A History and Proposed Development of the Sizewell B Level 2 PSA

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Abstract: For the first ~30 years of Sizewell B's operation, the continued use of a largely unchanged Level 2 PSA has been justified. In recent years however there has been a renewed focus on the Sizewell B Severe Accidents Safety Case. In response to new industry Operating Experience, a number of plant modifications to reduce the risk associated with severe accidents at Sizewell B have been undertaken. Passive Autocatalytic Recombiners and additional methods for Containment Water Injection have been installed. Activities are currently underway to develop a Filtered Containment Venting System alongside the development of an updated severe accident mitigation procedure.

Given the potential impact of these modifications to influence the station risk metrics, a holistic update of the Level 2 PSA is being developed. The intent is to update the Level 2 PSA to reflect the modified plant, modern phenomenological understanding and to consider timescales of ~7 days.

This paper describes the original Level 2 PSA for Sizewell B. The approach to develop and quantify the Containment Event Trees, and how the Level 2 PSA was used to support the original demonstration that station risk was As Low As Reasonably Practicable is described. The paper goes on to describe the development activities, including the approach to develop Containment Event Trees in RiskSpectrum, for which a number of significant modifications and assumptions are required compared to the original Level 2 PSA.

Keywords: PSA, PRA, Level 2, Severe Accidents, Containment Event Tree, ALARP.

ABBREVIATIONS

ALARP	As Low As Reasonably Practicable
CCI	Core Concrete Interaction
CET	Containment Event Tree
CFSS	Containment Fire Spray System
CWI	Containment Water Injection
DBUE	Deployable Back-Up Equipment
DoB	Degree of Belief
FCVS	Filtered Containment Venting System
LPSA	Living PSA
MAAP	Modular Accident Analysis Program
PCSR	Pre-Construction Safety Report
PDS	Plant Damage State
POSR	Pre-Operational Safety Report
PSA	Probabilistic Safety Assessment
RC	Release Category
RPV	Reactor Pressure Vessel
SAA	Severe Accident Analysis
STC	Source Term Category
SZB	Sizewell B
UCR	UnControlled Release
RCS	Reactor Coolant System
HSR	Heat Sink Recovery

1. INTRODUCTION

Sizewell B (SZB) was the first, and currently remains the only, Pressurised Water Reactor built in the United Kingdom. The design for Sizewell B started in the 1980s, and was based on the Westinghouse / Bechtel Standardised Nuclear Unit Power Plant System concept. The Pre-Operational Safety Report (POSR) was issued to the regulator in November 1992, seeking to obtain consent for 'fuel load' in 1994. One of the key tools utilised in order to demonstrate the safety of the design was the 'full scope' three level Probabilistic Safety Assessment (PSA) developed to support the POSR. The main objectives of PSA were to provide evidence that the design had reduced risks As Low As Reasonably Practicable (ALARP). While a number of different 'risk metrics' were considered, specifically this was considered to have been met if the risk of death to an individual off-site could be shown to be less than 10^{-6} per year. [1]

This paper describes the scope and development of the original Level 2 PSA. The approach to develop and quantify the Containment Event Trees (CETs), and how the Level 2 PSA was used to support the demonstration that station risk was ALARP is described. It goes onto describe the rationale and scope of modernisation efforts specific to the Level 2 PSA.

2. A HISTORY OF THE PSA AT SIZEWELL B

A PSA was recognised as being an essential tool early in the development of Sizewell B. Initially a Level 1 PSA considering a range of internal initiators at power was developed to support the Pre-Construction Safety Report (PCSR) submitted in 1987.

During the licensing process, the assessment against probabilistic targets resulted in a number of impacts upon the design. An overall probabilistic target during the design phase was that the frequency of UnControlled Release (UCR) should be less than 10^{-6} per year. The UCR frequency was representative of those fault sequences which gives rise to doses in excess of 100mSv at the site fence.

The first comprehensive three level PSA for Sizewell B was developed to support the POSR in 1992, and was unusual at the time being more thorough than typically produced for PWRs as it considered all initiators (faults and hazards) at all power levels (including shutdown). In addition to the expanded scope, because the POSR PSA analysis was being used to support the safety report, it needed to be fully justified and so made use of *bounding* rather than best estimate assumptions.

It was clear that following its successful use in supporting the licensing of the station, ongoing use of the PSA would be of benefit to the station. A 'Living PSA' (LPSA) model with reduced conservatisms (i.e. relaxations to success criteria) and re-platformed in the RiskSpectrum software package was produced in 1996. The LPSA model is regularly updated to account for changes to plant design, operations, maintenance, equipment reliability, and to account for operational experience. Primarily, the scope of updates is concerned with the Level 1 PSA. The Level 2 and 3 PSA have remained broadly static and remain based upon the same Containment and Radiological Analysis utilised for the POSR PSA. The conditional probability of a Plant Damage State (PDS) leading to the different consequences of concern is essentially utilised. Thus, the impact of revised PDS frequencies calculated by the updated Level 1 LPSA on the Level 2 and Level 3 risk metrics can be assessed.

3. SUMMARY OF THE (ORIGINAL) LEVEL 2 PSA APPROACH, METHODOLOGY AND USE

In order to support the construction schedule, it was necessary to develop the different Levels of the PSA in parallel and therefore a number of interfaces were defined for the original PSA. Thirty PDSs form the interface between the Level 1 and Level 2 PSA, and twenty-two Release Categories (RCs) form the interface between the Level 2 and Level 3 PSA. The Level 2 PSA is concerned with the performance of the remaining Containment barrier and how the PDSs develop into RCs.

- PDSs were defined to represent groups of accident sequences. PDSs were defined enable the grouping of sequences judged to have similar characteristics in terms of both the potential accident progression and the resultant release of activity to the environment.
- RCs were defined to cover the complete range of consequences from within the design basis through to a UCR. Each RC has an associated set of release fractions which result in a predetermined dose. The RCs are arranged into 9 bands relating to the mSv dose (at the site fence, or at a point 3km from the reactor) at order of magnitude intervals. These dose bands are further sub-divided based on release characteristics such as the duration of the release, the availability of warning time, volatile/involatile contribution and time delay.

For the 13 PDSs in which the Containment may provide a barrier to release, the Level 2 PSA must assess the potential accident progression, how the Containment may respond and identify the probability of the PDSs developing towards different RCs. This is assessed by understanding the loads which may arise in the accident and their impact on the Containment and the capability of the Containment with regards to those loads.

This is achieved by utilising a 20 node phenomenological CET to assesses the response of Containment for each of the 9 at power and 4 shutdown PDSs. The CET is a tool to assess the impact of the physical processes occurring in a severe accident that affect the Containment integrity. Questions are asked at each node, in chronological order, to characterise the different ways in which the accident could progress and influence the Containment response. The rationale for construction of the CET was to adopt a number of rules to allow for a systematic selection of key phenomenological and functional events. For example, phenomenological events are considered only if they can significantly affect the release characteristics of fission products or the integrity of the Containment. The rules form a rationale which achieves a consistent treatment of phenomenological events, system functional events and accident management procedures within the CET.

Within the CET, Time Frames are defined to mark the various important stages of severe accident progression and the times of the major changes in fission product release behaviour. Time frames consider the phenomena in different accident phases:

- Time Frame 1: from accident initiation to core degradation,
- Time Frame 2: from core degradation to just before Reactor Pressure Vessel (RPV) failure,
- Time Frame 3: from just before to an hour after RPV failure,
- Time Frame 4: One hour following RPV failure and beyond.

The same structure was used for each PDS that required a CET. The quantification of each PDS-CET differed, based upon insights from several sources of information. Based on the assimilation of information from sources, probabilities must be assigned to the CET pathways to enable its quantification. A ‘Degree of Believe’ (DoB) scheme was developed as a reference framework for this purpose. The DoB scheme has two extremes. If an event is judged highly unlikely, the least likely event considered in the CET, a value of 10^{-4} is assigned. For an event judged almost certain to occur on the other hand is assigned a value of 0.9999. Values of 0 or 1 are not used: severe accident phenomena are recognisably complex and even for a high confidence events there is some residual uncertainty. Quantifying the 13 CETs necessitated the assignment of ~1000 probabilities [2].

Table 1. The Sizewell B ‘Degree of Belief’ Scheme for Level 2 PSA Quantification

Probability	**Qualitative descriptor**
0.9999	Extremely Likely (i.e., almost certain)
0.9	Very Likely
0.7	Likely
0.5	Indeterminate
10^{-1}	Probable
10^{-2}	Unlikely
10^{-3}	Very Unlikely
10^{-4}	Extremely Unlikely

Where necessary, the assessment of certain issues was supported by decomposing the problem into a number of constituent issues according to the physics of a phenomenon. Tools such as Decomposition Event Trees or the even more detailed Risk Oriented Accident Analysis Methodology were used in assessment of issues such as hydrogen combustion and alpha-mode containment failure due to in-vessel steam explosions respectively. The extent of decomposition was judged on an issue by issue basis [2].

The CET is used to assess the performance of the Containment for the Level 2 PSA. As such, a probabilistic assessment of the performance of the Containment was required. This included the assessment of potential gross failures and leakage failure modes. Gross structural failure modes are defined as failures providing a large leak area capable of leading to rapid depressurisation of the Containment. Three critical failure modes were identified: liner tears at the equipment hatch, at the personnel airlock, and hoop membrane failure in the cylindrical portion of the containment Wall. Failure pressure probability distributions were derived at a range of temperatures. For the MAAP Severe Accident Analysis (SAA) assessing thermal-hydraulic conditions, the median failure pressures from the study at 300°C were used since it represented the upper limit of the Containment temperature for sequences leading to a slow pressurisation [3]. The outcomes therefore considered by the CET are no failure (Containment remains intact with only design basis leakage), enhanced leakage, or gross failure. In addition, basemat failure due to Core Concrete Interaction (CCI) is considered in the final node of the CET.

To enable the Level 2 PSA to link to the Level 3 PSA, firstly, each path through the CET is allocated to a Source Term Category (STC). STCs are utilised to enable pathways to be grouped based on the presence of similar phenomena (e.g., ongoing CCI). For each STC, a reasonably bounding source term was calculated to identify the timing and magnitude of fission product release to the environment. Secondly, the STCs are grouped into a set of pre-defined RCs on the basis of similar environmental consequences.

The categorisation procedure is more comprehensive than adopted in earlier PSAs, in which STCs were synonymous with the RCs and the large number of potential sequences were grouped into a relatively small number of RCs. This allowed a number of benefits, including for a wider range of phenomena and other issues important to the fission product behaviour to be represented [4]. The RCs were defined so as to enable the Level 2/3 interface to support the assessment of UCR against the target release frequency of 10^{-6} per year by the summation of all relevant RCs. During the licensing process, the regulator revised their Safety Assessment Principles to require assessment of UCR into two separate dose bands: 100-1000 mSv ('Small UCRs') and >1000 mSv ('Large UCRs'). Further, rather than simply being assessed as the dose at the site fence, they were to be assessed at either a distance of 1km or the nearest habitation, whichever was closer.

A summary of the Containment end states modes from the early Level 2 PSA is shown in Figure 1 [5]. Based on the Level 2 analysis of Core Damage PDSs, it was generally concluded that [6]:

- In comparison with other Level 2 PSA results for Large Dry Containments at the time, higher probabilities were seen for basemat failure and late Containment failure. This was attributed to the inclusion of shutdown states.
- For sequences at power (and shutdown without reduced water inventory) Containment failure is unlikely provided heat removal is available. The diversity and redundancy in Containment heat removal provisions resulted in a low frequency of PDSs involving no Containment safeguards.
- Without heat removal then late Containment failure is likely to occur. Given the size and strength of Containment, these failures were likely to be after about 2 or more days.
- For shutdown sequences involving reduced water inventories basemat melt through was more likely.

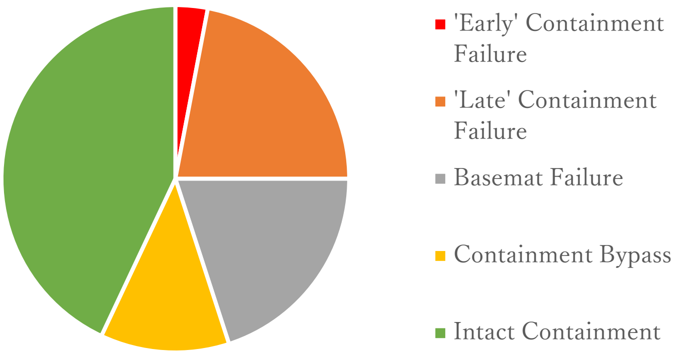

Figure 1: POSR PSA Summary of Containment End States Modes

As noted above, the underlying requirement of the POSR PSA as a whole was to demonstrate that the risk of death to any individual off-site was less than 10^{-6} per year. As such, the Level 2 analysis was developed so as to provide a robust and *conservative* analysis of the response to the Containment systems to show compliance with this target. [5]. The POSR PSA calculated that the total frequency of UCR exceeded the target, and thus further work was required to demonstrate that risk had been reduced ALARP. For Sizewell B, sensitivity studies were carried out to determine the effectiveness of water addition to the Containment by either recovering the Containment sprays or by the use of the Containment Fire Spray System (CFSS). A further 22-node CET to assess the benefit of such 'late spray recovery', as well as 'late safety injection', was developed. The benefit of Containment Water Injection (CWI) through late spray recovery can be illustrated by the representative 'TMI' PDS sequence which represents an intact circuit severe accident scenario, modelled as a complete station blackout and failure of safeguards except for the successful isolation of the Containment (i.e. Containment sprays and fan coolers are unavailable). With no water addition, 79% of the fault sequences led to Late Containment Failure and 21% led to Enhanced Leakage due to tearing of the liner. If it is assumed that the probability of being able to deliver CWI within 24 hours is 0.9, these percentages would be reduced to 8% and 3% respectively, so that in 89% of the sequences the Containment would be intact. [3]

Given their assessed benefit, it was therefore decided that the use of the CFSS as an additional Severe Accident Management measure should be adopted, and such additional actions were incorporated into the already comprehensive set of Station Operating Instructions (SOIs) [6]. With the benefit afforded by the proceduralisation of the CFSS, it was demonstrated that the frequency of Small UCR (100-1000 mSv) was less than the target frequency. Although also reduced, the Large UCR risk slightly exceeded the target frequency and thus it was necessary for the original safety case to be able to demonstrate that all reasonably practicable measures to reduce the Large UCR risk had been implemented.

Thus the PSA was also used to consider the reduction in risk associated with the provision of a Filtered Containment Venting System (FCVS). An FCVS could benefit those sequences resulting in Late Containment Over-pressure failure. An assessment was conducted which showed that for a proposed FCVS there was no reduction in the Individual Risk and the Societal Risk (defined as the frequency of more than100 fatal cancers) was reduced by 30%, to 4.9×10^{-6} per year. Since, the CFSS however could prevent Late Containment Over-pressure failure the sequences that could benefit from FCVS would be resolved by the proceduralisation of CFSS (assuming CWI could be maintained indefinitely until heat sink recovery occurred). In addition to this, the CFSS could benefit sequences that would otherwise result in the basemat melt-through. An assessment demonstrated that with proceduralisation of the CFSS, the Individual Risk was reduced by 11% from 1.9×10^{-7} to 1.7×10^{-7} per year and the Societal Risk (of 100 fatal cancers) by 49%, from 7.0×10^{-6} to 3.6×10^{-6} per year. With this use of the CFSS included in the SOIs, the potential benefit afforded by also installing an FCVS was therefore viewed to be small and installation of an FCVS was rejected on cost-benefit grounds at the start of station life [3].

4. REVIEW OF THE LEVEL 2 PSA

Since its development for the POSR, the Level 2 PSA and supporting analysis has been subject to a number of reviews. The general approach remains broadly consistent with state-of-the-art Level 2 PSAs and provides a logically convenient means of identifying when a significant radiological release to the environment would begin. Nevertheless, it is recognised that the Level 2 PSA was developed at a time when computational resources were much more restrictive than in more recent years. The methodology of applying probabilistic risk analysis to degraded core accidents was at an early stage of development with no internationally accepted guidelines on the application of the techniques. For these reasons, the simplifying assumptions were often made. For example, combining information about the state of the plant at the time of core damage into bounding scenarios to simplify the scope of analysis. While appropriate for conservatively bounding evaluations of offsite consequences, these simplifications may omit information which could be important for more detailed Level 2 PSA analysis and restrict the applicability or insights required by more detailed Level 2 analysis interested in characterising/measuring small changes in risk.

Further is the recognition of uncertainties associated with the data and in the predictions available at the time the Level 2 PSA was developed. Several aspects of the technical basis used to quantify the CET have advanced considerably since completing the SZB analysis in the late 1980s. Developments in experimental programmes and published analysis could now be utilised to refine the quantification of the CETs, not least supported by the use of up-to-date state of the art integral SAA codes such as MAAP. A number of comprehensive assessments of the impact such analysis and advancements in the state of knowledge would have on the Level 2 PSA has been undertaken. However, the conclusions from such assessments have typically focused on the potential to alter the claim that the PSA represents a bounding assessment of accident consequences.

On the whole, such reviews have concluded that the simplifications and assumptions remain appropriate for a conservatively bounding evaluation of potential offsite consequences, i.e. for continued demonstration of compliance with the original underlying requirement to demonstrate the risk of death to any individual off-site. More recently however it has been recognised that while the current use of the Level 2 PSA remains unchanged, standards and expectations have continued to develop as well as relevant modifications to the plant having occurred, particularly post-Fukushima.

For example, as a response to the learning from the events following the earthquake and tsunami at the Fukushima Dai-ichi nuclear power plant, EDF Energy established a response programme to review the resilience of its power stations. A number of modifications, such as temporary connection points to enable use of mobile Deployable Back-Up Equipment (DBUE) were implemented [7]. These modifications will have reduced the real risk at Sizewell B, although their benefit cannot be accounted for in the assessed risk without an update to the Level 2 PSA. Of particular relevance to the Level 2 PSA has been:

- Installation of Passive Autocatalytic Recombiners.

- The ability to connect a DBUE pump to inject borated water from the Refuelling Water Storage Tank into the Reactor Coolant System (RCS) while at shutdown. Although primarily aimed at preventing core damage, this could be utilised to inject water into Containment following core melt.
- The ability to connect a DBUE pump to enhance the existing provision of CWI as envisaged in the original safety case to be provided by the Containment sprays or CFSS.

The duration of events at Fukushima Dai-ichi also highlighted the need to consider aspects of the Level 2 PSA such as its temporal scope, and the rigour associated with certain claims. As noted above, the potential benefit of recovery of containment cooling during the late phase of the accident conditions was originally considered a reasonably practicable means of reducing the risk from late Containment over-pressure and basemat failure sequences. It was at the time judged that a relatively high level of confidence could be afforded to recovery of necessary equipment in the time available. Given the extent of damage evident at Fukushima Dai-ichi, a modern reassessment may necessarily be expected to reduce the original strength of any such recovery claim given the heightened burden of proof which may now be expected.

Nevertheless, it remains pessimistic in that existing base case Level 2 PSA does not model equipment recovery when naturally a large part of Severe Accident Management involves the need to prioritise and recover equipment which has previously failed. The provision of Resilience Enhancements post-Fukushima has resulted in a significant new inventory of DBUE which affords the potential provision of certain functions previously assumed to be unavailable in the base case analysis, for example CWI. Crediting the potential benefits afforded by recovery of such functions (through either the availability of DBUE or recovery of previously failed equipment) would remove a significant pessimism.

Further to plant modifications and updates to anticipated operations, the continued development of phenomenological understanding acts to further increase the gap with the original Level 2 PSA. For example: developments in phenomenological understanding now incorporated in updated MAAP models, such as lower plenum modelling not incorporated in MAAP3B, would be expected to lead to an extension in the time available between core degradation and RPV failure hence impacting ‘Time Frame 2’ of the CET. As noted previously, the systematic methodology for construction of the CET was based upon a number of rules which meant that Operator Actions were generally not considered in the Level 2 PSA. No credit was taken for the large RCS vent capacity provided by virtue of its three Pilot Operated Safety Relief Valves even though procedures existed in order to ensure that the RCS is depressurised prior to vessel failure [6]. Updated SAA indicating an increase in the in-vessel accident phase may allow for an increase in the reliability with which certain actions can be undertaken, and thus the existing ‘rules’ may now allow for the benefit to be claimed.

A further driver is the ambition to extend the operational life of Sizewell B power station by at least 20 years, from 2035 to 2055. With the potential extended operational life, it has been recognised that were the station to pursue Risk Informed initiatives, an updated Level 2 PSA could potentially provide additional benefit. In the context of extended operation, the installation of a FCVS as an additional measure of defence-in-depth is being considered, even though recent reviews have continued to uphold the POSR conclusion that the residual risk mitigated by an FCVS is very small and does not represent a shortfall in the safety case [8].

With the continued reliance on the original Level 2 PSA and SAA developed for POSR, the conditional risks from the Level 2 PSA remain unchanged. The potential need has therefore been recognised to develop an updated Level 2 PSA to provide a better estimate assessment in order to support the ambitions associated with extended operational life of Sizewell B.

5. PROPOSED MODIFICATIONS TO THE LEVEL 2 PSA

For the reasons discussed in the previous section, a number of modifications to the Sizewell B Level 2 PSA tools and methods are now being developed. The intent is to deliver a significant reassessment through an updated Level 2 PSA which reflects modern phenomenological understanding, reflects the relevant modifications to Sizewell B since the time of the POSR, and aligns with updated practice in the field. Prior to undertaking a holistic update of the Level 2 PSA, an approach was developed and implemented for a single PDS, as described below. The scenario selected was the ‘TMI’ PDS: an intact circuit severe accident scenario, modelled as a complete station blackout and failure of safeguards except for the successful isolation of the Containment.

The existing CET is developed over four Time Frames. Time Frame 4 was considered to cover all expected long-term Containment response options arising from use of installed mitigation systems, but without making specific claims on recovery of systems. It is proposed to include a new Time Frame 5 to differentiate between the existing long-term Containment response in Time Frame 4 and the possibility of maintaining Containment integrity 'indefinitely'. It is added at the end of the existing CET to represent the longer term accident management/progression, nominally 3 days to 7 days after the start of the accident. This affords a significant extension of the temporal scope of the Level 2 PSA and allows for specifically modelling the recovery of existing systems or use of the new mitigation systems. The extension to Time Frame 5 also requires an additional number of STC end points to be defined depending on the status of Containment.

A review of the new mitigation systems with the potential to influence the Containment response and source term analysis in the updated Level 2 PSA was undertaken. For example, the new Resilience Enhancement which affords the provision CWI by DBUE is considered. While the preferred option is to provide the function of CWI via the Containment Sprays, if unavailable the CFSS and DBUE are also potentially available. As noted previously, CWI has two roles in the safety case:

- To forestall the rise in Containment pressure
- To minimise the risk associated with basemat melt-through.

Given the redundancy and diversity in the plant design, for the majority of Severe Accident scenarios in which Containment safeguards have failed, CWI is highly unlikely to be recovered 'early' in the scenario. For a core melt sequence to occur in the first place, widespread plant failures would likely be required and thus the Containment Sprays are considered unlikely to be available without recovery actions. Similarly, operation of the CFSS is not an immediate action and DBUE would take some time to configure. It is therefore assumed that CWI would not be afforded in the CETs 'early' Time Frames prior to RPV failure.

Time Frame 3 considers events from just before RPV failure to an hour after RPV failure. It is considered however that any limited CWI at this point would have limited impact on the outcome of Time Frame 3. Therefore the availability of CWI was considered for Time Frame 4 and beyond. In order to quantify the new nodes to represent CWI, fault trees were used, allowing for differing quantification depending on the Time Frame. It models, based on expert judgement utilising the Degree of Belief (DoB) scheme:

- The likelihood that the water inventory in Containment from the previous time frame is inadequate to provide a sufficient depth of water in this timeframe (a depth corresponding to saturation associated with the source term benefit associated with the pool scrubbing effect is proposed). It is more likely to be true if CWI failed in the previous timeframe.
- The timely recognition by the operator to provide CWI such that it would affect the events in the Time Frame. Again, this is more likely to be true if CWI failed in the previous time frame.
- The systems available to provide initial CWI, modelled through a linked tree based on equipment reliabilities. For example, in Time Frame 4A the availability of DBUE to provide CWI shall not be considered as it is judged unlikely to be available on the required timescales.
- The systems available to provide extended (long term) CWI in the timeframe.

For the CET to be used to resolve the possibility of maintaining Containment integrity 'indefinitely', consideration of CWI alone is not sufficient. CWI will forestall a rise in Containment pressure but it is necessary for a Containment heat sink to be operating to remove the energy. Thus the possibility of Heat Sink Recovery (HSR) to enable Containment pressure to be reduced and avoid late Containment Failure shall be modelled in the updated CET. As discussed above, HSR could be achieved by restoring either the Containment Fan Coolers or Containment Sprays (in re-circulation mode).

Time Frame 3 was originally defined so as to represent the high energy events which may happen as a direct result of RPV failure. Therefore, although HSR may occur in this time frame, it is unlikely to impact on the outcome of TF 3. Once again then, the availability of HSR is proposed to be considered for Time Frame 4 and beyond. When HSR is successful, it is assumed that the Containment reaches a quasi-stable condition and no further events are considered in the CET.

The inputs to the new nodes concerning HSR could be modelled using system fault tree approach, similar to that described for CWI. For the initial assessment, they are to be modelled as Basic Events utilising expert judgement. The probabilities assigned to these nodes are not dependent on the path or sequence before the node, i.e. it is assumed that the recovery of a heat sink in any Time Frame does not depend on any previous event modelled in the CET.

The updates to the structure of the CET, as discussed above, result in a significant expansion of the CET. The original CET was created as a set of fifteen linked tree structures (subtrees) to account for quantification differences associated with different pathways due to preceding nodal dependencies. With the addition of nodes to represent new mitigation systems and the new Time Frame 5, a significant number of additional pathways through the model now exist. Consequently, the RiskSpectrum model for Updated 'TMI' CET consists of 93 linked event trees.

The quantification of the Updated 'TMI' CET was supported by an updated suite of SAA based on an updated MAAP5 model, thus utilising updated phenomenological understanding. This enabled the impact of other plant modifications not explicitly modelled in the CET, for example the impact of Passive Autocatalytic Recombiners, to be accounted for in the Level 2 PSA by adjusting nodal probabilities. Further, analysis of key scenarios based on updated phenomenological understanding was used to inform updated source term analysis. This was necessary to ensure the pathways through the Updated CET were allocated to the correct STC. Once the representative dose (or range of doses) are calculated for each STC, the remaining attributes are assigned to allow the RC allocation. Each RC has an associated conditional Individual Risk and Societal Risk associated with it, as derived when the Level 3 analysis was originally performed. Thus the impact of the Updated 'TMI' CET on the Level 3 metrics can ultimately be assessed.

For the 'base case' POSR Level 2 PSA, >99% of the conditional probability associated with the 'TMI' PDS was originally associated with STCs involving extended CCI and allocated to Late Containment failure. With the benefit of CWI and HSR credited in the Updated 'TMI' CET, the conditional probability of scenarios associated with high Containment pressure is significantly reduced. The conditional probability of Containment remaining intact, with leakage limited to that permitted under the design basis, is ~80%. Late Containment failure scenarios (in Time Frames 4 or 5) account for <19% of the conditional probability, with all other sequences contributing <1% of the conditional probability. Scenarios involving extended CCI were reduced to <5%. Although the Updated 'TMI' CET accounted for updated phenomenological understanding, the changes are predominantly associated with the introduction of consideration of CWI and HSR.

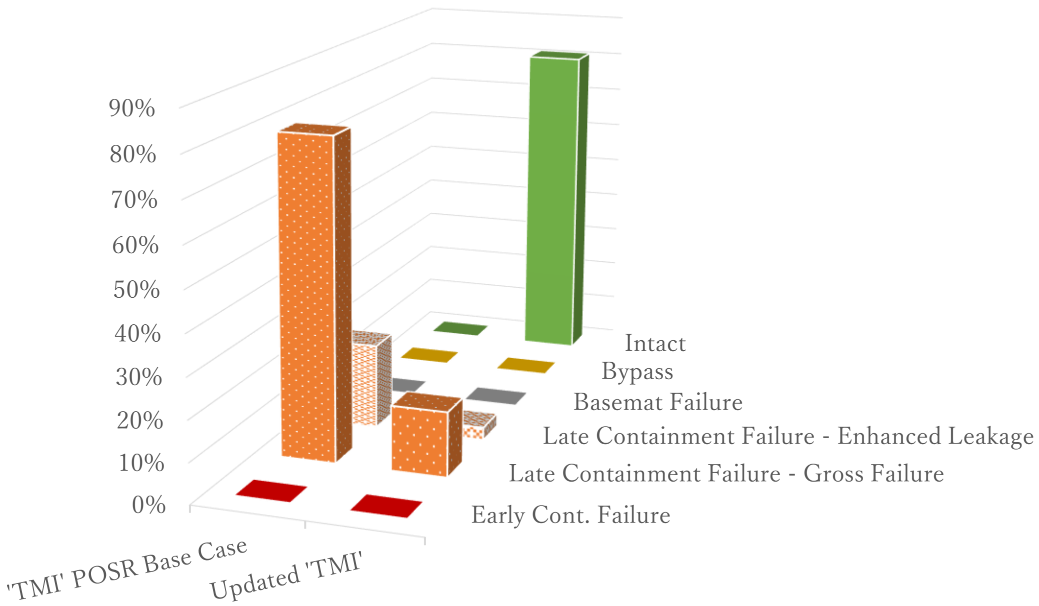

Figure 2: The Impact of the Updated 'TMI' CET on Containment End States

For the POSR Level 2 PSA, the most probable STCs assigned for the 'TMI' PDS were allocated to sequences associated with a 'Large' UCR, with doses in excess of 10^3mSv at 3km [4]. In the Updated CET, with the avoidance of prolonged CCI due to the action of CWI in >95% of the sequences, and given that now only ~20% of sequences result in Containment failure due to the potential for HSR, scenarios are typically allocated to much less onerous STCs. For the POSR base case analysis, the entire conditional probability associated with the 'TMI' PDS was allocated to Large UCR, while the Updated 'TMI' CET suggests this should be reduced to ~23% with the remainder allocated to Small UCR. This results in a predicted reduction in Individual Risk and Societal Risk (of 100 fatal cancers) to <30% and <20% of their original values. The POSR Level 3 analysis originally considered a wide number of societal risk metrics, and for example the Updated 'TMI' PDS would suggest a reduction to ~6% of the original accident cost (the cost being a valuation of the societal detriment caused by the accident, considering three broad categories (health effects, social disruption (e.g. relocation) and intervention (e.g. food bans)) assigned a monetary value using a cost framework, based on National Radiological Protection Board methodology).

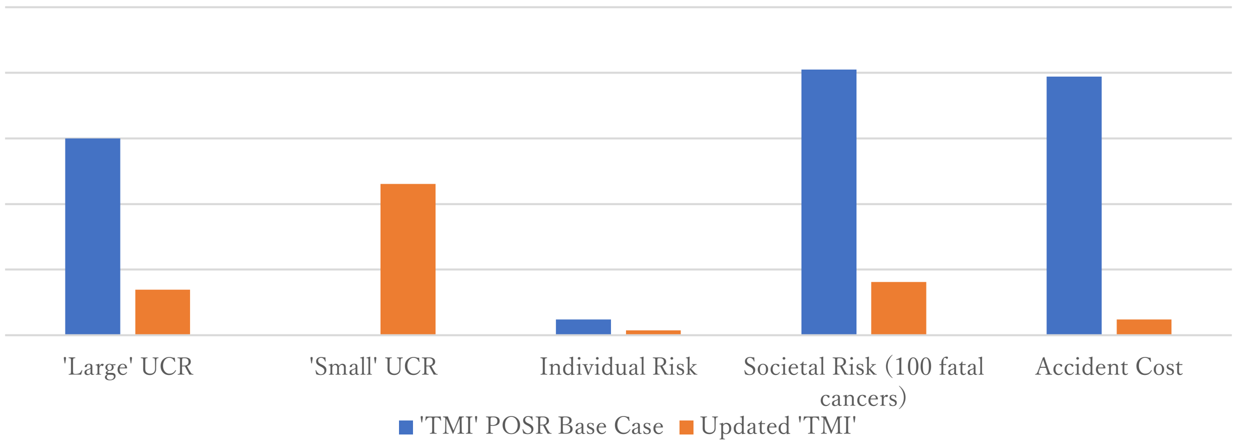

Figure 3: The Impact of the Updated 'TMI' CET on Risk Metrics

A review of the dominant minimal cut sets contributing to the most risk significant STCs was undertaken to assess the significance of assumptions and assignments within the Updated 'TMI' CET. As may be anticipated, this highlighted that significance of HSR in Time Frame 4B in determining an 'intact' or 'failed' (enhanced leakage / gross failure) end state for the Containment. The probabilities of HSR were estimated utilising the DoB scheme, based on expert judgement and a variety of data source (e.g., Level 1 PSA sequence contributors to the TMI PDS, plant design, etc.).

- A small increased DoB for the likelihood of HSR would result in a significant proportion of probability shifting from STCs associated with Late Containment Failure, with a corresponding modest increase in 'Intact' endpoints. (This reflects that, in the updated results, a relatively smaller proportion of sequences result in a Late Containment Failure end state compared to an Intact Containment end state.) Similarly, the shift results in a significant reduction in 'Large' UCR (and a modest increase in 'Small' UCR) as the Late Containment Failure end states are moved to 'Intact' Containment end states. This also acts to drive a large decrease in Societal Risk (of 100 fatal cancers), and a reasonable decrease in Individual Risk.
- Alternatively, a small decrease in the DoB associated for the likelihood of HSR would reallocate sequences in the opposite direction, resulting in a reasonable increase in Individual Risk and a large increase in Societal Risk (of 100 fatal cancers).

The changes resulting from the Updated 'TMI' PDS however are not necessarily likely to be reflected across all other PDSs, and thus are not representative of the cumulative impact upon the overall Risk Metrics. While some changes are specific to the 'TMI' PDS, a number would be relevant to other PDSs. The Core Damage PDSs represent the dominant contributors to both Individual Risk and Societal Risk (of 100 fatal cancers). Considering Individual Risk, the contribution from Core Damage PDSs is dominated by the PDS representative of scenarios in which Containment Isolation does not occur. Consequently, it is not assessed using a CET and its impact in terms of risk contribution is unlikely to change as a result of the changes previously discussed. Therefore overall, while Individual Risk may reduce, the dominant contributor would be largely unaffected. Containment bypass sequences are also significant contributors to Individual Risk and again are unlikely to be significantly affected. Considering Societal Risk (of 100 fatal cancers), 'TMI' is the principle contributor from the Core Damage PDSs, followed by the 'SMI' PDS (a small loss of coolant accident scenario). Similarly to the original assessment for the 'TMI' PDS, the dominant STCs for the 'SMI' PDS are associated with prolonged CCI and Late Containment Failure. Thus, it would be anticipated that if updated there would be a similar impact upon the reallocation of STCs and thus upon the risk metrics. Thus, updates may be expected to result in a reasonable reduction for the total Societal Risk (of 100 fatal cancers).

6. CONCLUSION

The POSR Level 2 analysis was developed so as to provide a robust and *conservative* analysis of the response to the Containment systems, to show compliance with the necessary targets. In this regard, the continued use of the POSR Level 2 PSA remains acceptable. However, it has been recognised that a number of drivers now exist to update the Level 2 PSA including to represent modifications to the plant, the desire to extend the lifetime of Sizewell B, pursue the installation of an FCVS, and the potential to better support future risk informed initiatives.

An initial demonstration of the impact of some of these changes has been performed for the 'TMI' PDS. While the assessment demonstrates a reduction in Large UCR, Individual Risk and Societal Risk (of 100 fatal cancers) metrics for the Updated 'TMI' CET, this is only one of the PDSs which inform the Risk Metrics. The impact on other PDSs is not necessarily to the same extent and will required individual consideration. Thus it can be seen that to adopt the Updated CET across the remaining 12 Core Damage PDSs would collectively require a significant level of effort in terms of the underwriting analysis and assessment to support the quantification.

With the modifications to the CET structures for the new mitigation system nodes, the Updated CET is very large, containing a large number of nodes and dependencies, hence, a large number of sub-trees. The original Sizewell B 'TMI' CET consisted of nine linked event trees. With the addition of nodes to represent new mitigation systems and Time Frame 5, a significant number of additional pathways through the model now exist. Consequently, the RiskSpectrum model for Updated 'TMI' CET now consists of 93 linked event trees.

While maintaining a consistent CET structure and level of detail is beneficial to both the understanding and accuracy of the model, with the extension of the CET, in some instances differences applied to different branches or subtrees can become difficult to justify (in probability or assigned consequences). In particular this is true where modern SAA would no longer predict as strongly the occurrence, and/or correlation, of certain phenomena. Further, quantifying branches within a CET can be difficult when sequences are shown to be extremely unlikely, or even unreproducible, in the SAA. Thus, further work is to be undertaken to review the potential for simplifications to the Updated CET with the aim to benefit the readability and, in some cases, enable easier quantification (for example by reviewing the validity of dependencies on earlier nodal outcomes). Additionally, although 'placeholder' nodes were included within the Updated CET for inclusion of an FCVS, these pathways must also be developed along with determination of new radiological assessments to characterise a filtered release.

The work reported herein is therefore the first step in developing an Updated Level 2 PSA for Sizewell B. Future work is required to develop and subsequently quantify the Updated CETs and the probabilistic assignments for each Core Damage PDS to account for current phenomenological understanding and plant design/operation. The updates to the CETs, and the future installation of an FCVS will primarily impact Core Damage PDSs. Nevertheless, to ensure a balanced view of risk the other PDSs will also require consideration (for example to claim the benefit of updated phenomenological understanding) as part of a holistic update to the Level 2 PSA.

Acknowledgements

The work reported in this paper required significant support from EDF Energy and contract partner colleagues, including colleagues at Sizewell B and within the Nuclear Operations Design Authority.

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