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# SEISMIC PSA OF ANGRA 2 NPP – MODELLING ASPECTS AND INSIGHTS ON BEYOND-DESIGN CAPACITIES

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## ABSTRACT

The present contribution deals with the Seismic PSA of the "Angra 2" unit, a four-loop pressurized water reactor (PWR) with a 1280 MWe capacity. A full-scope seismic PSA including a seismic walkdown and fragility analyses of varying levels of sophistication has been performed. For the risk integration with RiskSpectrum®, the augmented level 1 PSA model has been used in combination with preliminary seismic uniform hazard response spectra (UHRS) for the Angra site.
The contribution focusses on the modelling of seismic-induced initiating events and failures of safety-relevant systems, structures and components (SSC), within the existing level 1 PSA model of the unit. In particular, the modelling of the mutual dependence between seismic-induced failures of components belonging to the same safety system is discussed.
Fragility analyses were performed with the separation-of-variables method (Kennedy and Ravindra, 1984) for the civil structures (reactor building, emergency feed building, emergency power and chilled water supply building, switchgear building and service cooling water pump house), as well as for numerous components including those of the nuclear steam supply system (NSSS).
The paper addresses the resulting qualitative and quantitative insights on beyond-design capacities of safety-relevant SSC.

## INTRODUCTION

### *Plant Description and Context*

The nuclear power plant (NPP) Angra 2 is a four loop pressurized water reactor (PWR) with a rated power output of 1280 MWe, located on the Atlantic ocean at Itaorna bay (state of Rio de Janeiro, Brazil). Angra 2 was designed by the Kraftwerk Union AG (Siemens/KWU; now AREVA GmbH) and NUCLEN (now ELETRONUCLEAR); power operation started in the year 2000.
In order to comply with the conditions associated with the permanent operation authorization of Angra 2, a probabilistic safety analysis (PSA) including seismic-induced initiating events is required by the Brazilian nuclear authority (CNEN).

### *Seismic Design*

The external event design philosophy of Angra 2 is to ensure that the fundamental safety objectives (reactor shutdown and long-term sub criticality; residual heat removal; radioactivity confinement) are met

in the event of earthquakes, taking into account the site seismicity. Components required for the fulfilment of the fundamental safety objectives are classified into seismic class 1.
The seismic design concept of Angra 2 considers: i.) an operating basis earthquake (OBE)[1], ii.) a safe-shutdown earthquake (SSE), and additionally, iii) the load case SSB from combined effects of the SSE and a burst pressure wave (BPW) resulting from the loss of integrity of the main feedwater tank.
The seismic ground motion response spectra representing SSE are typical design spectra corresponding to the common practice at the onset of seismic design for Angra 2, proposed by Weston Geophysical Research, and similar to the design spectrum defined in the first version of the Regulatory Guide 1.60 (USAEC (1973)), with a plateau of maximum spectral acceleration in the 2 to 8.5 Hz frequency range. The horizontal peak ground acceleration (PGA) of the SSE was defined as 0.1 g; the vertical component is two thirds of the horizontal motion.
In case of SSB, the three fundamental safety objectives are ensured by the following front-line systems:

- Sub-criticality
    - Reactor trip by rod insertion (SCRAM)
    - Extra borating system
- Residual heat removal
    - Reactor coolant system integrity + extra borating system (inventory control)
    - Emergency feedwater system
    - Main steam system, from steam generator to main steam valve station
    - Main feedwater system, up to main feedwater isolation valves
    - Emergency residual heat removal trains
- Activity confinement
    - Containment

***Site Conditions***
The subsoil at the Angra 2 unit is composed of quaternary sediments; the main buildings and galleries of the plant are therefore supported by piles. In view of these conditions, the seismic design of SSC at Angra 2 was based on structural dynamics models accounting for the soil-structure-interaction (SSI).
Following the evolution of seismic hazard assessment methods, probabilistic seismic hazard analyses (PSHA) have been developed and updated for the Angra site over the last two decades.
For the SPSA discussed in this paper, preliminary hazard curves and uniform hazard spectra (ELETRONUCLEAR (2014)) are used.

## PSA-MODELLING

***Method***
An existing Probabilistic Safety Analysis (PSA) model for Angra 2 was the basis for the Seismic PSA (SPSA) discussed in the present paper. The model was originally developed in CAFTA® and afterwards transferred to RiskSpectrum®, which is the tool of choice for this analysis.

***Event Sequences***
Following the recommendations in ASME (2009), a hierarchical seismic event tree was adopted, as shown in the following Figure 1.

With the exception of the sequences “catastrophic failure of critical SSC” and “plant shutdown for inspection”, the fragility of the equipment whose failure leads to a seismic-induced sequence is based on generic fragilities in the literature (see e.g. Park et.al. (1998)) and on considerations based on the macroseismic-intensity scale (EMS (1998)). For the seismic-induced small LOCA, the fragility is based on SAND (1988).

[1] Formally, in the case of Angra 2 the term “design basis earthquake (DBE)“ is used instead of the term „OBE“.

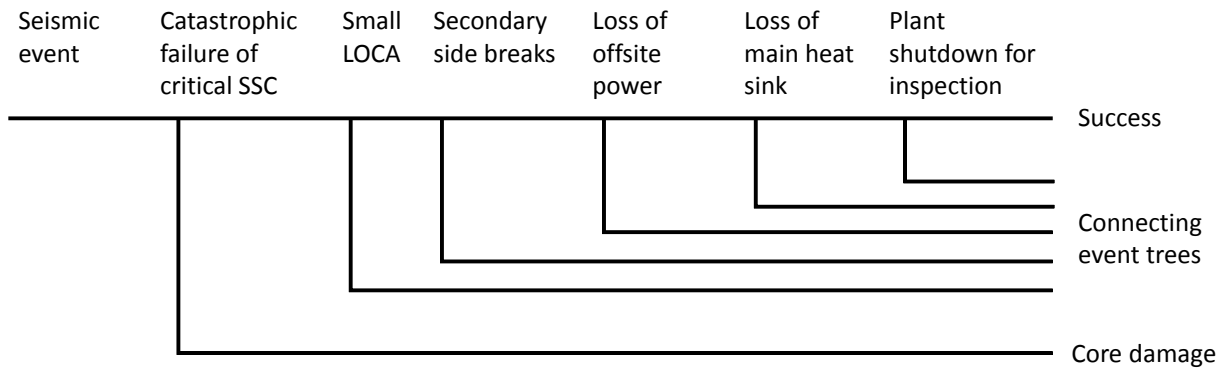


Figure 1. Hierarchical seismic event tree

The event sequences “Secondary side breaks” and “Loss of main heat sink” are assumed to be governed by the turbine building, which is not included in seismic class 1. The fragility is defined with respect to damage grade 2 (moderate damage). For this level of damage, loss-of-integrity of the main feedwater tank is not judged to be a credible scenario, considering its structural configuration (vertical support by five tank saddles with embedding angle of at least 180°; horizontal constraints by cleats welded to the tank in proximity of the tank saddles; constrained rotational motion along longitudinal axis). Therefore, a superposition of a consequential burst pressure wave is not considered for these event sequences.

***Seismic Equipment List***

In the SPSA for Angra 2, the Seismic Equipment List (SEL) has been developed based on the existing Level-1 (L1) PSA model for internal events. More specifically, in its first evolutionary stage the SEL was populated with systems/components whose failures contribute to the event sequences (see Figure 1) considered in the SPSA. In L1-PSA for internal events, failures of active components are modelled predominantly. In the second evolutionary stage, the SEL was augmented by passive components of the systems (or system parts) for which active components are modelled in the L1-PSA. Seismic events induce loads both in active and passive components. Thus, passive components are to be considered in the SPSA, too. In the following Table 1, the resulting number of individual equipment units are itemized for different equipment classes, adapted from the classification in EPRI (1991).

Table 1: Seismic equipment list - number of items per equipment class

| **Equipment Class (EPRI)** | | **Quantity** |
|---|---|---|
| 1 | Batteries | 8 |
| 2 | Battery chargers | 12 |
| 3 | Chiller | 8 |
| 4 | Control / instrumentation devices | 556 |
| 5 | Diesel generators | 8 |
| 6 | Fans | 26 |
| 7 | Instruments (on racks) | 237 |
| 8 | NSSS | 7 |
| 9 | Pumps (including reactor coolant pumps) | 60 |
| 10 | Switchgear devices | 226 |
| 11 | Tanks / Heat Exchangers | 78 |
| 12 | Transformers | 23 |
| 13 | Valves | 456 |
| | Total | 1705 |

***Fault Tree Modelling***

Seismic induced failures of systems and components are modelled by **dedicated seismic fault trees**, which are linked to the existing fault trees of **safety systems or functions** of the L1 PSA model. The seismic fault trees define the seismic induced failure of entire systems (or sub-systems), resulting in a better readability of the model, compared to the case in which individual fragility basic events are added to each non-seismic basic event. The seismic fault trees are connected to the system fault trees via transfer gates that are added to the OR-gate collecting the failure modes of the corresponding system. This is exemplified in the following Figure 2:

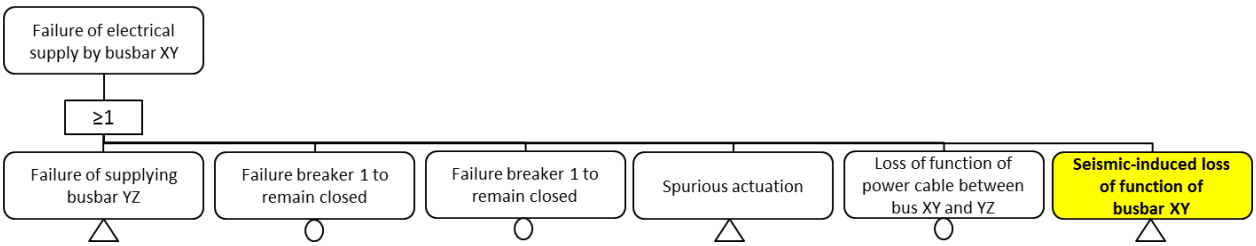


Figure 2. Exemplary fault tree with transfer to dedicated seismic fault tree

In Figure 2, transfer gates are indicated by triangular icons, whereas circular icons indicate basic events.
The seismic fault trees includes fragility basic events for the corresponding building, except for the structural failure of the reactor building, which is assumed to lead directly to core damage (event sequence "catastrophical failure of critical SSC"). If applicable, the seismic fault trees also include fragility basic events for relevant passive components, e.g. cable support structures for electrical systems.

***Correlations – General considerations for simplified configurations***

During a seismic event, <u>all</u> components are subject to inertial loads that can be traced back to a common cause, i.e. the ground motion. This raises the issue of mutual dependence between seismic induced failures, i.e. the question whether the seismic induced failure of a component is more probable, under the condition that another component experienced seismic induced failure. In the context of seismic fragility, the dependence is quantified in terms of correlations between the capacities of the individual components. See e.g. Budnitz et.al. (2015) or Pellissetti and Klapp (2011).

NPP safety systems typically have both a parallel and a serial structure, as represented in a simplified way in Figure 3. Different components of the same safety train (e.g. A1 and B1) are connected in series, whereas the two trains (1 and 2) are connected in parallel.
The relevance and consequences of various combinations of "binary correlations[2]" (i.e. either zero or total correlation) are discussed next.

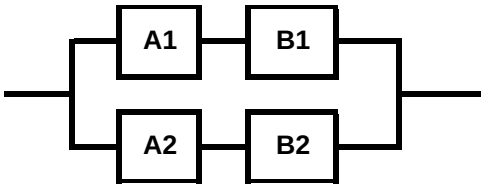


Figure 3. Simplified system diagram with components connected in series and in parallel

[2] The consideration of non-binary correlations (see Pellissetti et.al. (2015a)) requires extensive input data, such as probabilistic floor response spectra.

Four different cases of binary correlations can be distinguished:

1. The capacities of all four components are uncorrelated.
2. The capacities of all four components are fully correlated.
3. The capacities of the type-identical components (e.g. A1 and A2) are fully correlated; thus, the system has de-facto no redundancy. The capacities of different components (e.g. A1 and B1) are uncorrelated.
4. The capacities of the components in one train are fully correlated (e.g. A1 and B1), while there is no correlation between components of different trains (e.g. A1 and A2).

All four cases violate to some extent the common-sense based expectations regarding correlation:
The cases 1 and 4 assume that capacities of type-identical components are completely uncorrelated. This assumption is generally inadequate, because of the common source of the excitations (the same ground motion time history) and considering that for many safety systems, the individual trains are installed in the same building and with similar orientation.
In case 2, capacities of possibly very different components (e.g. a pump A1 and a switchgear B2) are assumed to be fully correlated. For components without any similarity full correlation is inadequate even these components are located on the very same floor slab, because the variability of the equipment response and – especially - of the component strength is uncorrelated[3].
In case 3, capacities of different components of the same train are assumed to be uncorrelated, even though their excitation is very similar (or identical). Case 3 leads to conservative results, because full correlation is detrimental for parallel components, and lack of correlation is beneficial for serial components.

***Correlations – adopted approach***
The configuration shown in Figure 3 is highly simplified. Actual safety systems involve a number of components of different type in each redundancy.
Therefore, the modelling adopted in the SPSA for Angra 2 is a combination of case 2 and case 3:

- Identical components of distinct trains are assumed to be fully correlated, i.e. redundancy is de-facto not credited.
- Component-specific fragilities were not introduced for all individual components. Instead, all seismic relevant components of a system are pooled – i.e. represented by a single basic event - if they are a.) of the same equipment <u>class</u> **and** b.) located <u>in the same building</u>. These components are then represented by a single basic event, which represents the seismic-induced failure of a pool of fully correlated components. Conservatively, the probability of this basic event is determined by the seismically most vulnerable component of the pool (e.g. the valve positioned at the highest elevation).

***Seismic-induced human error probability***
In general, each human task that is executed in case of earthquake fails with a certain Human Error Probability (HEP). This HEP is divided into a non-seismic part, that is independent from the earthquake, and a seismic part, that depends upon the characteristics of the earthquake.
Several models have been discussed in the literature, such as the Cumulative Absolute Velocity (CAV) model by Klügel (2007) and the “ramp-model” in Yokabayashi (1998).
In the seismic PSA for Angra 2, the seismic HEP has been modelled as a function of the PGA, enveloping the CAV model and the ramp model indicated in Figure 4.

[3] The total correlation is due to response correlation and strength correlation, see Pellissetti and Klapp (2011).

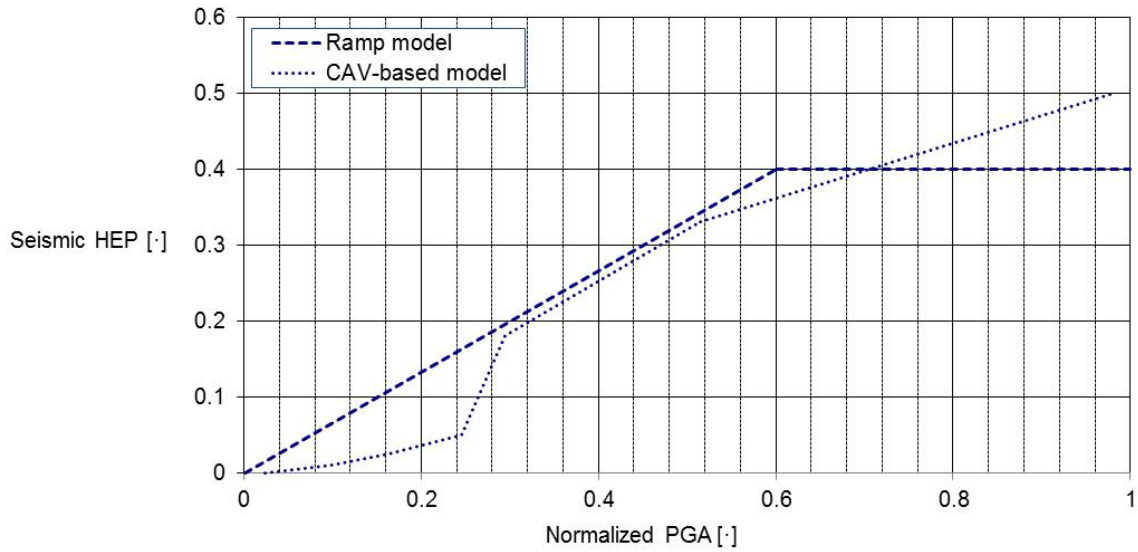


Figure 4. CAV model and ramp model for seismic human error probability

## FRAGILITY

### *Method*

Alternative methods were used by AREVA to determine the fragilities of SSC, with different level of detail (from low to high):

1. Generic fragilities based on earthquake experience and engineering judgment, EPRI (1991)
2. Plant specific fragilities based on existing design documentation
    a. Conservative deterministic failure margin (CDFM) method, EPRI (2009)
    b. Separation-of-variables, EPRI (1994)

For the safety relevant buildings, plant-specific fragilities are developed using the separation-of-variables method. For the individual equipment items, one of the three alternative methods is applied. The application of generic fragilities was preceded by a seismic walkdown conducted at the Angra 2 plant, following the guidance in EPRI (1991), ensuring that the equipment in the experience database is representative of the components encountered in the plant under investigation.

### *Structural Failure Modes*

The buildings considered in the SPSA of Angra 2 are as follows (see the site layout in Figure 5 below):

- Reactor building interior structure (UJA), reactor building annulus (UJB) and main steam and feedwater valve compartment (UJE)
- Switchgear building (UBA)
- Emergency power and chilled water supply building (UBP)
- Emergency feed building (ULB)
- Service cooling water pump house (UQB)

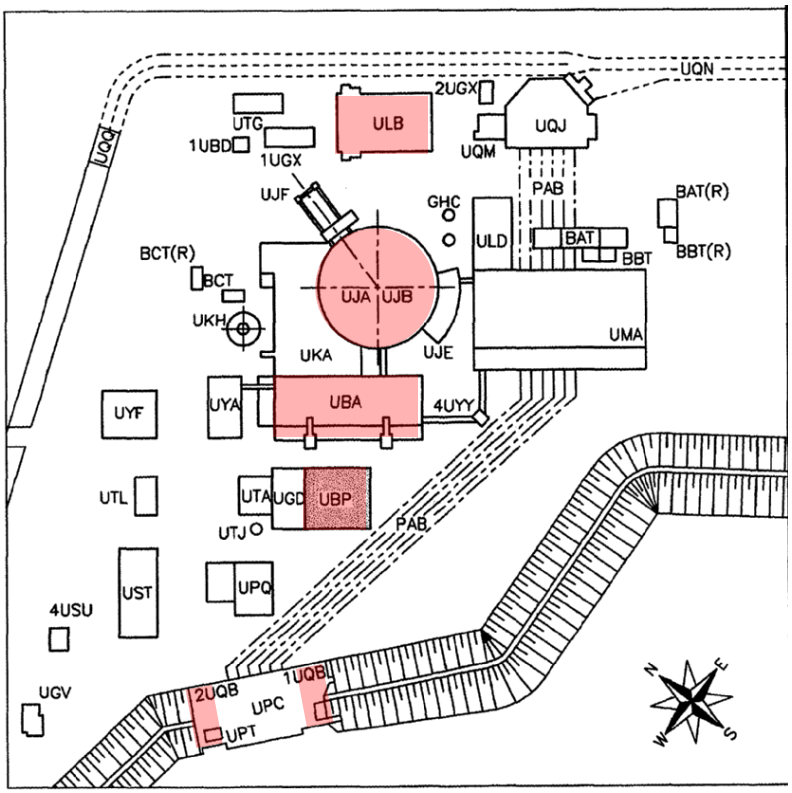


Figure 5. Arrangement of Angra 2 NPP buildings (red areas indicate buildings considered in SPSA)

The identification of the governing failure modes of the above buildings included the following steps:

- Review of fragility analysis of PWR plants with similar design.
- Review of available design documents / drawings (layout und loading).
- Identification / finding of the global load bearing system.
- Identification of critical loaded floors based on the maximum relative displacements resulting from dynamic analysis. Calculation of the displacements per floor (inter-story-drift ratio) based on the relative displacements.
- Identification of critical loaded floors based on internal forces of the building structure documented in available static and dynamic calculation reports

In the following Table 2, the normalized median strength factors of the governing failure modes are tabulated for each of the buildings considered in the SPSA. For concrete structures, the median strength factor is defined as the maximum scaling factor, by which the design ground motion can be multiplied ("scaled") without exceeding the ultimate capacity, under "median conditions". The meaning of "median conditions" is that all variables influencing the demand and capacity are set to the median value (50%-fractile). The strength factors of civil structures were estimated with the methods presented in EPRI (1994). Note that the cumulative fragility parameters depend also on other factors, namely the structural response factors and the inelastic energy absorption factors, which are out of the scope of the present paper.

Table 2: Normalized median strength factors of buildings considered in the SPSA

| UJA/B/E | UBA | UBP | ULB | UQB |
|---|---|---|---|---|
| 5.5 | 2.3 | 1.0 | 5.5 | 14.0 |

In the above table, the normalization was carried out with respect to the lowest strength factor, i.e. the one of the Emergency Power and Chilled Water Supply Building (UBP). The governing failure mode of that building is associated with an outer wall with several openings, at the elevation of the emergency diesel generators (see Figure 6 below). The horizontal load-carrying capacity is governed by the flexural capacity of the piers.

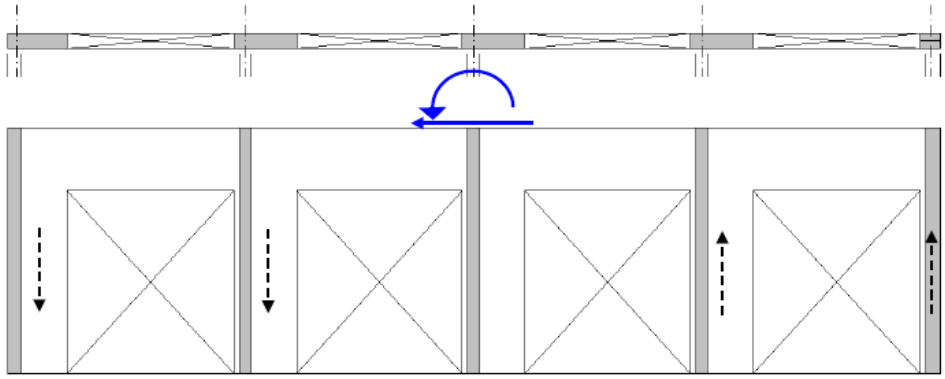

Figure 6. View of the outer wall of the UBP building

***NSSS Failure Modes***

An extensive review of the NSSS design reports was conducted, in order to identify the (sub-) components with the lowest margins, taking into account the extensive experience by AREVA in NSSS design. Based on this review, the fragility analysis focussed on the failure modes listed in the Table 3 below. The table indicates the consequence associated with the failure mode in the PSA and the **normalized** median strength factor.

In Table 3, the normalization was carried out with respect to the lowest strength factor, i.e. the one of the RPV (reactor pressure vessel) core barrel flange. The cumulative fragility parameters depend also on other factors, namely the structural response factors and the inelastic energy absorption factors.

Table 3: Normalized median strength factors of governing NSSS failure modes

| **NSSS (sub-)component** | **Consequence** | **Normalized median strength factor** |
|---|---|---|
| RPV support brackets | Catastrophic | 1.5 |
| RPV core barrel flange | ATWS[4] | 1.0 |
| Reactor core - fuel assembly spacer grid | ATWS | 1.4 |
| Main coolant line - cold leg nozzle no. 3 | Large LOCA[5] | 3.2 |
| Pressurizer spray line | Small LOCA | 1.1 |
| SG (steam generator) upper stage steam drier system | SGTR[6] | 1.8 |

Regarding the consequence ATWS: the strength factor of the RPV core barrel flange is based on linear scaling of design loads, whereas the strength factor of the fuel assembly spacer grid is based on non-linear limit analyses[7] similar to the one presented in Pellissetti et.al. (2015b). It is plausible to expect that a significantly higher strength factor would result from non-linear limit analyses of the core barrel flange.

[4] Anticipated transient without SCRAM
[5] Loss of coolant accident
[6] Steam generator tube rupture
[7] In a „limit analysis“, the seismic excitation is scaled until the failure criterion is violated, in the case of the fuel assembly spacer grids this is the maximum permissible deformation.

The strength factor of the pressurizer spray line is based on the margin with respect to the code allowable; the actual margin are expected to be significantly higher, see Kennedy (2002).

***Test-qualified Equipment Failure Modes***

A key factor in the fragility of test-qualified equipment is the ratio between the modified test response spectrum $TRS_C$ and the required response spectrum $RRS_C$. In the following Table 4, the ratio is displayed for the various categories of test-qualified equipment units considered in the SPSA. The values are normalized with respect to the category with the lowest ratio (LV Switchgear in UBA).

I&C devices exhibit a significantly higher ratio $TRS_C/RRS_C$ – and hence a larger seismic margin – as these are tested at a higher acceleration level, compared to the level adopted for the switchgear devices.

Table 4: Normalized ratios between test response ($TRS_C$) and required response ($RRS_C$)

| | Switchgear | | | | | | | Batteries | | I&C | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | MV | LV AC | | | LV DC | LV DC-24 | | | | | | |
| | 4.16 kV | 230V | 480V | | 220V | 24 V | | 24 V + 220 V | | | | |
| Building | UBA | UBA | UBA | ULB | UBA | UBA | ULB | UBA | ULB | UBA | UBP | ULB |
| Qualification method | sine-sweep (individual device) | | | | | | | sine-sweep (rack-battery-assembly) | | sine-sweep (individual device) | | |
| $TRS_C/RRS_C$ (normalized) | 1.5 | 1.4 | 1.1 | 1.3 | 1.0 | 1.3 | 1.6 | 3.4 | 3.6 | 8.8 | 14.0 | 12.4 |

The cumulative fragility parameters used by AREVA in the seismic PSA quantification consider in addition other factors, such as the structural response factors.

## CONCLUSIONS

Besides quantifying the seismic-induced core damage frequency and the importance metrics for the individual seismic risk contributors, the seismic PSA for Angra 2 provides several insights:

- The modelling of seismic induced failures of systems and components in dedicated *system seismic fault trees* resulted in a significantly **improved readability** of the model, compared to the case in which individual fragility basic events are added to each non-seismic basic event.
- The **structural failure mode** with the lowest median strength factor is associated with the outer wall of the **UBP building**, featuring large openings at the elevation of the emergency diesel generators. The horizontal load-carrying capacity is governed by the **flexural capacity of the piers**.
- For the NSSS failure modes, the **strength factor** of the **core barrel flange** is lower than that of the fuel assembly spacer grid and hence governing the ATWS fragility. However, it is noted that the fragility of the fuel assembly spacer grids is analysed with a **higher level of detail**, including non-linear limit analyses.
- The plant-specific fragility analyses by AREVA reveal **significant margins in the seismic design**. These data provide a sound basis for a realistic quantification of the seismic robustness with respect to design-exceeding seismic events.

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