.1) are identified in the "Core Damage" column in the SEL. Structures, systems and components whose failure is assumed to lead to large early releases, are identified in the "Large Releases" column in the SEL table. Other structures, systems and components that are judged to have an important impact on the PSA model but for which alternative success paths exist are shown in the "Model" column in the SEL table.

<u>Seismic Equipment List for Structures</u>

In the following Table, structures have been included which contain some equipment credited in the level 1 internal events PSA model for at power states, but supplied by normal AC power. For these cases, the equipment is judged unavailable due to the LOOP. However, the overall structure must not impact on SC1 classified SSCs.

Structure (Seismic Category)	PSA Impact Mapping		
	Core Damage	Large Releases	Model
Containment & Annulus (SC1)	X	X	
Containment Penetrations (piping, hatches etc.) (SC1)		X	
Reactor Bldg Internal Structure (SC1)	X		
IRWST (SC1)	X		
Core melt retention structure (SC1)	X		
Reactor Pit, Seal and Pools (SC1)	X	X	
Fuel Transfer Tube (SC1)	X	X	
Refuel Gates (SC1)	X	X	
Refuelling Machine (SC2)	X		
Polar Crane (SC2)	X		
Safeguards Buildings (SC1) (Train 1 through 4)	X	X	
EFW pools (SC1)	X		
RRI [CCWS] Surge Pools (SC1)			X
RBS [EBS] boric acid pools (SC1)			X
Control Room & Ceiling (SC1)	X		
Fuel Building (SC1)	X	X	
Spent Fuel Pool (SC1)	X	X	
Diesel Buildings (SC1)	X		
Ultimate Heat Sink Buildings (SC1)			X
Cable Duct & Shaft (SC1)	X		

Also, Access Building structures (possible interaction with divisions 3 and 4 of the Safeguards Buildings) and the Nuclear Auxiliaries Building stack (possible interaction with Reactor Building, Fuel Building and division 4 of Safeguards Building) are SC2 classified. The design of these SC2 structures must ensure that they will not collapse in such a way that they could impact on the safeguard buildings. The Nuclear Auxiliary Building (possible interaction with Fuel Building and division 4 of Safeguards Building) is SC1 classified and does not contain equipment on the success path, and must not fail in such a way that it could impact safeguard buildings. It is verified in the detailed design that this building is designed with sufficient margin. Thus, this building is not included in the SEL.

The Turbine Building (possible interaction with divisions 2 and 3 of Safeguards Buildings) is also SC2 classified. It contains equipment and supporting equipment fed by normal AC. This equipment is assumed to fail. The non-classified electrical equipment building in the Conventional Island contains electrical supporting systems, especially normal AC switchgear. All systems dependent on AC power in this building are not credited in the PSA, but the structures must not collapse in such a way as to impact safeguard buildings. The site specific elements of the Conventional Island will be designed with sufficient margins to the DBE to avoid an interaction threat. This will be confirmed as part of the justification of site specific design, and is outside the scope of GDA. Thus they are not included in the SEL.

Seismic Equipment List for Components and Systems

The following Table provides the SEL for systems and components and shows their impact on the PSA. Most of the components impact on the PSA model. The accident sequence analysis shows that if these components and systems fail, other success paths are available. Dependencies among seismic failures of different components are treated in the following manner:

- Total dependency of failure is assumed between identical components of redundant trains. In reality, identical pumps installed on different axis orientations might not fail at the same seismic level. This assumption is known to be strongly conservative, as stated in section 2.4.

- No correlation of failure is considered between components within the same structure. For example, components of identical category located in the same structure are generally assumed to fail at the same time in the seismic PSA. At this point, the fragility is calculated for individual elements of the SEL, and the equivalent fragility is specified for the system in the support PSA model for PSA-based SMA.

System/Component	PSA Impact Mapping		
	Core Damage	Large Releases	Model
Reactor Coolant System, Control Rods & Reactor Internals			
Reactor vessel supports	X		
Reactor core internals (failure should not prevent rod drop)			X
Fuel grid (fuel failure should not prevent rod drop)			X
Control rod drive mechanisms			X
Steam generators & supports (SG snubber, struts, support columns)	X		
Reactor coolant pumps (Support column, snubbers)	X		
Pressuriser Lower supports	X		
Pressuriser relief valves (including SOV)			X
Dedicated relief valves MOV			X
Pressuriser vent MOVs (open during SD)			X
Pressuriser Surge Line			
Secondary Coolant System			
Feedwater piping downstream of FWIV	X		
Main steam piping upstream of VIV [MSIV]	X		
VIVs [MSIVs] Oleo pneumatic (including SOVs)			X
FWIVs MOVs			X
FWIVs Full and Low Load Oleo pneumatic (including associated SOVs)			X
MSRVs Control MOV			X
MSRIVs steam operated (including typical SOVs)			X
MSSVs			X
Emergency Feedwater System			
Pumps			X
Isolation MOVs			X
Flow control valves			X
Pressure control valves			X
check valves			X
manual valves			X
Piping			X
Medium Head Safety Injection			

System/Component	PSA Impact Mapping		
	Core Damage	Large Releases	Model
Pumps			X
MOVs			X
check valves			X
manual check valves			X
manual valves			X
Piping			X
SI Accumulators			
Accumulator			X
MOV			X
check valves			X
safety valves			X
Piping			X
Low Head Safety Injection/RHR			
Pumps			X
Heat Exchangers			X
RRI [CCWS] pneumatic valve			X
MOVs			X
motor operated check valve			X
check valves			X
manual check valves			X
manual valve			X
safety valve			X
Piping			X
Severe Accident Containment Heat Removal			
Pumps			X
Heat Exchangers			X
Dedicated EVU Pump			X
Dedicated EVU HX			X
Dedicated SRU Pump			X

System/Component	PSA Impact Mapping		
	Core Damage	Large Releases	Model
MOVs			X
check valves			X
manual valves			X
Piping			X
Extra Borating System (RBS [EBS])			
Pumps			X
MOVs			X
Tank			X
manual valves			X
safety valves			X
check valves			X
Piping			X
Building Ventilation (fans, dampers, ducts, coolers, filters etc.)			X
Reactor Coolant Pump Seal Integrity			
Reactor Coolant Pump shaft, seals and Standstill seal			X
Reactor Coolant Pump breaker (GMPP)			X
DEA [SSSS] N2 supply SOV			X
DEA [SSSS] N2 discharge MOV			X
Seal 1 SOV			X
Seal 2 MOV			X
Seal 3 MOV			X
Thermal barrier SOV and MOV			X
Pneumatic valves (TB and Leak off)			X
check valves			X
safety valves (TB)			X
manual valves			X
Piping			X
Component Cooling Water			

System/Component	PSA Impact Mapping		
	Core Damage	Large Releases	Model
Pumps			X
Heat Exchangers			X
MOVs			X
Pneumatic Valves			X
check valves			X
manual valves			X
safety valves			X
Piping			X
Essential Service Water			
SEC [ESWS] Pumps			X
Cooling Tower Fans & Equipment			X
check valves			X
manual valves			X
filters and strainers			X
Piping			X
Building Ventilation (fans, dampers, ducts, coolers, filters etc.)			X
Emergency Diesels			
Diesel generator and controls (LHP/LHQ/LHR/LHS)			X
Fuel oil day tanks			X
Fuel oil storage tanks			X
Air start compressors			X
Air start receivers			X
Diesel heat exchangers (air cooled)			X
Building Ventilation (fans, dampers, ducts, coolers, filters etc.)			X
SBO Diesels			
Diesel generator (LJP/LJS)			X
Fuel Pool Cooling			

System/Component	PSA Impact Mapping		
	Core Damage	Large Releases	Model
Pumps			X
Heat Exchangers			X
RRI [CCWS], EVU dedicated MOV			X
RRI [CCWS] dedicated manual valve			X
MOVs			X
manual valves			X
check valves			X
Piping			X
Building Ventilation (fans, dampers, ducts, coolers, filters etc.)			X
Containment Isolation Valves			
Ventilation			X
Gaseous Waste			X
Reactor Bldg Primary & Secondary Drain MOVs			X
Containment area sump, floor drain			X
Letdown isolation valves			X
SG Blowdown			X
RCP [RCS] vessel vent			X
Electrical Buildings Ventilation			
Supply fans			X
Exhaust fans			X
Pumps DEL			X
Chillers DEL			X
Chillers DER			X
Pumps DER			X
Check dampers			X
MO dampers			X
Piping, ducting			X
Control Room Emergency Ventilation			
Pre, HEPA, Carbon filters			X

System/Component	PSA Impact Mapping		
	Core Damage	Large Releases	Model
Fan			X
Chillier Cooling coil			X
Supply Air Filter			X
Normal AC (LOOP) (non-seismic)			Initiating Event
Emergency AC & DC			
Transformer 400kVAC/10kVAC (TS/TA)			X
10 kV switchgear (Lei, Lei)			X
10kV-690V, 690V-400V AC Transformer (Lei)			X
Transformer, Voltage Regulated (Lois)			X
690V emergency AC bus (Lei) or normal AC bus (LIi)			X
400V emergency AC bus (LLi) or normal AC bus (LKi)			X
400V regulated AC (LOi)			X
400V uninterruptible AC (LVi)			X
220V uninterruptible DC (LAi)			X
Electrical Panel Boards (120V AC, 24V DC)			X
Batteries & racks (220V DC)			X
Batteries Chargers 220V DC (2 hrs and 12 hrs)			X
AC/DC Converters (for 220V DC or 24V DC)			X
Inverters with manual maintenance bypass switch (for 400V AC)			X
Inverters			X
Breakers (10kV, 690V, 400V)			X
EDG breaker (qualified as part of cabinet)			X
SBO diesel breaker (qualified as part of cabinet)			X
Cable trays			X
			X
I&C			
Relays			X

System/Component	PSA Impact Mapping		
	Core Damage	Large Releases	Model
Sensor & transmitters in the field (input signals to PS, RCSL, SAS, PAS) - Low voltage at 10.0 kV - SG level - SG pressure - Pressuriser pressure - Activity sensor - EFW pump Flow - Reactor Coolant Pump Speed - Cold leg temperature elements - Hot leg pressure elements - RCP [RCS] loop level - Neutron flux source range - RCCA rod position Reactor trip check back (RGL [CRDM])			X
PS Reactor Protection Cabinets, Racks, Modules, Fiber Optics (TXS)			X
PS Reactor trip cabinets (breakers, contactors) (TXS)			X
RCSL cabinets (reactor control) (TXS)			X
SAS cabinets (safety automation system) (TXP)			X
PAS cabinets (process automation system) (TXP)			X
PICS cabinets (operator displays, digital control and screens)			X
SICS cabinets (safety control room)			X
PACS cabinets (ESF, priority module actuators, solid state modules) (TXS)			X

4.3. CONTAINMENT PERFORMANCE

Analysis of detailed radiological release due to seismic event is not performed as part of the PSA-based SMA within GDA. However, the containment performance is studied to ensure that key elements claimed to mitigate radiological releases in the level 2 PSA assessment are included in the SEL. The internal events level 2 PSA model was reviewed, and the important components identifies are summarised below:

- The Reactor Building (external wall and liner), including the penetrations and hatch: the RB contributes to the containment function;
- Multiple containment isolation valves and supporting equipment: these valves must close to ensure the containment function;
- Severe Accident Dedicated Valves: this equipment is used for Bleed function and depressurisation of the RCP [RCS] in a severe accident;
- Containment Heat Removal System, EVU [CHRS], and supporting equipment: this equipment ensures residual heat removal from containment and limits thermal and pressure loads acting on the containment building;
- The passive cooling line of the EVU [CHRS] used to flood the basemat, and the drain valve from the IRWST to core catcher, including the passive opening device, which must not prematurely fail due to a seismic event since it could impact on the availability of the IRWST volume to support the operation of the safety injection system (RIS [SIS]): this equipment is necessary to ensure residual heat removal from the corium in the spreading compartment;
- The core melt retention structure is included in the SEL as it is considered part of the internal containment structure;
- The Annulus Ventilation System (EDE [AVS]): the ventilation system is SC1 classified;
- The combustible gas control system (ETY) inside containment: Passive Autocatalytic Recombiners and “2-room” isolation valves are SC2 classified and are not credited in the SMA analysis. They are not included in the SEL.

4.4. SHUTDOWN STATES

Most of the systems used for mitigating initiating events occurring in shutdown states are also used for mitigation of accidents in at-power states. Therefore, shutdown states are not described in detail in the seismic PSA model. However, a review of the level 1 PSA modelling of shutdown states was performed to identify systems and components that should be added to the SEL. It was found that as expected many systems credited in the shutdown state level 1 PSA were already identified in the review of the PSA for at-power states. One key difference was that loss of LHSI in the RHR mode was identified as a new initiating event. However this system is already included on the SEL. Similarly, loss of offsite power was identified as an important initiating event, but this was already identified as an important initiator for power operation. However, based on this evaluation, several structures, systems and components were added to the SEL, as summarised below:

- Reactor Pit and connected pools,
- Fuel Transfer Tube and Gate Valve,
- Refuelling Gates,
- Refuelling Machine and Polar Crane (must not tip over, drop heavy loads etc),
- Spent Fuel Pool,
- Spent Fuel Pool Cooling System and supporting equipment,
- Pressuriser Vent Isolation MOVs.

The effect of the seismic capacities of these structures and components on the results of this SMA is discussed at the end of this sub-chapter, in section 7.

4.5. SEISMICALLY INDUCED INTERNAL HAZARDS

Internal Hazards due to a seismic event, such as fires or internal flooding, are not addressed in detail in the present SMA study. Indeed, the fragility evaluation is based on the assumption that equipment will be installed according to design, and components and structures whose failure might initiate such hazards are expected to have a high seismic capacity. Also, the plant layout, the physical barriers, and the classification principles applied to equipment ensures a very low likelihood that a seismically induced fire or flood will impact on safety-related equipment.

The classification principles described in Sub-chapter 3.2, section 6 assign equipment whose failure may have a damaging impact on safety-equipment (e.g. tanks and piping containing oil, water carrying pipes with potential for flooding) into seismic class SC2. The corresponding mechanical analyses demonstrating integrity under DBE loads will therefore be carried out with conservative assumptions in compliance with the analyses for SC1 equipment. Therefore, similar HCLPF capacities to those for equipment included in the SEL can be expected.

Furthermore, physical separation between divisions will ensure a reduced impact of such hazards.

Therefore, it is expected that an explicit consideration of fires or floods due to earthquakes would not challenge the results presented in the SMA. The stated assumptions will be verified by a confirmatory plant walkdown when the plant is built (see section 5.2).

4.6. RELAY CHATTER

Solid state relays are generally used in the EPR. These solid state switching devices are inherently immune to chatter. Electro-mechanical relays, if used, will be tested for the floor response generated using the 0.25g DBE as ground motion input. At the GDA stage there is limited information on the location and qualification of relays; however systems and equipment that depend on electromechanical relays rather than solid state devices would be seismically qualified, and will have adequate margins to ensure that relays have capacities greater than the SME. Therefore, contact chatter has been screened out and is not included in the SEL. The detailed design must confirm the use of seismically rugged and qualified electro-mechanical relays where they could impact the seismic PSA model.

4.7. NON-SEISMIC RANDOM FAILURES AND HUMAN ACTIONS

Non-seismic equipment failure modes (random failures) and human action failures are modelled in the development of the seismic PSA model as described in section 4.1.

With regard to random equipment failures, there are four trains of redundancy for ASG [EFWS], MHSI, LHSI, Accumulators and their supporting systems. There are two trains of redundancy for the EVU [CHRS] and its supporting systems. There is also additional redundancy in the model as described in section 4.1. For example, primary feed and bleed systems provide backup to the ASG [EFWS] for residual heat removal. As a result, non-seismic random failures of equipment do not result in unacceptable risk or reduce seismic margins. Non-seismic random failure modelling is based on the level 1 PSA for internal events.

The impact of human action failures is also small because the systems and functions on the primary success paths respond automatically as described in section 4.1. It is only after the failure of a reliable automatic function that a human action would be required. The following summarises the important operator actions identified:

- Manual initiation of Feed & Bleed (OP_FB_120M_MDEP and OP_BLEED_120MN): this operator action is required within 2 hours after LOOP with failure of partial cooldown with the ASG [EFWS]. It is considered that 2 hours is a sufficient grace period for a post seismic event operator action performed from the control room. Therefore, this operator action is judged to be of adequate reliability.

- Manual initiation of Fast Secondary Cooldown (OP_FSCD_30MN): this operator action must be performed within 30 minutes of a small LOCA (RCP [RCS] leakage or failure of Reactor Coolant Pump seals) with loss of MHSI in order to reach LHSI injection pressure. The probability of requiring this operator action is expected to be low.

  In the event of a small LOCA, partial cooldown is automatically initiated to achieve MHSI injection pressure. A recovery operator action (OPE_PCD) can be credited if performed within 15 minutes. This operator action is not judged to be important as the probability of it being required is low.

- Manual start and control of ASG [EFWS] in the event of I&C failure (OP_EFWS and OP_EFWS30): the aim of this operator action, which is performed from the main control room, is to compensate for the failure of the RPR [PS] to provide ASG [EFWS] regulation.

  This operator action is not judged to be important as the probability of it being required is low and the protection system is SC1 classified.

- Cross connection of the SGs and opening of the VDAs [MSRTs] (OP_EFW/MSRT_2H LOCAL): the cross connection of ASG [EFWS] trains in a SBO is necessary to perform secondary cooldown with injection trains 1 and 4, and the action must be performed within 2 hours. The probability of this operator action being required is expected to be low given the high capacity (>SME) and the redundancy of EDGs. Also, this operator action is judged to be quite reliable.

- Manual start of the SBO diesels must be performed within 2 hours (OP_SBODG2H) in the case of loss of EDGs (without LOCA), within 30 minutes in the case of LOCA (OP_SBODG30M) and within 15 minutes (OP_SBODG15M) if PCD is required.

  This action can be performed from the control room, or locally and manually (OP_SBODG_LOCAL) in case of battery failure. The probability of this operator action being required is expected to be low due to the high seismic capacity (>SME) and redundancy of the EDGs. Also, this action is judged reliable as the diagnosis of a SBO event is relatively easy. Also, the seismic capacity of the SBO diesels is high, and above the SME.

- Manual initiation of IRWST cooling (OPE_52) is required in the case of a small LOCA or Feed & Bleed, to remove residual heat from the containment. The grace period for this action is more than 4 hours, and the action is thus judged very reliable.

- Manual start of the RBS [EBS] (OPE_EBS 60MIN) is required in the event of failure of ATWS signal. The probability of this operator action being required is low as the signal is redundant. Although the operator action is required within 1 hour, it is expected to be reliable as it is requested early in the procedures.

Random failures and operator action failures are discussed further in section 6. It is shown therein that due to the availability of seismically qualified automatic protection actions, there are no cutsets involving only operator action failures that result directly in core damage following a seismic event. Therefore operator action failures do not influence the seismic capacity of the plant calculated using the current PSA-based SMA methodology.

5. SEISMIC FRAGILITY EVALUATION

The fragility evaluation has been performed for the components and structures of the UK EPR based on design information available at this time. As described in section 3, the median ground response spectrum for different sites (as described in section 2.1) is conservative for the design ground motion spectrum (EUR ground response spectra). The median ground response spectra were taken from HSE document [Ref-1].

5.1. SEISMIC MARGIN CALCULATIONS

Within GDA, fragility analysis is limited to using seismic design criteria and available qualification methods normally applied in the nuclear industry to estimate seismic capacities. In some cases, this can result in conservatively low estimates of seismic margin. In these cases, increased capacities based on the additional margins typically available in design and qualification testing are considered. These estimations result in what are known as “reasonably achievable” fragilities. Where applicable, these fragilities have been reviewed against the recommended values in the literature. Spectral shape factors were estimated based on the spectra for sites 2 and 4.

The following approach was used to estimate the fragilities of structures and equipment. Guidance was also obtained from the methodology described by EPRI [Ref-1].

- Building Fragilities

  Building fragilities are calculated for the critical elements/failure modes using the specific design information as available. In the absence of such information, design criteria and generic material strength data are used to estimate the median capacity and variability. From these, the HCLPF capacities of buildings are calculated; these are all higher than the target of 0.4g PGA. Results of fragility calculations for buildings and structures are presented in a detailed seismic fragilities report [Ref-2].

- Equipment Fragilities

  The equipment fragilities are calculated for the critical elements/failure modes using specific design information as available. In the absence of such information, design criteria and qualification procedures are used to estimate the median capacity and variability. From these, the HCLPF capacities of equipment are calculated; these are all higher than the target of 0.4g PGA. Results of fragility calculations for buildings and structures are presented in detailed reports [Ref-2] [Ref-3].

  Where this procedure resulted in lower seismic margins, reasonably achievable fragilities are assigned based on past earthquake experience and results of seismic PSAs. In addition, minimum performance or design requirements are specified when appropriate to ensure that the as-built configuration will achieve the plant seismic margin. Examples are requirements for equipment anchorages to have adequate margin, requirements to prevent seismic spatial interactions from occurring, etc.

The following Table shows the assigned seismic capacities of the various equipment categories (taken from the detailed seismic fragilities of structures and equipment report [Ref-2]). The results of the analysis are presented in the summary of seismic capacities of Structures, Systems and Components in Sub-chapter 15.6 - Table 1 and Sub-chapter 15.6 - Table 2. The detailed numerical values are subject to changes that may arise from changes in design details, refinements in analysis, and specific design criteria.

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5.2. WALKDOWN FOCUS

The fragility evaluation at the GDA stage is based on the assumption that equipment will be installed as designed and that there are will be no potential spatial interaction concerns in the as-built configuration (e.g. adjacent cabinets are assumed to be bolted together, collapse of non-seismically designed equipment or masonry walls on to safety-related equipment is precluded, and there is assumed no likelihood of seismically induced fire or flood impacting on safety-related equipment). After the plant is built and the equipment, piping systems, cable trays and HVAC ducts are fully installed, a confirmatory walkdown of the plant will be conducted.

With modern design methods, rigorous QA requirements, and lessons learned from analysis and design of nuclear power plants in the last forty years, there are not expected to be many instances of system interactions and improper installations. However, this confirmatory walkdown is still deemed important.

The walkdown will focus on verifying that the anchorages of equipment and distribution systems are properly positioned and installed and that the equipment modelled in the seismic PSA are not exposed to any potential spatial seismic system interactions. Details of the seismic walkdown procedures can be found in ANS standard [Ref-1]. Examples of items to be considered during the confirmatory walkdowns are as follows:

- Special care should be taken to ensure that the anchor bolts are properly installed and adequate edge distances are maintained to preclude concrete bursting failures (e.g. concrete piers for heat exchanger supports).

- Where pre-tensioning of the anchor bolts is specified by the equipment vendor, it should be ensured that such pre-tensioning is done and documented.

- Noting of any field modifications to equipment anchorages (e.g. excessive shimming) that may have impact on the anchorage capacity.

- Review of proper installation of equipment items that are significant for performance of equipment modelled in the PSA; e.g. EDG lube oil tank and silencer.

- Review to show that the HVAC ducts are properly supported near the air handling units and that the fabric joints have adequate flexibility to accommodate potential relative movements between the ducts and the air handling units.

- Opening of the air handling units to ensure that the fans are not mounted on spring vibration isolators and that the cooling coils are properly bolted to the housing.

- If equipment is mounted on a raised floor, confirmation that the equipment is properly anchored to the structural slab and that the raised floor does not pose a collapse hazard.

- Ensuring that piping supports are properly positioned.

6. SEISMIC MARGIN EVALUATION

The seismic margin evaluation for PSA-based SMA is performed using the methodology described in section 2.4.

The assessed HCLPF capacity for the plant is 0.60g PGA for rock sites and 0.61g PGA for soil sites, which is more than double the DBE. The HCLPF for the plant of the UK EPR is thus at least 50% above the target SME (0.4g PGA). The critical seismic component or system failures that could lead directly to core damage are identified as seismic failure of AC power switchgear, I&C and the ASG [EFWS]; however their HCLPF is higher than SME.

The fuel grid seismic capacity was found to be relatively low but higher than the SME, especially for a soil site, where it is estimated as 0.40g PGA, based on available information and the assumptions in the study. The fuel grid capacity does not govern the fragility of the plant, as a failure of rod drop due to mechanical blockage leads to an ATWS sequence for which EPR is tolerant in the short term and RBS [EBS] injection ensures reactivity control in the medium term. Also, the failure criterion considered for the fuel grid capacity calculation is the initiation of buckling of a grid in a fuel assembly, which should not prevent rod drop. More detailed analysis of the ability to insert control rods is expected to show larger margins at the stage of detailed design.

A specific analysis was performed for the LOOP, small LOCA and ATWS scenario. For these accident sequence models, the list of cutsets was evaluated. The single cutsets and cutsets with operator action failure or random non-seismic component failure are analysed in particular. The following procedure was followed to evaluate the accident sequence:

The frequency of the initiating events SLOOP, S SLOCA and S ATWS was set to 1.0 to consider their occurrence following the seismic event. The seismic LOOP event tree is linked to the seismic ATWS model, as shown in Sub-chapter 15.6 - Figure 1.

The system seismic fragility was taken into account by introducing basic events for the fragilities in the model with a bounding high failure probability of 0.1, so that the cutsets appear at the top of the list when the ET is evaluated. The following table presents the basic events introduced in the seismic model of the SMA. These basic events represent the governing seismic failure mode of the system. In order to conservatively take into account the common cause failure of identical equipment of a given system, the same basic event was used for all trains of the same system.

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Also, the following equipment was set to failure:

- RCV [CVCS] for its Reactor Coolant Pump seal injection function, and RCP [RCS] makeup,
- AAD [SSS] for the Feedwater function,
- AC Power for normal power feeding. No recuperation is considered after a seismic event,
- GCT [MSB], used for partial cooldown.

The probability of failure of human error was set to 1.0 in the model to ensure that operator actions were included in the cutsets at the top of cutset lists to allow individual analysis.

Seismic LOOP

The following table shows single element seismic failure cutsets and main cutsets that include operator error and random failure contributing to seismic LOOP accident sequences down to a frequency of about 1.0E-3/yr, which represents an acceptable level of conditional probability of failure of mitigation. Only single element seismic failure cutsets are shown except as required to show human failure contribution and random non seismic failure.

Seismic Failures	Random Equipment Failures	Human error	Remarks
SEIS I&C			No automatic action or instrumentation for the operators
SEIS AC			No cooling possible (ASG [EFWS] or F&B)
SEIS BAT			No power for I&C (AC or DC)
SEIS EFWS		OP_BLEED_120MN	Failure of secondary cooling and failure of operator to initiate F&B within 120 min
SEIS EDG		OP_SBODG2H	No AC Power
SEIS SEAL LOCA and SEIS EDG		OP_EFW/MSRT_2H_LOCAL	Seal LOCA without initiation of secondary cooling by the operators
SEIS EFWS and SEIS SADV	LHP_DFR or LHS_DFR	OP_SBODG2H	No secondary cooling possible and failure of Bleed
SEIS EFWS and SEIS SADV	LHQ_DFR or LHR (i.e. one EDG failure)		No secondary cooling possible and failure of Bleed
SEIS EDG	RCP_SEAL (#1 and #2)	OP_EFW/MSRT_2H_LOCAL	Severe seal damage after EDG failure without initiation of secondary cooling by the operators
SEIS SEAL LOCA and (SEIS ESWS or SEIS CCWS)		OPE_52	Seal LOCA without LHSI HX cooling, and operator fails to initiate IRWST cooling with EVU [CHRS] within 4 hrs
SEIS SEAL LOCA and (SEIS LHSI or SEIS EDG)		OPE_52	Seal LOCA without LHSI for IRWST cooling and operator fails to initiate IRWST cooling with EVU [CHRS] within 4 hrs
SEIS EFWS and SEIS LHSI		OPE_52	Failure of secondary cooling, and LHSI for IRWST cooling and operator fails to initiate IRWST cooling with EVU [CHRS] within 4 hrs
SEIS SEAL LOCA and (SEIS MHSI or SEIS EDG or SEIS ESWS or SEIS CCWS)		OP_FSCD_30MN	Failure of MHSI and failure of the operator to initiate FSCD within 30 min
SEIS EDG and SEIS SEAL LOCA		OP_SBODG30M	Seal LOCA in SBO condition and operator failure to start SBO DG within 30 min

Failure of I&C for automatic action of the start of SBO diesel generators or manual initiation of secondary cooldown is backed up in the model by operator action. In the case of manual start and control of ASG [EFWS] for secondary heat removal following PS failure, I&C availability is required for the operator action. In the present study, it is then considered that seismic failure of the I&C will lead to core damage and thus resulting in a single seismic cutset.

There are no operator error single element cutsets. This means that a demand for an operator action only occurs after the failure of an automatic action or of a system. The effective seismic capacity of the resulting cutset will be higher than the capacity of the system as operator reliability should be taken into account in the HCLPF of the cutset.

However, I&C, batteries and AC power are particularly important as they form seismic failure single element cutsets. The fragility calculation shows a HCLPF above the SME for I&C, batteries and AC switchgear. Also, the capacity of I&C components is assessed globally in the current study. It will be possible to assess the local dependency of I&C and switchgear failures more precisely when more detailed information is available on the plant design during the UK EPR plant construction phase.

The first cutset not involving any seismic equipment failures following the seismic LOOP initiating event involves the following failures:

- LHP__DFR_D-ALL: Common Cause Failure to run of the EDGs, and
- OP_SBODG: the operators fail to start the SBO diesel generators after the failure of the EDGs.

Failure of AC power prevents residual heat removal by secondary cooldown or Feed & Bleed. The conditional probability of this undesired situation is about 1.2E-3 per event. However, the frequency of a seismic LOOP is relatively low, and the manual startup of the SBO diesels by the operator is expected to be a reasonably reliable action (as the SBO event diagnosis is obvious). These considerations ensure that the contribution from non seismic failures during seismic initiators is low.

Seismic Small LOCA

The following table shows the single element seismic failure cutsets and top level cutsets showing operator error and random failure contributions for the seismic small LOCA accident sequence (1.0E-3 frequency cut-off). Only single element seismic failure cutsets are shown except as required to show human failure contribution and random non-seismic failure.

Seismic Failures	Random Equipment Failures	Human error	Remarks
SEIS BAT			No power for I&C and diesels start up
SEIS I&C			No automatic action or instrumentation for the operators
SEIS AC			No cooling possible (ASG [EFWS] or F&B)
SEIS EFWS			Failure of partial secondary cooldown
SEIS MSRT			Failure of partial secondary cooldown

Seismic Failures	Random Equipment Failures	Human error	Remarks
SEIS EDG or SEIS CCWS or SEIS MHSI or SEIS ESWS		OP_FSCD_30MN	Failure of MHSI and failure of the operator to initiate FSCD within 30 min
SEIS EDG		OP_SBODG2H(/30MIN /15M)	Failure of all AC power (no EDG or SBO start)
SEIS LHSI or SEIS ESWS or SEIS CCWS	LHS_DFR and LHP_DFR	OP_SBODG2H	Failure of IRWST cooling with LHSI or EVU [CHRS]
SEIS ESWS or SEIS LHSI or SEIS EDG or SEIS CCWS		OPE_52	Operator fails to initiate IRWST cooling with EVU [CHRS] within 4 hrs after loss of LHSI (injection or cooling)
SEIS EDG	LJP_DFR or LJS_DFR or PM		Failure of all AC power

For the seismic small LOCA, there are no single element operator failure cutsets.

As for the seismic LOOP scenario, failures of I&C, batteries and AC power are important as they appear in single seismic failure cutsets. In addition, the seismically induced failures of the ASG [EFWS] or the VDA [MSRT] appear in single seismic failure cutsets, since the RCV [CVCS] is conservatively assumed to be unavailable as it is not SC1 classified. For all single seismic failure cutsets the HCLPF is higher than SME.

The first cutsets on the list not involving any seismic equipment failures for the seismic SLOCA initiating event have a conditional frequency of about 1.2E-3 per event. These cutsets include the following failures:

- LHP__DFR_D-ALL: Common Cause Failure to run the EDGs, and
- Operator action:
  - OP_SBODG30M: the operators fail to start SBO diesel generators, or
  - OP_FSCD_30MN: the operators fail to initiate Fast Secondary Cooldown within 30 minutes.

The loss of AC power in a LOCA+LOOP event (EDG and SBO) induces failure of LHSI injection and hence core uncovery. If the EDGs fail, the SBO diesel generators must be started to provide ultimate AC power to perform LHSI injection and remove the residual heat. Operator actions are necessary to start the SBO diesels, initiate fast cooldown to reach the LHSI injection pressure and, in the longer term, to initiate IRWST cooling. The frequency of a seismic S LOCA is expected to be lower than that of a seismic S LOOP. The manual start of SBO is the most critical action as the grace period for electrical power recovery is short. However, the probability of failure of EDGs is low, and the SBO event diagnosis is obvious. Therefore the risk due to these cutsets is expected to be low.

Seismic ATWS

There are three single element seismic cutsets identified in the analysis of seismic ATWS following LOOP. They correspond to the seismic failure of:

- I&C,
- batteries, or
- seismic failure of AC switchgear.

If no I&C reactor trip signal is generated (due to failure of I&C or failure of electrical support), the control rods will not insert and no ATWS signal will be produced. The control rods will not insert as the RGL [CRDM] are powered by uninterruptible supplied powered by 2 hour batteries. These batteries are SC1 classified but are located in Conventional Buildings: they are conservatively assumed not to fail because of the seismic event. Therefore after a seismic LOOP, with no reactor trip signal, the batteries are assumed to keep the RGL [CRDM] withdrawn. The seismic capacity of the I&C is above SME. It will be possible to assess the local dependency of I&C failures more precisely when more detailed information is available on the plant design during the UK EPR plant construction phase.

The following table presents the main cutsets showing operator error and random failure contributions for the seismic ATWS accident sequence (1.0E-3 frequency cut-off).

ATWS initiator after seismic LOOP	Seismic Failures	Random failure	Remarks
	SEIS BAT		No power for I&C and diesels start up
	SEIS I&C		No automatic action (reactor trip) or instrumentation for the operators
	SEIS AC		No AC power for boron injection and secondary cooldown
SEIS RT MB	SEIS RBS		Failure of Boron injection
SEIS RT MB	SEIS PSRV		Failure of overpressure protection
SEIS RT MB	SEIS EFWS		Failure of heat removal by secondary cooldown
SEIS RT MB	SEIS EDG		No AC power for boron injection or secondary cooldown
SEIS RT MB		LHP_DFR and LHS_DFR	No AC power for boron injection and secondary cooldown

For seismic LOOP ATWS, there are no single element operator failure cutsets. Also, the probability of failure of the PS reactor trip signal or actuators is remote and the seismic ATWS is more likely to occur in practice due to failure of control rod insertion due to mechanical blockage of the rods. The EPR is tolerant to ATWS in the short term, and RBS [EBS] injection ensures reactivity control in the medium term. The seismic capacity of the RGL [CRDM] with respect to blockage is just above SME for rock sites and soil site: however it is seen that mechanical blockage will not by itself lead to core damage. The only single element seismic failure cutsets involve failure of I&C, batteries and AC power: it can be seen that ASG [EFWS], the PSRV, RBS [EBS] and EDG are key systems, as their failure to mitigate seismic ATWS after seismic LOOP leads to core damage.

The highest level non-seismic cutset has a very low conditional frequency and is not significant for the S ATWS accident sequence. It is concluded that ATWS mitigation is robust and high margins are available with respect to seismic risk due to ATWS.

7. CONCLUSIONS

The seismic capacity of the EPR has been assessed using a PSA-based SMA approach, using the Level 1 PSA for internal events and elements of the Level 2 PSA.

The analysis results in a seismic capacity of the plant of 0.60g PGA for rock sites and 0.61g PGA for soil sites.

Therefore, it has been demonstrated that the seismic capacity of the UK EPR is higher than the Seismic Margin Earthquake defined as 1.6 times the Design Basis Earthquake (corresponding to 0.4g PGA). The Seismic Margin Assessment has shown that there are no cliff edge effects for seismic events with magnitudes above that assumed in the design basis.

The SMA analysis has provided the following insights:

- After a seismic initiating event, there is no single operator action whose failure would lead to core damage. Operator actions are only required following failure of automatic protection actions, which rely on safety systems that have a seismic capacity that is above the SME.

- The PSA-based SMA has also shown that the risk related to random failure of systems is expected to be low, as non-seismic random failures which dominate the risk have a low frequency.

For both sites (rock and soil), the plant HCLPF is governed by a single element seismic failure cutset, namely SEIS I&C (0.60g) for the rock site and SEIS AC (0.61g) for the soil site. The capacities of all other systems or functions are equal to or higher than these capacities (for the respective site), with the exception of a reactor trip due to mechanical blockage and SEIS RT MB (0.4g), which does not appear in the single element seismic failure cutsets.

This implies that, due to the methodology characteristics for the seismic margin assessment of the entire plant (see section 2.4), the obtained plant HCLPF is a lower bound with respect to accident sequences involving seismic failures (of components present in the SEL) that are not explicitly modelled in the PSA model for this SMA.

In particular, this applies to seismic events during shutdown states. In accordance with the above statements, all components which are specifically relevant to shutdown states have a

HCLPF capacity equal to or higher than the respective plant HCLPF capacity. It can then be concluded that:

- the plant HCLPF for shutdown states cannot be lower than that for power states,
- the conclusions derived from this SMA cannot be challenged by the extension of the fault tree based analysis to shutdown states.

The seismic capacities of Structures, Systems and Components assessed within GDA are conservative assessed based on available design information available at the GDA stage. When further design details and site specific details are available the seismic capacity calculations are expected to show additional margins.

SUB-CHAPTER 15.6 – TABLE 1

Summary of Seismic Capacity of Structures of the SEL of UK EPR

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SUB-CHAPTER 15.6 – TABLE 2

Summary of Seismic Capacity of Systems and Components of the SEL of UK EPR (Rock Site and Soil Site)

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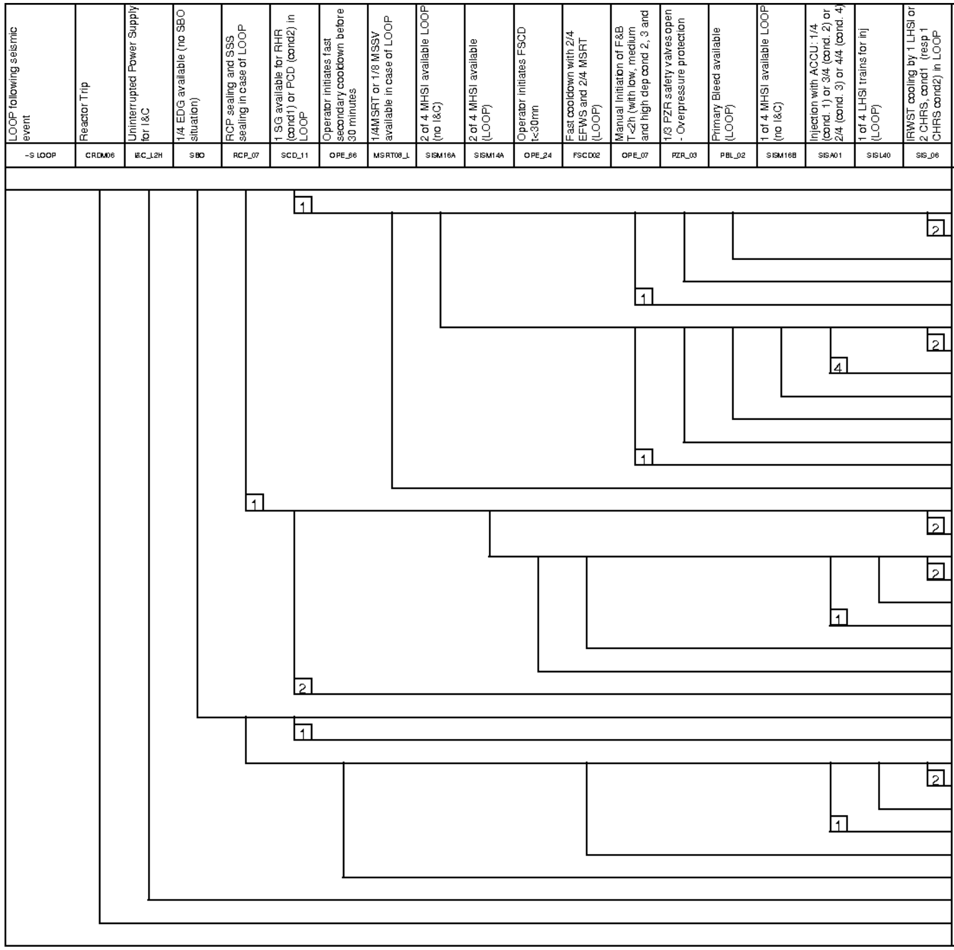
SUB-CHAPTER 15.6 - FIGURE 1

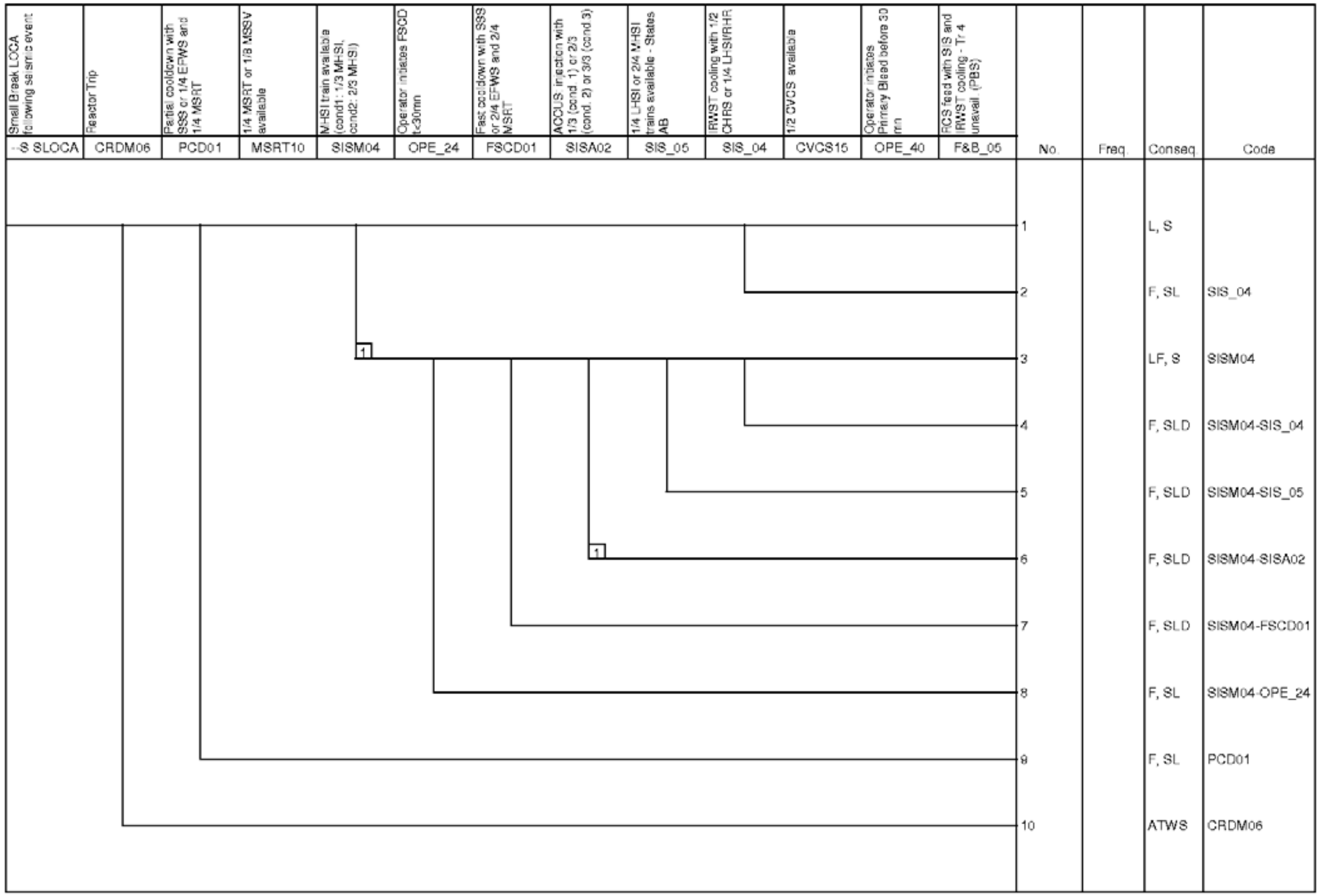
Plant Response Event Tree for seismic LOOP: S LOOP



No.	Freq.	Conseq.	Code
1		S, T	
2		LF-LOOP, S	SCD_11
3		F, F LOOP, TP	SCD_11-SIS_06
4		F, F LOOP, TP	SCD_11-PBL_02
5		F, F LOOP, TP	SCD_11-PZR_03
6		F, F LOOP, TP	SCD_11-OPE_07
7		LF-LOOP, S	SCD_11-SISM16A
8		F, F LOOP, TP	SCD_11-SISM16A-SIS_06
9		F, F LOOP, TP	SCD_11-SISM16A-SISA01
10		F, F LOOP, TP	SCD_11-SISM16A-SISM16B
11		F, F LOOP, TP	SCD_11-SISM16A-PBL_02
12		F, F LOOP, TP	SCD_11-SISM16A-PZR_03
13		F, F LOOP, TP	SCD_11-SISM16A-OPE_07
14		F, F LOOP, TP	SCD_11-MSRT08_L
15		S, L-LOOP	RCP_07
16		F, F LOOP, SP	RCP_07-SIS_06
17		LF-LOOP, S	RCP_07-SISM14A
18		F, F LOOP, SPD	RCP_07-SISM14A-SIS_06
19		F, F LOOP, SPD	RCP_07-SISM14A-SISL40
20		F, F LOOP, SPD	RCP_07-SISM14A-SISA01
21		F, F LOOP, SPD	RCP_07-SISM14A-FSCD02
22		F, F LOOP, SPD	RCP_07-SISM14A-OPE_24
23		F, F LOOP, SP	RCP_07-SCD_11
24		S	SBO
25		F, F LOOP	SBO-SCD_11
26		S	SBO-RCP_07
27		F, F LOOP	SBO-RCP_07-SIS_06
28		F, F LOOP	SBO-RCP_07-SISL40
29		F, F LOOP	SBO-RCP_07-SISA01
30		F, F LOOP	SBO-RCP_07-FSCD02
31		F, F LOOP	SBO-RCP_07-OPE_66
32		F, F LOOP, SP	I&C_L2H
33		ATWS	CRDM06

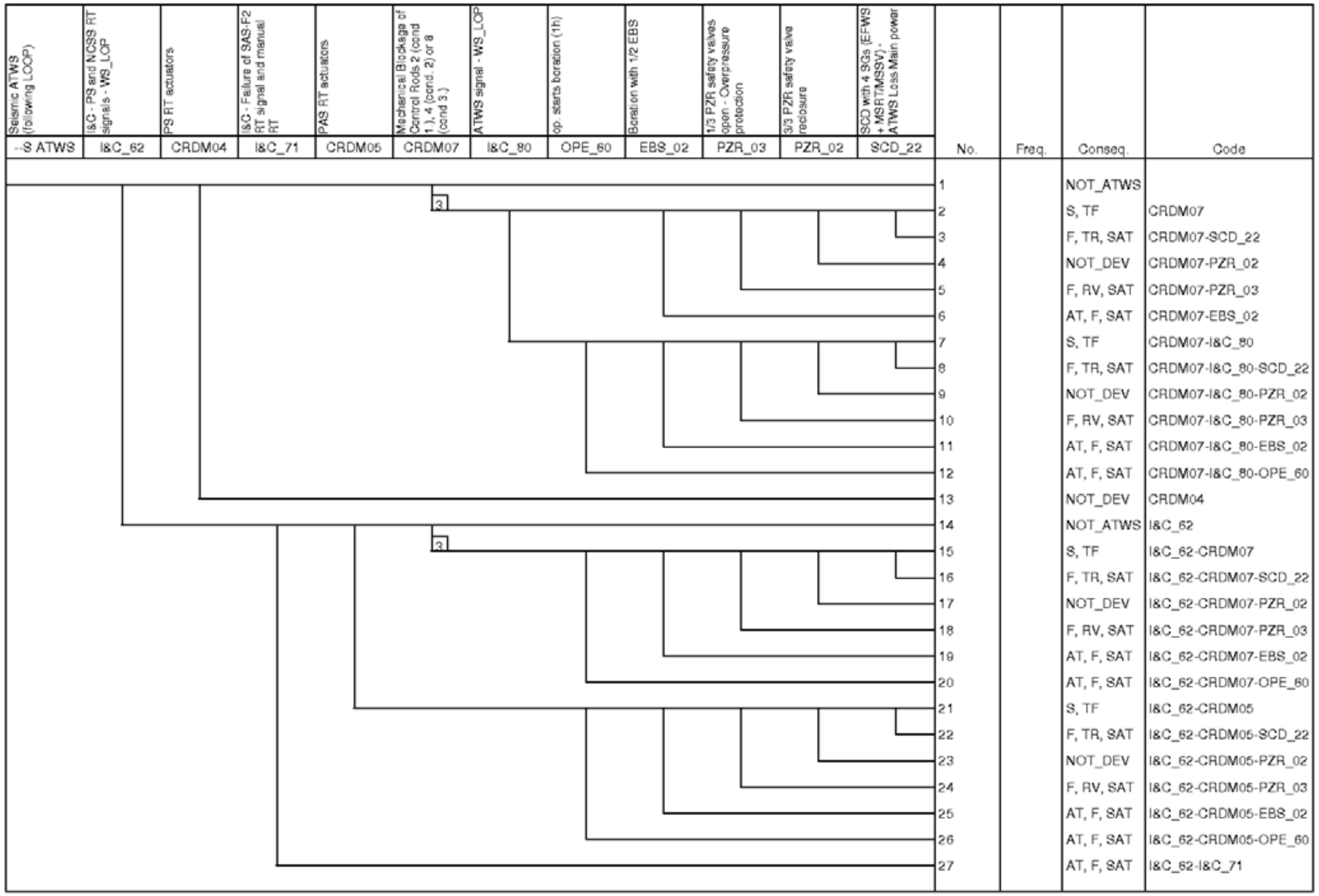
SUB-CHAPTER 15.6 - FIGURE 2

Plant Response Event Tree for seismic small LOCA, cumulated to LOOP: S SLOCA



SUB-CHAPTER 15.6 - FIGURE 3

Plant Response Event Tree for seismic ATWS following LOOP: S ATWS



SUB-CHAPTER 15.6 - REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

1. OBJECTIVE OF THE SEISMIC MARGIN ASSESSMENT (SMA) FOR THE UK EPR

[Ref-1] UK Health and Safety Executive (HSE). "Safety Assessment Principles for Nuclear Facilities". 2006 Edition Revision 1. January 2008. (E)

[Ref-2] UK Health and Safety Executive (HSE). Nuclear Safety Directorate. Technical Assessment Guide. "External Hazards". TAG T/AST/013.
and associated SAPs: (P16, P21, P33, P42 to P44, P72, P119 to P140). TAG T/AST/013. (E)

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2. GENERAL METHODOLOGY

2.2. IDENTIFICATION OF THE SEISMIC EQUIPMENT LIST AND SEISMIC FRAGILITY EVALUATION

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[Ref-2] UK EPR GDA Seismic Fragilities of Primary Equipment for Use in Seismic Margin Assessment. NEEL-F DC 87 Revision A. AREVA. May 2009. (E)

[Ref-3] Budnitz et al.. An Approach to the Quantification of Seismic Margins in Nuclear Power Plants. NUREG/CR-4334. U.S. Nuclear Regulatory Commission. (E)

2.4. SEISMIC MARGIN EVALUATION FOR PLANT

[Ref-1] Watanabe, Y., Oikawa, T., Muramatsu, K. Development of the DQFM method to consider the effect of correlation of component failures in seismic PSA of nuclear power plant, Reliability Engineering and System Safety, Vol. 79 (2003), 265-279, 2003. (E)

3. SEISMIC HAZARD ANALYSIS

Median Ground Response Spectrum

[Ref-1] UK Health & Safety Executive (HSE) Nuclear Directorate Assessment Report – New Build – Step 2 EPR Civil Engineering and External Hazard Assessment. (E)

5. SEISMIC FRAGILITY EVALUATION

[Ref-1] UK Health & Safety Executive (HSE) Nuclear Directorate Assessment Report – New Build – Step 2 EPR Civil Engineering and External Hazard Assessment. (E)

5.1. SEISMIC MARGIN CALCULATIONS

[Ref-1] J W Reed and R P Kennedy. Methodology for Developing Seismic Fragilities. EPRI TR-103959. Electrical Power Research Institute. Palo Alto. CA. June 1994. (E)

[Ref-2] M.K. Ravindra. “Seismic Fragilities of Structures and Equipment of UK European Pressurized Reactor for Use in Seismic Margin Assessment”. Revision 0. March 2009. (E)

[Ref-3] UK EPR GDA Seismic Fragilities of Primary Equipment for Use in Seismic Margin Assessment. NEEL-F DC 87 Revision A. AREVA. May 2009. (E)

5.2. WALKDOWN FOCUS

[Ref-1] External Events Standard. ANSI-ANS-58.21-2003. American Nuclear Society (ANS). (E)