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REVISION HISTORY

Issue	Description	Date
00	First issue for INSA review	11/02/08
01	Integration of co-applicant and INSA review comments	29/04/08
02	Integration of level 2 PSA for low power and shutdown states and incorporation of other comments from co-applicant and INSA	30-06-08
03	PCSR June 2009 update including PSA model modifications. Update of Level 2 PSA due to Level 1 PSA changes: - Change in description of plant operating states. - Change in reliability data used in the PSA. - Integration of preventative maintenance in the base line PSA model. - Update the initiating events list: o 2A-LOCA in the base line PSA model, o Medium LOCA frequency updated. Update of Level 2 PSA model: - CDES (Core Damage End States) for interfacing system LOCA during shutdown states (Ca, Cb, D and E) were combined: new CDES called IS SD. - Modelling of hatch closure in all none LOCA sequences in states Ca and Cb. - Update of containment isolation model.	30-06-09
04	Removal of RESTRICTED marking and addition of CCI marking	18-06-2010

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REVISION HISTORY (Cont'd)

Issue	Description	Date
05	Consolidated Step 4 PCSR update: - Minor editorial changes - Clarification of text - Update and addition of references - Inclusion of information on soda injection in severe accident conditions, and associated PSA modelling (§3.5.3) - Update of Level 2 PSA consistent with Level 1 PSA changes. - Update of Level 2 PSA model: o Update of the containment isolation modelling	31-03-11
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 - Update of references - Minor updates for clarification / consistency (§3.5.3, §4.4.2 and §4.4.5) - Sub-section 15.4.3.5 – Table 3 minor typographical correction - Sub-section 15.4.4.3 – Tables 1, 4 and 5 and related Sub-section 15.4.4.4 - Figures 1, 3 to 8 updated (input data and typographical correction (e.g. RC SFP / FREL (2a))	21-11-2012

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SUB-CHAPTER 15.4 – LEVEL 2 PSA

1. LEVEL 2 PSA - INTRODUCTION

The objective of the Level 2 PSA is to assess the response of the containment and its related systems to potential loads and to assess the characteristics of radiological releases from severe core damage accidents. The Level 2 PSA calculates the probability, composition, magnitude, and timing of fission product releases from the plant. It has been performed using a combination of deterministic and probabilistic analyses consisting of the following:

- Integration of the Level 1 and Level 2 PSA analyses through the definition of Core Damage End States (CDES). The CDES from Level 1 PSA provide the “initiating events” for the Level 2 PSA analysis.
- Identification of physical phenomena important to containment integrity that could occur during the course of a severe accident.
- Accident progression analysis and detailed phenomenological evaluation to support development of the Containment Event Trees (CETs) and determination of branch probabilities.
- Level 2 PSA systems analysis.
- Identification of the human action for severe accident mitigation and the linked human error probabilities.
- Development of Release Category (RC) bins to characterise fission product releases to the environment.
- Determination of the source terms for key nuclides in each RC.
- Uncertainty and sensitivity evaluations.

The study is based on the US EPR Level 2 PSA submitted to the US NRC in support of EPR Design Certification. However, significant updates were made to the study for the UK application, including:

- Changes necessary to address design differences between the US and UK EPRs, including new systems analyses,
- Changes required to integrate the Level 2 PSA model into a single RiskSpectrum® model with the UK EPR Level 1 PSA,
- Improvements to the source term analyses,
- Addition of analysis covering shutdown plant states.

2. LEVEL 2 PSA - SCOPE

The Level 2 PSA study considers all of the severe accidents identified within the Level 1 PSA and for the following conditions/states:

- Accidents occurring from “at-power” (plant states A and B) initial conditions,
- Accidents occurring from shutdown plant states C, D and E,
- Internal events and hazards, such as internal fire and flood,
- External hazards as treated in Level 1 PSA (Sub-chapter 15.2), and
- Accidents occurring in the Spent Fuel Pool.

3. LEVEL 2 PSA - METHODOLOGY

3.1. INTRODUCTION AND MAJOR TASKS

This section summarises the main elements of the Level 2 PSA, referencing the sub-sections of this sub-chapter in which further details can be found.

Interface with Level 1 PSA

A set of Core Damage End States (CDES) is defined which enable the core damage events identified in the Level 1 PSA study to be grouped in a way that facilitates the accurate treatment of severe accident phenomena in the Level 2 PSA study. The CDES are equivalent but defined differently from “traditional” Plant Damage States (PDSs) due to the integrated modelling of Level 1 and Level 2 PSAs in Risk Spectrum, which obviates the need to include system information in the transition CDES. The CDES definition is described in section 3.2.

Phenomenology

Detailed phenomenological evaluations are performed to identify and analyse the phenomena important in determining the split fractions in the containment event trees.

Seven families of phenomena were considered:

- Induced RCP [RCS] ruptures including hot leg and SG tube ruptures,
- Fuel-Coolant Interactions (FCI), in-vessel and ex-vessel,
- Hydrogen generation, distribution and combustion, including flame acceleration and Deflagration to Detonation Transition (DDT),
- In-vessel recovery (i.e. evaluation of potential in-vessel corium quench before vessel failure)

- Direct Containment Heating (DCH), and vessel rocketing,
- Long-term containment challenges, including Molten Core-Concrete Interaction (MCCI) and long-term overpressurisation of the containment.

In addition, further supporting evaluations and analyses were performed, including:

- Containment fragility evaluation:

The structural capability of the containment to withstand the pressure loads was discussed and the probabilistic fragility curves for the containment were developed [Ref-1].

- Human Reliability Analysis (HRA):

Human actions are identified based on the AREVA-NP OSSA severe accident management guidance package [Ref-2]. Modelling of these actions and the evaluation of human reliability is performed [Ref-3] using the US SPAR-H method [Ref-4].

- Equipment and system survivability evaluation:

Equipment and systems credited within the Level 2 PSA study are identified and studied to evaluate their survivability during a severe accident and within a harsh environment.

- Supporting severe accident analysis:

Supporting severe accident analysis is performed in order to support the phenomenological evaluations, and the CET quantification. The analysis is performed with the MAAP4.0.7 code [Ref-5], with EPR-specific models including a 27-volume containment model [Ref-6].

- Evaluation of phenomenology for shutdown states:

Phenomenological evaluations developed for at-power plant states A and B are reviewed and modified where necessary to provide the required split fraction information for the shutdown state CETs.

The phenomenological evaluation and the additional supporting analyses are described in section 3.3.

Systems Analysis, Accident Sequence Analysis and Containment Event Trees

Systems considered in Level 2 PSA, in addition to those covered in Level 1 PSA, include:

- Containment isolation,
- Primary system depressurisation (extension to cover severe accident depressurisation),
- PARs / hydrogen control,
- Corium stabilisation system (“core catcher”),
- Severe accident spray system EVU [CHRS], including all functions (i.e. passive debris flooding, containment spray and active recirculation cooling).

Reliability analysis needed in order to quantify the CET is performed where necessary.

The CETs are developed and quantified, based on the results of the foregoing tasks. The EPR CETs use three timeframes, and contain nodes related to phenomena, systems and human actions.

Systems Analysis, Accident Sequence Analysis and Containment Event Trees are described in section 3.4.

Release Category Definition and Source Term Analysis

Fission product Release Categories (RCs) are defined. The purpose is to group severe accident sequences into categories, each of which can be represented by a single source term. A representative severe accident sequence is chosen for each RC, and a source term (the magnitude, isotopic composition and timing of the release to the environment) was determined for each one. The source term analysis was performed using the MAAP4.0.7 code. RC definition and source term analysis are covered in section 3.5.

Sensitivity and Uncertainty Analysis

The uncertainty analysis allows for the propagation of uncertainty distributions through the CET. The importance of chosen systems, actions or phenomena was investigated through the use of sensitivity analyses. Sensitivity and uncertainty analyses are described in section 3.6.

Computer Codes

The following computer codes are used in the Level 2 PSA:

- Risk Spectrum integrated Level 1-Level 2 PSA model (Level 1 PSA can be interrogated from Level 2 PSA) [Ref-7],

- MAAP 4.0.7, with an EPR-specific parameter file and EPR models (e.g. core catcher) [Ref-5],

- Crystal Ball [Ref-8], for Monte-Carlo simulations.

3.2. INTERFACE WITH LEVEL 1 PSA

In previous Probabilistic Safety Assessment (PSA) studies, Level 1 PSA results were grouped ("binned") into Plant Damage States (PDSs). These PDSs were selected according to plant characteristics that define the status of the reactor, the containment and the core cooling systems at the time of core damage. This was done in order to ensure that systems which are important to core damage in the Level 1 PSA event trees, and dependencies between containment and other systems, were handled consistently. A PDS therefore represented a grouping of sequences that were considered likely to yield similar accident progressions.

These PDSs allow for accident sequences from the Level 1 PSA analysis to be binned into a set of states that define both the physical and systems-related characteristics. However, when the PDSs were quantified, the results for each PDS had to be collapsed into a single frequency, and information relating to component failures was lost. The definition and quantification of PDSs in this way therefore typically required either the use of so-called “bridge trees” or the extension of the Level 1 PSA core damage sequences; these techniques were required because the Level 1 PSA sequences focus on core damage only and do not include all of the information that can affect the post-core damage accident progression (and hence the Level 2 PSA).

Using the RiskSpectrum PSA software, the UK EPR Level 2 PSA has been able to link the Level 1 PSA core damage model directly to the Level 2 PSA Containment Event Tree (CET). This approach offers a number of advantages in comparison to the methodology used in previous PSA studies:

- Quantification of the model from initiating event, through core damage, to release category offers the opportunity to determine systems and equipment and human action importance regarding Large Release Frequencies (LRF) and Level 2 PSA Release Categories (RCs).

- Integrated quantification of the Level 1 and Level 2 PSA models allows for simplifications to be made in the set of end states to provide the transition from the Level 1 to Level 2 PSA analyses.

- Linking the Level 1 and Level 2 PSA analyses allows the seamless transfer of dependency information. All cutset level unavailabilities are carried forward into the Level 2 PSA analysis during the integrated quantification process. A consistent treatment of operator dependencies between the Level 1 and Level 2 PSA models is therefore facilitated.

- Since the additional information on system status can be interrogated in the Level 2 PSA CET model (by the incorporation of relevant fault trees into the model) the extension of the Level 1 PSA analysis to facilitate binning into traditional PDSs can be avoided.

While this approach also offers some challenges, mainly concerning the complexity of the model, these are outweighed by the advantages obtained.

3.2.1. Core Damage End State Definition

Core Damage End States (CDES) are named as such in order to differentiate them from the Plant Damage States used in previous PSA studies. They provide the interface between the Level 1 and Level 2 PSA analyses – i.e. between core-damage accident sequences and fission product RCs.

The CDES are designed to link the Level 1 PSA core damage event trees to the Level 2 PSA containment event trees by bringing together core damage sequences with similar characteristics, and to use those sequences as the initiating events for the appropriate CET. CDES have been established for each of the Level 1 PSA event trees.

The principal reason for establishing CDES is to facilitate the accurate treatment of severe accident phenomena in the CET. Phenomenological evaluation and containment failure probabilities (split fractions) are assigned on a “per CDES” basis.

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The integrated Level 1/Level 2 PSA model makes the end state distinctions according to system unavailability unnecessary. The CDES distinguish between significant groups of core damage sequence types by including the following information from the Level 1 PSA event trees:

- Types of sequences (Transients, LOCAs, etc.),
- Condition of the containment (e.g. no bypass, SGTR, interfacing system LOCA),
- System related plant status:
    - Off-site power,
    - Feedwater,
    - Steam generator pressure and isolation.

When the status of Containment Isolation, Safety Injection, Containment Heat Removal, and other systems is required by the Level 2 PSA analysis, this status can be found by interrogating system related top events in the Containment Event Trees.

3.2.1.1. Core Damage End State Definitions for At-Power Plant States

The Level 1 PSA study, Sub-section 15.4.3.2 - Table 1 shows the CDES into which the Level 1 PSA sequences are divided, which CDES link tree and CET they are sent to, and how they are treated in the CDES link trees and CETs. The description of CET treatment is simplified, the aim of the Sub-section 15.4.3.2 - Table 1 being rather to show the impact of each CDES on the Level 2 PSA analysis. A full description of the CETs is presented in section 3.4. In addition, Sub-section 15.4.3.2 - Table 2 shows the main assumptions used in the assignment of CDES.

It is important to note that a combination of systems that does not meet the Level 1 PSA success criteria can guarantee to a limited extent the core damage in the Level 2 PSA. The Containment Event Tree “Limited Core Damage” is used in such cases. Another significant point is that some CDES (SS, TR, SSD) are split into several CDES (SS SB1, SS CCALL…etc). This splitting allows for the correct boundary conditions to be passed onto the CET. In the same way, duplicated CETs are used (see section 3.4).

3.2.1.2. Core Damage End State Definitions for Shutdown Plant States

For the Level 1 PSA study, Sub-section 15.4.3.2 - Table 3 shows the CDES into which the Level 1 PSA sequences from shutdown states are divided, which CDES link tree and CET they are sent to, and how they are treated in the CDES link trees and CETs. The description of CET treatment is simplified, the aim of the Sub-section 15.4.3.2 - Table 3 being rather to show the impact of each CDES on the Level 2 PSA analysis. A full description of CET is presented in section 3.4. In addition, Sub-section 15.4.3.2 - Table 4 shows the main assumptions used in the CDES assignment for shutdown states.

The same CDES types used for at-power states (SS, SL, TP, TR, IS…) are used for shutdown states. The shutdown state CDES codes are the same as for the at-power CDES codes, but completed with the shutdown state (Ca, Cb, D or E) in brackets. For example. TR (CA) is the code for core damage from transient sequences or not isolated homogeneous boron dilution sequences in state Ca.

The distinction between at-power and shutdown CDES allows, among other things, a stand-alone quantification of the Level 2 PSA results for shutdown, or even a stand-alone quantification of the individual shutdown states (Ca, Cb, D, and E).

As for the at-power states, some CDES are split into several CDES (TR-CCALL, TR-LOOPS etc) to allow for the correct boundary conditions to be passed onto the CET. In the same way, duplicated CETs are used (see section 3.4).

In shutdown states, the containment and RCP [RCS] status, which are important accident sequence attributes, depend on the plant operating state. The assignment of the containment and RCP [RCS] status to each sequence is consistent with Section 15.1.3 – Table 1:

- Shutdown state C:

Shutdown state C consists of five sub-states – Ca1, Ca2, Ca3, Ca4 and Cb. The core damage end state definitions distinguish sub-states Ca1 to Ca4 (coded as CA in the CDES definitions) from sub-state Cb (coded as CB in the CDES definitions) as follows:

  - o “CA” consists of sub-states Ca1 to Ca4: RCP [RCS] closed, equipment hatch status depends upon the state,
  - o “CB” consists of sub-state Cb: RCP [RCS] partially open, equipment hatch open for part of the time.

- Shutdown states D and E:

In shutdown states D and E, the vessel head is removed, which means that the RCP [RCS] is open. As a consequence, a flag is inserted at link tree level to pass the correct boundary condition “RCS open”. This applies to all accident sequences except IS LOCA sequences for which this boundary condition is not needed. In states D and E respectively the equipment hatch is considered as closed and open.

The different related states (i.e. corresponding to the shutdown states C to E) were then used in the CET code as shown in Sub-section 15.4.3.2 – Table 3.

SUB-SECTION 15.4.3.2 - TABLE 1

Core Damage End States and their treatment in CDES link trees and CETs – At-Power Plant States

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
AT	Any sequence where reactivity is not controlled by safeguards systems and which is not assigned to ATI	#CDES-AT	- Label	CET1 HI Pressure	CET1 HI PRESSURE considers specific early time frame phenomena for high pressure sequences, including manual or RCP [RCS] rupture depressurisation mechanisms. If the sequence is depressurised, it is sent to the CET LO PRESSURE tree. If it is not, it is sent to the CET2 HI PRESSURE tree.
ATI	Any sequence with failure of reactivity control and where the failure to control reactivity may lead directly to challenge containment integrity which is assumed to fail containment.	#CDES-ATI	- Label	CET CF	CET CF takes the sequence directly to containment failure before vessel breach.
IS	Core Damage from Interfacing System LOCA sequences	#CDES-IS	- Label	CET ISL	CET ISL determines whether or not there is water available to cover the break outside containment and scrub the fission products released from the leak.
LL	Core damage from Large LOCA sequences	#CDES-LL	- Label - if LHSI is not successful it is sent to CET-LO pressure tree - if LHSI is successful, it is sent to limited core damage	CET-LO Pressure CET LIMITED CD	CET LO Pressure analyses the core melt and containment failure progression when the primary system is depressurised. CET LIMITED CD analyses the limited core damage sequences

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
			tree		
ML	Core damage from Medium LOCA sequences	#CDES-ML	Similar to LL	CET-LO Pressure CET LIMITED CD	Similar to LL
RV	Core damage from reactor pressure vessel brittle fracture sequence or from reactor vessel rupture following failure to control pressure during ATWS.	#CDES-RV	- Label	CET LO Pressure	CET LO Pressure analyses the core melt and containment failure progression when the primary system is depressurised. In-vessel core recovery is not relevant for this CDES.
SG	Steam Generator Tube Rupture sequences with ASG [EFWS] isolated.	#CDES-SG	- Label	CET SGTR	CET-SGTR treats the sequence as an unscrubbed release
SG2	Steam Generator Tube Rupture sequences with ASG [EFWS] not isolated.	#CDES-SG2	- Label, - if ASG [EFWS] is not available, it is sent to CET SGTR. - If ASG [EFWS] is available to the ruptured SG, it is sent to CET SGTR FW.	CET SGTR CET SGTR FW	CET-SGTR treats the sequence as an unscrubbed release CET-SGTR FW treats the sequence as a scrubbed release

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SL	Core damage from small LOCA sequences where fast cooldown is not demanded or where operator fails to initiate it.	#CDES-SL	- Label	CET1 HI Pressure.	See AT for CET1 HI Pressure description.
SLD	Core damage from small LOCA sequences where fast cooldown has been successfully initiated by the operator.	#CDES-SLD	- Label	CET1 HI Pressure.	See AT for CET1 HI Pressure description. The depressurisation of at least one SG makes the conditional probability of induced SGTR/induced hot leg rupture higher.
SP	Core damage from sequences initiated by a long loss of offsite power (consequential LOOP included) with Seal LOCA and where fast cooldown is not demanded or operator fails to initiate it.	#CDES-SP	- Label	CET1 TP HI Pressure	Same treatment as in CET1 HI Pressure. CET1 TP HI Pressure is dedicated to sequences with offsite power not available. Transfer CETs following this CET consider recovery of offsite power in the late timeframe. Seal LOCA and Small LOCA are distinguished to best consider the induced SGTR phenomenon.
SPD	Core damage from sequences initiated by a long loss of offsite power (consequential LOOP included) with Seal LOCA where fast cooldown has been successfully initiated by the operator	#CDES-SPD	- Label	CET1 TP HI Pressure	Similar to SP. Note: CET1 interrogates the sequence to determine the method of depressurisation The depressurisation of at least one SG makes the conditional probability of induced SGTR/induced hot leg rupture higher.
SS	Core damage from seal LOCA sequences with	#CDES-SS	- Label	CET1 HI Pressure	See AT for CET1 HI Pressure description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
	offsite power available and where fast cooldown is not demanded or operator fails to initiate it.				Seal LOCA and Small LOCA are distinguished to best consider the induced SGTR phenomenon.
SS-CC1+2	"SS" sequences (see above) with RRI [CCWS] trains 1 and 2 unavailable.	#CDES-SS CC1+2	-Label	CET1 CC1+2 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC1+2 HI Pressure is dedicated to sequences with RRI [CCWS] trains 1 and 2 unavailable.
SS-CC2	"SS" sequences (see above) with RRI [CCWS] train 2 unavailable.	#CDES-SS CC2	-Label	CET1 CC2 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC2 HI Pressure is dedicated to sequences with RRI [CCWS] train 2 unavailable.
SS-CC2+3	"SS" sequences (see above) with RRI [CCWS] trains 2 and 3 unavailable.	#CDES-SS CC2+3	-Label	CET1 CC2+3 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC2+3 HI Pressure is dedicated to sequences with RRI [CCWS] trains 2 and 3 unavailable.
SS-CCALL	"SS" sequences (see above) with all RRI [CCWS] trains unavailable.	#CDES-SS CCALL	-Label	CET1 CCALL HI Pressure	See AT for CET1 HI Pressure description. CET1 CCALL HI Pressure is dedicated to sequences with all RRI [CCWS] trains unavailable.
SS-LOOPS	"SS" short LOOP sequences	#CDES-SS LOOPS	-Label	CET1 LOOPS HI Pressure	See AT for CET1 HI Pressure description. CET1 LOOPS HI Pressure is dedicated to short LOOP

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
					sequences
SS-SB1	"SS" sequences (see above) after initiating event IH F/FL_SB1_AB.	#CDES-SS SB1	-Label	CET1 SB1 HI Pressure	See AT for CET1 HI Pressure description. CET1 SB1 HI Pressure is dedicated to sequences after initiating event IH F/FL_SB1_AB.
SS-SWGB	"SS" sequences (see above) after initiating event IH F SWGB_AB	#CDES-SS SWGB	-Label	CET1 SWGB HI Pressure	See AT for CET1 HI Pressure description. CET1 SB1 HI Pressure is dedicated to sequences after initiating event IH F SWGB_AB
SSD	Core damage from seal LOCA sequences where fast cooldown has been successfully initiated by the operator and offsite power is available.	#CDES-SSD	- Label	CET1 HI Pressure	Similar to SS. The depressurisation of at least one SG makes the conditional probability of induced SGTR/induced hot leg rupture higher.
SSD-CC1+2	"SSD" sequences (see above) with RRI [CCWS] trains 1 and 2 unavailable.	#CDES-SSD CC1+2	-Label	CET1 CC1+2 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC1+2 HI Pressure is dedicated to sequences with RRI [CCWS] trains 1 and 2 unavailable.
SSD-CC2	"SSD" sequences (see above) with RRI [CCWS] train 2 unavailable.	#CDES-SSD CC2	-Label	CET1 CC2 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC2 HI Pressure is dedicated to sequences with RRI [CCWS] train 2

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
					unavailable.
SSD-CC2+3	"SSD" sequences (see above) with RRI [CCWS] trains 2 and 3 unavailable.	#CDES-SSD CC2+3	-Label	CET1 CC2+3 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC2+3 HI Pressure is dedicated to sequences with RRI [CCWS] trains 2 and 3 unavailable.
SSD-CCALL	"SSD" sequences (see above) with all RRI [CCWS] trains unavailable.	#CDES-SSD CCALL	-Label	CET1 CCALL HI Pressure	See AT for CET1 HI Pressure description. CET1 CCALL HI Pressure is dedicated to sequences with all RRI [CCWS] trains unavailable.
SSD-SB1	"SSD" sequences (see above) after initiating event IH F/FL_SB1_AB.	#CDES-SSD SB1	-Label	CET1 SB1 HI Pressure	See AT for CET1 HI Pressure description. CET1 SB1 HI Pressure is dedicated to sequences after initiating event IH F/FL_SB1_AB.
SSD-SWGB	"SSD" sequences (see above) after initiating event IH F SWGB_AB	#CDES-SSD SWGB	-Label	CET1 SWGB HI Pressure	See AT for CET1 HI Pressure description. CET1 SB1 HI Pressure is dedicated to sequences after initiating event IH F SWGB_AB
TP	Core damage from sequences initiated by a long loss of offsite power (consequential LOOP included)	#CDES-TP	- Label	CET1 TP HI Pressure	Same treatment as in CET1 HI Pressure. CET1 TP HI Pressure is dedicated to sequences with offsite power not available. Transfer CETs following this CET consider recovery of

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
					offsite power in the late timeframe.
TR	Core damage from transient sequences or from not isolated homogeneous boron dilution sequences	#CDES-TR	- Label	CET1 HI Pressure	See AT for CET1 HI Pressure description.
TR-CC1+2	"TR" sequences (see above) with RRI [CCWS] trains 1 and 2 unavailable.	#CDES-TR CC1+2	-Label	CET1 CC1+2 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC1+2 HI Pressure is dedicated to sequences with RRI [CCWS] trains 1 and 2 unavailable.
TR-CC2	"TR" sequences (see above) with RRI [CCWS] train 2 unavailable.	#CDES-TR CC2	-Label	CET1 CC2 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC2 HI Pressure is dedicated to sequences with RRI [CCWS] train 2 unavailable.
TR-CC2+3	"TR" sequences (see above) with RRI [CCWS] trains 2 and 3 unavailable.	#CDES-TR CC2+3	-Label	CET1 CC2+3 HI Pressure	See AT for CET1 HI Pressure description. CET1 CC2+3 HI Pressure is dedicated to sequences with RRI [CCWS] trains 2 and 3 unavailable.
TR-CCALL	"TR" sequences (see above) with all RRI [CCWS] trains unavailable.	#CDES-TR CCALL	-Label	CET1 CCALL HI Pressure	See AT for CET1 HI Pressure description. CET1 CCALL HI Pressure is dedicated to sequences with all RRI [CCWS] trains unavailable.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TR-CTM	"TR" sequences (see above) in case of fire in the containment.	#CDES-TR CTM	-Label	CET1 CTM HI Pressure	See AT for CET1 HI Pressure description. CET1 CTM HI Pressure is dedicated to sequences with fire in the containment.
TR-SB1	"TR" sequences (see above) after initiating event IH F/FL_SB1_AB.	#CDES-TR SB1	-Label	CET1 SB1 HI Pressure	See AT for CET1 HI Pressure description. CET1 SB1 HI Pressure is dedicated to sequences after initiating event IH F/FL_SB1_AB.
TR-SWGB	"TR" sequences (see above) after initiating event IH F SWGB_AB	#CDES-TR SWGB	-Label	CET1 SWGB HI Pressure	See AT for CET1 HI Pressure description. CET1 SB1 HI Pressure is dedicated to sequences after initiating event IH F SWGB_AB
TRD	Core damage from non isolated or non isolable Steam Line Break sequences with failure of heat removal.	#CDES-TRD	- Label	CET1 HI Pressure	See AT for CET1 HI Pressure description. The depressurisation of at least one SG makes the conditional probability of induced SGTR/induced hot leg rupture higher.

SUB-SECTION 15.4.3.2 - TABLE 2

Core Damage End States for At-Power Plant States – Main Assumptions

N°	Bin or Event Tree	Assumptions
1	**ATI**	The sequences identified as ATI are steam line breaks inside the containment with at least one steam line unisolated and reactor power uncontrolled or heterogeneous boron dilution. In both cases the assumption of direct containment failure may be conservative. This assumption can be re-evaluated if needed.
2	**TR, SS**	High pressure CDES (TR, SS…) are applied to the circumstances where the operator initiates feed and bleed, the bleed is available, MHSI is available but LHSI/RHR is not. This decision is judged to be conservative, since with MHSI available but no long term injection core damage may result at low pressure (if the PSV is opened and not re-closed). This assumption can be re-evaluated when detailed operating procedures are available.
3	**Short LOOP**	Short LOOP sequences, where offsite power is recovered within 2 hours, are not treated as LOOP sequences. The CDES SS LOOPS is a sub-case of CDES SS used to include the specific case of 2 hours mission time in the Level 2 PSA model.
4	**SSD**	CDES “D” such as SSD are assigned to sequences where the operator initiates fast cooldown and fast cooldown fails due to system failure. This decision is conservative since the SG depressurisation impacts unfavourably on Level 2 PSA results.
5	**TRD**	CDES TRD is assigned to sequences of steam line break outside containment. TRD is assumed to be the most unfavourable CDES even in the case of additional reactivity transient.
6	**SG/SG2**	CDES SG/SG2, related to the EFW state, is assigned following the current operating procedures that include the ASG [EFWS] isolation and ASG [EFWS] locking in case of Steam Generator isolation.

SUB-SECTION 15.4.3.2 - TABLE 3

Core Damage End States and their treatment in CDES link trees and CETs – Shutdown Plant States

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TR (CA)	Core damage from transient sequences or not isolated homogeneous boron dilution sequences in state Ca	# CDES-TR (CA)	<u>RCP [RCS] status:</u> RCP [RCS] is closed in all cases. <u>Label:</u> Label TR <u>Containment status:</u> Should the equipment hatch be open, its re-closure is considered in the link tree.		
			If containment (equipment hatch) is closed (or re-closed), it is sent to CET1 HI PR HC C.	CET1 HI PR HC C	CET1 HI PR HC C considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is sent to CET2 HI PR HC C. In this event tree containment failure is evaluated. If the RCP [RCS] is depressurised, it is sent to the CET LO PR HC C.
			If containment (equipment hatch) is open, it is sent to CET1 HI PR HO C.	CET1 HI PR HO C	CET1 HI PR HO C considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is send to CET2 HI PR HO C. If the RCP [RCS] is depressurised, it is sent to the CET LO PR HO C.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TR (CB)	Core damage from transient sequences or not isolated homogeneous boron dilution sequences in state Cb	# CDES-TR (CB)	RCP [RCS] status: RCP [RCS] is open. (Its re-closure is considered in the link tree). Label: Label TR if RCP [RCS] is re-closed Label ML if RCP [RCS] is open Containment status:		
			If containment (equipment hatch) is closed, it is sent to either CET1 HI PR HC C or CET LO PR HC C depending on RCP [RCS] re-closure.	CET1 HI PR HC C CET LO PR HC C	See TR (CA) for CET1 HI PR HC C and CET LO PR HC C description.
			If containment (equipment hatch) is open, it is sent to either CET1 HI PR HO C or CET LO PR HO C depending on RCP [RCS] depressurisation.	CET1 HP HO C CET LO PR HO C	See TR (CA) for CET1 HI PR HO C and CET LO PR HO C description.
TR (D)	Core damage from transient sequences or not isolated homogeneous boron dilution sequences in state D	# CDES-TR (D)	RCP [RCS] status: RCP [RCS] is open in all cases. Label: Label LL is assigned as RCP [RCS] is open (vessel head removed). Containment status: Containment (equipment hatch) is assumed to be closed.	CET LO PR. (D)	CET LO PR. (D) analyses the core melt and containment failure progression when the RCP [RCS] is depressurised and the equipment hatch is closed.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TR (E)	Core damage from isolated ISLOCA sequences in state E	# CDES-TR (E)	RCP [RCS] status: RCP [RCS] is open in all cases. Label: Label LL is assigned as RCP [RCS] is open (vessel head removed). Containment status: Containment (equipment hatch) is assumed to be open	CET LO PR. (E)	CET LO PR. (E) analyses the core melt and containment failure progression when the RCP [RCS] is depressurised and the equipment hatch is open.
TR-CCALL (CA)	" TR (CA)" sequences with all RRI [CCWS] trains unavailable	# CDES-TR-CCALL (CA)	RCP [RCS] status: RCP [RCS] is closed in all cases. Label: Label TR Containment status: If containment (equipment hatch) is closed (or re-closed), it is sent to CET1 CCALL HIP HC C.	CET1 CCALL HIP HC C	CET1 CCALL HIP HC C is dedicated to sequences with all RRI [CCWS] unavailable. It considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is send to CET2 CCALL HIP HC C. In this event tree containment failure is evaluated. If the RCP [RCS] is depressurised, it is sent to the CET CCALL LOP HC C.
			If containment (equipment hatch) is open, it is sent to CET1 CCALL HIP HO C.	CET1 CCALL HIP HO C	CET1 CCALL HIP HO C is dedicated to sequences with all RRI [CCWS] unavailable. It considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is send to CET2 CCALL HIP HO C. If

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
					the RCP [RCS] is depressurised, it is sent to the CET CCALL LOP HO C.
TR-CCALL (CB)	" TR (CB)" sequences with all RRI [CCWS] trains unavailable	# CDES-TR-CCALL (CB)	<u>RCP [RCS] status:</u> RCP [RCS] is open (Its re-closure is considered in the link tree). <u>Label:</u> Label TR if RCP [RCS] closed Label ML if RCP [RCS] open <u>Containment status:</u> If containment (equipment hatch) is closed (or re-closed), it is sent to either CET1 CCALL HIP HC C or CET CCALL LOP HC C depending on RCP [RCS] depressurisation. If containment (equipment hatch) is open, it is sent to either CET1 CCALL HIP HO C or CET CCALL LOP HO C depending on RCP [RCS] depressurisation.	CET1 CCALL HIP HC C CET CCALL LOP HC C CET1 CCALL HIP HO C CET CCALL LOP HO C	See TR CCALL (CA) for CET1 CCALL HIP HC C and CET CCALL LOP HC C description. See TR CCALL (CA) for CET1 CCALL HIP HO C and CET CCALL LOP HO C description.
TR-CCALL (D)	" TR (D)" sequences with all RRI [CCWS] trains unavailable	# CDES-TR-CCALL (D)	<u>RCP [RCS] status:</u> RCP [RCS] is open in all cases. <u>Label:</u> Label LL is assigned as RCP [RCS] is open (vessel head removed). <u>Containment status:</u> Containment (equipment hatch) is assumed to be closed.	CET LO PR. (D)	See TR (D) for CET LO PR. (D) description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TR-LOOPS (CA)	“TR” short LOOP sequences in state Ca.	# CDES-TR-LOOPS (CA)	RCP [RCS] status: RCP [RCS] is closed in all cases. Label: Label TR Containment status:		
			If containment (equipment hatch) is closed (or re-closed), it is sent to CET1 LOOPS HIP HC C.	CET1 LOOPS HIP HC C	CET1 LOOPS HIP HC C is dedicated to sequences with loss of power. It considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is send to CET2 LOOPS HIP HC C. In this event tree containment failure is evaluated. If the RCP [RCS] is depressurised, it is sent to the CET LOOPS LOP HC C.
			If containment (equipment hatch) is open, it is sent to CET1 LOOPS HIP HO C.	CET1 LOOPS HIP HO C	CET1 LOOPS HIP HO C is dedicated to sequences with loss of power. It considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is send to CET2 LOOPS HIP HO C. If the RCP [RCS] is depressurised, it is sent to the CET LOOPS LOP HO C.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TR-LOOPS (CB)	"TR" short LOOP sequences in state Cb.	# CDES-TR-LOOPS (CB)	RCP [RCS] status: RCP [RCS] is open (Its re-closure is considered in the link tree). Label: Label TR if RCP [RCS] closed Label ML if RCP [RCS] open Containment status: If containment (equipment hatch) is closed or re-closed, it is sent to either CET1 LOOPS HIP HC C or CET LOOPS LOP HC C depending on RCP [RCS] depressurisation.	CET1 LOOPS HIP HC C CET LOOPS LOP HC C	See TR LOOPS (CA) for CET1 LOOPS HIP HC C and CET LOOPS LOP HC C description.
			If containment (equipment hatch) is open, it is sent to either CET1 LOOPS HIP HO C or CET LOOPS LOP HO C depending on RCP [RCS] depressurisation.	CET1 LOOPS HIP HO C CET LOOPS LOP HO C	See TR LOOPS (CA) for CET1 LOOPS HIP HO C and CET LOOPS LOP HO C description.
TR-LOOPS (D)	"TR" short LOOP sequences in state D.	# CDES-TR-LOOPS (D)	RCP [RCS] status: RCP [RCS] is open in all cases. Label: Label LL is assigned as RCP [RCS] is open (vessel head removed). Containment status: Containment (equipment hatch) is assumed to be closed.	CET LOOPS LO PR (D)	CET LOOPS LO PR. (D) analyses the core melt and containment failure progression in case of loss of power and when the RCP [RCS] is depressurised and the equipment hatch is closed.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SS-CCALL (CA)	Core damage from seal LOCA sequences with offsite power available and with all RRI [CCWS] trains unavailable in state Ca.	# CDES-SS-CCALL (CA)	<u>RCP [RCS] status:</u> RCP [RCS] is closed. <u>Label:</u> Label SS <u>Containment status:</u> Containment (equipment hatch) is assumed to be non-closable. However the model follow the same structure than the TR CCALL (CA) sequences to ease any future modification.		
			DUMMY: If containment (equipment hatch) is closed, it is sent to CET1 CCALL HIP HC C.	CET1 CCALL HIP HC C	DUMMY: See TR CCALL (CA) for CET1 CCALL HIP HC C and CET CCALL LOP HC C description.
			If containment (equipment hatch) is open, it is sent to CET1 CCALL HIP HO C.	CET1 CCALL HIP HO C	See TR CCALL (CA) for CET1 CCALL HIP HO C and CET CCALL LOP HO C description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SS-LOOPS (CA)	Core damage from seal LOCA sequences initiated with short LOOP in state Ca.	# CDES-SS-LOOPS (CA)	<u>RCP [RCS] status:</u> RCP [RCS] is closed in all cases. <u>Label:</u> Label SS <u>Containment status:</u> Containment (equipment hatch) is assumed to be non-closable. However the model follow the same structure than the TR LOOPS (CA) sequences to ease any future modification. DUMMY (N/A): If containment (equipment hatch) is closed, it is sent to CET1 LOOPS HIP HC C. If containment (equipment hatch) is open, it is sent to CET1 LOOPS HIP HO C.	N/A CET1 LOOPS HIP HO C	DUMMY (N/A): See TR LOOPS (CA) for CET1 LOOPS HIP HC C and CET LOOPS LOP HC C description. See TR LOOPS (CA) for CET1 LOOPS HIP HO C and CET LOOPS LOP HO C description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SS-LOOPS (CB)	Core damage from seal LOCA sequences initiated with short LOOP in state Cb.	# CDES-SS-LOOPS (CB)	RCP [RCS] status: RCP [RCS] is open. Its re-closure is considered in the link tree. Label: Label SS if RCP [RCS] closed Label ML if RCP [RCS] open Containment status:		
			If containment (equipment hatch) is closed, it is sent to either CET1 LOOPS HIP HC C or CET LOOPS LOP HC C depending on RCP [RCS] depressurisation.	CET1 LOOPS HIP HC C CET LOOPS LOP HC C	See TR LOOPS (CA) for CET1 LOOPS HIP HC C and CET LOOPS LOP HC C description.
			If containment (equipment hatch) is open, it is sent to either CET1 LOOPS HIP HO C or CET LOOPS LOP HO C depending on RCP [RCS] depressurisation.	CET1 LOOPS HIP HO C CET LOOPS LOP HO C	See TR LOOPS (CA) for CET1 LOOPS HIP HO C and CET LOOPS LOP HO C description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SL (CA)	Core damage from small LOCA sequences in state Ca	# CDES-SL (CA)	<u>RCP [RCS] status:</u> RCP [RCS] is closed in all cases. <u>Label:</u> Label SL <u>Containment status:</u> Containment (equipment hatch) is assumed to be non-closable. However the model follow the same structure than the TR (CA) sequences to ease any future modification.		
			DUMMY (N/A): If containment (equipment hatch) is closed, it is sent to either CET1 HI PR HC C or CET LO PR HC C depending on RCP [RCS] depressurisation.	N/A	DUMMY (N/A): See TR (CA) for CET1 HI PR HC C and CET LO PR HC C description.
			If containment (equipment hatch) is open, it is sent to either CET1 HI PR HO C or CET LO PR HO C depending on RCP [RCS] depressurisation.	CET1 HP HO C CET LO PR HO C	See TR (CA) for CET1 HI PR HO C and CET LO PR HO C description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SL (CB)	Core damage from small LOCA sequences in state Cb	# CDES-SL (CB)	RCP [RCS] status: RCP [RCS] is open. Its re-closure is considered in the link tree. Label: Label SL if RCP [RCS] closed Label ML if RCP [RCS] open Containment status:		
			If containment (equipment hatch) is closed, it is sent to either CET1 HI PR HC C or CET LO PR HC C depending on RCP [RCS] depressurisation.	CET1 HP HC C CET LO PR HC C	See TR (CA) for CET1 HI PR HC C and CET LO PR HC C description.
			If containment (equipment hatch) is open, it is sent to either CET1 HI PR HO C or CET LO PR HO C depending on RCP [RCS] depressurisation.	CET1 HP HO C CET LO PR HO C	See TR (CA) for CET1 HI PR HO C and CET LO PR HO C description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SL (D)	Core damage from small LOCA sequences in state D	# CDES-SL (D)	<u>RCP [RCS] status:</u> RCP [RCS] is open in all cases. <u>Label:</u> Label LL is assigned as RCP [RCS] is open (vessel head removed). <u>Containment status:</u> Containment (equipment hatch) is assumed to be closed.	CET LO PR. (D)	See TR (D) for CET LO PR. (D) description.
SL (E)	Core damage from small LOCA sequences in state E	# CDES-SL (E)	<u>RCP [RCS] status:</u> RCP [RCS] is open in all cases. <u>Label:</u> Label LL is assigned as RCP [RCS] is open (vessel head removed). <u>Containment status:</u> Containment (equipment hatch) is assumed to be open.	CET LO PR. (E)	See TR (E) for CET LO PR. (E) description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TP (CA)	Core damage from sequences initiated by a long LOOP in state Ca	# CDES-TP (CA)	<u>RCP [RCS] status:</u> RCP [RCS] is closed in all cases. <u>Label:</u> Label TP <u>Containment status:</u>		
			If containment (equipment hatch) is closed or re-closed, it is sent to CET1 TP HIP HC C.	CET1 TP HIP HC C	CET1 TP HIP HC C is dedicated to sequences with long term loss of power. It considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is send to CET2 TP HIP HC C. In this event tree containment failure is evaluated. If the RCP [RCS] is depressurised, it is sent to the CET TP LOP HC C.
			If containment (equipment hatch) is open, it is sent to CET1 TP HIP HO C.	CET1 TP HIP HO C	CET1 TP HIP HO C is dedicated to sequences with long time loss of power. It considers specific early time frame phenomena for high pressure sequences, and RCP [RCS] depressurisation. If the pressure stays high it is send to CET2 TP HIP HO C. If the RCP [RCS] is depressurised, it is sent to the CET TP LOP HO C.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
TP (CB)	Core damage from sequences initiated by a long LOOP in state Cb	# CDES-TP (CB)	RCP [RCS] status: RCP [RCS] is open. Its re-closure is considered in the link tree. <u>Label:</u> Label TP if RCP [RCS] closed Label ML if RCP [RCS] open <u>Containment status:</u>		
			If containment (equipment hatch) is closed or re-closed, it is sent to either CET1 TP HI PR HC C or CET TP LO PR HC C depending on RCP [RCS] depressurisation.	CET1 TP HI PR HC C CET TP LO PR HC C	See TP (CA) for CET1 TP HI PR HC C and CET TP LO PR HC C description.
			If containment (equipment hatch) is open, it is sent to either CET1 HI PR HO C or CET LO PR HO C depending on RCP [RCS] depressurisation.	CET1 TP HP HO C CET TP LO PR HO C	See TP (CA) for CET1 TP HI PR HO C and CET TP LO PR HO C description.
TP (D)	Core damage from sequences initiated by a long LOOP in state D	# CDES-TP (D)	RCP [RCS] status: RCP [RCS] is open in all cases. <u>Label:</u> Label LL is assigned as RCP [RCS] is open (vessel head removed). <u>Containment status:</u> Containment (equipment hatch) is assumed to be closed.	CET TP LO PR. (D)	CET TP LO PR. (D) analyses the core melt and containment failure progression when the RCP [RCS] is depressurised and the equipment hatch is closed.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SP (CA)	Core damage from seal LOCA sequences initiated with long LOOP in state Ca.	# CDES-SP (CA)	<u>RCP [RCS] status:</u> RCP [RCS] is closed in all cases. <u>Label:</u> Label SP <u>Containment status:</u> Containment (equipment hatch) is assumed to be non-closable. However the model follow the same structure than the TP (CA) sequences to ease any future modification.		
			DUMMY (N/A): If containment (equipment hatch) is closed, it is sent to CET1 TP HIP HC C.	N/A	DUMMY (N/A): See TP (CA) for CET1 TP HIP HC C description.
			If containment (equipment hatch) is open, it is sent to CET1 TP HIP HO C.	CET1 TP HIP HO C	See TP (CA) for CET1 TP HIP HO C description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
SP (CB)	Core damage from seal LOCA sequences initiated with long LOOP in state Cb.	# CDES-SP (CB)	RCP [RCS] status: RCP [RCS] is open. Its re-closure is considered in the link tree. Label: Label SP if RCP [RCS] closed Label ML if RCP [RCS] open Containment status:		
			If containment (equipment hatch) is closed, it is sent to either CET1 TP HI PR HC C or CET TP LO PR HC C depending on RCP [RCS] depressurisation.	CET1 TP HI PR HC C CET TP LO PR HC C	See TP (CA) for CET1 TP HI PR HC C and CET TP LO PR HC C description.
			If containment (equipment hatch) is open, it is sent to either CET1 TP HI PR HO C or CET TP LO PR HO C depending on RCP [RCS] depressurisation.	CET1 TP HP HO C CET TP LO PR HO C	See TP (CA) for CET1 TP HI PR HO C and CET TP LO PR HO C description.

Bin	Description of sequences in Bin	Link Tree	Treatment in link tree	CET	Treatment in CET
ATI (CA)	Any sequence with failure of reactivity control and where the failure to control reactivity may lead directly to challenge containment integrity which is assumed to fail containment. State Ca.	# CDES-ATI (CA)	<u>RCP [RCS] status:</u> RCP [RCS] is closed in all cases. <u>Label:</u> Label ATI <u>Containment status:</u> We consider that in all cases where the containment (equipment hatch) is closed or re-closed, then it is sent to CET CF HC C.	CET CF HC C	CET CF HC C considers containment failure before vessel breach equipment hatch close (RC304). Note that as per the hypothesis provided in Sub-section 15.4.3.2 - Table 4, the ATI severe accident sequences consider a direct containment failure. It was therefore assumed that in all sequences the containment hatch is closed and the containment failed, rather than having an open containment.
IS (C)	Core Damage from non isolated Interfacing System LOCA sequences in state C	#CDES-IS SD	<u>RCP [RCS] status:</u> RCP [RCS] is closed in states Ca and open in state Cb. <u>Label:</u> Label IS <u>Containment status:</u> Equipment hatch status has no impact on the sequence treatment and assumed RC.	CET ISL SD	CET ISL SD determines whether or not there is water available to cover the break outside containment and scrub the fission products released from the leak.

SUB-SECTION 15.4.3.2 - TABLE 4

Core Damage End States for Shutdown Plant States – Main Assumptions

N°	Bin or Event Tree	Assumptions
1	ATI (CA)	The sequences identified as ATI are heterogeneous boron dilution sequences. In both cases the assumption of direct containment failure may be conservative. This assumption can be re-evaluated if needed.
2	Short LOOP	Short LOOP sequences, where offsite power is recovered within 2 hours, are not treated as LOOP sequences. The CDES SS-LOOPS (CA), SS-LOOPS (CB), TR-LOOPS (CA), TR-LOOPS (CB), TR-LOOPS (D), are sub-cases of CDES SS or TR used to include the specific case of 2 hours mission time in the Level 2 PSA model.
3	Short LOOP	In the case of Level 1 PSA sequences initiated by short LOOP where the accident progression is not detailed, a seal LOCA has been assumed. This assignment aims to give conservative results. However it has to be noted that seal LOCA occurrence, even if possible, is highly improbable.

3.3. PHENOMENOLOGICAL EVALUATIONS AND OTHER SUPPORTING EVALUATIONS

3.3.1. Introduction

Phenomenological evaluations (PEs) are performed to develop the plant specific phenomenological information needed to quantify the CET. The PEs address those severe accident phenomena judged to be significant in determining the eventual outcome of a severe accident. Each PE evaluates the current state of knowledge concerning the phenomenon and considers inputs from available sources, including experiments, industry studies, and plant-specific accident progression analyses.

The PEs develop the probability values and uncertainty distributions used in the Level 2 PSA models. The probability values and uncertainty distributions are input to the basic events used in the CET top events (or supporting fault trees). In some cases, the PEs developed Decomposition Event Trees (DETs), which are small event trees produced and calculated independently of the CET, to produce probability values for use in the CET models.

The following PEs have been developed for the UK EPR PSA2:

- Induced rupture of the reactor system pressure boundary
- Fuel coolant interactions.
- In-vessel core recovery.
- Phenomena at vessel failure.
- Hydrogen deflagration, flame acceleration, and deflagration-to-detonation transition.
- Long-term containment challenges.

These physical phenomena are described in sections 3.3.2 to 3.3.13 below.

In addition to phenomenological evaluations, the following supporting evaluations have been performed:

- Containment Fragility Evaluation (section 3.3.14)
- Equipment Survivability Evaluation (section 3.3.15)
- Human Error Probability Evaluation (section 3.3.16)
- Supporting Severe Accident (MAAP) Analysis (section 3.3.17).

Phenomenological Evaluations were performed for at-power plant states (states A and B). In order to adapt these for the shutdown states (C, D and E), a review of the analyses and an update of the split fraction calculations was performed. The review of phenomena for shutdown states is described in section 3.3.18.

3.3.2. Induced Rupture of the RCP [RCS] Pressure Boundary - Phenomenology

Following core uncovery, natural circulation of superheated steam (and hydrogen) can occur in the reactor vessel and RCP [RCS]. Natural circulation is a result of small differences in gas density between various regions in the reactor vessel and reactor coolant system as a result of heat losses to the structures in each region. Experiments have been performed in the U.S., using a 1/6th scale model of a PWR reactor coolant system [Ref-1]. These tests have shown that three distinct natural circulation patterns can be established for an event occurring at high system pressure in this type of system.

These circulation patterns are: (1) between the core region and upper plenum of the reactor vessel, (2) between the upper plenum of the reactor vessel and the SG inlet plenum, and (3) between the inlet plenum and outlet plenum of the SG.

The natural circulation flows have been shown to be a strong function of system pressure, with the flow decreasing to nearly zero at pressures below approximately 12 MPa. The natural circulation flows are also quickly disrupted by forced circulation flows, such as the opening of the pressuriser relief or safety valves; however, the natural circulation flow is rapidly re-established when the forced circulation flow is terminated.

Natural circulation of gases in the reactor system during the core degradation phase is important since it transports heat away from the overheating core, and into the structures of the upper plenum, hot leg and SG tubes. The heat transport has two major effects:

- It slows the heat-up rate of the core, and causes the degradation to proceed more uniformly; however, the heat removal by this process is not large enough to arrest core degradation.

- It causes the heat-up of the reactor system structures in contact with the circulating gas flow. This heat-up can be sufficient in certain cases to cause failure of the reactor system pressure boundary before vessel failure. This potential failure may occur in any part of the system exposed to the heat-up effects of the gas circulation–principally the hot leg, surge line or SG tubes

For a high pressure transient or SLOCA, residual water present in the crossover legs and in the lower plenum of the reactor vessel is expected to ‘block’ full loop natural circulation of gases. This is what was observed in the experiments. However, in some sequences, clearance of these loop seals could occur, in which case the preferential natural circulation pattern would be that shown in Sub-section 15.4.3.3 - Figure 1: Natural Circulation Flow Paths in the Primary System, (i.e. the ‘normal’ full loop circulation path). Though less likely, this situation must be considered since it gives rise to higher gas flow rates, and in principle to structural heating rates. For example, in the case of a break in the cold leg, including pump seal leakage, a unidirectional circulation flow, instead of a counter-current flow, may prevail with resulting increased heat transfer to the structures. As a consequence, higher temperature in the SG tubes will occur, especially if these tubes are not cooled by water from the secondary side.

The probability of RCP [RCS] failure depends on:

- The temperature of the structure. The temperature is higher close to the RPV and may be considerably lower for the SG tubes.

- The pressure differential across the structure (because the failure temperature of the material decreases with increasing pressure).The pressure difference is higher for the pipes of the hot leg than for the tubes because the pressure on the secondary side could be up to approximately 10 MPa.

- The duration of high temperature. The time period corresponds to the period from the beginning of core heat-up until core slumping. Under certain circumstances a late phase increase of structural temperature may occur just before vessel failure.

Induced RCP [RCS] structure failure is important for two reasons:

- Failure of the SG tubes. SG tube failure may lead to containment bypass in the case that the SG cannot be isolated and a closure of the main steam valves is not possible. This failure mode is of most concern in Level 2 PSA, because it leads to the potential for large early release.

- Failure of the hot leg close to the RPV (hot leg nozzle) or surge line (surge line nozzle). RCP [RCS] piping failure prior to reactor vessel failure can have a substantial effect on other in-vessel and ex-vessel degraded core phenomena. Hydrogen production can be increased due to water flashing in the bottom head of the reactor vessel which passes through the overheated core, or by the discharge of accumulator water onto the overheated core. Further, the reactor coolant system pressure at the time of reactor vessel failure is near the containment pressure, thus affecting the potential for degraded core phenomena associated with high pressure reactor vessel failure events (e.g. core debris dispersion and direct containment air heating). Also, the fission product releases to containment are substantially increased due to the creation of a large blowdown from the RCP [RCS] near the time of fission product release from the core.

It is important to note that the above failure modes are mutually exclusive. Once failure occurs at any location, the resulting depressurisation and reduction in stress on other components precludes subsequent failures.

This phenomenological evaluation [Ref-2] uses analyses performed with MAAP4.0.7 to investigate various high pressure accident sequences, and to evaluate the sensitivity of the induced rupture phenomena to various key parameters, including:

- Natural circulation flow rate.

- Rupture location.

- Different initiators.

- Degraded tubes.

- SG pressure.

- Seal leaks and SLOCAs and behaviour of loop seals.

- Creep correlation fitting parameters.

3.3.3. Probabilistic Evaluation of Induced Rupture

The Level 2 PSA provides a probabilistic evaluation of the potential for rupture of either the RCP [RCS] loop or the SG tubes for applicable (high pressure) situations. The probabilistic evaluation is performed by developing probability distributions for the key uncertain parameters, and performing Monte Carlo simulations to determine the predicted times to hot leg, SG tube, and vessel rupture.

This CET top event is only evaluated for cases where the primary system has not been depressurised using the dedicated severe accident depressurisation valves. The probability of depressurisation failure is evaluated separately. For cases with no primary depressurisation via the pressuriser, the strongest sensitivity observed is to SG pressure.

If the SGs remain pressurised, there is no risk of tube failure for any case analysed. Hot leg rupture is, however, highly likely (probability >0.9). The location of hot leg rupture is predicted to be at the nozzle to hot leg pipe weld. This is important for some sequences because it leads to break flow discharge to the reactor pit.

If the SGs are depressurised, either due to failure of one or more secondary relief or safety valves, or due to operator action, the situation is more severe, because SG tube failure is predicted to occur first with a probability of around 4E-4 for transients and up to 0.84 for sequences involving seal failure or small LOCAs.

3.3.4. Fuel-Coolant Interactions - Phenomenology

The key fuel-coolant interaction is steam explosion. Steam explosions may occur, and are potentially significant, in both the ex-vessel and in-vessel phases of a nuclear reactor accident. In-vessel steam explosions are postulated as potentially failing the upper or lower head of the reactor pressure vessel. A possible consequence of upper head failure, if sufficiently energetic, is containment failure. Ex-vessel steam explosions may cause local damage to internal containment structures.

The initial condition from which a steam explosion process would start in an accident scenario is core melt and relocation. Core melt can occur at high or low RCP [RCS] pressure. Eventually, following extensive core melting and slumping, a large mass of molten material falls into the lower head, where water is present. This is the in-vessel steam explosion scenario. For the ex-vessel scenario the initial condition would be a pour of molten corium into an ex-vessel water pool.

When hot molten material enters into a volatile coolant, explosive interactions are a possibility. The steam explosion process can be broken down into a series of sequential phases. These phases include: (1) initial coarse mixing phase (pre-mixing), (2) trigger phase, (3) detonation propagation phase and (4) hydrodynamic expansion phase. These four phases are described below.

1. Initial Coarse Mixing Phase: During the initial premixing phase, the molten corium entering the coolant undergoes fragmentation (i.e. vapour generation causes breakup of the jet or drops into smaller diameter drops and depends on breakup due either to acceleration or velocity difference between molten material and coolant). The breakup increases the surface area for heat transfer and, therefore, steam generation increases. However, a quasi-stable state is reached because steam can settle into a stable blanket around the fragments and the cooling of the corium (and, therefore, steam production) rate is lowered by this isolating vapour film

2. Triggering Phase: Triggering starts when the quasi-stable vapour film collapses due to local perturbation. This allows (liquid) water to come into (closer) contact with the molten corium. Heat transfer is thus enhanced and the local steam production rate and local steam velocity increases. The next phase, detonation propagation, is entered.

3. Detonation Propagation Phase: In the detonation propagation phase, sharp micro-interaction zones propagate through the mixing zone. The process escalates as the molten corium is further fragmented, meaning that there is a rapid increase in the surface area for heat transfer and, therefore, further increased steam production. Intensive steam generation could generate shock waves.

4. Hydrodynamic Expansion Phase: In the expansion phase, thermal energy is converted into mechanical energy which acts on its surroundings (upper head, lower head, internal or ex-vessel structures). This leads either to missile generation or lower head failure in the in-vessel scenario (a slug of water becomes a high-energy missile which transfers its energy to the upper head and then to the containment) or to loads on internal containment structures (possibly dynamic loads) in the ex-vessel scenario.

3.3.5. Probabilistic Evaluation of Fuel-Coolant Interactions

The probabilistic evaluation addresses steam explosions in-vessel and ex-vessel [Ref-1]. The evaluations involve the use of Monte Carlo simulations.

3.3.5.1. In-Vessel Steam Explosion

For the in-vessel scenario, the probabilistic evaluation centres on a comparison between steam explosion loads in terms of the mechanical energy generated and a threshold above which the energy is sufficient to cause containment failure. Both the load and the threshold are treated as uncertain parameters, although it was conservatively assumed that any load sufficient to cause upper head failure would also cause containment failure. The probabilistic evaluation was performed for two scenarios, these being (1) core melt at low pressure, and (2) core melt at high pressure. These two scenarios were evaluated separately because triggering is generally considered more likely at low pressure, whereas the conversion ratio of thermal to mechanical energy is expected to be higher at high pressure.

The loads resulting from an in-vessel steam explosion were calculated by multiplication of the following factors to give the resulting energy of a molten slug potentially affecting the upper head:

- The total mass of the core.
- The fraction of the core material in the lower head that participates in pre-mixing.
- The thermal energy stored in the core materials per unit mass of core (i.e. it is assumed that the composition of the molten core in the lower plenum maintains the same proportions of materials as in the core as whole).
- The conversion ratio for thermal to mechanical energy.
- The fraction of the mechanical energy that is transmitted to the slug (i.e. there are expected to be losses due to venting around the slug during the expansion phase).

Each of the above factors (except the total core mass which was modelled by a single value) was assigned a probability distribution. The probability distributions were generated from a review of various references such as NUREG-1116 [Ref-1] and SKI Report 02:16 [Ref-2] containing information about and assessments of steam explosions (mostly non-probabilistic). The use of Monte Carlo simulation enables the distributions on the above basic parameters to be propagated through the multiplicative model described above to give a probability distribution for the load on the upper head.

The strength of the upper head (stated in energy load terms) was based on generic estimates of this strength in NUREG-1524 [Ref-3]. The median value used for the strength of the upper head was 1GJ. This value was treated as an uncertain parameter and assigned a probability distribution, centred on 1GJ.

The load and strength distributions were compared in the Monte Carlo simulation to generate the probability of containment failure given a steam explosion occurring in-vessel (for low-pressure and high-pressure scenarios). The final result for an in-vessel steam explosion leading to containment failure also factors in the probability that a steam explosion might occur which is not modelled by the factors described above. The assessment generates the following approximate values for the probability of in-vessel steam explosion causing containment failure:

- A. A value of 2.3E-05 for a high-pressure core melt scenario.
- B. A value of 5.6E-06 for a low-pressure core melt scenario.

A further possible consequence of an in-vessel steam explosion that was investigated is lower head failure. Where lower head failure is assessed as occurring, damage in the reactor pit is assumed without taking credit for the distribution of energy loads that the pit structures would actually experience or the capacity of the pit to withstand these. This approach is somewhat conservative. It should also be noted that the CET modelling assumes that the impact of pit damage on the progression of the postulated severe accident would be early release of melt from the pit into the spreading area. Since such a release is not the design pathway for the EPR melt stabilisation approach, it is assume (also conservatively) that molten core concrete interaction (MCCI) would not be prevented in such a case.

The assessment of the lower head failure probability closely followed the procedure outlined above for the upper head failure (leading to containment failure). The difference between the two evaluations is that the factor for the fraction of the mechanical energy that is transmitted to the slug that impacts the upper head was not applied for the lower head evaluation. Rather, 100 percent of the mechanical energy was assumed to impact the lower head. This assumption is conservative.

The resulting probabilities of a steam explosion causing failure of the lower head given a core melt and relocation were approximately as follows:

- A value of 8.6E-04 for a high pressure core melt scenario.
- A value of 2.5E-05 for a low pressure core melt scenario.

3.3.5.2. Ex-Vessel Steam Explosion

Ex-vessel steam explosions were evaluated for scenarios in which molten corium could be released from the vessel into a wet pit. In general, the EPR pit is expected to be dry. However, two scenarios were identified in which this may not be the case and ex-vessel steam explosions were, therefore, considered for the following cases:

1. Pour of molten corium into an ex-vessel pool at vessel failure for a sequence that has the RCP [RCS] depressurised due to an induced hot leg rupture (located at the RPV nozzle) leading to the spillage of water into the reactor pit. In this case the flow of corium into the pool is at the rate occurring at the time of vessel failure. MAAP analyses confirmed that in this scenario, with failure at the RPV nozzle (this being the most likely failure location), a water pool (approximately 4m deep) develops in the reactor pit.

2. Pour of molten corium into an ex-vessel water pool in the longer term, after vessel failure, due to the long-term melting of the remaining core material not in the lower head at the time of vessel failure. In this case, the pour may be into an ex-vessel pool that has accumulated because of safety injection water which is lost into the pit after vessel breach. In this case, it is considered likely that the remaining core material in the vessel would freeze rather than melt and fall into the ex-vessel water pool. The pour rates are also anticipated to be lower than in Case 1 above. It was therefore considered acceptable to bound this scenario (Case 2) using the results of Case 1 which predicts low probabilities of an ex-vessel steam explosion causing pit damage (see below, results obtained for Case 1).

The ex-vessel steam explosion analysis is based on a comparison of impulse loading on the cavity structures to their strengths. The impulse loading is evaluated in two steps. The first step was to evaluate the mechanical energy release following a similar process to that used for the in-vessel steam explosion. Specifically the mechanical energy release was evaluated by multiplication of:

1. The total mass of corium in premixing.

2. The thermal energy stored in the core materials per unit mass of core. (It is assumed that the composition of the molten core in the lower plenum maintains the same proportions of materials as in the core as whole.)

3. The conversion ratio for thermal to mechanical energy.

As in the case of the in-vessel steam explosion analyses, the total load was evaluated probabilistically using Monte Carlo simulations. Items (2) and (3) were evaluated using the same distributions as for the in-vessel steam explosion. The total mass of corium in pre-mixing was, however, re-evaluated for the ex-vessel scenario to take into account the expected flows into the ex-vessel water pool and the depth of this pool.

The second step was to evaluate the impulse loading by translating the mechanical energy release to an impulse. This was performed by use of a correlation [Ref-1] relating energy release to peak overpressure and duration.

Finally, the impulse loading probability distribution was compared to the impulse loading capacity of the reactor pit structures. As in the cases of in-vessel steam explosions, the capacity of the structures was assigned a probability distribution. It is expected that the major structures of the EPR reactor pit (including the plug) are likely to withstand an impulse loading of at least 10 kPa.s. The probability distribution assigned also contemplates lower values, with capacities in the range 5-10 kPa.s being considered possible (but not probable) and a residual probability being assigned to allow for capacities as low as 2-5 kPa.s. An upper capacity of 20 kPa.s [Ref-2] was taken in developing the probability distribution.

The evaluation generated a probability of structural damage due to an ex-vessel steam explosion of approximately 2.6E-05.

As mentioned above, this value was generated by comparing a distribution of loads to a distribution of pit capacity. This comparison was carried out using Monte Carlo simulation. The result was generated based on the conditions expected for a hot leg rupture with discharge of water into the pit and release of molten corium into that water at vessel failure. To simplify the CET modelling, this value is also conservatively applied to model the effects of an ex-vessel steam explosion occurring due to the release of long term melt into an ex-vessel water pool in the period after vessel failure.

3.3.6. In-Vessel Recovery - Phenomenology

The principal cause of core heat-up in a severe accident is the lack of cooling water. Depending on the time when safety injection (SI) is recovered, the accident progression can be stopped or delayed. Thus the SI recovery time has a direct impact on the RCP [RCS] and containment conditions after injection is initiated to a degraded core. Depending on the injection flow rate, the hot corium can either be quenched or not. With an insufficient flow, the accident progression is delayed, but reactor vessel failure is not prevented.

The effects of the re-flooding of a damaged core as presented in IAEA Workshop [Ref-1] include enhanced oxidation leading to temperature escalation and high hydrogen peaks. Flooding a damaged core can also lead to the formation of a debris bed due to thermal shock collapse of the upper fuel rods located above the core molten pool, as happened in the Three Mile Island (TMI) accident.

A severe accident starts with insufficient cooling conditions in the core followed by continuous heat-up of the fuel. The heat transferred from the fuel rods to the steam is not sufficient to remove all decay heat, but is able to heat up the steam close to the highest temperature of the fuel rods that normally occurs at the top of the core. Core exit temperature of the steam is therefore a measure of the early accident progression and is therefore used as a criterion for initiating dedicated bleed (at approximately 650ºC).

To mitigate further accident progression, in particular the consequences of a high pressure core melt scenario, the RCP [RCS] depressurisation strategy aims at opening the depressurisation valves to allow injection of cooling water from the available safety injection system and accumulators before the start of core melt. If the depressurisation and the injection of the RIS [SIS] accumulator or the LHSI are not successful, fuel element degradation will continue.

The exothermic reaction of the superheated steam with the Zirconium (Zr) of the fuel rods produces hydrogen, which is transported with the remaining steam through the RCP [RCS] into the containment. The production rate is governed by the diffusion of the steam through the boundary layer of hydrogen that establishes around the fuel rods and through the oxidic layer to the unoxidised Zr. When the temperature has reached approximately 1200ºC the oxidation reaction becomes significant and dominates the heat-up of the fuel, which is significantly accelerated because the reaction is strongly exothermic. The availability of steam influences the production rate. The rate can be limited in the late phase, when water level and heat transferred to the water are low (steam starvation) and, on the other hand, enhanced in case of re-flood, particularly when the core is already exposed to high temperature.

The core melt starts with eutectic interactions between core materials, then relocation of cladding, structural materials and fuel, with formation of blockages near the bottom of the core and a molten pool. Natural convection in a volumetrically heated molten pool leads first to a sideward relocation through the heavy reflector to the lower head, which occurs earlier than a downward relocation through the thick core support plate.

The interaction of the melt with water in the lower plenum could result in mechanical loads being applied to the RPV, and, in cases of RPV failure, also to the containment shell. Dispersion of (or a part of) the melt within the RCP [RCS] could also occur. As a result of the latter process, heat sources are distributed along the RCP [RCS] piping with the potential consequences of thermal failure and also re-vaporisation of deposited fission products.

Corium heat-up in the lower plenum after the first relocation into the water consists of dryout of debris which subsequently re-melts, and which, in combination with the gradually relocating corium, forms a molten pool. The pool may develop crusts on the top and along the vessel wall. If no water injection is available, this debris bed at the bottom of the RPV may grow to a large size melt pool. Convection within this pool will transport heat to the top of the pool with the expected consequence of a lateral failure of the RPV at an elevation close to the surface of the pool. This failure mode competes with failure at the bottom of the vessel, where, although heat fluxes are much lower, a high pressure local failure of the RPV, possibly before a large pool of molten material has developed, can be postulated.

Vessel failure can be due to several possible mechanisms:

- The molten metal located on top of the oxidic melt, which thermally attacks and weakens the vessel wall and causes failure due to the internal residual pressure.
- Weight of the corium and thermal loads resulting in creep rupture.
- A jet impingement occurring in the relocation phase causing localised ablation of the lower head.

3.3.7. Probabilistic Evaluation of In-Vessel Core Recovery

The approach used in the Level 2 PSA [Ref-1] considers the beginning of the severe accident as the onset of core heat-up and the end of the in-vessel accident progression as vessel failure.

The probability of successfully arresting the core heat-up in-vessel, $P_{success}$, is the product of the probability to quench the core P_{quench} from a thermodynamic point of view multiplied by the probability to succeed in the quenching as obtained from experimental studies:

$$P_{success} = P_{quench} * P_{recovery}$$

where:

P_{quench} = probability that the amount of water brought to the degraded core is sufficient to remove the decay heat, the stored energy, the vaporisation energy and the oxidation energy when applicable at a given time t.

$P_{recovery}$ = conditional probability to quench the corium at a given time t, given sufficient water for heat removal.

The process of quenching the core begins at the time when primary depressurisation is initiated. The time that it takes to quench the core t_{quench}, is calculated using a spreadsheet analysis that uses a mass and energy balance to determine how long it will take to quench the core. This spreadsheet analysis uses a single LHSI pump as the source of injection, and uses the severe accident discharge valves (SADVs) as the mode of depressurisation. The analysis evaluates this energy balance over a range of times during each phase of the event, and calculates $P_{success}$ for each of these times.

- Phase 1: Core Heat-up to Core Melt Onset
  During this phase the core is in a coolable geometry, and the injection in-vessel would recover the core cooling in most cases. During this phase there is no molten core material. Once heat removal exceeds heat generation, the core will begin to cool and maintain a coolable geometry. The maximum quenching mission time is considered to be 24 hours.

  - If the calculated time to quench the core is less than 24 hours, then $P_{recovery}$ = 1, otherwise $P_{recovery}$ = 0.

  - In all cases P_{quench} is the value of the average of the values of P_{quench} at the end of the depressurisation and the end of quench

- Phase 2: Core Melt Onset to Relocation into the Lower Head of the Vessel
  During this phase, the corium is above the support plate. Water is assumed to be available in the lower plenum but not in contact with the hot material. .

  - During this phase core geometry changes may continue while the core material is molten. If heat removal exceeds heat input during this phase, the time to relocation could be extended. However, the extension of this time is conservatively ignored and a limiting time is calculated as the time from depressurisation to the end of the phase.

  - If the time needed to quench the core is less than the time to the end of Phase 2, then $P_{recovery}$ = 1 and P_{quench} is the average of the values of P_{quench} at the end of the depressurisation and at the end of the quench.

  - If the calculated time needed to quench the core is greater than the time to end of Phase 2 but less than 24 hours, then $P_{recovery}$ = 1 and P_{quench} takes an average value between reference P_{quench} at the end of the depressurisation and at the end of quenching, with a minimum value of 0.1. If the calculated time needed to quench is larger than 24 hour, then $P_{recovery}$ = 0 and P_{quench} = 0.1.

- Phase 3: Relocation into the Lower Head of the Vessel to Vessel Failure
  At the start of this phase, the corium will fall into the water, which experiences a boiling off phase. This event depends on the amount of water present in the lower plenum. If hot material is quenched by the water in the lower plenum, the probability of successfully restoring core cooling, based on the injection in-vessel at this time and until the corium reheats, is 1. After boil off, the corium will again eventually melt and the same evaluation as in Phase 2 is performed, except that the oxidation rate of the Zr is neglected, and the water required to refill the core is reduced. The presence of a molten pool at the bottom of the vessel will increase the probability of failure to recover the core.

3.3.8. Phenomena at Vessel Failure

3.3.8.1. Introduction

The phenomenological assessment [Ref-1] performed considered the following phenomena at vessel failure:

- Overpressurisation of the reactor pit due to release of gases from the vessel at vessel failure (high RCP [RCS] pressure).

- Rocketing of the vessel, due to reaction forces on the vessel when it fails at high RCP [RCS] pressure.
- Direct Containment Heating (DCH) due to entrainment of debris into the main containment volumes with concurrent rapid heat transfer from the debris to the containment atmosphere and generation and combustion of hydrogen following vessel failure at high pressure.

An additional consideration was to assess the likely failure modes of the vessel (and in particular the size of the failure) to the extent that these can impact downstream events in the CET, including those events assessed in this phenomenological assessment.

The events described above were considered for inclusion into the CET since they have the potential to lead to containment failure and an associated release of radionuclides, or otherwise impact the accident progression. The overpressurisation of the reactor pit may lead to damage that potentially affects the subsequent accident progression (i.e. retention, spreading and cooling of corium ex-vessel).

An outline of the phenomenology associated with each of the items introduced above is presented in the following sections:

3.3.8.2. Vessel Failure Modes

The different vessel failure modes that are considered to be possible following a core damage accident are:

1. An off-centre tear of the lower head.
2. A rupture of the lower head at its lowest point.
3. An ablation failure of the lower head due to jet impingement.
4. A complete circumferential failure of the lower head.

The first failure mode noted, an off-centre tear of the lower head, has been seen in the EU FOREVER experiments [Ref-1] and is anticipated due to high heat loads expected to result at the top of corium pools in the lower head. If the corium relocates to the lower head without a prompt jet-impingement failure (discussed later), high heat loads can arise at the top of the pool if (a) the melt constituents are well mixed and there is strong convection within the pool, or (b) the metallic and oxide phases separate when the corium is in the lower head, in which case the upper metal layer could lead to a “focusing” effect whereby the highest heat fluxes occur at the top of the melt pool.

The second failure mode noted, lower head rupture, could occur if the pool in the lower head forms a static, but mixed, configuration. In this case, the highest heat fluxes will occur at the base of the pool since there is a radiation heat removal mechanism at the pool surface. This pool configuration is generally considered much less likely than convective or stratified behaviour.

The third failure mode noted, ablation failure due to jet impingement, may occur as a result of a sideways relocation mode or a bottom failure of the crust in which a "jet" of molten debris is generated, leading to jet impingement and an ablation failure. Such a failure would be prompt, but localised. One mechanism by which this relocation mode could occur is a side breach of the debris crust layer which forms during the in-vessel melt progression, opening a path through the baffle (heavy reflector for the UK EPR) and allowing molten material to reach the lower head. A vertical pour with a jet is also possible; in this case, it is postulated that the crust failure occurs at the base, with a small opening, leading to a debris jet impinging on the lower head wall. Wall ablation is postulated to occur due to enhanced convective heating during the pour process. This failure mode is unlikely because of the narrow range of jet diameters over which it may be postulated.

The fourth failure mode noted, complete circumferential failure of the lower head, could be postulated if the vessel failure occurs at the top of a corium pool in the lower head, either in the convective mixing scenario or the stratified melt scenario. A circumferential failure might be postulated either (a) due to a situation with highly symmetric head loads and vessel wall strength, or (b) following a localised tear at the top of the pool which subsequently propagates (rapidly) around the lower head. This failure mode has not been observed experimentally, even though convective pools have been studied and the localised tear failure mode has been observed. It is considered of negligible probability if the vessel fails by jet impingement and ablation, since jet impingement is expected to lead to the smallest, most localised failure.

3.3.8.3. Overpressurisation of the Reactor Pit

This phenomenon may occur when the blowdown rate of the vessel exceeds the venting capability of the reactor pit at a relatively low pressure (i.e. gases from the failed RPV discharge rapidly into the pit and the flow paths out of the pit are not sufficiently large for the blowdown gases to exit the cavity without resulting in pressurisation). The pressurisation of the pit is expected to be more likely for larger failure sizes of the RPV, since this would imply a more rapid inflow of gases into the pit which is more likely to overwhelm the pressure relief capacity of flow paths out of the pit.

The potential consequences of overpressurisation of the reactor pit are expected to be structural damage. The structural damage potentially resulting is expected to be more likely to result in an impact on downstream nodes in the containment event tree than to result in direct containment failure. A possible example of a downstream impact would be the impact on severe accident melt stabilisation.

3.3.8.4. Rocketing of the Vessel

Rocketing of the vessel was originally proposed as a failure mechanism for the containment in the WASH-1400 study. Rocketing would be credible if, at the time of vessel failure, upward forces on the vessel exceed the hold-down capability of vessel supports by a margin sufficiently great so as to cause transfer of enough energy to the vessel such that it becomes an energetic missile able to fail the containment.

3.3.8.5. Direct Containment Heating

The postulated sequence of events for direct containment heating (DCH) includes:

- The RPV fails at high pressure.
- Molten core material (UO2 and zircaloy) and molten steel are forced out of the vessel at high pressure and this material becomes highly fragmented into small particles.

- There is therefore a large surface area for interactions and energy exchange with the containment atmosphere.

- Heat from the fragmented debris is transferred to the containment atmosphere, pre-existing hydrogen burns and more hydrogen is generated and burns due to the chemical reactions of zircaloy and steel with steam in the containment.

- The resultant energy input into the containment atmosphere results in a rapid pressure increase, and possible containment failure.

More recent experimental and modelling investigations [Ref-1] have tended to result in lower estimates of the peak pressures from DCH than earlier evaluations. The main reasons have been the mitigating influence of lower containment compartments where debris may be retained and limitations on the interaction zone inside the containment for heat exchange and chemical reactions. The NUREG/CR-6338 study [Ref-2] presents a resolution of the DCH issue for large dry containment design U.S. PWRs. While resolution is formally stated as meaning merely that the CCFP given a core damage accident is less than 0.1, the results of this study strongly suggest very large margins between the containment strengths and the potential loads from DCH. This implies that, from a Level 2 PSA perspective, containment failure probabilities from DCH could be relatively small.

3.3.9. Probabilistic Evaluation of Vessel Failure

3.3.9.1. Vessel Failure Modes

The probabilistic evaluation of vessel failure modes was performed by developing a decomposition event tree (DET) containing the following headers:

- Location of crust breach - side or base: This considers two mechanisms of melt relocation:

    - A side jet/pour where the breaching of the debris crust layer which forms during the in-vessel melt progression occurs at the side, and a path opens through the heavy reflector for the UK EPR;

    - A vertical jet/pour, in which it is postulated that the crust failure occurs at the base. The first mechanism was evaluated as the more probable of the two mechanisms.

- Prompt vessel wall failure by jet impingement: This considers jet impingement on the vessel wall which could result in enhanced heat transfer from the jet to the wall location and thus in rapid wall ablation and localised prompt failure. Based on a review of recent investigations (SKI Report 01:23 [Ref-1] and 00:53 [Ref-2]), this vessel failure mode was evaluated as an unlikely scenario. It was also noted that in the case of a base crust penetration, the melt will either fall into water (leading to possible break-up of the jet) or if not, the jet will eventually be submerged in the melt pool which accumulates in the lower plenum. Thus, prolonged direct contact of the jet and the wall is more likely in cases where a side failure of the crust occurs under the preceding header leading to a reduction in the assigned probability for a base location of crust breach.

- Pool state: This considers which of the following classes of pool would be expected to form in the lower head following relocation:

- Phase separation and metal layer with focusing of heat towards the top of the pool
- Fully mixed convective pool, leading to higher heat loads at the top of the pool due to convective flows.
- A fully mixed static pool, with highest heat loads at the base of the vessel. Of the three configurations, the fully mixed static pool was assigned the lowest probability, implying that it was judged to be more likely that the highest heat loads would be at the top of the pool.

- Vessel failure: This considers the mode of wall failure and breach area. Specifically, the following failure modes and characteristics were addressed:
    - “Small base" or "Small base/side", local failure modes due to jet impingement and ablation of the wall (the base/side variant was used for the case that the jet impingement results from a sideways relocation);
    - “Base", a localised failure due to the formation of a fully mixed static pool, expected at the bottom centre of the lower head, and assigned a probability of 1.0 conditional on the formation of a fully mixed static pool;
    - “Side tear”, a failure mode where the initial wall breach is near the top of a relocated debris bed, but where it is not postulated that the entire circumference of the wall fails simultaneously;
    - Complete vessel breach (CBV), a rapid gross cross-sectional failure of the lower head, which applies only to convective pool or separated phase situations, and for which creep strain is postulated to be exactly equal all around the vessel wall. When failure is postulated to occur, the entire vessel head is instantaneously detached (this failure mode is considered unlikely since the expected presence of non-uniformities in the melt, and also possibly the wall material, would favour an initial localised failure, as seen experimentally).

The outcomes of the DET were classified according to failure mode of the RPV, resulting in the following overall outcomes:

Failure Diameter	Failure Mode	Probability of Failure Mode (given that a failure occurs)
0.1m	Small base, Small base/side	0.04
0.1m – 0.5m	Base	0.048
0.5m – 1.0m	Side tear	0.902
4.87m	CBV	0.010

3.3.9.2. Direct Containment Heating

The probabilistic evaluation of DCH consisted of the development of a model for the DCH pressure rise, based on the NUREG/CR-6338 TCE model [Ref-1] together with the use of dispersion factors based on experimental information [Ref-2], to model the specific dispersion properties of the EPR reactor pit. A Monte Carlo simulation was used to generate a probability distribution representing the uncertainty on the DCH pressure rise [Ref-3]. This probability distribution was compared to the EPR containment fragility curve to generate an overall probability of failure of the containment by DCH, given a high pressure vessel failure.

The adaptation of the NUREG/CR-6338 DCH loads was based on the pressure rises predicted by the NUREG model compared to the initial or baseline pressure conditions. Initial pressure conditions for the phenomenological analysis of DCH for the EPR were taken from U.S. EPR MAAP analyses [Ref-4], to ensure EPR specific initial conditions.

The other parameters accounted for in calculating the DCH pressure rise for the EPR were:

- Dispersion.
- Zircaloy mass (total in core).
- Steel mass in lower plenum at vessel failure.
- UO2 Mass (total in core).
- Coherence Multiplier.
- Containment Volume.

The above parameters were chosen because they are the main parameters that varied between the different plants and were also judged qualitatively to be those most likely to significantly influence the DCH loads.

The probabilistic evaluation of DCH concluded that the probability of containment failure, given a DCH event with the vessel failing at high pressure, is 5.5E-4.

3.3.9.3. Cavity Overpressure

The probabilistic evaluation of a cavity overpressure is centred on the comparison of potential loads on the cavity for a range of vessel failure sizes with the structural capacity of the cavity. The loads (overpressure) were estimated using a series of MAAP runs for the vessel failure sizes evaluated in the vessel failure modes DET described above.

Based on the above analyses, and an assessment of the pressure capability of the cavity, cavity overpressure following a high pressure vessel failure was evaluated as possible, with a conditional probability of 0.02, for the case of a high pressure vessel failure resulting in a complete breach of the vessel (CBV). However, the analysis of vessel failure modes in section 3.3.9.1 indicated that the probability of the CBV failure mode was low (0.01), leading to an overall probability (given a vessel failure) of 2E-4 when multiplied by the probability of a CBV occurring. The expected point of failure was assessed to be the melt plug (gate). However, it should be noted that a containment failure due to vessel rocketing would be expected for the CBV failure mode. Cavity failure was also assessed as having a small probability of occurrence of 2.3E-6 in the case of the largest side tear failure of 1m equivalent diameter (as assessed in section 3.3.9.1).

3.3.9.4. Vessel Rocketing

Rocketing of the vessel was assessed by use of the so-called “Rocket equation” [Ref-1] which evaluates the total rocketing upward force as the sum of a momentum term (due to the exiting flow) and a pressure term (due to the net upwards pressure on the vessel with a hole in the lower part of the vessel.

Based on this assessment, together with an assessment of the total hold-down force on the vessel (due to the cold legs), rocketing was discounted for small hole sizes (0.1 m and 0.5 m diameter breaches) on the basis that the restraining forces exceed the maximum possible rocket thrust force in these cases. In the case of a 1 m hole size, it was also seen that the rocketing forces would not exceed the hold-down forces, although the calculated margin was lower in this case; it is noted that the location of the 1m diameter (side tear) failure precludes rocketing in any case, since forces would be sideways not upwards. For the complete circumferential rupture of the vessel (CBV case), which is assessed as an unlikely failure mode, with a probability of 0.01 in high pressure sequences, rocketing is expected, as the restraining forces are exceeded by nearly an order of magnitude. The CET models assume containment failure in this case.

3.3.10. Hydrogen Phenomena

A deflagration is a combustion form in which the combustion front travels at sub-sonic speed relative to the unburned gas. If the flame speed is small compared to the speed of sound, the pressure rise is expected to be uniform throughout the containment volume and the loads will be quasi-static in character. Loadings from deflagration can be estimated by (1) assessing the heat input to the containment atmosphere arising from combustion (based on heats of reaction) and (2) evaluating the final peak pressure of the mixture at the resulting gas temperature, based on the thermal properties of the constituent gases and the heat input. When this calculation is based on assumptions of complete combustion of all reacting gases and no heat losses to structures (etc), it is referred to as an Adiabatic Isochoric Complete Combustion (AICC) calculation. Codes such as MAAP and MELCOR [Ref-1] also include models where losses are taken into account and deflagrations are allowed to propagate through different volumes in the containment, tending to lead to lower calculated pressure rises than those arising from the AICC method, which can be seen as an upper bound for deflagrations.

Detonation is a form of combustion where the flame travels at supersonic speed (≈2000 m/s) relative to the unburned gas. In this case, a shock wave is formed, and, depending on the time constants of the containment structure and the detonation pulse, the structural load is determined either by the peak pressure or the impulse of the detonation pressure wave, or by a combination of these two items.

The peak pressure from a detonation is expected to be in the range of 12 to 20 times the base containment pressure [Ref-2]. This implies high containment failure probabilities given the occurrence of a detonation. The effective pressure (i.e. the static pressure that would give a load equivalent to the dynamic detonation load) due a deflagration-to-detonation transition (DDT) is in the region of 1.5 to 2 times the pressure that would arise from a slow deflagration. Nuclear power plant (NPP) containment structural response natural frequencies are in the range 5-25 (or 5-50) Hz (i.e. characteristic times of 20-200 ms), with the effective pressure factor quoted above appropriate for this range.

An accelerated flame can also lead to structural loads which have time scales shorter than the structural response time and therefore correspond to higher effective pressures. In the range of NPP containment structural response frequencies, the effective pressure from an accelerated flame is in the region of 1.5 to 2 times the pressure that would arise from a slow deflagration (i.e. a similar ratio to that obtained for the case of a DDT). Flame acceleration is essentially a pre-condition for DDT since direct initiation of a detonation is considered very unlikely. Occurrence of an accelerated flame, followed by DDT is a more likely scenario in a nuclear power plant containment.

Based on the above discussion, it can be seen that deflagration, flame acceleration and DDT should all be considered as potentially unfavourable loadings for the containment of an NPP during a severe accident. This is different to the historical position regarding destructive failure modes, where, in the past, only DDT was considered a potential containment challenge. Recent references are however clear that the loads from fast flames may approach or even exceed those from DDT.

3.3.11. Probabilistic Evaluation of Hydrogen Phenomena

The phenomenological assessments performed for containment loads derived from hydrogen combustion processes addressed containment failure due to overpressure from hydrogen deflagration or because of dynamic loads from “destructive” combustion modes (flame acceleration or deflagration-to-detonation transition, DDT).

3.3.11.1. Deflagrations

The deflagration assessment was performed on a global basis, based on the global AICC pressure. The main parameters considered in the assessment were as follows:

- In-vessel hydrogen production.
- Ex-vessel hydrogen production.
- Steam concentration.

Consumption of hydrogen and oxygen by recombiners was accounted for by reference to the MAAP analyses performed. Consumption of hydrogen by random hydrogen burns at lower concentrations was, conservatively, ignored. In-vessel hydrogen production was assessed as being in the range 48 percent to 82 percent equivalent zircaloy oxidation [Ref-1].

This assessment of deflagrations in the U.S. EPR containment identified two scenarios as having non-zero probabilities of containment failure:

- Deflagration during the in-vessel phase of a high pressure core damage transient, resulting in a probability of containment failure of 2.0E-06.
- Deflagration during the in-vessel phase of a high pressure core damage transient following a hot leg rupture and the consequent release of hydrogen into the containment. The resulting probability of containment failure is 1.38E-04.

The above results were based on bounding assessments in terms of hydrogen and steam conditions (i.e. top of range hydrogen concentrations and steam concentrations close to inert conditions).

The probability of hydrogen deflagration leading to containment failure at the time of vessel failure was dismissed as being of negligible probability, as was the probability of a long-term hydrogen deflagration causing containment failure. The arguments presented in reaching this conclusion for long-term hydrogen deflagration include a justification for not expecting oxygen leakage back into containment (and resultant de-inerting of the containment atmosphere) to occur.

3.3.11.2. Destructive Combustion Modes

An analysis of potential local concentrations was carried out for a range of scenarios. Containment nodes and time periods of potential susceptibility to flame acceleration were identified and assessed based on MAAP analyses for these scenarios. This required the assessment of the mixture property histories for all 27 MAAP nodes for 26 MAAP analysis cases. For each node, a limiting hydrogen concentration for flame acceleration was dynamically calculated (as a function of oxygen and steam concentrations) and compared to the calculated hydrogen concentration histories. The limits used were based on the recent OECD/NEA State-of-the-art report on hydrogen [Ref-1].

A number of nodes were identified as presenting mixture properties that were susceptible to flame acceleration for short periods during the scenarios analysed. These nodes and time frames were grouped into the scenarios (cases) listed below, together with the assessed probabilities that flame acceleration would cause local damage or global containment failure:

- Case 1. Transients at high pressure, in-vessel phase, period of discharge from RCP [RCS] via pressuriser valves:
    - o Assessed probability of local damage in lower equipment rooms or middle equipment rooms (MAAP nodes 3 and 5) = 0.016.
    - o Assessed probability of containment failure due to flame acceleration loads = 0.016.
- Case 2. Transients at high pressure at approximately the time of Induced Hot Leg Rupture:
    - o Assessed probability of local damage in middle equipment rooms (level 2 to 4) or upper equipment rooms (level 2 to 4) (MAAP nodes 6 and 10) = 0.00125.
    - o Assessed probability of containment failure due to flame acceleration loads = 0.00125.
- Case 3. Transients at high pressure, at approximately the time of vessel failure:
    - o Assessed probability of local damage in middle equipment rooms (level 2 to 4), upper equipment rooms (level 2 to 4), Level 1 upper equipment rooms, or staircase south (MAAP nodes 6, 10, 7, 23) = 0.0056.
    - o Assessed probability of containment failure due to flame acceleration loads = 0.0056.
- Case 4a. Low pressure scenarios with short term fast MCCI following vessel failure:
    - o Assessed probability of containment failure due to flame acceleration loads = 0.00045.
- Case 4b. Scenarios without recombiner damage/impairment, ongoing long-term MCCI (dry spreading area):
    - o Assessed probability of containment failure due to flame acceleration loads = 0.0001.
- Case 4c. Similar to Case 4b but with damaged recombiners (75 percent efficiency):
    - o Assessed probability of containment failure due to flame acceleration loads = 0.0005.

Where a destructive combustion mode was assessed to occur without leading to containment failure, the possibility of localised damage to recombiners was considered. This implies loss of some recombiners in Case 1, Case 2 and Case 3. Cases 4a to 4c have no local consequences, since global containment failure was assessed to have a probability of 1.0, given the occurrence of an accelerated flame, making local consequences irrelevant.

3.3.12. Long Term Containment Challenge Mechanisms

This evaluation [Ref-1] deals with potential long-term challenges to the containment integrity, starting at the time of core debris arrival in the spreading area. The important phenomena include containment pressurisation due to steaming during quench, or in the longer term, containment pressurisation due to the absence of heat removal, and molten core concrete interactions.

This evaluation identifies and decomposes the identified phenomena, using the results of the analyses performed using MAAP4.07. The MAAP4.07 analyses modelled the UK EPR core melt retention device and the EVU [CHRS], because these systems are key to the maintenance of containment integrity in the long term.

The U.K. EPR melt stabilisation process involves the following phases:

- In-vessel melt progression and release from the RPV
- Temporary retention and accumulation of the melt in the reactor cavity with a subsequent failure of the cavity retention gate.
- Melt spreading and distribution.
- Flooding, quenching and long term cooling of melt in the lateral spreading compartment.
- Containment heat removal

The following challenge mechanisms are identified based on review of the melt stabilisation process:

- Melt quench in the core spreading area.
- Incomplete transfer of core debris to the spreading area.
- Failure of passive flooding and molten core concrete interaction.
- MCCI after passive flooding.
- Damage to the reactor pit.
- Containment pressurisation.

These mechanisms have been organised into the DET shown in Sub-section 15.4.3.3 - Figure 2 Decomposition Event Tree for Long Term Challenges. This tree provides the framework for performing the probabilistic evaluation described below.

3.3.13. Probabilistic Evaluation of Long Term Containment Challenges

The probabilistic evaluation of the long term challenges consists of the quantification of the failure probability expected due to the failure mechanisms listed in the DET. The DET headers that are quantified elsewhere in the Level 2 PSA study and are not included in this discussion are:

- Success / failure of passive flooding (essentially a passive system analysis – covered in systems analysis models).
- EVU [CHRS] spray availability (covered by system analysis and HRA).
- Active cooling availability (covered by system analysis and HRA).

The remaining DET headers are discussed below.

3.3.13.1. DET Header: No Containment Overpressure Failure due to Debris Quench

The following are considered as key uncertain parameters for the containment overpressure analysis:

- The fraction of the core debris which is quenched, fq.
- The pressure increase in the containment per fraction of debris quenched, P.
- The base (initial) containment pressure at the time of debris flooding, Pco.

The peak containment pressure resulting from corium quench is determined by the formula:

Pc_{peak} = Pco + fq * P

This pressure is compared with the fragility curve developed in the Containment Fragility analysis (section 3.3.14), and the CCFP is calculated using Monte Carlo simulation analysis.

For the fraction of core debris quenched (fq), the MAAP4.07 model uses a distribution describing the fraction of the debris quenched assuming heat transfer is limited by heat conduction through a solid crust. This distribution has a median at 10 percent and lower and upper bounds at 0 and 80 percent, respectively. This treatment assumes that crack formation and water ingress during quench is impossible. While it may be likely that a stable crust will form, at least initially, it is not considered impossible that crust cracking could occur during quenching. A modified distribution has been developed using the following assumptions:

- A likely situation is that a stable crust will form and heat transfer will be limited by the conduction rate. In the distribution, a probability of 0.45 is assigned for quenching between 8 and 12 percent of the debris.
- Another likely configuration would be debris cracking and water ingress during debris quench, resulting in a critical heat flux limited heat transfer rate, which could allow quenching of close to 100 percent of the debris. In the distribution, a probability of 0.45 is assigned for quenching between 96 and 100 percent of the debris.
- All other physical situations of crust and water interaction are assumed to be equally likely. A uniform distribution with a total probability of 0.1 is assigned to these.

For the probabilistic analysis of the pressure increase during the quench, in order to avoid potential non-conservatisms, the distribution for containment pressure rise for 100% debris quenched, is developed based on MAAP results with fixed values of FCHF (the flat plate critical heat flux (CHF) Kutateladze number) for the LLOCA sequence. The basis for this distribution is:

- Most likely value (from FCHF=0.1 case): 3.7 bar pressure increase.

- Upper bound (from FCHF=1.0 case): 4.3 bar pressure increase
- Distribution type: symmetric triangular. The triangular distribution is chosen because the value FCHF = 1.0 is seen as very unlikely and it is desirable to give a greater weight to the median value
- The same distribution is used for all CDES since this value is not expected to be dependent on the initiator.

The following values are chosen for the base pressures (Pco), in each CDES, with a uniform distribution between the upper and lower bounds

CDES	Expected Value (bar abs)	Expected Value (bar g)	Upper and lower bounds (bar)
TP/TR:	3.1	2.1	± 0.5
PL:	2.3	1.3	± 0.5
SL / ML / SS / LL	1.9	0.9	± 0.5

The results of the Monte Carlo simulation using 1 million samples show a conditional probability of containment failure of 0.0 for CDES PL, SL, ML, SS, LL, and 3E-6 for CDES TP/TR.

3.3.13.2. DET Header: No Significant MCCI

This header is evaluated only if passive flooding succeeds. If passive flooding fails, a significant MCCI is assumed to occur. When passive flooding succeeds, the potential for MCCI beneath flooded debris is judged to be of very low probability based on AREVA-NP studies of melt spreading and corium heat transfer in connection with the design of melt stabilisation measures in the reactor containment. Conservatively, the conditional probability for failure at this node is assumed to be 1.0E-3, based on engineering judgment.

3.3.13.3. DET Header: No Containment Overpressure Failure before Basemat Penetration

This header is only evaluated for the case of significant MCCI in a dry spreading area with sprays unavailable. Currently it is assumed that overpressure failure does not occur for MCCI in a flooded spreading area. Results from analysis of the containment pressurisation rate during MCCI show a rate of approximately 1 bar in 40 hours, or 0.025 bar/h. At 60 hours, the pressure is approx. 4 bar (abs). Thus to reach the median failure pressure of 11.6 bar (g), or 12.6 bar (abs), would take approximately

(12.6 – 4.0) / 0.025 + 60 = 404 hours, or about 17 days.

The rate of ablation in the spreading area is approx. 0.5 m in 30 hours [Ref-1], or 0.017 m/hr. The thickness of the basemat below the spreading area is taken from the containment general arrangement drawing and is approximately 4.4 m. The time to penetrate the basemat is therefore, approximately:

(4.4 – 1.5) / 0.017 + 60 = 230 hours = 9.5 days

Although approximate, this calculation indicates that the first failure mode to occur due to sustained MCCI would be basemat penetration. If it is further assumed that penetration of the basemat would prevent further pressure increase, then the probability of overpressure failure should be taken as a low value.

Based on the above discussion, in cases where there is ongoing MCCI, basemat melt through is expected first. Therefore, containment overpressure is judged as very unlikely and assigned a probability of 0.01.

3.3.13.4. DET Header: No Basemat Penetration

This header is evaluated for significant MCCI where sprays are available, and where sprays are not available but overpressure failure does not occur. Theoretically, due to the large spreading area, the possibility exists that even a dry core debris bed may cool sufficiently for MCCI to be arrested before the basemat was penetrated. Physically, this is possible if heat generated in the melt can be conducted away into the concrete with a delta-T below that required to sustain the concrete decomposition temperature. Success at this header precludes containment overpressure as well, so that if MCCI did arrest then this would also preclude the overpressure failure due to generation of non-condensables. Therefore, end states with success of this header are classified as "no failure".

However, considering the ablation area and the debris temperatures during MCCI, and considering the values calculated previously, the split fraction is assigned a success conditional probability of 0.01 (failure conditional probability of 0.99).

3.3.13.5. DET Header: Containment Overpressure Failure due to Incomplete Melt Transfer

For cases with passive flooding and active cooling started later, should any debris be still present in the reactor pit or transfer tube, there is the possibility that the water in these regions would not be cooled by the EVU [CHRS] and that boiling and steam pressurisation could occur. Numerous design features of the debris stabilisation system make this possibility unlikely. In particular, the concept of the melt plug arrangement itself and the composition of the sacrificial concrete are chosen to condition the core debris/concrete melt mixture properties such that a complete transfer of core debris to the spreading area is assured. There is little data regarding this potential failure mode. Nonetheless, a conditional probability of 1E-2 for failure has been estimated.

During high pressure CDES sequences, there is a high likelihood that Hot Leg Rupture will result in flooding of the reactor pit. Upon vessel failure, there is the possibility that part of the debris will quench and remain in the pit while the remainder of the debris transfers to the spreading area. In this case, no matter what the status of EVU [CHRS], there is a risk of overpressurisation of the containment because of boil-off of the water in the pit. Containment overpressure could occur because the pit is not in the main cooling circuit of the EVU [CHRS] and is maintained at the same level as the spread area / IRWST, thus the pit is constantly replenished.

The coolability of the corium in the pit is highly uncertain, because the debris will form a very deep pool which is not likely to be coolable. Due to this high uncertainty a split fraction of 0.5 is assigned.

3.3.13.6. Summary – Long Term Challenges

The results of the long term challenge evaluation are summarised in Sub-section 15.4.3.3 - Table 1, Summary of Long Term Challenges Probabilistic Evaluation.

3.3.14. Containment Fragility Evaluation

The Level 2 PSA study identifies, evaluates and quantifies loads on the containment structure that can occur as a result of a severe accident. In order to assess the probability that a given load will result in failure of the containment structure (also part of the Level 2 PSA study [Ref-1]), knowledge of the capacity of the structure to withstand loads is needed. Most containment structures are conservatively designed, and when their capacity is assessed realistically, they are found to have considerable margin above design conditions. It is, for example, often found (even on existing plants) that a containment structure can withstand around two times its design internal pressure before failure would be expected to occur.

This capacity information is generally used in the form of a composite fragility curve, which shows the probability of failure at less than or equal to a pressure p, as a function of p. Thus it is a cumulative distribution function, differentiation of which leads to the probability density function. It is important to note that, unlike in design space, a PSA uses best estimate approaches, with consideration of the uncertainties. Thus the median of the fragility distribution represents the best estimate failure pressure, while the uncertainties around this value are represented by the probability distribution. It is also important to realistically characterise any failures, particularly by selecting justified failure modes, and expected leak or rupture areas. These are used in the source term calculations.

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The fragility curve is used to estimate containment failure probability given certain loads. The loads are determined (for different phenomena and for different classes of sequences) in the Level 2 PSA phenomenological evaluations and uncertainties in the loads are considered by representing the loads as probability density functions. More details of the analyses carried out for each phenomenological event are given in the sections above.

3.3.15. Equipment Survivability Evaluation

This evaluation addresses the survivability of equipment credited in the CET models under severe accident conditions. During the severe accident, conditions of high temperature, humidity, pressure and radiation are expected inside the containment. Systems that are inside the containment will be exposed to these conditions. There is also the possibility that containment failure could affect the continued operation of systems used for source term mitigation. This may be dependent on the location of containment failure; containment failure at a particular location could have the potential (dependent on the containment failure modes and plant geometry) to cause release of hot gases into equipment rooms.

Since the CET model may include the actuation or continued operation of such systems, it is necessary to assess the likelihood that the systems will operate or continue to operate under these conditions.

The following functions have been identified as requiring evaluation for qualification during severe accident conditions:

- Reactor Coolant System RCP [RCS] depressurisation.
- Hydrogen mitigation.
- Melt stabilisation.
- Containment heat removal.
- Monitoring activity distribution within the containment and potential releases to the environment.

The review of equipment survivability is documented in Sub-section 15.4.3.3 - Table 3, Evaluation of Equipment Survivability for Level 2 PSA.

The following headers in the CET were also reviewed, but are not relevant for equipment survivability:

- No induced hot leg rupture.
- RCP [RCS] pressure remains high in small LOCA sequences.
- No reactor pit damage due to lower head failure following an in-vessel steam explosion.
- Reactor pit not damaged by ex-vessel steam explosion.

The review of the CET and assessment of equipment credited in light of plans for equipment qualification for severe accidents has concluded that, with the exception of the hydrogen recombiners, none of the equipment credited in the CET models should be considered affected by the severe accident conditions expected to occur during the progression through the Level 2 PSA CET. Consequential damage to the recombiners due to accelerated flame phenomena is considered in the CET model.

3.3.16. Human Error Probability Evaluation

3.3.16.1. Issues Related to Classical HRA Methodologies

A number of features of severe accident management create difficulties when modelling human actions using "classic" HRA methodologies. In particular, the following are noted:

- Guidance versus Procedures:

  Severe accident management packages are usually guidance, and not procedures. There is no requirement that the Technical Support Centre (TSC) and operators follow verbatim the "instructions", as is the case in Emergency Procedures. The guidance users may decide to deviate from the guidance, depending on their evaluation of the situation. They may decide not to perform a recommended task because they evaluate the potential negative consequences as too severe, or they may choose to adopt a different mitigation strategy from that proposed in the guidance. This evaluation process and its potential outcomes are difficult to model using "classic" HRA techniques

- Emergency Organisation:

  In EOP space, the evaluation of plant conditions, selection of appropriate strategies, and implementation of those strategies is the responsibility of the reactor emergency response team. In severe accident management space, it is normal that responsibilities are assigned to different parts of the emergency organisation. In particular, the TSC is typically much more involved in determining which strategies are recommended, whereas the operations team remains responsible for implementing the strategies. While this may help by reducing the importance of human error dependencies, it raises issues such as the modelling of communication between the different teams involved.

- Negative Impacts:

  As noted above, strategies taken during implementation of the severe accident management guidance may have negative or positive impacts on the situation. For example, quenching and cooling core debris may lead to significant steam pressurisation of the containment. The TSC will evaluate the expected severity of potential negative impacts and may change or modify the recommended strategies, or disregard them.

3.3.16.2. UK EPR PSA2 Human Reliability Analysis Methodology

In order to accommodate the issues identified above, the UK EPR PSA2 developed an approach to evaluate the Human Error Probabilities (HEPs) for severe accident management strategies [Ref-1] based on a well known and validated approach known as SPAR-H [Ref-2]. The SPAR-H approach was adapted to the particularities of the severe accident guidance (OSSA), to the severe accident emergency organisation, and to the evaluation of negative and positive impacts that is required in the case of a severe accident.

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3.3.17. Supporting Severe Accident (MAAP) Analysis

Deterministic severe accident analysis – the simulation of the progression of a severe accident sequence – is a key input to a Level 2 PSA in two areas:

- To assist in developing the containment event tree and understanding the most likely event progression for the important sequences within a damage state bin;
- To assist in quantifying the containment event tree by aiding in understanding the important phenomena and resulting loads on containment resulting from the severe accident. EPR The analysis has been performed for accident sequences fulfilling one or more of the following criteria:
- The sequence is representative of a dominant sequence in each Core Damage End State (CDES)
- The sequence involves an initiator which would not otherwise be included
- Analysis of the sequence is needed to support phenomenological evaluation(s)
- The analysis is needed to verify that Level 1 PSA core damage sequences do actually involve core damage (as a check on accident sequence analysis)
- The analysis was specifically requested by one of the Level 2 PSA team (to resolve a specific issue).

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3.3.18. Phenomena in Accidents Occurring from Shutdown States

3.3.18.1. Introduction

This section discusses how the phenomenological evaluations which were performed for full power (states A and B) were adapted for use in shutdown states C, D and E. The following evaluations were reviewed and, where deemed necessary, adapted for use in the Level 2 PSA for shutdown:

- Containment fragility
- Induced RCP [RCS] rupture
- Fuel coolant interactions
- In-vessel recovery
- Loads at vessel failure

- Hydrogen
- Long term challenges
- Equipment survivability
- Other events included in the Level 2 PSA model
- Shutdown state Ca was subdivided into sub-states Ca1, Ca2, Ca3 and Ca4 for the Level 2 PSA in order to distinguish the possible combinations of RCP [RCS] status and containment status that may arise, but also the different times of sub-states for which the equipment hatch could be open. These sub-states are summarised below: Sub-state Ca1 of state C: the RCP [RCS] and containment are both closed in this sub-state.
- Sub-state Ca2 of state C: the RCP [RCS] is closed in this sub-state and the containment is partially open (35% of the sub-state).
- Sub-state Ca3 of state C: the RCP [RCS] is closed and the containment is partially open (40% of the sub-state).
- Sub-state Ca4 of state C: the RCP [RCS] is closed and the containment is closed.
- Sub-state Cb of state C: the RCP [RCS] is partially open and the containment is partially open (12% of the sub-state).
- State D: the RCP [RCS] is open in this state and the containment is assumed to be closed for all sequences in state D for a more realistic evaluation.
- State E: the RCP [RCS] is open with the reactor pit flooded and the containment is partially open (80% of the sub-state). For model simplification we conservatively assume that the containment is open. No containment reclosure is modelled.

Each of the phenomenological evaluation reviews is discussed in the subsequent sections taking account of the above states and sub-states.

3.3.18.2. Containment fragility

The containment fragility curve used for shutdown conditions is the same as the curve used for full power conditions. The only independent parameter considered in the fragility evaluation performed to support the full power PSA was temperature. The temperature of 170 deg C assumed for the full power evaluation is considered to be adequately bounding for shutdown conditions. Due to lower (or similar) decay heat levels during shutdown, it is expected that the containment ambient temperatures at the time of key challenges to the containment would not exceed those arising during accidents initiated from full power conditions.

3.3.18.3. Induced RCP [RCS] rupture

Two modes of induced RCP [RCS] failure are considered in the full power Level 2 PSA. These are: (i) induced steam generator tube rupture, and (ii) induced hot leg rupture. These induced rupture modes are both due to creep rupture, which is a temperature and pressure dependent phenomenon.

Induced rupture is not possible in states Cb, D and E, since the RCP [RCS] is open and the pressure cannot therefore increase to levels at which this phenomenon is possible. An evaluation was performed for states Ca in which the RCP [RCS] is closed with a pressure setpoint of 5.6 MPa. Under these conditions it was concluded that hot leg rupture would not be considered in the containment event tree but that induced SGTR would be retained with a much reduced probability.

The rationale for the above conclusions was as follows:

- Pressure effect: a review of the Larson-Miller curves for creep rupture concluded that times to rupture would be greatly extended in shutdown transients (the curves shift by three orders of magnitude on the time axis at the reduced RCP [RCS] pressure). The transients from full power were not formally evaluated for shutdown conditions, but it was qualitatively concluded that induced primary ruptures would be less likely at the reduced pressure available in shutdown states. Nevertheless, this was not judged to be a sufficient basis for ruling out induced rupture by itself.

- Temperature effect: it was also noted that natural circulation flows in the RCP [RCS] are a strong function of pressure (reducing with lower pressure) and this would lead to lower temperatures as well as lower pressures. Calculations were performed to estimate the magnitude of this effect.

It was considered that the combined effect of the lower expected temperatures and pressures were sufficient to justify the removal of induced hot leg rupture from the CET models for states Ca. Since hot leg rupture is a beneficial failure (leading to RCP [RCS] depressurisation) it may be slightly conservative to completely remove this failure from the models, but it is not expected that the PSA results would be very sensitive to this as the EPR containment is very resilient to challenges arising from high pressure core damage sequences.

The induced SGTR is not a beneficial failure, and this was therefore retained in the models, but with a reduced probability to account for the much lower likelihood of tube-threatening temperatures and pressures arising. This is potentially conservative, but it is not expected to distort the PSA results significantly as ISGTR is seen to be a small contributor.

3.3.18.4. Fuel-coolant interactions

As discussed in the following paragraphs, the results of the phenomenological evaluations for fuel-coolant interactions are applied to shutdown states without modification.

<u>In-vessel steam explosions which fail the containment</u>

For the case of in-vessel steam explosions which fail the containment, the original assessments for full power conditions are considered to be bounding or close to the values which would be obtained by formally adjusting the evaluations to shutdown conditions. Considering each of the parameters involved in the probabilistic evaluation of in-vessel steam explosions leads to the conclusions listed below and justifies the use of unchanged probabilities for these events:

- The total mass of the core and the strength of the upper head are unchanged for shutdown rather than full power initial conditions.

- The thermal energy stored in the core materials per unit mass of core at the time of relocation is not expected to vary significantly. The thermal energy stored is largely determined by the mechanical characteristics of the core and the melting points of the constituent components.

- The fraction of the core material in the lower head that participates in pre-mixing is expected to be unaffected, since the relocation is expected to depend more on the core physical characteristics than the power level (relocation timing may be affected but the nature of the relocation is expected to be similar). If there were any variation in this parameter for shutdown rather than full power initial conditions it seems reasonable to expect that it would be conservative to assume the full power distribution for this parameter, since lower decay heat during shutdown seems more likely to lead to a slower (less massive) relocation if there is any variation.

- The only identified impact on conversion ratio for thermal to mechanical energy is pressure. For low pressure accident sequences the pressure is identical in shutdown compared to full power. High pressure scenarios for shutdown initial conditions would be at a lower pressure than for full power initial conditions due to the lower RCP [RCS] setpoint. Since higher conversion ratios were judged more likely at higher pressure, using the distribution for conversion ratio for high pressure conditions at full power for high pressure conditions at shutdown is a conservative choice.

- The fraction of the mechanical energy that is transmitted to the slug is not expected to be dependent on the initial reactor power level conditions.

- The probability of steam explosion given a melt pour is taken to be a function of pressure in the full power assessment. Lower probabilities were used in the high pressure scenario than in the low pressure scenario. These probabilities were assumed to be unchanged in the shutdown assessment and it was assumed that the reduced pressure of the high pressure scenario (6 MPa compared to 16 MPa at full power) would not increase the steam explosion probability. In making these choices it was noted that the steam explosion probabilities assumed in the full power assessment were judged to be conservative choices as they did not take any credit for the experimentally observed low triggerability of typical corium mixtures.

<u>In-vessel steam explosion causing lower head failure and reactor pit damage</u>

Since the assessment of the lower head failure probability closely followed the procedure used for upper head failure (leading to containment failure), the application of the results of the full power evaluation is justified, based on the same arguments outlined above for the in-vessel steam explosion case.

<u>Ex-vessel steam explosion causing reactor pit damage</u>

This phenomenological evaluation was based on a bounding case of the pour of molten corium into an ex-vessel pool at vessel failure for a sequence that has the RCP [RCS] depressurised due to an induced hot leg rupture (located at the RPV nozzle) leading to the spillage of water into the reactor pit. In this case the flow of corium into the pool is at the rate occurring at the time of vessel failure. A water pool approximately 4 m in depth develops in the reactor pit in this scenario. Thus a bounding release rate of corium and a bounding depth of water (maximising the melt mass in premixing) was taken for the evaluation and this is considered bounding for shutdown conditions, justifying the application of the resulting values in the shutdown Level 2 PSA.

3.3.18.5. In-vessel recovery

The in-vessel recovery probability evaluations for at-power states were applied directly to shutdown without modification. It is considered that these values are bounding for shutdown conditions, since the decay heat levels during shutdown start at levels which are initially similar to (but not greater than) those for full power conditions.

3.3.18.6. Loads at vessel failure

The results of this phenomenological evaluation for at-power states were reviewed for applicability to the shutdown and the conclusions are summarised below.

Vessel Failure Modes

The details of the assessment of different vessel failure modes are not influenced by the reactor operating temperature and pressure or decay heat level. Therefore the same probabilities for the reactor vessel failure modes are used in shutdown as in full power conditions.

Overpressurisation of the Reactor Pit

The results of this evaluation can be conservatively applied for shutdown. The likelihood of overpressurisation is dependent on the pressure inside the vessel at the time of breach and the mode of vessel failure. As discussed above, the expected modes of vessel failure were judged to be unchanged compared to the full power case and the pressure in the vessel at the time of breach will be the same, or lower than, for full power, due to a reduced system setpoint during shutdown states.

Rocketing of the Vessel

The probabilities for vessel rocketing at full power are applied unchanged for shutdown accident sequences. The upthrust on the vessel at vessel breach is a combination of momentum (upwards reaction forces due to the outflow of material from the vessel) and pressure terms (upwards force due to differential pressure on the vessel). In the full power evaluation, for a complete circumferential vessel breach at 16 MPa initial pressure, the combined upward force from pressure and momentum effects exceeds the restraining force by an order of magnitude. It is therefore not expected that this upwards force would be sufficiently low to avoid vessel rocketing when the initial RCP [RCS] pressure is reduced to 6 MPa.

Direct Containment Heating

The full power evaluation was performed for 16 MPa initial RCP [RCS] pressure. The results of this evaluation are conservatively applied to shutdown and no credit is taken for the reduced initial RCP [RCS] pressure of 6 MPa which would apply for high pressure shutdown accident sequences.

3.3.18.7. Hydrogen

The full power assessment of hydrogen loads on the containment addressed quasi-static loads from deflagrations and dynamic loads due to accelerated flames. It should be noted that no codes for analysis of hydrogen generation and distribution are available for shutdown conditions and therefore the applicability of full power results to the shutdown Level 2 PSA is assessed on the basis of a qualitative review, which is discussed below.

It was judged that the analysis of hydrogen deflagration could be applied for shutdown conditions. The full power assessment was developed using bounding rather than best estimate assumptions about the mass of hydrogen which would be present in containment but nevertheless resulted in low probabilities of containment failure. While it is conceivable that the absolute masses of hydrogen generated may be higher during shutdown than at power, it is also expected that shutdown transients would proceed more slowly, which would lead to increased take-up of hydrogen by the recombiners. Overall it was therefore considered reasonable to use the full power probabilities unadjusted for shutdown conditions and in making this decision it was noted that the probabilities of containment failure due to hydrogen deflagration are very low in absolute terms, meaning that the overall results of the Level 2 PSA would only be sensitive to large changes in their values.

The results of the full power assessment for dynamic loads (accelerated flames) were used in the shutdown Level 2 PSA without adjustment. It is recognised that flows within containment, and hence hydrogen distribution, may vary between full power and shutdown conditions. However, it was observed that the situations identified as potentially susceptible to flame acceleration in the full power study are the result of localised flows from the RCP [RCS] or ex-vessel corium into a particular compartment, rather than being the result of flows within containment causing preferential distribution of hydrogen into a particular volume. These situations are as follows:

Case 1. Transients at high pressure, in-vessel phase, period of discharge from RCP [RCS] via pressuriser valves

Case 2. Transients at high pressure at approximately the time of Induced Hot Leg Rupture:

Case 3. Transients at high pressure, at approximately the time of vessel failure:

Case 4a. Low pressure scenarios with short term fast MCCI following vessel failure (release into compartments adjacent to corium undergoing MCCI)

Case 4b. Scenarios without recombiner damage/impairment, ongoing long-term MCCI (release into compartments adjacent to corium undergoing MCCI)

Case 4c. Similar to Case 4b but with damaged recombiners (75 percent efficiency) - (release into compartments adjacent to corium undergoing MCCI)

Given that all the above cases involve accumulation of hydrogen in volumes adjacent to the in-vessel or ex-vessel release point, it was judged that the mixing behaviour would be analogous under shutdown conditions and hence it was considered reasonable to apply the probabilistic results from the full power evaluation to shutdown conditions.

3.3.18.8. Long term challenges

The individual elements of the long term challenges phenomenological assessment for full power are reviewed for their applicability to the shutdown below:

<u>No Containment Overpressure Failure due to Debris Quench</u>

The following parameters were identified as relevant to debris quenching causing overpressure in the full power assessment:

- The fraction of the core debris which is quenched; it is expected that this fraction would be unchanged in shutdown, since the stored heat in the debris would be a function of the melt physical characteristics rather than the decay heat level.

- The pressure increase in the containment per fraction of debris quenched; this parameter is expected to be unaffected by the plant state provided the containment is closed.

- The base (initial) containment pressure at the time of debris flooding; the base containment pressure may be lower during shutdown

Based on the above no change to the corresponding phenomenological probabilities in the Level 2 PSA model were made, since the original evaluation is considered generally applicable with some conservatism in some areas.

Significant MCCI

The likelihood of MCCI occurring during shutdown is expected to be similar or lower than at full power, due to the falling decay heat level during shutdown. MCCI is expected to be unlikely even at full power (if the debris is successfully flooded) and it was not considered worthwhile performing analysis to justify reducing the probability values for shutdown. The full power values were therefore, conservatively, applied for shutdown.

Containment Overpressure Failure due to non-condensables, Basemat Penetration or no failure

In the unlikely event of an accident sequence occurring in which MCCI is ongoing but the sprays, active cooling or safety injection system are preventing long term overpressure by steaming, the full power phenomenological assessment considers whether basemat melt-through or overpressure due to non-condensables would happen first. (If steaming is not controlled, overpressure was clearly seen to be the first failure mode, but that is not discussed here). Based on an extrapolation it was seen that there was a considerable margin in favour of basemat penetration being the first failure mode in these circumstances. It is expected that in shutdown, due to reducing decay heat levels, both basemat melt-through and overpressure due to non-condensables would be delayed but there is no reason to expect a significant shift in the relative timing of the two failure modes.

The full power probabilities for this event were therefore maintained without modification. It is noted that the probability of neither failure mode occurring may increase during shutdown (due to the mentioned lengthening of the basemat erosion and overpressure transients). However, no credit was taken for this effect, as it is not expected that CET sequences involving either basemat erosion or overpressure due to non-condensable generation would be significant in the overall results.

Containment Overpressure Failure due to Incomplete Melt Transfer

The probabilities assessed for the full power study were maintained without change for the shutdown Level 2 PSA. Only limited information is available on this potential phenomenon which was identified in the full power Level 2 PSA and it is difficult to assess the impact of power level on this. Under normal circumstances with a dry reactor pit the phenomenon is rather speculative and hence it is not considered justified to change the assigned probability. In the case of a hot leg rupture, for which a flooded reactor pit is expected, the full power study assigned a high probability to this event due to the limited investigations which were performed; it is not considered worthwhile or justified to vary that probability for shutdown conditions given that sensitivity to a value which is already high is not expected to be significant, even if it were concluded that a changed value was appropriate.

3.3.18.9. Equipment survivability

This evaluation is unaffected by the power status of the reactor.

3.3.18.10. Other events included in the Level 2 PSA model

The main impact on other events in the Level 2 PSA model is via the timing of sequences, which would impact human reliability and offsite power recovery events. Due to the lower decay heat levels which develop during shutdown, transient development is expected to be slower, implying that the time available for human actions and recoveries would be increased.

The human reliability and offsite power recovery events were not re-evaluated for shutdown conditions, however, meaning that no credit was taken for the increase in available time. The values used for these events in the shutdown Level 2 PSA are therefore considered to be conservative.

SUB-SECTION 15.4.3.3 - TABLE 1

Summary of Long Term Challenges Probabilistic Evaluation

Phenomenon	Conditions		Conditional Failure Probability
	CDES	Other - Applicable DET path - outcome DET Header	
DET Header - No containment overpressure failure due to debris quench	TP, TR	Passive flooding successful	3E-06
	PL, SL, ML, SS, LL		0.0
DET Header - No significant MCCI	all	Passive flooding unsuccessful	1.0
		Passive flooding successful	1E-3
DET Header - No containment overpressure failure before basemat penetration	all	Passive flooding unsuccessful	1E-2
DET Header - No basemat penetration	all	Flooding not effective AND Significant MCCI	0.99
		Flooding effective AND Significant MCCI AND EVU [CHRS] sprays not available AND Active cooling available	
		Flooding effective AND Significant MCCI AND EVU [CHRS] available	
DET Header – Containment overpressure failure due to incomplete melt transfer	all	Flooding effective AND EVU [CHRS] Active cooling available and actuated AND No hot leg rupture	1E-2
	TR, TP, SS, SL	Flooding effective AND EVU [CHRS] Active cooling available and actuated AND Hot leg rupture	0.5

SUB-SECTION 15.4.3.3 - TABLE 2

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SUB-SECTION 15.4.3.3 - TABLE 3A

Evaluation of Equipment Survivability for Level 2 PSA

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
Containment isolation system	Containment isolation except failure to reclose the equipment hatch(shutdown states)	None inside containment Note: For each of the containment penetrations, the isolation valves are supplied from 400V buses that are located in the auxiliary building for the applicable train. Pneumatically operated dampers on ventilation penetrations fail closed on loss of pneumatic supply or power to the pilot solenoids.	With the containment successfully isolated all pathways to the active components of this system are isolated from the containment environmental conditions. In the event of any other containment failure, the operation or otherwise of this system is irrelevant. All signals modelled (in the fault tree model) required for actuation of the containment isolation system are present before the onset of core damage and therefore are not subjected to severe accident conditions. Evaluation of survivability: Therefore the CET models assume no impact of severe accident conditions on the operation of this system.
	Failure to reclose the equipment hatch (shutdown states)	Support system for reclosing the equipment hatch Note: In case of loss of power closure of the dished cover is possible due to backup power supply from emergency diesel generators; otherwise the equipment hatch can be closed manually.	The closure of the equipment hatch in relevant shutdown states is required for complete containment isolation. Time window available before the containment condition becomes unacceptable for equipment hatch closure are evaluated for the severe accident sequences. Equipment hatch closure time is compared to these different time windows to determine if this reclosure is feasible. Evaluation of survivability: The closure of the equipment hatch is performed in two hours for most of the relevant sequences.
	Ventilation (annulus,	None inside containment	In order to model the failure of ventilation for fission

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
	reactor building or fuel building ventilation)	Note: For each of the containment penetrations, the isolation valves are supplied from 400V buses that are located in the auxiliary building for the applicable train. Pneumatically operated dampers on ventilation penetrations fail closed on loss of pneumatic supply or power to the pilot solenoids.	product mitigation in containment intact sequences, a dedicated branching was added under this header to separate sequences which were previously RC101 to RC101 and RC102 according to success or failure of ventilation header. Evaluation of survivability : The failure of any individual ventilation system leads to containment failure.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
Pressuriser safety valves Severe accident depressurisation valves	Depressurisation before vessel failure	None inside the containment: Note: The pressuriser safety valves are pilot operated valves with power supplied from 120V buses that are located in the auxiliary building for the applicable train. The Severe Accident Depressurisation valves are Motor-Operated Valves (MOVs) with power supplied from 400V buses that are located in the auxiliary building for the applicable train.	There are two functional requirements on these valves: (1) The pressuriser safety valves are required during transients at high pressure to control the RCP [RCS] pressure at the system setpoint. This function involves repeated cycling of the valves. NUREG-1150 [Ref-1] considered a failure mode of these valves sticking open during repeated cycling for high pressure sequences with core melt (P=0.5), resulting in a (beneficial) depressurisation of the RCP [RCS]. (2) The pressuriser safety valves and the severe accident depressurisation valves are used to perform the RCP [RCS] depressurisation function. Fulfilment of this function requires the valves to open and to remain open. Opening may be because of a prior opening for feed and bleed (prior to core damage) or may be after the onset of core damage, cued by high temperature (from 650ºC to 1050ºC at the core outlet.) Evaluation of survivability: These systems are to be reviewed for usage in severe accident conditions. Therefore the Level 2 PSA assumes no impact of accident conditions on equipment survivability. Qualification will include any connecting/controlling cables for actuation.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
Secondary relief and safety valves	No Induced steam generator tube rupture (SGTR)	None inside containment – these valves are located in the main steam line “bridge” areas, that are physically separated from the Reactor Building	These valves will not be subject to severe accident temperatures or pressures, as the temperature and pressure conditions are controlled by the valve setpoint pressure and therefore as in Level 1 PSA. Evaluation of survivability: Therefore the Level 2 PSA assumes no impact of accident conditions on equipment survivability and only normal “failure to reclose” probabilities will be modelled.
Hydrogen recombiners	Operation is implicitly assumed for the following headers: No containment failure before vessel breach No containment failure at the time of vessel breach No late containment failure due to hydrogen deflagration or FA/DDT	No support systems – these hydrogen recombiners are passive catalytic media that require no motive power or other support.	The hydrogen recombiners are required throughout the accident. Their operation is assumed to mitigate hydrogen concentrations as shown in the basis MAAP calculations. Evaluation of survivability: This system will be qualified for severe accidents. However, there are a number of recombiners in the MAAP containment nodes 3, 5, 6, 7, 10, and 23 that have a small susceptibility to the phenomenon of flame acceleration. The phenomenological evaluation for hydrogen includes the susceptibility of the recombiners to this failure mode. Otherwise, the CET models will assume that the performance of this system is not degraded or impacted by severe accident conditions.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
Safety Injection	Melt retention in-vessel Containment Steam Pressurisation Controlled	No support systems inside containment The MHSI and LHSI systems are normally lined up for injection into the primary system, and there are no motor operator valves inside containment that need to operate in order that safety injection succeeds.	For the CET model, this system is required to operate for in-vessel recovery, and it acts as a back-up for the active cooling mode of Containment Heat Removal System (EVU [CHRS]). For injection, the system must actuate, provide flow and continue to provide flow and heat removal in order to retain a coolable debris configuration in-vessel. In general once the core melt has been stabilised and heat removal assured, the containment conditions will improve (lower temperature and pressure). Therefore, any limiting considerations are likely to arise close to the time of actuation of the system rather than later. The alignment used to provide the backup cooling function for EVU [CHRS] is the same as for injection. If the system is in the injection line-up, the function only requires a pump start. If the system is not aligned for injection, the motor-operated valves (MOVs) in the injection path are outside containment. Evaluation of survivability: The active, electrically actuated components in this system are not exposed to severe accident conditions. The system connects directly to the RCP [RCS] but is protected by non-return valves in the case that it is not operating. Therefore there is no impact of severe accident conditions on the operation of the system. The system model for Safety Injection System (SIS) also includes failure probabilities for the clogging of the suction strainers during accident conditions. These probabilities are considered reasonable for severe accident conditions.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
EVU [CHRS] passive flooding	Melt stabilisation ex-vessel	No support systems inside containment	Required to operate a few hours after vessel failure. Operation is one time only - not continuous. This is a simple opening of a passively operated valve. Evaluation of survivability: This system will be qualified for severe accidents. Furthermore, the passive nature of the operation of the system reduces any potential susceptibility to adverse environmental conditions. On this basis, the CET models will assume that the performance of this system is not degraded or impacted by severe accident conditions.
EVU [CHRS] active flooding	Melt stabilisation ex-vessel Containment steam pressurisation controlled No basemat failure (implicitly assumes continued operation of melt stabilisation)	No support systems inside containment The valves that operate to initiate active flooding are MOVs with power supplied from a 400V bus located in the Trains 1 and 4 in the auxiliary building	This system is required to operate during the late phase of the accident. It should continuously operate following operation of EVU [CHRS] sprays for melt stabilisation. Evaluation of survivability: This system will be qualified for severe accident conditions. Therefore the CET models will assume that the performance of this system is not degraded or impacted by severe accident conditions.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
EVU [CHRS] Sprays	Melt stabilisation ex-vessel Containment steam pressurisation controlled No basemat failure (implicitly assumes continued operation of melt stabilisation)	No support systems inside containment The valves that operate to initiate active flooding are MOVs with power supplied from a 400V bus located in the Trains 1 and 4 in the auxiliary building	This system performs three functions: (1) To assist in initial stabilisation of the melt ex-vessel following passive flooding. The system ensures the condensation of water vapour in the containment atmosphere which is then provided as cooling water to the melt. This function requires suction from the In-Containment Refuelling Water Storage Tank (IRWST) and operation of the EVU [CHRS] heat exchanger. (2) To control containment pressure. Assumption in PSA: sprays are manually actuated {CCI removed} [a]. (3) To mitigate the source term. Survivability for this condition is considered as a separate line item below. Evaluation of survivability: This system will be qualified for severe accidents. Therefore the CET models will assume that the performance of this system is not degraded or impacted by severe accident conditions.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
EVU [CHRS] sprays (continued operation following containment failure)	Melt stabilisation ex-vessel EVU [CHRS] sprays actuated to control source term	The intermediate train of EVU [CHRS] provides cooling water to the EVU [CHRS] Heat Exchanger. This intermediate train is supported by a SRU [UCWS] train. EVU [CHRS] and its support components are supplied by the 400V and 690V networks of electrical Division 1 and 4, and are provided with power from the Division 1 and 4 Emergency Diesel Generators and the Station Blackout (SBO) Diesel Generators.	In this case, operation of the sprays is required to continue after containment failure for control of the leakage rate from containment and scrubbing of the release. Evaluation of survivability: This system will be qualified for severe accidents. Furthermore, the containment is expected to fail at the base of the dome, a location that will not lead to releases into compartments containing EVU [CHRS] components.
EVU [CHRS] active flooding (continued operation following containment failure for continued melt stabilisation)	Melt stabilisation ex-vessel	The intermediate train of the EVU [CHRS] provides cooling water to the EVU [CHRS] Heat Exchanger. This intermediate train is supported by a SRU [UCWS] train. EVU [CHRS] components are supplied by the 400V and 690V networks of electrical Division 1 and 4, and are provided with power from the Division 1 and 4 Emergency Diesel Generators and the Station Blackout Diesel Generators.	In this case operation is required to continue after containment failure to maintain melt stable and avoid source term contribution from MCCI. Evaluation of survivability: This system will be qualified for severe accidents. As discussed above, containment failure is not expected to lead to releases into compartments containing EVU [CHRS] components, nor to components of its support systems.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
Safety injection (continued operation with isolation failure of containment or very early containment failure)	Melt retention in-vessel	The RRI [CCWS] and SEC [ESWS] support the LHSI heat exchanger for all four trains, and the LHSI Trains 2 and 3 and MHSI motor pumps and the corresponding sealing fluid. The cooling coils of the LHSI pump motor and seals Trains 1 and 4 are supplied from the air cooled safety chilled water system DEL (SCWS) if RRI [CCWS] is lost. RIS [SIS] components are supplied by the 400V and 690V networks of electrical Divisions 1- 4, and are provided with power from the division Emergency Diesel Generators.	In this case the continued operation of safety injection is addressed for the following CET sequences: (1) Sequences with a containment isolation failure, for which in-vessel retention is subsequently assessed in the CET; (2) Sequences for which a very early containment failure occurs with a containment isolation failure, for which in-vessel retention is subsequently assessed in the CET. Evaluation of survivability: The evaluation performed for in-vessel recovery applies here, except as follows: (1) The possibility of long term water loss with a failed containment is considered to be unimportant since once sub-cooled conditions are achieved in the RCP [RCS] there will be no further water loss. (2) As with the EVU [CHRS] system, containment failure is not expected to lead to releases into compartments containing RIS [SIS] components, nor to components of its support systems.

System	Relevant CET Headers	Support Systems	Comments and Evaluation of Survivability
Instrumentation	Depressurisation before vessel failure Melt retention in-vessel Melt stabilisation ex-vessel Containment steam pressurisation controlled EVU [CHRS] sprays actuated to control source term	No support systems inside containment	The following Severe Accident Instrumentation is required to support the following operator actions: Depressurisation of RCP [RCS] Core outlet thermocouples RCP [RCS] wide and narrow range pressure Depressurisation valve actuation and position Actuation of safety injection for in-vessel core cooling Core outlet thermocouples RCP [RCS] wide and narrow range pressure IRWST level and temperature Actuation of EVU [CHRS] sprays, active flooding Containment pressure EVU [CHRS] pump inlet and outlet pressure EVU [CHRS] volumetric flow rate EVU [CHRS] passive flooding, active flooding, and spray line valve position Evaluation of survivability: These instruments should be qualified to the temperatures, pressures, and to the doses expected during their Severe Accident mission time, and are judged to be adequate in supporting their function in the Level 2 PSA CET.

SUB-SECTION 15.4.3.3 - TABLE 3B

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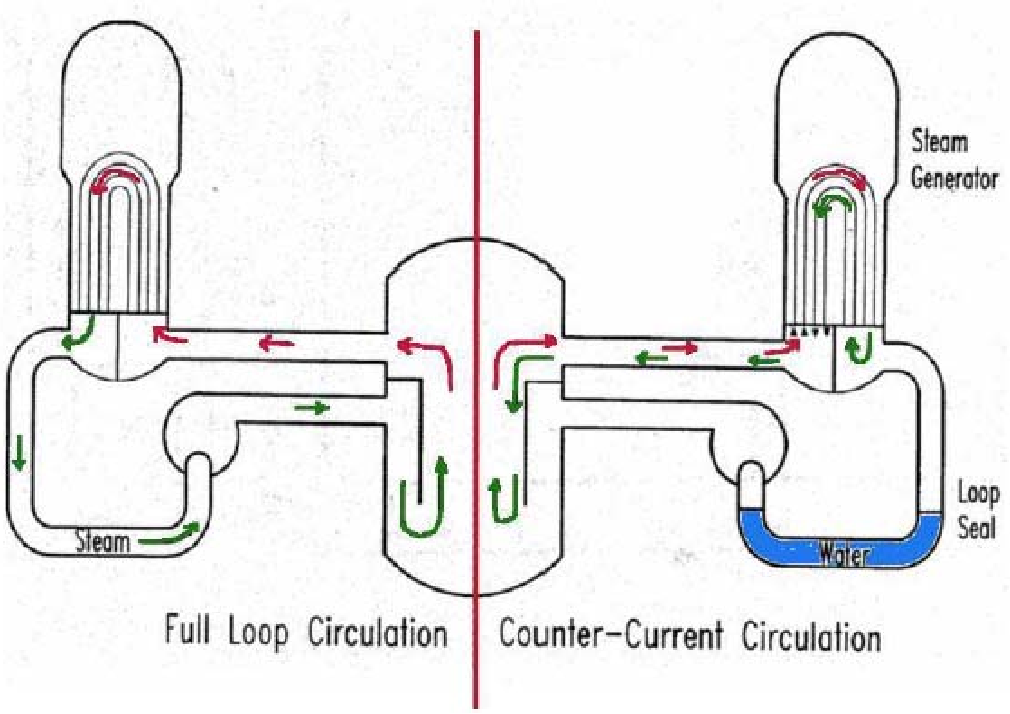
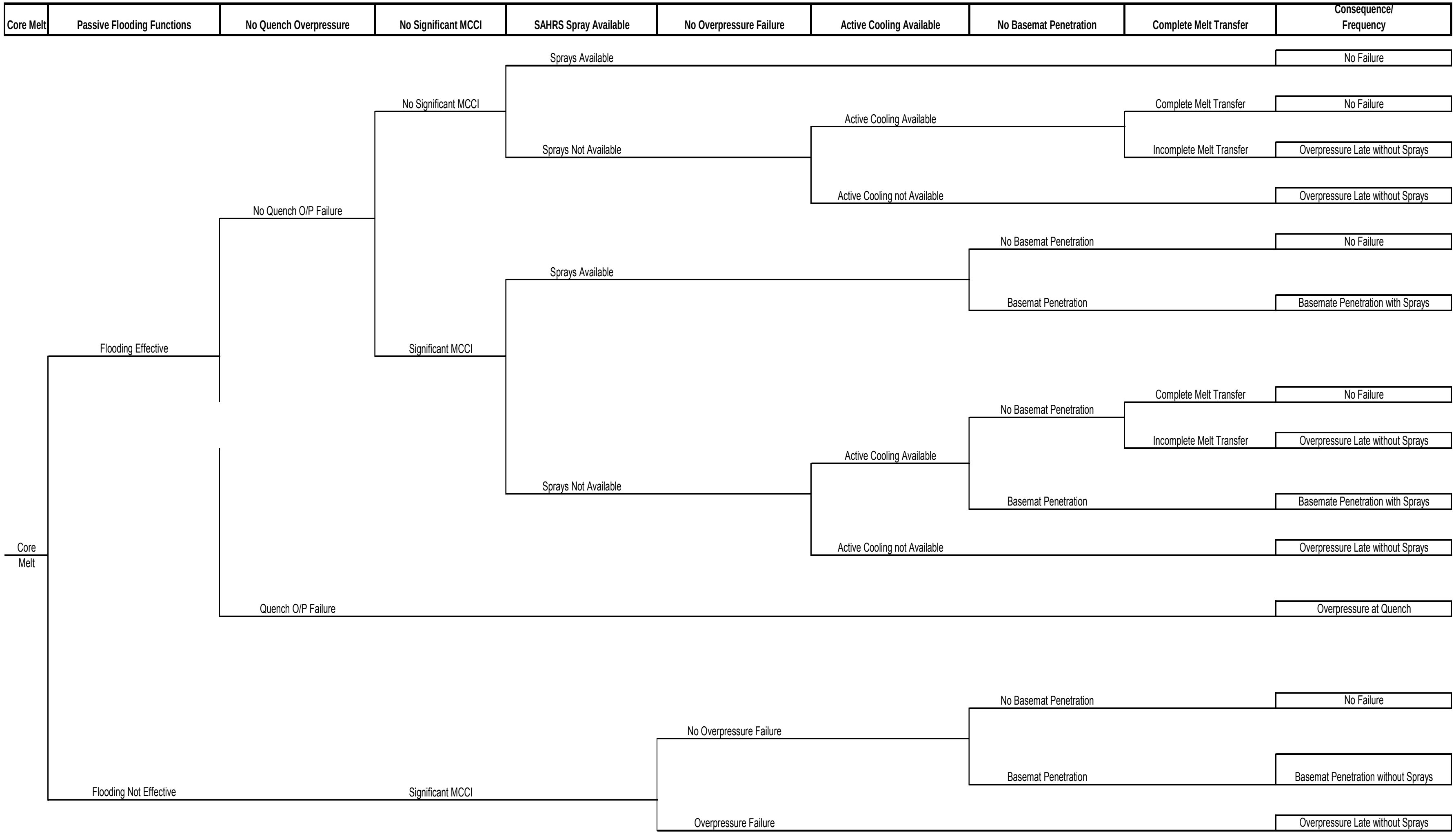
SUB-SECTION 15.4.3.3 - FIGURE 1

Circulation Flow Paths in the Primary System



SUB-SECTION 15.4.3.3 - FIGURE 2

Decomposition Event Tree for Long Term Challenges



SUB-SECTION 15.4.3.3 - FIGURE 3

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3.4. ACCIDENT SEQUENCE ANALYSIS AND CONTAINMENT EVENT TREES

3.4.1. Containment Event Trees for Events Initiated from At-Power Plant States A and B

The UK EPR Level 2 PSA consists of a basic structure of ten CETs. A number of variant CETs are also used, following the structure of three of the basic CETs [Ref-1]. The purpose of these variant CETs is to facilitate model quantification and the correct transfer of "boundary conditions" (which set house events) from the Level 1 PSA event trees to the Level 2 PSA CETs. There are no structural differences between the variant and basic CETs and so, to avoid extensive repetition, these CETs are not described in detail here. For convenience, this discussion focuses on the ten basic CETs.

Sub-section 15.4.3.4 – Table 1 lists the ten basic CETs and provides a description of each. This table also provides a column which relates the variant CETs to the corresponding basic CET on which it is based.

Sub-section 15.4.3.4 –Table 1 is supplemented by Sub-section 15.4.3.4 – Table 2 to Table 11. These provide further details on the headers included in each CET and the input events used. Figures presenting the ten basic CETs are shown as Sub-section 15.4.3.4 - Figure 1 to Sub-section 15.4.3.4 - Figure 10.

The top events included in the UK EPR CETs address the phenomenological events, the systems, and the human actions credited to mitigate the severe accident. The top events included are those which are expected to have a significant impact on the severe accident progression, meaning that they can affect, directly or indirectly, either the likelihood of containment failure or bypass or the magnitude of the source term. For convenience, the events considered within the CETs are grouped into different time frames, as follows:

- Timeframe 1 (TF1), which considers the period from the onset of core damage up to the time of vessel failure (if this occurs).
- Timeframe 2 (TF2), which considers the period from the time of vessel failure to the start of melt transfer to the spreading area.
- Timeframe 3 (TF3), which considers long term events from the time of melt transfer to the spreading area.

Relevant events considered in Timeframe 1 include containment isolation, induced RCP [RCS] failures, depressurisation of the RCP [RCS] by the operators, and hydrogen combustion. Feedwater to any steam generator is also considered as a header in this phase since it affects the likelihood of induced steam generator tube rupture. System models for annulus and building ventilation are included for containment-intact sequences in this time frame due to the impact on the source term.

Relevant events in Timeframe 2 include in-vessel steam explosion (failing containment or damaging the reactor pit), melt retention in-vessel, ex-vessel steam explosion (damaging the reactor pit), and loads at vessel failure leading to containment failure (DCH, hydrogen or vessel rocketing).

Relevant events considered in timeframe 3 include melt transfer to the spreading area, initial stabilisation of melt ex-vessel, steam overpressure during quenching leading to containment failure, hydrogen combustion, long term overpressure or basemat failure due to core concrete interaction, and sprays for source term mitigation.

The phenomena involved in the CETs for the above timeframes and their analyses are discussed in detail in section 3.3.

The linkage of the CETs to the Level 1 PSA is via the use of Core Damage End States, which are described in section 3.2. The CDES are not, however, directly transferred to Level 2 PSA CETs. Rather, each individual end state is transferred through an intermediate event tree, referred to as a CDES link event tree, prior to transfer to a Level 2 PSA CET. The use of these CDES link event trees provides a consistent structure for linking the Level 1 and Level 2 PSA models, allows separation of limited core damage sequences from severe core damage sequences, and also allows some technical aspects of the linked model to be implemented.

The CDES link event trees generally have a simple structure consisting of an initiating event (which takes the CDES consequence as input) and a labelling event which adds a basic event (flag) to the sequence to identify it according to its CDES. The labelled sequence is transferred on to the corresponding CET. An example CDES link tree is shown in Sub-section 15.4.3.4 - Figure 11.

Three CDES link event trees contain an additional function and do not follow the simple structure described above. These three event trees are presented in Sub-section 15.4.3.4 – Figure 12 to 14. These are the trees for LL CDES (Large LOCA), ML CDES (medium LOCA) and SG2 CDES (steam generator tube rupture with ASG [EFWS] not manually isolated). The LL and ML trees both have the same structure and cover the availability of 1 out of 3 trains of LHSI; success implies a limited core damage end state, which is transferred to a limited CD CET. The SG2 event tree covers the availability of emergency feedwater to the ruptured steam generator; a transfer to the SGTR CET with feedwater availability is used if this function is successful.

Once the incoming sequences from the Level 1 PSA have passed through the CDES link trees they are transferred to the appropriate CET model. Of the ten basic CETs used in the UK EPR Level 2 PSA, six receive a direct transfer from the CDES link event trees. The eighth, ninth and tenth CETs (the second stage CET for high pressure sequences, with and without long LOOP, and the second stage low pressure CET for long LOOP) only receive transfers from the corresponding first stage CETs for high pressure sequences.

A fission product release category (RC) is assigned to each end point of the CETs. All sequences within a single RC have similar release characteristics (source terms). A detailed discussion of the release categories is provided in the next section (3.5).

Low pressure, bypass and guaranteed containment failure sequences only pass through a single CET before assignment to a Release Category. The release category assignments are marked on the end of each CET sequence. High pressure sequences (without bypass or guaranteed containment failure) enter an initial CET, identified as #CET1 HI PRESSURE (or a variant CET based on this structure). This CET uses further transfers to other CETs and three outcomes are possible for sequences entering this CET. The possible outcomes are: (1) assignment of the end state to a release category due to induced SGTR, (2) transfer to the low pressure CET if depressurisation occurs (due to deliberate operator action or thermally-induced hot leg rupture), (3) transfer to the second stage high pressure CET for sequences not falling into category (1) or (2).

3.4.2. Containment Event Trees for Events Initiated from Shutdown Plant States C, D and E

The following sections describe the states and sub-states of the shutdown as used for the Level 2 PSA and provide an overview of the CET models developed for each state.

3.4.2.1. Shutdown states and sub-states defined for Level 2 PSA

Shutdown state Ca was subdivided into sub-states C1, C2, C3 and C4 for the Level 2 PSA in order to distinguish the possible combinations of RCP [RCS] status and containment status that may arise, but also the different times of sub-states for which the equipment hatch could be open. These sub-states are summarised below:

- Sub-state C1 comprises state Ca1 where the RCP [RCS] and containment are both closed (primary pressure below 30 MPa);
- Sub-state C2 comprises state Ca2where the RCP [RCS] is closed and the containment is partially open (35% of the sub-state);
- Sub-state C3 comprises states Ca3 where the RCP [RCS] is closed and the containment is partially open (40% of the sub-state);
- Sub-state C4 comprises states Ca4 where the RCP [RCS] and the containment are both closed (primary pressure below 0.5 MPa).

States Cb, D and E were not sub-divided for the level 2 PSA. These states have the following relevant characteristics:

- In state Cb the RCP [RCS] and the containment are both partially open (12% of the sub-state for the equipment hatch);
- In state D the RCP [RCS] is open and the containment is closed;
- In state E the RCP [RCS] is open with the reactor pit flooded and the containment is open.

3.4.2.2. Description of Containment Event Trees for Shutdown

The CET models for shutdown states C, D and E are based on those developed for the full-power study. A number of adjustments were made to these models to adapt them to the characteristics of the shutdown states, as described in section 3.3.18. The following paragraphs summarise these changes and provide an overview of the CET models for shutdown.

Based on the review of the phenomenological evaluations for full power and the consideration of differences in RCP [RCS] conditions for shutdown, the induced SGTR probabilities applied in the high pressure CETs were adjusted for shutdown. This was a numerical change which did not require any change to the structure of the high pressure CET. The induced hot leg rupture was completely removed from the high pressure CET and this change was implemented by a corresponding modification to the structure of the CET.

The other structural change that was applied to the CET models was applicable for states in which the containment is open, i.e. sub-states C2 and C3 of state C and state E. In these cases many sequences in the original full power CETs are no longer applicable and were therefore removed. The sequences retained in these cases were those corresponding to RC201 to RC205, i.e. the large loss of isolation cases, which were taken as equivalent source terms for shutdown cases with the containment open. The necessary changes were implemented in the second stage high pressure CETs and the low pressure CET for sub-states C2 and C3 and state E.

Other points to note regarding the CETs for shutdown states are as follows:

- High pressure CETs are not applicable for states with the RCP [RCS] open (all CDES link trees for RCP [RCS] open states transfer directly to low pressure versions of the CETs);

- Where necessary the input fault trees for the shutdown CETs were adjusted to reflect shutdown system configurations, based on the system fault tree models as used in the Level 1 PSA. These adjustments were implemented by changing the input fault tree for the affected CET headers to one which used the appropriate transfer fault trees.

- The shutdown Level 2 PSA model uses the containment isolation fault tree model developed for the LCHF analysis for shutdown modes. This fault tree model includes additional failure modes related to the containment sweep vent system.

- Failure of closure of the equipment hatch is modelled using an operator error for sequences with adequate containment conditions throughout the period up to two hours after the initiating event. Reclosure is not considered in sequences where containment conditions are not adequate for the full two hour period.

The correspondence of shutdown CET models and shutdown state of the Level 2 PSA is shown in Sub-section 15.4.3.4 - Table 12. This table also refers to figures which show the key CETs for the shutdown states. These CETs are shown in Sub-section 15.4.3.4 – Figure 15 to Figure 21.

SUB-SECTION 15.4.3.4 - TABLE 1

Summary descriptions of Level 2 PSA Containment Event Trees for At-Power Plant State Initial Conditions

CET ID	Description of CET	Variant[3] CETs also using this CET structure
#CET CF	This CET is used for core damage sequences assigned the ATI CDES. Entry is via the link tree for the ATI CDES. Sequences in this CDES are steam line breaks inside containment (SLBI) with failure to fulfil the Level 1 PSA reactivity control success criteria and heterogeneous boron dilution accidents. The latter are postulated to fail the containment. The SLBI core damage sequences are assumed to be an accident at full reactor power with blowdown of the secondary side directly into containment. It is assumed that the steam generation and pressurisation of containment in such a scenario would overpressure the containment causing its failure. For both scenarios, the sequences in this CET are assigned directly to an early containment failure release category. This modelling may be conservative. It is noted that calculations [Ref-1] show that containment-threatening pressures are not reached in SLBI sequences where reactivity control is achieved (such sequences are not transferred into this CET).	None
#CET ISL	This CET is used for core damage sequences assigned the IS CDES (IS LOCA). A header is included to assess whether or not the break location is scrubbed due to an overlying water pool. Note that an assessment performed concluded that a conditional probability of 1.0 of no overlying water pool for scrubbing should be used for IS LOCA sequences.	None

[3] Variant CETs have the same structure as the CET they are a variant of except, with the only difference being that boundary conditions are set at the entry to the CET to activate house events which model specific unavailabilities associated with the incoming Level 1 sequence.

CET ID	Description of CET	Variant[3] CETs also using this CET structure
#CET LIMITED CD	This CET is used for sequences which are identified as being limited core damage cases in the CDES link trees. In these cases as in-vessel arrest of the core damage process and in-vessel retention are assured, the only relevant question is whether or not the containment is isolated.	None
#CET LO PRESSURE	Entry to this CET is via transfers from CET1 HI PRESSURE or directly for low pressure CDES. This CET models the remaining applicable phenomena for low pressure sequences (these being those that are low at core damage or become low in the CET1 HI PRESSURE).	#CET CC1+2 LO PRESS (RRI [CCWS] trains 1 & 2 failed) #CET CC2 LO PRESS (RRI [CCWS] train 2 failed) #CET CC2 +3 LO PRESS (RRI [CCWS] trains 2 & 3 failed) #CET CCALL LO PRESS (All RRI [CCWS] trains failed) #CET CTM LO PRESS (Fire in containment) #CET LOOPS LO PRESS (Short LOOP) #CET SB1 LO PRESS (Fire in Safeguards Building 1) #CET SWGB LO PRESSURE (Fire in switchgear)
#CET SGTR	This CET passes the incoming sequences through to RC702 (unscrubbed SGTR).	None
#CET SGTR FW	This CET passes the incoming sequences through to RC701 (scrubbed SGTR).	None

CET ID	Description of CET	Variant[3] CETs also using this CET structure
#CET TP LO PRESSURE	#CET TP LO PRESSURE is used for core damage sequences following a long term Loss of Offsite Power (Long LOOP). This CET is similar to #CET LO PRESSURE except that an additional header is included to model recovery of offsite power before 31 hours (the earliest time by which containment overpressure is high enough to result in non-negligible containment failure probabilities). Recovery of offsite power is modelled by applying conditional recovery and non-recovery probabilities on the event tree branches under this header. On subsequent sequences where recovery has occurred, a house event is set which cancels the LOOP condition which is present on entry to this event tree.	None
#CET1 HI PRESSURE	This CET is the initial entry point to the CET model for CDES which are initially at high pressure. This CET asks questions corresponding to phenomena occurring during the initial in-vessel phase (timeframe 1, excluding containment isolation, which is addressed in CET2 HI PRESSURE) of the severe accident. Depressurisation performed by the operators, depressurisation due to an induced hot leg rupture and induced steam generator tube rupture are assessed. For small LOCAs the proportion of these sequences remaining at high pressure (at the time of vessel failure) is also assessed; in the current model it is conservatively assumed that 100% of these sequences remain at high pressure. The outcomes of this initial tree are either release category RC702 (unscrubbed SGTR) or a transfer to the low pressure CET (for sequences depressurised by a hot leg rupture or operator depressurisation) or a transfer to the 2nd stage high pressure CET (sequences without depressurisation or induced SGTR).	#CET1 CC1+2 HI PRESS (RRI [CCWS] trains 1 & 2 failed) #CET1 CC2 HI PRESS (RRI [CCWS] train 2 failed) #CET1 CC2+3 HI PRESS (RRI [CCWS] trains 2 & 3 failed) #CET1 CCALL HI PRESS (All RRI [CCWS] trains failed) #CET1 CTM HI PRESS (Fire in containment) #CET1 LOOPS HI PR (Short LOOP) #CET1 SB1 HI PRESS (Fire in Safeguards Building 1) #CET1 SWGB HI PRESS (Fire in switchgear) #CET1 TP HI PRESSURE (Long LOOP) Note: #CET1 TP HI PRESSURE (long LOOP) has the same structure as #CET1 HI PRESSURE as recovery events for long LOOP are only considered in the transfer CETs (#CET2 TP HI PRESS and #CET TP LO PRESSURE).

CET ID	Description of CET	Variant[3] CETs also using this CET structure
#CET2 HI PRESSURE	Entry to this CET is via transfers from CET1 HI PRESSURE. This CET models the remaining applicable phenomena for high pressure sequences (which have not depressurised due to the phenomena addressed in #CET1 HI PRESSURE).	#CET2 CC1+2 HI PRESS (RRI [CCWS] trains 1 & 2 failed) #CET2 CC2 HI PRESS (RRI [CCWS] train 2 failed) #CET2 CC2+3 HI PRESS (RRI [CCWS] trains 2 & 3 failed) #CET2 CCALL HI PRESS (All RRI [CCWS] trains failed) #CET2 CTM HI PRESS (Fire in containment) #CET2 LOOPS HI PRESS (Short LOOP) #CET2 SB1 HI PRESS (Fire in Safeguards Building 1) #CET2 SWGB HI PRESS (Fire in switchgear)
#CET2 TP HI PRESSURE	Entry to this CET is via transfers from #CET1 TP HI PRESSURE. This CET models the remaining applicable phenomena for high pressure sequences (which have not depressurised due to the phenomena addressed in #CET1 HI PRESSURE) following Long LOOP in Level 1 PSA. Recovery events for offsite power are included as described in the case of #CET TP LO PRESSURE.	None

SUB-SECTION 15.4.3.4 - TABLE 2

Containment Event Tree #CET CF – CET for sequences leading to direct containment failure

Event-Tree Top Event		Input Events	Description of input events
#CET CF	Entry from CDES with containment overpressurised	#CET CF (Consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees.
#DUMMY		L2FLDUMMY (Basic event)	The input event is a dummy event represented by a flag with numerical value of 1.0.

SUB-SECTION 15.4.3.4 - TABLE 3

Containment Event Tree #CET ISL – CET for IS LOCA sequences

Event-Tree Top Event		Input Events	Description of input events
#CET ISL	Input: from IS CDES tree	#CET-ISL (consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees.
#CET ISL LABEL	Label sequences for CET ISL	L2FLCET ISL	The input event is a flag with numerical value of 1.0.
#IS BL	IS LOCA break location is covered by water	Alt 1 – L2CP ISL BL WATER Alt 2 – L2CP ISL BL NO WATER	Events representing the presence or absence of an overlying pool of water for fission product scrubbing. (Event probabilities sum to 1.0)

SUB-SECTION 15.4.3.4 - TABLE 4

Containment Event Tree #CET LIMITED CD – CET for sequences identified as limited core damage cases in CDES link trees

Event-Tree Top Event		Input Events	Description of input events
#CET LIMITED CD	Entry: sequences identified as limited CD in CDES link trees	#CET LIMITED CD (consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees.
#CET LIMITED CD LABEL	Label sequences for CET LIMITED CD	L2FLCET LIMITED CD (Basic event)	The input event is a flag with numerical value of 1.0.
#T1 CI	Containment isolated	Alt 1 – GL2 CONT ISOL3+ Alt 2 – GL2 CONT ISOL3- Alt 3 – GL2 CONT ISOL SUC (All fault tree top gates)	Inputs are described in #CET LO PRESSURE DESCRIPTION

SUB-SECTION 15.4.3.4 - TABLE 5

Containment Event Tree #CET LO PRESSURE – CET for low pressure CDES or Depressurised HI CDES

Event-Tree Top Event		Input Events	Description of input events
#CET LO PRESSURE	ENTRY: Low pressure CDES or Depressurised HI CDES	#CET LO PRESSURE (consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees or from #CET1 HI PRESSURE (which also transfers here)
#CET LO PRESS LABEL	Label sequences for CET LO PRESS	Alt 1 – L2FLDELETE Alt 2 – L2FLCET LO PRESSURE	The input events are flags. L2FLDELETE has a value of zero (marks sequences which are not used). L2FLCET LO PRESSURE has a value of 1.0; it is used to mark sequences passing thru this CET.
#T1 CI	Containment isolated	Alt 1 – GL2 CONT ISOL3+ Alt 2 – GL2 CONT ISOL3- Alt 3 – GL2 CONT ISOL SUC (All fault tree top gates)	GL2 CONT ISOL3+ represents failures leading to a loss of containment isolation of 3” or greater diameter. GL2 CONT ISOL3- represents failures leading to a loss of containment isolation of less than 3” diameter. GL2 CONT ISOL SUC implements a delete term for the success path. Delete terms are added automatically by the PSA software for two branch event tree nodes, but the user has to implement these manually on three branch nodes.
#T1 CF	No cont. fail before vessel breach	Alt 2 – GL2 TF1 VEARLY CF(L) (Fault tree top gate)	Fault tree model for very early containment failure due to hydrogen combustion. Alt 2 is used as the input on the low pressure CET (Alt 1 is used in the high pressure CET).

Event-Tree Top Event		Input Events	Description of input events
#T2 CFIVSE	No cont fail due to in-vessel steam explosion	GL2 TF1 STM EXP-1 (Fault tree top gate)	Fault tree modelling containment failure due to in-vessel steam explosion. Dependency on CDES category and depressurisation status (high pressure versus low pressure sequence) is addressed by the use of logic within the fault tree.
#T2 PFIVSE	No reactor pit damage due to lower head failure by in-vessel steam explosion	GL2 TF2 STM EXP2-1 (Fault tree top gate)	Fault tree modelling failure of the lower head by a steam explosion which is assumed to lead to reactor pit damage. Dependency on CDES category and depressurisation status (high pressure versus low pressure sequence) is addressed by the use of logic within the fault tree.
#T2 VB	Melt retention in-vessel	Alt 1 – GL2 TF2 VB-1 Alt 2 – GL2 TF2 VB=N (Fault tree top gates)	The Alt 1 fault tree input is used on the failure path. This fault tree represents the availability of LHSI to provide injection and the operator actions required to manually actuate LHSI. Phenomenological failure is also modelled in the fault tree using the failure probabilities derived from the Phenomenological Evaluation. The Alt 2 fault tree is used on the success path for this event. Since the probability of failure at this node is relatively high (>0.05) it is necessary to manually add a quantitative assessment of the success probability. The fault tree uses the numerical complement of the in-vessel recovery failure probabilities.
#T2 PFXVSE	Reactor Pit not damaged by ex-vessel steam explosion or (for hi press sequences) pit overpressure at VF	GL2 TF2 STM EXP EXV (Fault tree top gate)	A single fault tree is used for the low pressure CET and for the high pressure CET. Logic is set up within the fault tree to add in the pit overpressure failures which are applicable only for high pressure vessel failures. A single bounding event is used in both cases for the probability of steam explosion causing pit failure.

Event-Tree Top Event		Input Events	Description of input events
#T2 CF	No containment failure at the time of Vessel Breach	GL2 TF2 EARLYRUPT-1 (Fault tree top gate)	The same fault tree model is used in the high pressure and low pressure CETs. Logic is set up within the fault tree to select the relevant failures for low pressure and high pressure cases. DCH and rocketing are specific failure events for the high pressure vessel failure case. Failures due to hydrogen combustion are modelled in both CETs, with different probabilities.

Event-Tree Top Event		Input Events	Description of input events
#T3 MSXV	Melt stabilisation ex-vessel	Alt 1 – GL2 TF3 CCI Alt 2 – GL2 TF3 CCI 01 Alt 3 – GL2 TF3 CCI 01=N (Fault tree top gates)	The Alt 1 fault tree is used on pathways through the CET where there has been no damage to the reactor pit, meaning that melt transfer to the spreading area will occur in an orderly manner, according to the design intent. This fault tree therefore models two failure paths: (i) a residual probability of phenomenological failure under normal circumstances, (ii) failure of the passive basemat flooding which leaves the corium in dry conditions. Note that active EVU [CHRS] and EVU [CHRS] sprays are not required for melt stabilisation success. This is because (as shown by MAAP analysis) if passive flooding is successful, dryout of the spreading area would not occur for over 72 hours. Furthermore, in the absence of active EVU [CHRS]/sprays and wet conditions in the spreading area, overpressure of the containment would occur before 72 hours, making MCCI irrelevant. The Alt 2 fault tree represents a guaranteed failure (100% probability of MCCI occurring). It is used on sequences where the pit was damaged by previous events on the sequence – it is conservatively assumed that any pit damage resulting in “bypass” of the reactor pit initial melt retention provision will prevent proper melt stabilisation and lead to MCCI with a probability of 1.0. The Alt 3 input is used to model the success path when Alt 2 is used on the failure path. The fault tree consists of a single basic event with a probability of zero. This deletes all cutsets on this path and is used because the Alt 2 fault tree is a guaranteed failure.

Event-Tree Top Event		Input Events	Description of input events
#T3 CFH2	No late containment failure due to hydrogen deflagration or FA or quench spiking	Alt 1 – GL2 TF3 CF H2-1 Alt 2 – GL2 TF3 CF H2 01-1 (Fault tree gates)	These fault trees model failure of the containment due to hydrogen combustion or a pressure peak during quenching of the corium ex-vessel (for some CDES). Logic is set up in the fault tree to select the correct hydrogen combustion failure probabilities for high and low pressure sequences and to apply the containment failure probability due to quench spiking only for CDES where it is applicable. The Alt 1 and Alt 2 variants are used on pathways with and without MCCI occurring. This influences the hydrogen combustion failure probabilities.
#T3 STMCNTL	Containment steam pressurisation controlled	GL2 TF3 STM PCNTRL (Fault tree top gate)	This fault tree models the use of active EVU [CHRS] flooding and sprays to control the steam pressurisation of the containment. Relevant operator actions are incorporated into the fault tree. The fault tree also models phenomenological failures which may lead to retention of some corium in the reactor pit, which may not be outside the circulation path of the water being cooled by EVU [CHRS]. This situation requires the operators to actuate cooling using LHSI to inject into the reactor pit; these actions are modelled in the fault tree.
#T3 LTCF=NO/OP/BMT	Long term CF, Branches: 1 = No fail; 2=OP fail due to non-condensables; 3=Basemat fail	Alt 1 – L2PH LATE-CCI-CF=N Alt 2 – L2PH LATE-CCI-CF=OP Alt 3 – L2PH LATE-CCI-CF=BMT (Basic events)	These events represent the probabilities of basemat melt through, no failure and overpressure due to non-condensables, as assessed in the phenomenological evaluation. The events sum to 1.0 and are only applied on pathways where MCCI is ongoing.

Event-Tree Top Event		Input Events	Description of input events
#T3 SPR	EVU [CHRS] sprays actuated to control source term	Alt 1 – GL2-TF3 SAHRS SPR ST Alt 2 – GL2 TF3 SAHRS SP(CI) (Fault tree top gates)	These fault trees represent failure of the sprays for source term mitigation (including operator failures). The Alt 1 event is used except following a loss of containment isolation has occurred, where the Alt 2 event is used. Different operator failure events are modelled in the two variant fault trees.
#TF1 VENTILATION	Failure of annulus, fuel building or auxiliary building ventilation	Alt 1 - GL2 SYS VENTILATION Alt 2 - GL2 SYS VENTILTN=Y (Fault tree top gates)	Alt 1 is used on the failure path under this header. It is modelled as failure of annulus ventilation (EDE [AVS]) or failure of buildings isolation (DWL). “OR” logic is used since the source term calculations for the success path assume availability of annulus ventilation and buildings ventilation. The gate used as input Alt-2 switches the top “OR” gate to a “NOR” gate. This ensures deletion of cutsets that are impossible on the success path for this header.

SUB-SECTION 15.4.3.4 - TABLE 6

Containment Event Tree #CET SGTR - CET for SGTR sequences

Event-Tree Top Event		Input Events	Description of input events
#CET SGTR	Entry: from SG CDES, sequences with FW not running	#CET SGTR (consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees.
#CET SGTR LABEL	Label sequences for CET SGTR	L2FLCET SGTR (Basic event)	Flag event, value = 1.0.

SUB-SECTION 15.4.3.4 - TABLE 7

Containment Event Tree #CET SGTR FW - CET for SGTR sequences, FW running

Event-Tree Top Event		Input Events	Description of input events
#CET SGTR FW	CET for SGTR sequences, FW running	#CET SGTR FW (consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees.
#CET SGTR FW LABEL	Label sequences for CET SGTR FW	L2FLCET SGTR FW (Basic event)	Flag event, value = 1.0.

SUB-SECTION 15.4.3.4 - TABLE 8

Containment Event Tree #CET TP LO PRESSURE - CET for Long LOOP sequences with depressurisation

Event-Tree Top Event		Input Events	Description of input events
#CET TP LO PRESSURE	ENTRY: Low pressure CDES or Depressurised HI CDES	#CET TP LO PRESSURE (consequence)	Consequence is used to mark the transfer into this CET from #CET1 TP HI PRESSURE
#CET LO PRESS LABEL	As #CET LO PRESSURE		
#T1 CI			
#T1 CF			
#T2 CFIVSE			
#T2 PFIVSE			
#T2 VB			
#T2 PFXVSE			
#T2 CF			
#T3 MSXV			
#T3 CFH2			
#T3 LOOP REC	LOOP Recovery before 31 hours	Alt –1 - GL2 LOOP RECOVER<31H Alt-2 - GL2 LOOP NO RECOVER (Fault tree top gates)	Appropriate LOOP recovery and non-recovery probabilities are used under the input gates shown.

Event-Tree Top Event		Input Events	Description of input events
#T3 STMCNTL	Containment steam pressurisation controlled	Alt-1 - GL2 TF3 STM PCNTRL Alt-2 - GL2 TF3 STM PCNTRL Alt-2 input sets boundary condition LOOP RECOVERED (Fault tree top gate)	This fault tree models the use of active EVU [CHRS] flooding and sprays to control the steam pressurisation of the containment. Relevant operator actions are incorporated into the fault tree. The fault tree also models phenomenological failures which may lead to retention of some corium in the reactor pit, which may not be outside the circulation path of the water being cooled by EVU [CHRS]. This situation requires the operators to actuate cooling using LHSI to inject into the reactor pit; these actions are modelled in the fault tree. In LOOP sequences with recovery of offsite power input Alt-2 is selected. This causes a house event to be set which cancels the LOOP condition which was active on entry to this CET.
#T3 LTCF=NO/OP/BMT	As #CET LO PRESSURE		
#T3 SPR			
#TF1 VENTILATION			

SUB-SECTION 15.4.3.4 - TABLE 9

Containment Event Tree #CET1 HI PRESSURE - Initial CET for High Pressure CDES

Event-Tree Top Event		Input Events	Description of input events
#CET1 HI PRESSURE	ENTRY: CD from a high pressure CDES	#CET1 HI PRESSURE (consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees.
#CET HI PRESSURE LBL	Label sequences for CET HI PRESSURE	Alt 1 - L2FLDELETE Alt 2 - L2FLCET1 HI PRESSURE (Basic events)	The input events are flag events. L2FLDELETE has a value of zero, and is used to delete cutsets on unused sequences. L2FLCET1 HI PRESSURE marks the cutsets as coming from this CET and has a value of 1.0.
#TF1-RCS.DEP	Depressurisation before induced SGTR occurs	GL2 SYS DEPRESS-1 (Fault tree top gate)	This fault tree models the failure of the operator to depressurise the primary system according to cues in the EOPs (at 650 °C core outlet temperature) or in the OSSA (cued by 1050 °C in the core outlet). The failures modelled are hardware failures of the depressurisation valves, operator failures and consequential failures arising from some initiating events.
#FW ANY SG	Feedwater (and heat removal) to any SG	GL2 SYS EFW 4/4 (Fault tree top event)	This fault tree models failure of feedwater and heat removal to all steam generators. If heat removal to any SG is available, challenge to the SG tubes is avoided and there will be no thermally induced SGTR.

Event-Tree Top Event		Input Events	Description of input events
#TF1-ISGTR	No induced SGTR	Alt 1 - GL2 TF1 ISGT=Y Alt 2 - GL2 TF1 ISGT=N (Fault tree top gates)	The Alt 1 fault tree models the induced SGTR probabilities as assessed in the phenomenological evaluation. Logic in the fault tree selects the correct values for the different CDES entering this CET. A fault tree model is also included as an input to this header to model stuck open steam generator valves causing depressurisation of an SG, as this influences the induced SGTR probability. The Alt 2 fault tree is used to assign the appropriate probability to the success path at this node. It is required since the failure probabilities may be high in some cases. Selection logic is included to assign the correct probabilities according to the cases entering this node.
#TF1-IHLR	No Induced Hot Leg Rupture	Alt 1 - GL2 TF1 IHLR=N Alt 2 - GL2 TF1 IHLR=Y (Fault tree top gates)	The Alt 2 fault tree models the induced hot leg probabilities as assessed in the phenomenological evaluation. Logic in the fault tree selects the correct values for the different CDES entering this CET. The Alt 1 fault tree is used to assign the appropriate probability to the success path at this node. It is required since the failure probabilities may be high in some cases. Selection logic is included to assign the correct probabilities according to the cases entering this node.

Event-Tree Top Event		Input Events	Description of input events
#TF1-LOCADEP	RCP [RCS] pressure remains high in small LOCA sequences	Alt 1 - L2PH LOCA-DEPRESS=N Alt 2 - L2PH LOCA-DEPRESS=Y (Basic events)	L2PH LOCA-DEPRESS=N is used to assign the probability of the pressure remaining high in Small LOCA sequences. Its value is set to 1.0; it is assumed that all incoming sequences remain at high pressure. L2PH LOCA-DEPRESS=Y takes a value of 0.0 - no depressurisation is assumed.
#LBLDPR	Label according to reason for depressurisation - HLR or Operator	Alt 1 - L2FLOP DEPRESS Alt 2 - L2FLHLR DEPRESS Alt 3 - L2FLNAT DEPRESS (Basic events)	These events are flag events. They are used to mark cutsets with the mechanism of depressurisation, as identified according to the event tree sequence. The flags are used for results interpretation. Note: Input Alt-2 is not used under this header.

SUB-SECTION 15.4.3.4 - TABLE 10

Containment Event Tree #CET2 HI PRESSURE - CET for low pressure CDES or Depressurised HI CDES

Event-Tree Top Event		Input Events	Description of input events
#CET2 HI PRESSURE	Transfer CET - non-depressurised High Pressure CDES	#CET2 HI PRESSURE (consequence)	Consequence is used to mark the transfer into this CET from the CDES link event trees or from #CET1 HI PRESSURE which also transfers here.
#CET2 HI PRESS LABEL	Label sequences for CET2 HI PRESS	Alt 1 - L2FLDELETE Alt 2 - L2FLCET2 HI PRESSURE (Basic events)	The input events are flag events. L2FLDELETE has a value of zero, and is used to delete cutsets on unused sequences. L2FLCET2 HI PRESSURE marks the cutsets as coming from this CET and has a value of 1.0.
#T1 CI	Containment isolated	Alt 1 – GL2 CONT ISOL3+ Alt 2 – GL2 CONT ISOL3- Alt 3 - GL2 CONT ISOL SUC (All fault tree top gates)	As explained in Low Pressure CET description.
#T1 CF	No cont. fail before vessel breach	Alt 1 - GL2 TF1 VEARLY CF(H) (Fault tree top gate)	Fault tree model for very early containment failure due to hydrogen combustion. Alt 1 is used as the input on the high pressure CET (Alt 2 is used in the low pressure CET).
#T2 CFIVSE	No cont fail due to in-vessel steam explosion	GL2 TF1 STM EXP-1 (Fault tree top gate)	As explained in Low Pressure CET description.

Event-Tree Top Event		Input Events	Description of input events
#T2 PFIVSE	No reactor pit damage due to lower head failure by in-vessel steam explosion	GL2 TF2 STM EXP2-1 (Fault tree top gate)	As explained in Low Pressure CET description.
#T2 PFXVSE	Reactor Pit not damaged by ex-vessel steam explosion or (for hi press sequences) pit overpressure at VF	GL2 TF2 STM EXP EXV (Fault tree top gate)	As explained in Low Pressure CET description.
#T2 CF	No containment failure at the time of Vessel Breach	GL2 TF2 EARLYRUPT-1 (Fault tree top gate)	As explained in Low Pressure CET description.
#T3 MSXV	Melt stabilisation ex-vessel	Alt 1 - GL2 TF3 CCI Alt 2 - GL2 TF3 CCI 01 Alt 3 - GL2 TF3 CCI 01=N (Fault tree top gates)	As explained in Low Pressure CET description.
#T3 CFH2	No late containment failure due to hydrogen deflagration or FA or quench spiking	Alt 1 - GL2 TF3 CF H2-1 Alt 2 - GL2 TF3 CF H2 01-1 (Fault tree gates)	As explained in Low Pressure CET description.
#T3 STMCNTL	Containment steam pressurisation controlled	GL2 TF3 STM PCNTRL (Fault tree top gate)	As explained in Low Pressure CET description.

Event-Tree Top Event		Input Events	Description of input events
#T3 LTCF=NO/OP/BMT	Long term CF, Branches: 1 = No fail; 2=OP fail due to non-condensables; 3=Basemat fail	Alt 1 - L2PH LATE-CCI-CF=N Alt 2 - L2PH LATE-CCI-CF=OP Alt 3 - L2PH LATE-CCI-CF=BMT (Basic events)	As explained in Low Pressure CET description.
#T3 SPR	EVU [CHRS] sprays actuated to control source term	Alt 1 - GL2-TF3 SAHRS SPR ST Alt 2 - GL2 TF3 SAHRS SP(CI) (Fault tree top gates)	As explained in Low Pressure CET description.

SUB-SECTION 15.4.3.4 - TABLE 11

Containment Event Tree #CET2 TP HI PRESSURE - Continuation CET for High Pressure TP (long LOOP) CDES

Event-Tree Top Event		Input Events	Description of input events
#CET2 TP HI PRESSURE		#CET2 TP HI PRESSURE (consequence)	Consequence is used to mark the transfer into this CET from #CET1 TP HI PRESSURE which also transfers here.
#CET2 HI PRESS LABEL	Label sequences for CET2 HI PRESS	As #CET LO PRESSURE	
#T1 CI	Containment isolated		
#T1 CF	No cont. fail before vessel breach		
#T2 CFIVSE	No cont fail due to in-vessel steam explosion		
#T2 PFIVSE	No reactor pit damage due to lower head failure by in-vessel steam explosion		
#T2 PFXVSE	Reactor Pit not damaged by ex-vessel steam explosion or (for hi press sequences) pit overpressure at VF		
#T2 CF	No containment failure at the time of Vessel Breach		
#T3 MSXV	Melt stabilisation ex-vessel		

Event-Tree Top Event		Input Events	Description of input events
#T3 CFH2	No late containment failure due to hydrogen deflagration or FA or quench spiking		
#T3 LOOP REC	LOOP Recovery before 31 hours	Alt –1 - GL2 LOOP RECOVER<31H Alt-2 - GL2 LOOP NO RECOVER (Fault tree top gates)	Appropriate LOOP recovery and non-recovery probabilities are used under the input gates shown.
#T3 STMCNTL	Containment steam pressurisation controlled	Alt-1 - GL2 TF3 STM PCNTRL Alt-2 - GL2 TF3 STM PCNTRL Alt-2 input sets boundary condition LOOP RECOVERED (Fault tree top gate)	This fault tree models the use of active EVU [CHRS] flooding and sprays to control the steam pressurisation of the containment. Relevant operator actions are incorporated into the fault tree. The fault tree also models phenomenological failures which may lead to retention of some corium in the reactor pit, which may not be outside the circulation path of the water being cooled by EVU [CHRS]. This situation requires the operators to actuate cooling using LHSI to inject into the reactor pit; these actions are modelled in the fault tree. In LOOP sequences with recovery of offsite power input Alt-2 is selected. This causes a house event to be set which cancels the LOOP condition which was active on entry to this CET.
#T3 LTCF=NO/OP/BMT	Long term CF, Branches: 1 = No fail; 2=OP fail due to non-condensables; 3=Basemat fail	As #CET LO PRESSURE	

Event-Tree Top Event		Input Events	Description of input events
#T3 SPR	EVU [CHRS] sprays actuated to control source term		

SUB-SECTION 15.4.3.4 - TABLE 12

Summary descriptions of Level 2 PSA Containment Event Trees for Shutdown Plant State Initial Conditions

Shutdown State	Main CETs	Variant CETs	Notes
C	#CET CF HC C #CET CF HO C #CET1 XXX HI PR HC C #CET1 XXX HI PR HO C #CET XXX LO PR HC C #CET XXX LO PR HO C XXX : applicable for CCALL, LOOPS, TP and TR #CET IS C	 #CET2 XXX HI PR HC C #CET2 XXX HI PR HO C #CET XXX LO PR HC C #CET XXX LO PR HO C XXX : applicable for CCALL, LOOPS, TP and TR	In state C, the CET models are analogous to those for at-power conditions, except that the high pressure CETs are modified to remove the induced hot leg rupture, which is not credited for shutdown states. The CET includes the potential for equipment hatch to be in closed or open position (i.e. HC or HO). When the equipment hatch is open the second stage high pressure CET and the low pressure CET are modified to treat the containment open status by assigning release categories RC201 to RC205 to the end states. As described in section 3.4.22 this was carried out by removing sequences from the original full power CET which are no longer relevant given the initial containment open status. Sub-section 15.4.3.4 – Figures 15 and 17 show the CET1/2 HI PR HC C in which the induced hot leg rupture has been removed (for equipment hatch closed). Sub-section 15.4.3.4 – Figures 16 and 18 show CET1/2 HI PR HO C in which the induced hot leg rupture has been removed (for equipment hatch open). When the sub-state is an RCP [RCS] open state, no high pressure CETs is included in the model. Sub-section 15.4.3.4 – Figure 19 shows CET LO PR HC C, CET for equipment hatch closed. Sub-section 15.4.3.4 – Figure 20 shows CET LO PR HO C, CET for equipment hatch open.

Shutdown State	Main CETs	Variant CETs	Notes
			Variant[4] CETs are provided for loss of all CCW, short loop and long LOOP (TP). The reasons for using variant CETs are as described for the full power model. Note that the number of variant CETs is reduced compared to the full power case since a number of initiating events which generate core damage accident sequences for full power initial conditions are not present in the shutdown core damage analysis.
D	#CET IS #CET LO PR. (D)	#CET LOOPS LO PR (D) #CET TR LO PR (D)	State D is an RCP [RCS] open state. Therefore there are no high pressure CETs. The low pressure CETs for state D follow the same structure as those developed for full power. Sub-section 15.4.3.4 – Figure 21 shows #CET LO PR. (D).
E	#CET IS #CET LO PR. (E)	None	State E is an RCP [RCS] open state. Therefore there are no high pressure CETs. The containment is open is state E, therefore the low pressure CET is simplified to take account of the initial containment open status in the same way as described for sub-state C2. Sub-section 15.4.3.4 – Figure 22 shows #CET LO PR. (E).

[4] As in the case of the full power Level 2 PSA, variant CETs have the same structure as the CET of which they are a variant, the only difference being that boundary conditions are set at the entry to the CET to activate house events which model specific unavailabilities associated with the incoming Level 1 sequence. The naming convention used for variant CETs is based on the same conventions as for full power - for example where LOOPS is included in a variant CET name, the CET is being applied for short LOOP sequences.

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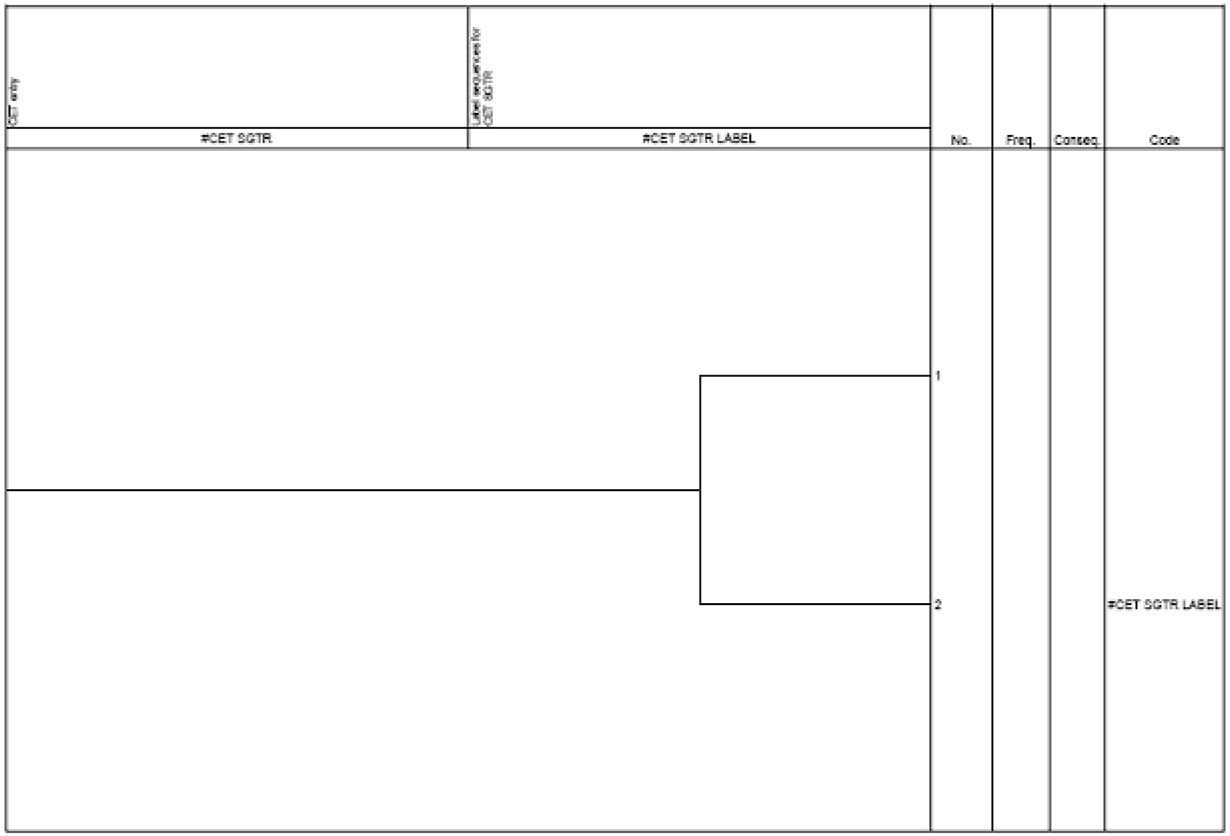
SUB-SECTION 15.4.3.4 - FIGURE 5

Containment Event Tree #CET SGTR



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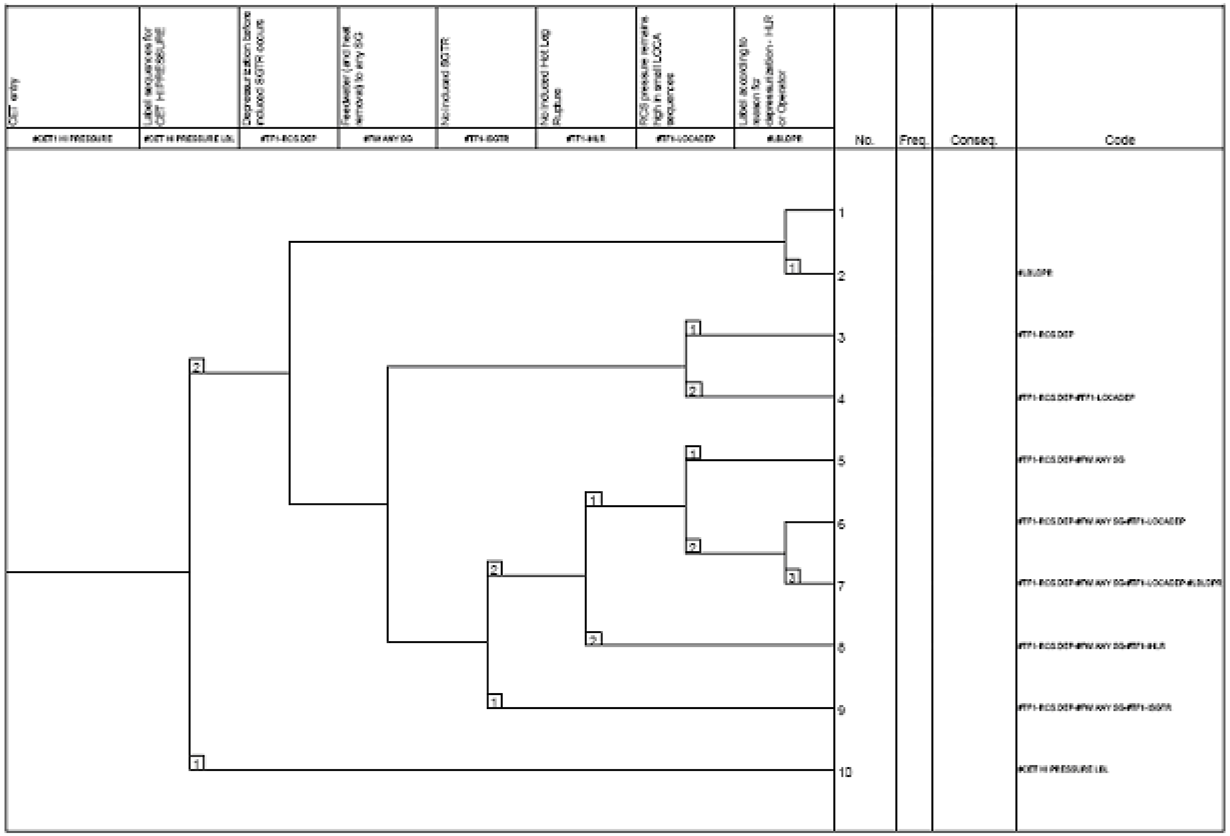
SUB-SECTION 15.4.3.4 - FIGURE 8

Containment Event Tree #CET1 HI PRESSURE



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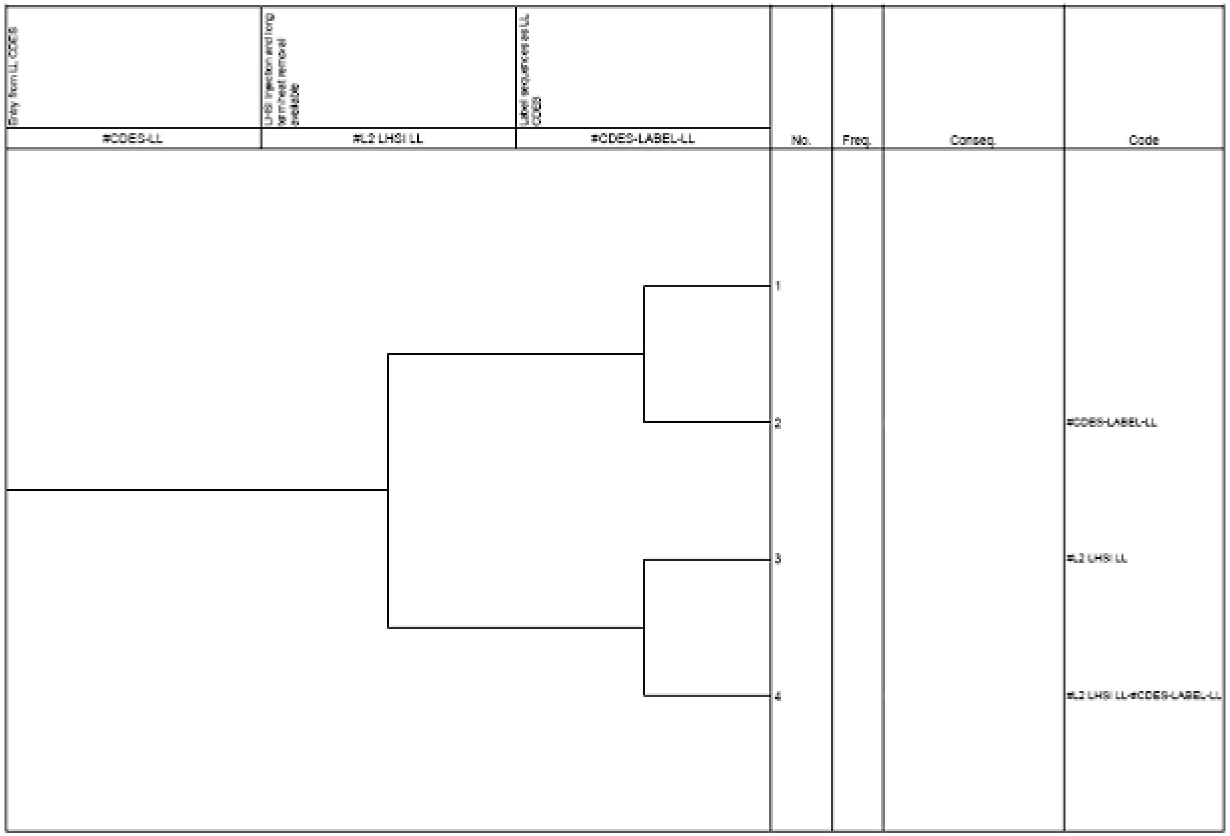
SUB-SECTION 15.4.3.4 - FIGURE 12

CDES Link Event Tree for LL CDES - #CDES-LL



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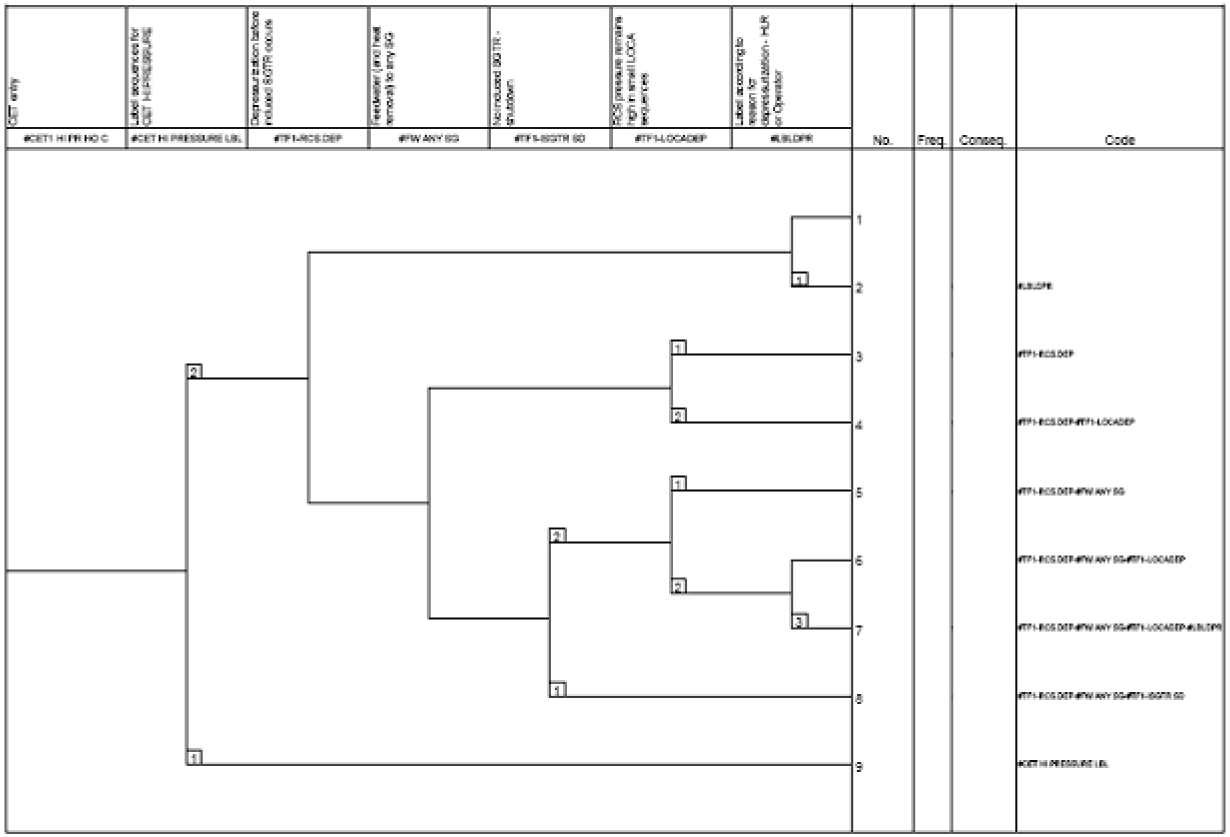
SUB-SECTION 15.4.3.4 - FIGURE 16

Containment Event Tree #CET1 HI PR HO C



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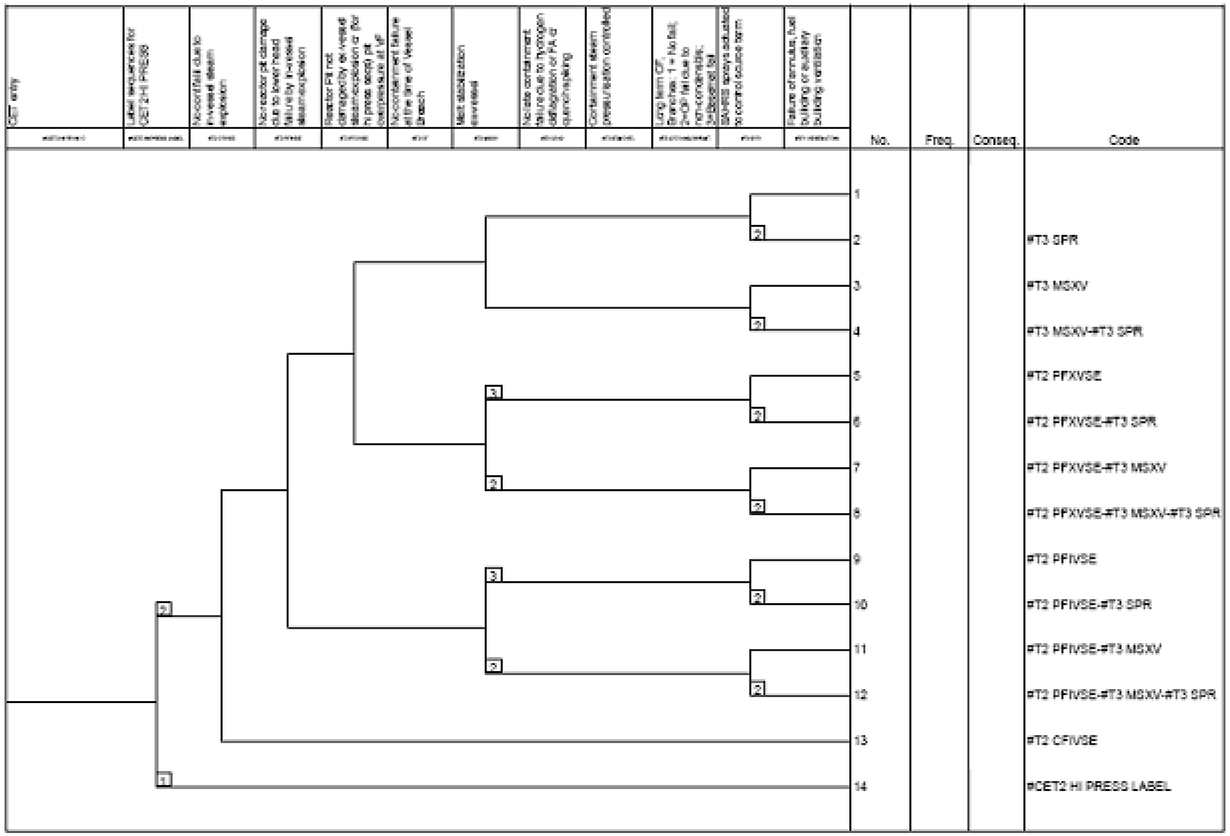
SUB-SECTION 15.4.3.4 - FIGURE 18

Containment Event Tree #CET2 HI PR HO C



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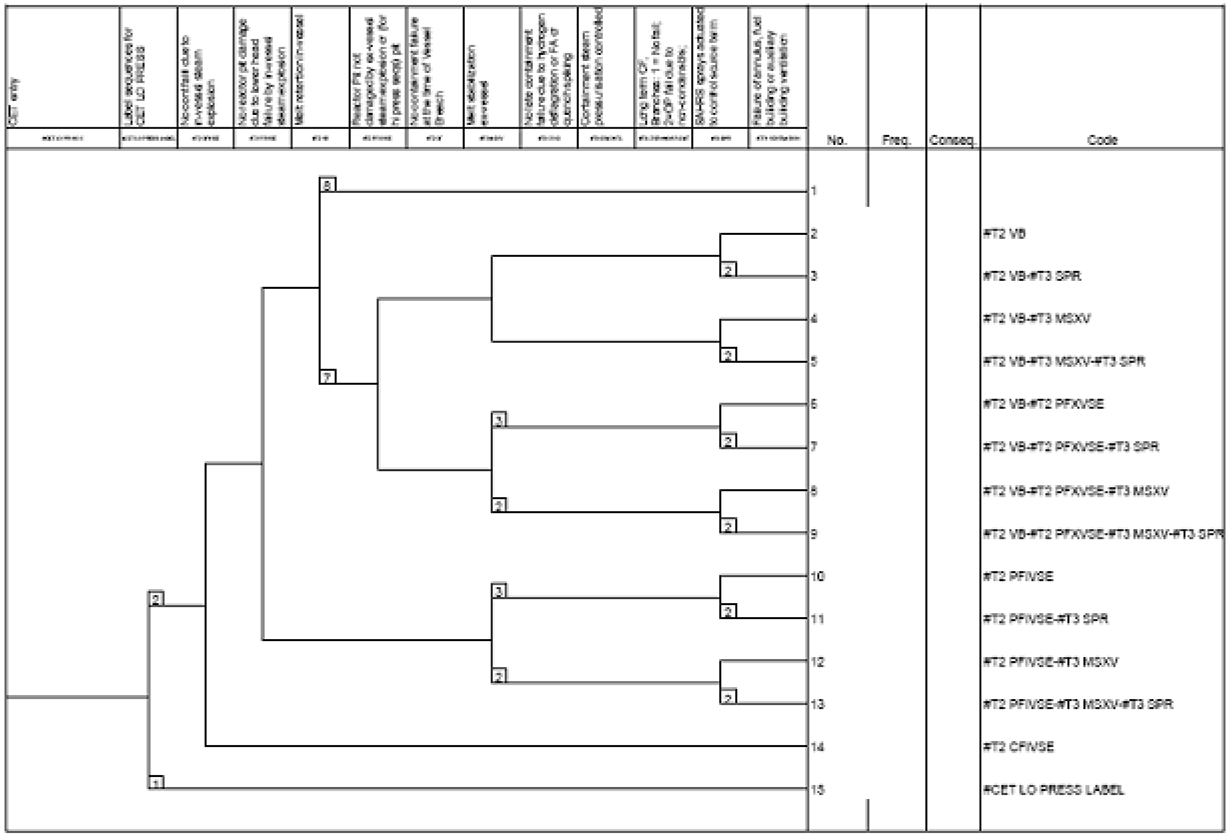
SUB-SECTION 15.4.3.4 - FIGURE 20

Containment Event Tree #CET LO PR HO C



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3.5. RELEASE CATEGORY DEFINITION AND SOURCE TERM ANALYSIS

3.5.1. Release Category Definition

A total of 29 release categories have been defined for the U.K. EPR Level 2 PSA. Seven (7) attributes of the accident sequences have been considered in defining these release categories.

The paragraphs below discuss the attributes used to define the release categories.

Attribute – Containment bypass versus no bypass

This heading is used to separate sequences in which the containment is bypassed from sequences in which there is no bypass. Bypass sequences are:

- ISLOCAs (with no isolation of the break)
- SGTRs
- SGTRs induced by creep rupture due to high temperature and pressure during the severe accident

Attribute – Time frame in which containment failure occurs

This heading separates sequences with containment failure according to the time frame in which they occur. The time frames are:

TF1 - period from the onset of core damage up to the time of vessel failure

TF2 - period approximately at the time of vessel breach, up to the melt transfer to the spreading area

TF3 - long term, the period after melt transfer to the spreading area (including sequences where quench should have occurred but did not, due, for example, to the failure of severe accident passive flooding)

The types of failure possible in each time frame are identified in the next attribute.

Attribute – Containment failure category

Classifies the category of failure according to the following:

- For TF1 – the failure may be a loss of isolation or a rupture (alpha-mode - in-vessel steam explosion - failures are grouped as ruptures under this header)
- For TF2 - only a rupture of the containment is possible
- For TF3 - the failure may be a rupture or a basemat melt through
- For bypass sequences, this header separates Steam Generator Tube Rupture sequences from Interfacing System LOCA sequences.

Attribute – Melt retained in-vessel

This header splits out sequences with and without vessel breach (success or failure of melt retention in-vessel).

Attribute – MCCI occurs

This header separates sequences having extended Molten Core Concrete Interaction (MCCI) from sequences with no MCCI.

Attribute – Melt flooded ex-vessel (covered by water)

This attribute identifies whether or not the melt is covered by water ex-vessel.

Attribute – Source term mitigated by sprays or scrubbing (for bypass sequences)

Sprays are considered for source term mitigation in all categories with containment failure, except for cases in which the vessel has not breached. Calculations assume no sprays in the latter cases. For bypass sequences (SGTR and ISLOCA events) this characteristic represents whether or not the release is scrubbed by an overlying water pool.

The resulting release categories are provided in Sub-section 15.4.3.5 - Table 1, Release Category Definitions.

3.5.2. Source Term Analysis

3.5.2.1. Source Term definition

The source term represents the release to the environment, as a function of time, for the different isotope groups considered in the model. The source term analysis was performed using the MAAP4.0.7 code, which includes EPR specific-models. This analysis is documented in the AREVA-NP source term methodology document [Ref-1].

In MAAP, fission products are organised into 12 groups as follows:

GROUP 1 VAPOR (V): Noble gases (Xe + Kr), and Aerosol (A): All non-radioactive inert aerosols

GROUP 2 V & A: CsI + RbI

GROUP 3 V & A: TeO2

GROUP 4 V & A: SrO

GROUP 5 V & A: MoO2

GROUP 6 V & A: CsOH + RbOH

GROUP 7 V & A: BaO

GROUP 8 V & A: La2O3 + Pr2O3 + Nd2O3 + Sm2O3 + Y2O3

GROUP 9 V & A: CeO2 + NpO2 + PuO2

GROUP 10 V & A: Sb

GROUP 11 V & A: Te2

GROUP 12 V & A: UO2

Where: V = vapour, A = aerosol

The source term is the result of the MAAP analysis and presents the fraction of the initial core inventory which is released to the environment as a function of time.

3.5.2.2. Source Term Analysis Objectives

The objectives of the source term analysis are to:

- Characterise the source term associated with each release category.
- Perform analysis to determine the sensitivity of the source term to a number of key variables.

3.5.2.3. MAAP4.07 Analysis Specification

To achieve these objectives, a number of sequences were identified for analysis using MAAP4.0.7.

The source term analysis [Ref-1] was performed using the MAAP4.0.7 code, which includes US EPR specific models. The US analysis was supplemented by additional MAAP analyses performed specifically for the UK PSA2.

The first step in the source term analysis was to review the quantification results for the US EPR Level 2 PSA, to determine how best to model each Release Category with MAAP, and to specify the additional MAAP runs needed to achieve an optimum set of MAAP results for each Release Category.

The second step was to specify and perform the additional MAAP runs necessary to achieve the optimum MAAP results for each Release Category. These additional MAAP runs were performed, and are documented in the severe accident source term analysis [Ref-2].

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This additional modelling includes:

- Adjustment for differences between the US and UK allowable containment normal leakage rates
- Generation of organic iodine within containment.
- Reduction in the offsite releases due to operation of the annulus ventilation system
- Reduction in the offsite releases due to operation of the fuel/safeguards building ventilation systems
- Reduction in the offsite releases due to fission product deposition in the annulus and the fuel/safeguards buildings when ventilation is not operating

- Fission product scrubbing due to submergence of the point of release for both Large and Small ISLOCAs

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3.5.4. Source Term Release Fractions

The release fractions for all of the Release Categories except for Spent Fuel Pool accidents were calculated using MAAP results The post processing factors described above were then applied to the calculated fission product releases from the MAAP analyses.

The release fractions for Spent Fuel Pool accidents are obtained from Table A4-3 of NUREG-1738 [Ref-1]. These in turn come from NUREG/CR-4982 [Ref-2]. The smaller source term for La and Ce from NUREG/CR-4982 (compared with that from NUREG/CR-6451) are used based on the discussion and results in NUREG-1738:

SFP Release Fractions from NUREG-1738		
Nuclide	Symbol	Release Fraction
Noble Gases	Xe, Kr	1
Iodine	I	1
Caesium	Cs	1
Tellurium	Te	2.00E-02
Strontium	Sr	2.00E-03
Ruthenium	Ru	2.00E-05
Lanthanum	La	1.00E-06
Cerium	Ce	1.00E-06
Barium	Ba	2.00E-03

The species in the table above have been adjusted in the table below to match the fission product groups reported by MAAP. In addition, Group 5 (Mo), 10 (Sb) and 12 (UO2 etc) are not represented in the NRC studies. The release fraction for these groups is based on the Large LOCA case for RC802a. The table below provides a summary of the Spent Fuel Pool accident release fractions reported in Sub-section 15.4.4.3 - Table 1.

SFP Release Fractions adjusted to MAAP Groups		
Group	**Contents**	**Release Fraction**
1	Nobles	1
2	CsI, RbI	1
3	TeO2	2.00E-02
4	SrO	2.00E-03
5	MoO2	5.00E-02
6	CsOH, RbOH	1
7	BaO	2.00E-03
8	LaO2 etc	1.00E-06
9	CeO2	1.00E-06
10	Sb	1.00E-01
11	Te2	2.00E-02
12	UO2 etc	2.00E-05

The Release Fractions for all of the events analysed in the UK EPR PSA2 are summarised in Sub-section 15.4.4.2 -Table 1.

SUB-SECTION 15.4.3.5 - TABLE 1

Release Category Definitions

Release Category	Description
RC101	No containment failure, credit taken for deposition in the annulus and fuel/safeguards buildings
RC102	No containment failure, credit taken for filtration in the annulus and fuel/safeguards building ventilation systems
RC200	Containment fails before vessel breach due to isolation failure, melt retained in vessel, with containment sprays
RC201	Containment fails before vessel breach due to isolation failure, melt retained in vessel, without containment sprays
RC202	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex-vessel, with containment sprays
RC203	Containment fails before vessel breach due to isolation failure, melt released from vessel, with MCCI, melt not flooded ex-vessel, without containment sprays
RC204	Containment fails before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex-vessel with containment sprays
RC205	Containment failures before vessel breach due to isolation failure, melt released from vessel, without MCCI, melt flooded ex-vessel without containment sprays
RC206	Small containment failure due to failure to isolate 2” or smaller lines
RC301	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex-vessel, with containment sprays
RC302	Containment fails before vessel breach due to containment rupture, with MCCI, melt not flooded ex-vessel, without containment sprays
RC303	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex-vessel, with containment sprays
RC304	Containment fails before vessel breach due to containment rupture, without MCCI, melt flooded ex-vessel, without containment sprays
RC401	Containment failures after breach and up to melt transfer to the spreading area due to containment rupture, with MCCI, without debris flooding, with containment spray
RC402	Containment failures after breach and up to melt transfer to the spreading area due to containment rupture, with MCCI, without debris flooding, without containment spray
RC403	Containment failures after breach and up to melt transfer to the spreading area due to containment rupture, without MCCI, with debris flooding, with containment spray
RC404	Containment failures after breach and up to melt transfer to the spreading area due to containment rupture, without MCCI, with debris flooding, without containment spray
RC501	Long term containment failure during and after debris quench due to rupture, with MCCI, without debris flooding, with containment sprays
RC502	Long term containment failure during and after debris quench due to rupture, with MCCI, without debris flooding, without containment sprays

Release Category	Description
RC503	Long term containment failure during and after debris quench due to rupture, without MCCI, with debris flooding, with containment sprays
RC504	Long term containment failure during and after debris quench due to rupture, without MCCI, with debris flooding, without containment sprays
RC601	Long term containment failure due to basemat failure, without debris flooding, with containment sprays
RC602	Long term containment failure due to basemat failure, without debris flooding, without containment sprays
RC701	Steam Generator Tube Rupture with Fission Product Scrubbing
RC702	Steam Generator Tube Rupture without Fission Product Scrubbing
RC801	Large Interfacing System LOCA with Fission Product Scrubbing
RC802	Small or Large Interfacing System LOCA, without Fission Product Scrubbing, direct release to environment
RC802a	Small or Large Interfacing System LOCA, without Fission Product Scrubbing, fission product deposition in fuel/safeguards building
RC802b	Small or Large Interfacing System LOCA, without Fission Product Scrubbing, fission product filtration in annulus and fuel/safeguards building ventilation systems

SUB-SECTION 15.4.3.5 - TABLE 2

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SUB-SECTION 15.4.3.5 - TABLE 3

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3.6. SENSITIVITY AND UNCERTAINTY TREATMENT

3.6.1. Containment Event Tree

The sensitivity studies performed in support of the UK EPR Level 2 PSA focused on the phenomenological aspects of the model. These sensitivity studies were carried out by investigating the impact on the results of assuming that a modelled phenomenon was either sure to occur (probability set equal to 1.0) or sure not to occur (probability set equal to 0.0). The base probabilities assigned to phenomenological events in the PSA represent a level of confidence (or degree of belief) in the phenomena occurring or not. These base probability values were assigned as a result of the detailed phenomenological evaluations performed (see section 3.3). The sensitivity study approach adopted therefore aims to provide an indication of the relative importance of these evaluations and the corresponding probability assignments incorporated into the UK EPR Level 2 PSA model. For the purposes of reporting, events are judged to be significant if they can lead to a factor of two increase or decrease in the target risk metric when set equal to 1 or 0. The risk metrics targeted for these studies were Large Release and Large Early Release frequencies (the definition of these metrics is discussed in section 4.4.).

A number of sensitivity studies were also carried out to assess the impact of groups of human actions or particular functional events in the CETs. Note that studies of the impact of maintenance unavailabilities on Large Release and Large Early Release frequencies are presented in Sub-chapter 15.7

As a complement to the sensitivity studies performed, an integrated uncertainty analysis was also carried out. This uncertainty assessment was performed by propagation of all parameterised uncertainties, including phenomenological uncertainties and other basic event parametric uncertainties. The uncertainty propagation was performed using the uncertainty analysis capability in RiskSpectrum and used the integrated Level 1 – Level 2 PSA linked fault tree model. Parametric uncertainties associated with both Level 1 and Level 2 PSA basic events were incorporated in this uncertainty assessment.

The basis for the input uncertainty distributions for systems related basic events and operator actions is discussed in the relevant sections of the Level 1 PSA description.

The basis for the input uncertainty distributions for the phenomenological events follows directly from the results of the phenomenological evaluations performed to support the Level 2 PSA. Phenomenological events are identified in the PSA database by use of the prefix “L2PH”. Discrete uncertainty distributions were used for these basic events. The distribution form chosen is double delta, implying that a probability is assigned for each of two deterministic outcomes for this type of basic event; there is a probability that the event is sure to occur (relative frequency of one) and another that it is sure not to occur (relative frequency of zero). For each event, the probability of the “sure occurrence” outcome is, therefore, equal to the mean value of the basic event.

The double-delta distribution type described above is appropriate for Level 2 PSA phenomenological events since, generally, these events are expected to have deterministic, but unknown, outcomes rather than random outcomes. The results of the sensitivity and uncertainty evaluations are presented in section 4.5.

3.6.2. Source Term Sensitivity Analysis

Two sensitivity cases were identified which investigated:

- Importance of passive deposition in the annulus for intact containment sequences
- Importance of passive deposition in fuel / safeguards buildings for ISLOCAs

3.6.2.1. Importance of passive deposition in the annulus for intact containment sequences

Sub-section 15.4.3.6 - Figure 1 shows the importance of passive deposition in the annulus for intact containment sequences. This figure compares the CsI release fraction for the containment intact source term MAAP run (ST1.10 at 20 hours), where containment leakage is released directly to the environment, against the release fraction for the containment intact source term MAAP run (ST1.10g at 20 hours), where containment leakage is released into the annulus, no ventilation is running, and only deposition is credited.

The stable release fraction for the direct release to the environment is 3.0E-5, while the release fraction crediting annulus deposition is 1.33E-6. This gives an effective filtration factor for annulus deposition of 23.

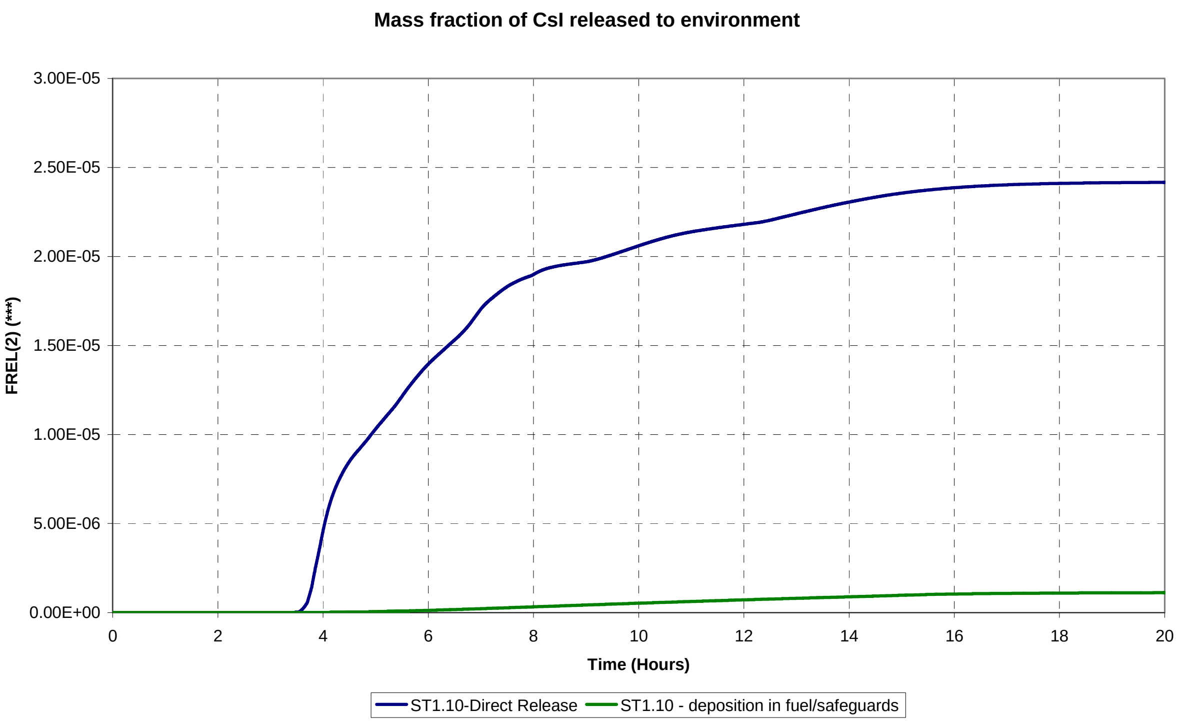
3.6.2.2. Importance of passive deposition in fuel / safeguards buildings for ISLOCAs

Sub-section 15.4.3.6 - Figure 2 shows the importance of passive deposition in the fuel/safeguards buildings for ISLOCA sequences. This figure compares the release fraction for CsI for the containment intact source term MAAP run (ST3.1), where the ISLOCA is released directly to the environment, against the release fraction for the ISLOCA source term MAAP run (ST3.1c),where containment leakage is released into the fuel/safeguards buildings, no ventilation is running, and only deposition is credited.

The stable release fraction for the direct release to the environment is 0.94, while the release fraction crediting annulus deposition is 0.15. This gives an effective filtration factor for annulus deposition of ~6.

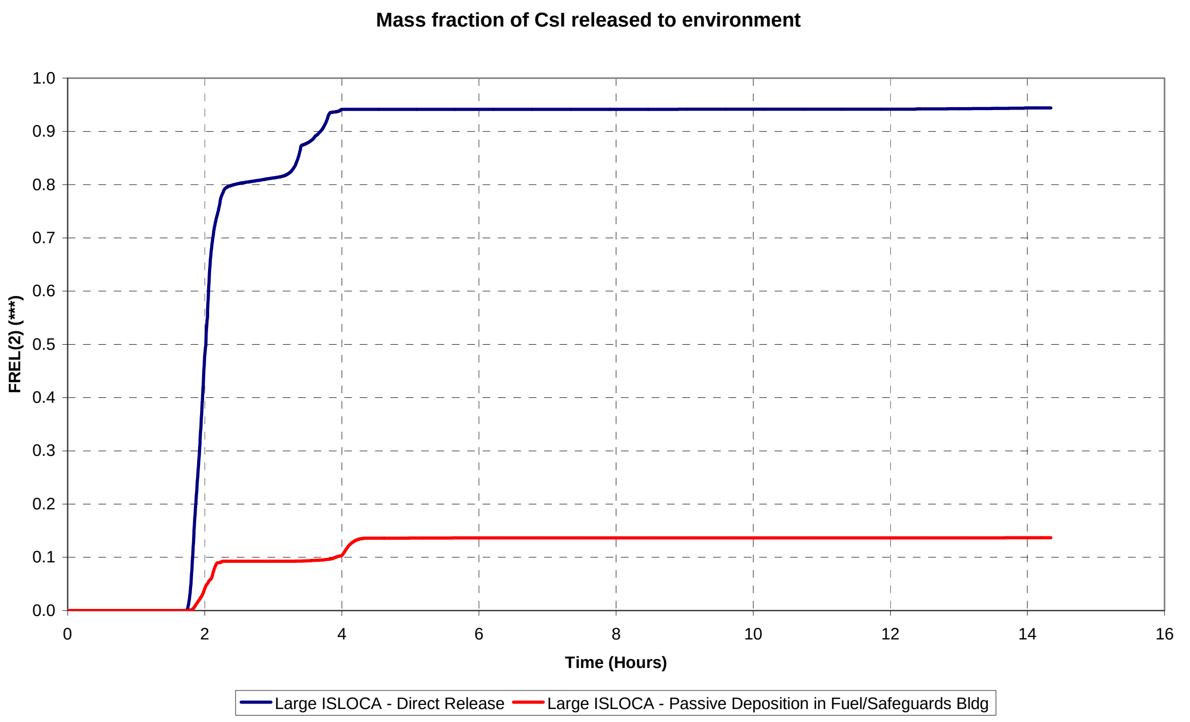
SUB-SECTION 15.4.3.5 - FIGURE 1

UK EPR PSA2 – Containment Intact with/without Passive Deposition



SUB-SECTION 15.4.3.5 - FIGURE 2

UK EPR PSA2 – ISLOCA with/without Passive Deposition



4. LEVEL 2 PSA – RESULTS

4.1. CET QUANTIFICATION AND RELEASE FREQUENCIES – PLANT STATES A AND B

4.1.1. Introduction and overview of quantification approach

The CET quantification for operating states A and B was performed by quantifying the end state frequencies associated with each release category (RC). This quantification was implemented in the RiskSpectrum Professional software used for development of the UK EPR PSA by defining MCS and consequence analysis cases[5] for each RC. The consequence analysis cases were used to generate a minimal cutset list for all the sequences in each RC. MCS analysis cases were used to post-process the results for each RC to generate the final minimal cutset list and frequencies for each RC. The post-processing step was used to implement dependency processing for cutsets containing multiple human errors.

Sub-section 15.4.4.1 - Table 1 presents the frequencies obtained for each RC.

Quantification of the frequencies of the individual CDES contributing to the total core damage frequency for states A and B was also performed. The results of the CDES quantification are presented in Sub-section 15.4.4.1 - Table 2.

All the percentages presented in this sub-section are linked to the CDF at power.

4.1.2. CDES results

As seen in Sub-section 15.4.4.1 - Table 2 the highest contributing CDES are SS (38.5% of CDF), TR (21.3% of CDF), SL (14.7% of CDF), TP (8.3% of CDF), SP (3.7% of CDF), RV (2.4% of CDF) and AT and TRD (2% of CDF each)

SS is a high pressure CDES, representing seal LOCA core damage sequences. Sub-CDES are defined for SS to account for unavailabilities inherent in the Level 1 PSA core damage sequences. In the case of SS, the main contributing sub-CDES are:

- SS-SB1 (SS following internal hazard affecting safeguard building 1), which has a 11.5% contribution to CDF;
- SS-CCALL (SS with loss of all RRI [CCWS] trains), which has a 9.3% contribution to CDF;
- SS-CC2 (SS with loss of train 2 of RRI [CCWS]), which has a 9.3% contribution to CDF;
- and SS-LOOP (SS with short LOOP), which has a 6.7% contribution to CDF.

[5] Here the terms "MCS Analysis Case" and "Consequence Analysis Case" are used in a very specific sense to refer to the corresponding types of analysis provided by the RiskSpectrum software. A consequence analysis allows generation of a single minimal cutset list for all sequences assigned to a particular end state. MCS analysis cases can take the minimal cutsets generated by a Consequence Analysis Case and provide a post-processed output after applying a user-defined set of rules to the input minimal cutset list.

TR is a high pressure CDES, with the RCP [RCS] intact. Sub-CDES are defined for TR to account for system unavailabilities inherent in the Level 1 PSA core damage sequences. In the case of TR, the main contributing sub-CDES are:

- TR (TR without other unavailabilities), which has a 13.8% contribution to CDF;
- TR-CCALL (TR with all RRI [CCWS] trains failed), which has a 3.1% contribution to CDF;
- TR-SWGB (TR following fire in the switchgear building), which has a 2.2% contribution to CDF;
- and TR-CC2 (TR with RRI [CCWS] train 2 unavailable), which has a 1.6% contribution to CDF.

The CDES summarised above are the main contributors to the core damage profile in terms of frequency. A further few CDES, contributing only 2% of CDF and below are not discussed in this section.

4.1.3. Release category results

At power the release category frequencies presented in Sub-section 15.4.4.1 - Table 1 show that the conditional probability of the containment remaining intact, isolated and not bypassed following core damage is 89.2%. The remaining 10.8% of CDF consists of 1% of CDF (6.60E-9 /yr) with failure to isolate the containment, 2% of CDF (1.29E-8 /yr) with the containment bypassed, 2% of CDF (1.25E-8 /yr) with containment failure due to overpressure or energetic events in the short term (before melt release from the reactor pit), and 5.7% of CDF (3.65E-8 /yr) with containment failure due to overpressure or energetic events in the longer term (after melt release from the reactor pit).

Note that the results presented indicate a conditional probability of 94.8% of the containment remaining intact for 24 hours after the initiating event.

SUB-SECTION 15.4.4.1 - TABLE 1

Summary of Release Category Frequency Results for Operating States A and B

Release category	Frequency	%	Not intact	**LERF** – note (1)	**LRF** - note (1)	Bypass	Not isolated
RC101	1.44E-07	22.64%					
RC102	4.24E-07	66.59%					
RC200	2.94E-10	0.05%	2.94E-10	2.94E-10	2.94E-10		2.94E-10
RC201	9.80E-11	0.02%	9.80E-11	9.80E-11	9.80E-11		9.80E-11
RC202	1.67E-12	0.00%	1.67E-12	1.67E-12	1.67E-12		1.67E-12
RC203	2.77E-13	0.00%	2.77E-13	2.77E-13	2.77E-13		2.77E-13
RC204	1.29E-09	0.20%	1.29E-09	1.29E-09	1.29E-09		1.29E-09
RC205	3.74E-10	0.06%	3.74E-10	3.74E-10	3.74E-10		3.74E-10
RC206	4.54E-09	0.71%	4.54E-09	4.54E-09	4.54E-09		4.54E-09
RC301	7.86E-12	0.00%	7.86E-12	7.86E-12	7.86E-12		
RC302	5.47E-12	0.00%	5.47E-12	5.47E-12	5.47E-12		
RC303	9.99E-09	1.57%	9.99E-09	9.99E-09	9.99E-09		
RC304	2.47E-09	0.39%	2.47E-09	2.47E-09	2.47E-09		
RC401	2.64E-11	0.00%	2.64E-11	2.64E-11	2.64E-11		
RC402	8.37E-12	0.00%	8.37E-12	8.37E-12	8.37E-12		
RC403	1.21E-09	0.19%	1.21E-09	1.21E-09	1.21E-09		
RC404	1.09E-09	0.17%	1.09E-09	1.09E-09	1.09E-09		
RC501	5.66E-13	0.00%	5.66E-13				
RC502	3.83E-11	0.01%	3.83E-11		3.83E-11		
RC503	1.19E-09	0.19%	1.19E-09				
RC504	3.24E-08	5.09%	3.24E-08		3.24E-08		
RC602	5.72E-10	0.09%	5.72E-10		5.72E-10		
RC701	4.14E-09	0.65%	4.14E-09	4.14E-09	4.14E-09	4.14E-09	
RC702	5.01E-09	0.79%	5.01E-09	5.01E-09	5.01E-09	5.01E-09	
RC802	3.70E-09	0.58%	3.70E-09	3.70E-09	3.70E-09	3.70E-09	
Total (frequency)	6.36E-07	100.00%	6.85E-08	3.43E-08	6.73E-08	1.29E-08	6.60E-09
Total (frequency)	100.00%	100.00%	10.76%	5.39%	10.58%	2.02%	1.04%

Notes: (1) LERF (Large Early Release Frequency) and LRF (Large Release Frequency), based on a CS-137 release > 100TBq. See section 4.3.

SUB-SECTION 15.4.4.1 - TABLE 2

Summary of CDES Contributions to CDF for Operating States A and B

CDES	Key Characteristics of CDES	Contribution (% of CDF at power)	Contributing sub-CDES (if applicable - % of CDF at power)
SS	High pressure, seal LOCA (Sub-CDES account for system unavailabilities)	38.5%	SS-SB1 (11.5% of CDF), SS-CCALL (9.3%), SS-CC2 (9.3%), SS-LOOP (6.7%)
TR	High pressure, RCP [RCS] intact (Sub-CDES account for system unavailabilities)	21.3%	TR (13.8% of CDF), TR-CCALL (3.1%), TR-SWGB (2.2%), TR-CC2 (1.6%)
SL	High pressure, small LOCA	14.7%	N/A
TP	High pressure, RCP [RCS] intact, long LOOP	8.3%	N/A
SP	High pressure, seal LOCA, long LOOP	3.7%	
RV	Low pressure, RPV ruptured	2.4%	N/A
AT	High pressure, RCP [RCS] intact, ATWS	2%	N/A
TRD	High pressure, RCP [RCS] intact, secondary depressurised	2%	N/A
SPD	High pressure, seal LOCA, long LOOP, secondary depressurised	1.6%	N/A
ML	Low pressure, medium LOCA	1.5%	N/A
SSD	High pressure, seal LOCA, secondary depressurised (Sub-CDES account for system unavailabilities)	1.5%	SSD-SB1 (0.7% of CDF), SSD-CC1 (0.5%), SSD-CCA (0.2%)
IS	Containment bypass	0.6%	N/A
SG	Containment bypass	0.7%	N/A
SG2	Containment bypass	0.7%	N/A
SLD	High pressure, small LOCA, secondary depressurised	0.2%	N/A
LL	Low pressure, large LOCA	0.2%	N/A
ATI	High pressure, containment overpressure sequence	0.1%	N/A

4.2. CET QUANTIFICATION AND RELEASE FREQUENCIES – PLANT STATES C, D AND E

4.2.1. Introduction

The CET quantification for operating states C, D and E used the same method as described for states A and B, except that separate results were generated for each of these states, whereas for the full power Level 2 PSA, combined results were generated for states A and B.

Sub-section 15.4.4.2 - Table 1 presents the frequencies obtained for each RC for each of states C, D and E.

All the percentages expressed in this sub-section are linked to the CDF in shutdown states (state by state).

4.2.2. Release category results

The release category frequencies presented in Sub-section 15.4.4.2 - Table 1 show that the following conditional probabilities of the containment remaining intact, isolated and not bypassed following core damage in states C and D

- 82.95% in state C (the equipment hatch is open during some sub-states of state C but in most of the sequences can be reclosed)

- 98.85% in state D (the equipment hatch is closed during state D).

Note that the containment is open throughout state E.

In both state C and state D the conditional probability of the containment not being intact is dominated by the large loss of isolation release categories, which are also used to model the containment open status in sub-states Ca2 and Ca3 of state C.

SUB-SECTION 15.4.4.2 - TABLE 1

Summary of Release Category Frequency[6] Results for Operating States C, D and E

Release Category[7]	Plant states C	Plant state D	Plant state E
RC101	3.46E-09	1.43E-09	
RC102	2.79E-08	3.24E-08	
RC200[8]	4.61E-10	2.21E-11	1.25E-10
RC201	1.54E-10	7.35E-12	4.15E-11
RC202	6.96E-13	2.46E-14	2.09E-13
RC203	2.41E-14	1.47E-15	1.76E-15
RC204	5.13E-10	2.49E-11	1.27E-10
RC205	6.95E-11	4.25E-12	3.43E-12
RC206	6.35E-11	5.09E-12	
RC301	2.03E-13		
RC302	1.78E-13	1.88E-13	1.56E-15
RC303	2.04E-10		
RC304	4.51E-09		
RC401	2.65E-13		
RC402	2.43E-15		
RC403	2.08E-11		
RC404	1.87E-12		
RC501	5.46E-14	3.00E-14	
RC502	5.23E-13	7.61E-13	
RC503	6.14E-11	1.95E-11	
RC504	2.79E-10	2.40E-10	
RC602	3.35E-11	5.19E-11	
RC802	6.95E-11	1.59E-11	3.97E-11

[6] RCs which have zero frequency in all shutdown states are omitted from the table.
[7] Shaded rows are large releases. Darker shading (categories RC502, 504 and 602) indicates those RCs which are large release but not large early release due to timing.
[8] Frequency generated for RC200 by applying the same proportionality as in the at power case.

4.3. SOURCE TERM ANALYSIS RESULTS

4.3.1. Release fractions

The results of the source term analysis for the UK EPR PSA2, for all of the Release Categories in the analysis, are contained in Sub-section 15.4.4.3 - Table 1.

4.3.2. Fission Product Inventories and Source Term

In order to provide an effective interface between the Level 2 and Level 3 PSA analyses, the release fractions summarised above can be combined with the fission product inventories of the core and spent fuel pool, so that the total activity of the fission products released from the containment can be calculated.

4.3.2.1. UK EPR Core Inventory

The fission product inventory for the UK PSA2 is derived from the spreadsheet entitled “Core Inventory for EPR Level 3 PSA.xls” that was supplied as part of the documentation package for the US EPR Level 3 PSA for DC [Ref-1]. The description of how these cases were derived is presented in section 2.3.3.4.

The core inventory is presented in Sub-section 15.4.4.3 - Table 2. This spreadsheet presents a summary of the isotopic content for both the bounding and equilibrium cases. The Bounding Case was chosen as the “highest of the highest” isotopic activities for all combinations of enrichment and fuel burnup expected for the EPR. The Equilibrium Core Inventory represents more of a “best estimate” value for each isotope presented, and is therefore appropriate for use in PSA.

The remainder of the table provides values for the nominal core inventory for NPP TMI-1 [Ref-2], and the inventory for the same plant were it to be operated at 4612 MW. The final three columns in the table provide values for the ratio of the bounding to equilibrium core inventory, as well as a comparison of the bounding and equilibrium core inventories to the corrected TMI-1 values. These ratios are provided to allow benchmarking of the UK EPR core inventory.

4.3.2.2. Spent Fuel Pool Inventory

The fission product inventory for the Spent Fuel Pool (SFP) accident is calculated in Sub-section 15.4.4.3 - Table 3.

The basic inventory data in this table comes from Table A4-1 of NUREG 1738 [Ref-1]. Since the reference inventories in that table are for Millstone 1, they are scaled up in accordance with the ratio of the nominal power levels of the reactors. The scale up factor of 1.7 x 4500/3441 is composed of two parts. The factor of 1.7 as used to ratio the original Millstone power level to that of the “large BWR” with a power level of 3441 MW discussed in Appendix 4 of NUREG 1738 [Ref-1]. The factor of 4500/3441 is used to ratio the power level of that BWR to that of the UK EPR.

There is no adjustment for complete fuel unload as the SFP accident is assumed to occur while fuel is in core. The values for the 30 day and 1 year SFP inventories correspond to a spent fuel pool load after the 11th refuelling outage, with 1/3 core offloaded each outage. The "30 day" values assume that 30 days has elapsed since last discharge, and this value is adjusted for core power in the corresponding column.

The column called "Fresh discharge" calculates values for fission product inventory to be used as the source term for the SFP accident. The value in this column uses the 1 year reference inventory, adjusts it for the EPR power level, and adds 1/3 of the MAAP core inventory to account for the portion of the core discharged to the SFP immediately prior to the SFP accident.

Only the active nuclides are considered in this analysis. Thus, when the associated MAAP fission product groups are indicated in this table, only specified isotopes are compared.

The last two columns of the table calculate the ratio of the 30 day adjusted and fresh discharge fission product inventories, respectively, with the MAAP core inventory. This has been done to facilitate direct conversion of the MAAP core inventory to the SFP inventories in subsequent analyses.

4.3.2.3. UK EPR Fission Product Source Term

The release fraction of Sub-section 15.4.4.3 - Table 1 can be combined with core and spent fuel pool inventories in Sub-section 15.4.4.3 - Table 2 and Sub-section 15.4.4.3 - Table 3, respectively, and the fission product source term for each release category can be calculated.

This has been done using an Excel spreadsheet, and the results of this integration are contained in Sub-section 15.4.4.3 - Table 4.

4.3.3. Source Term Release Energies and Locations

The release energies and locations and timings associated with the source term in each release category are presented in Sub-section 15.4.4.3 - Table 5. This information is provided as an input to the Level 3 PSA evaluation (Sub-chapter 15.5).

The release start and end times are derived from analysis of the source term MAAP run results, as well as the phenomenological evaluation governing that failure mode.

The energy release rate is conservatively assumed to be the highest energy release rate observed during the release duration. This energy release rate is assumed to be constant for the duration of the release.

SUB-SECTION 15.4.4.3 - TABLE 1

UK EPR PSA2 – Release Fractions

Release Category	Containment Failure Mechanism	Debris flood	Sprays	FREL(1) Xe. Kr	FREL(2) CsI. RbI	FREL(2a) CH3I	FREL(3) TeO2	FREL(4) SrO	FREL(5) MoO2	FREL(6) CsOH. RbOH	FREL(7) BaO	FREL(8) La etc 03	FREL(9) CeO2. PuO2. NpO2	FREL(10) Sb	FREL(11) Te2	FREL(12) UO2
RC 101	None - deposition in annulus & building	Yes	No	2.81E-03	1.34E-06	4.21E-06	1.59E-06	3.39E-07	2.07E-06	1.09E-06	1.13E-06	1.01E-08	4.20E-08	2.80E-06	4.02E-10	5.74E-11
RC 102	None - annulus and building ventilation	Yes	No	9.41E-03	4.90E-07	1.41E-07	6.15E-08	1.01E-08	5.31E-08	8.32E-08	2.88E-08	3.40E-10	8.79E-10	7.36E-07	1.05E-08	2.94E-12
RC 200	Isolation failure - in-vessel recovery	Yes	Yes	2.42E-01	3.11E-02	3.64E-04	2.70E-03	1.80E-05	3.70E-04	2.67E-02	1.15E-04	6.77E-07	3.15E-06	2.44E-03	0.00E+00	0.00E+00
RC 201	Isolation failure - in-vessel recovery	Yes	No	3.63E-01	1.03E-01	5.44E-04	7.64E-03	7.85E-05	1.06E-03	9.55E-02	4.13E-04	3.37E-06	1.72E-05	9.58E-03	0.00E+00	0.00E+00
RC 202	Isolation failure	No	Yes	9.16E-01	3.99E-03	1.37E-03	1.65E-03	1.15E-03	3.13E-03	2.41E-03	4.19E-03	6.02E-05	2.64E-04	1.06E-02	1.36E-06	2.35E-05
RC 203	Isolation failure	No	No	9.75E-01	1.12E-01	1.46E-03	3.77E-02	1.17E-02	1.79E-02	9.83E-02	2.50E-02	5.03E-04	1.22E-03	4.31E-02	1.41E-05	1.67E-05
RC 204	Isolation failure	Yes	Yes	9.92E-01	1.32E-02	1.49E-03	6.95E-03	1.05E-03	3.60E-03	7.02E-03	5.37E-03	5.62E-05	2.92E-04	2.70E-02	6.33E-07	2.19E-05
RC 205	Isolation failure	Yes	No	1.00E+00	1.16E-01	1.50E-03	3.81E-02	8.74E-03	2.12E-02	1.00E-01	1.96E-02	3.78E-04	8.01E-04	1.01E-01	4.04E-07	1.35E-05
RC 206	All small isolation failures (<2")	Yes	No	1.85E-01	5.60E-03	2.78E-04	7.65E-03	1.24E-03	7.25E-03	4.98E-03	4.20E-03	5.49E-05	1.80E-04	8.99E-03	5.13E-07	3.42E-07
RC 301	Early failure	No	Yes	9.74E-01	2.26E-02	1.46E-03	2.14E-02	1.44E-04	8.38E-03	1.04E-02	1.86E-03	1.38E-05	1.01E-04	7.95E-02	4.37E-05	1.36E-05
RC 302	Early failure	No	No	9.81E-01	5.10E-02	1.47E-03	1.52E-02	8.03E-05	3.43E-03	3.06E-02	1.39E-03	5.18E-06	3.00E-05	9.35E-02	2.56E-04	1.27E-06
RC 303	Early failure	Yes	Yes	9.99E-01	3.13E-02	1.50E-03	9.29E-03	3.97E-05	1.81E-03	1.95E-02	5.04E-04	4.93E-06	7.87E-05	7.33E-02	2.40E-06	9.44E-06
RC 304	Early failure	Yes	No	1.00E+00	6.85E-02	1.50E-03	2.42E-02	1.41E-04	5.94E-03	4.12E-02	2.24E-03	1.59E-05	2.23E-04	1.36E-01	2.00E-05	2.27E-05
RC 401	Intermediate failure	No	Yes	9.78E-01	1.22E-02	1.47E-03	2.62E-03	1.57E-04	2.79E-03	6.25E-03	1.52E-03	1.62E-05	1.07E-04	7.61E-02	4.65E-05	1.39E-05
RC 402	Intermediate failure	No	No	9.84E-01	1.47E-02	1.48E-03	6.69E-03	2.81E-04	3.86E-03	8.07E-03	3.88E-03	3.67E-05	1.99E-04	1.82E-01	1.27E-04	1.38E-05
RC 403	Intermediate failure	Yes	Yes	1.00E+00	8.35E-03	1.50E-03	2.40E-03	1.36E-04	1.32E-03	3.36E-03	2.64E-03	1.46E-05	6.35E-05	2.71E-02	1.32E-08	6.78E-08
RC 404	Intermediate failure	Yes	No	1.00E+00	2.47E-02	1.50E-03	6.07E-03	1.22E-04	3.46E-03	1.10E-02	1.97E-03	1.49E-05	2.26E-04	1.11E-01	1.71E-05	2.30E-05
RC 50*	Late failure (phase 1)	N/A	N/A	6.29E-04	1.97E-05	2.96E-08	3.52E-05	8.32E-06	4.41E-05	1.83E-05	2.37E-05	2.74E-07	6.45E-07	5.16E-05	0.00E-00	0.00E-00
RC 501	Late failure	No	Yes	7.82E-01	5.72E-05	1.17E-03	3.69E-05	8.44E-06	4.52E-05	2.69E-05	2.67E-05	2.88E-07	8.43E-07	1.04E-03	1.17E-06	4.41E-09
RC 502	Late failure	No	No	9.95E-01	7.80E-04	1.49E-03	5.30E-05	7.38E-06	4.36E-05	4.11E-04	2.42E-05	2.16E-07	7.03E-07	1.73E-02	1.27E-05	5.98E-09
RC 503	Late failure	Yes	Yes	1.00E+00	1.08E-04	1.50E-03	3.96E-05	8.45E-06	4.43E-05	2.71E-05	2.40E-05	2.83E-07	7.32E-07	2.38E-04	3.78E-06	2.45E-09
RC 504	Late failure	Yes	No	1.00E+00	4.08E-04	1.50E-03	5.12E-05	8.45E-06	4.43E-05	6.94E-05	2.40E-05	2.83E-07	7.32E-07	6.13E-04	8.76E-06	2.45E-09
RC 602	Basemat failure	No	No	9.95E-01	7.80E-04	1.49E-03	5.30E-05	7.38E-06	4.36E-05	4.11E-04	2.42E-05	2.16E-07	7.03E-07	1.73E-02	1.27E-05	5.98E-09
RC 701	SGTR scrubbed	Yes	No	1.09E-01	4.21E-03	8.17E-05	5.74E-03	6.00E-04	4.80E-03	4.35E-03	2.72E-03	2.25E-05	1.12E-04	6.94E-03	2.25E-07	5.30E-08
RC 702	SGTR unscrubbed	Yes	No	1.09E-01	8.41E-02	1.63E-04	1.15E-01	1.20E-02	9.60E-02	8.70E-02	5.45E-02	4.49E-04	2.24E-03	1.39E-01	4.51E-06	1.06E-06
RC 801	ISLOCA scrubbed	Yes	No	1.00E+00	9.43E-03	1.50E-04	9.35E-03	1.43E-03	6.94E-03	9.40E-03	4.26E-03	6.30E-05	4.60E-04	9.49E-03	6.96E-07	2.59E-06
RC 802	Large ISLOCA unscrubbed. no deposition	Yes	No	1.00E+00	9.43E-01	1.50E-03	9.35E-01	1.43E-01	6.94E-01	9.40E-01	4.26E-01	6.30E-03	4.60E-02	9.49E-01	6.96E-05	2.59E-04
	Small ISLOCA unscrubbed. no deposition	Yes	No	9.05E-01	9.17E-01	1.36E-03	7.91E-01	9.28E-02	5.73E-01	8.64E-01	3.92E-01	5.00E-03	2.33E-02	9.01E-01	2.75E-04	1.96E-05
RC 802a	Large ISLOCA. unscrubbed. deposition in building	Yes	No	9.84E-01	1.37E-01	1.48E-03	1.05E-01	1.60E-02	2.60E-02	1.37E-01	4.62E-02	6.57E-04	2.77E-03	6.19E-02	6.13E-06	1.99E-05
	Small ISLOCA. unscrubbed. deposition in building	Yes	No	8.18E-01	1.78E-01	1.23E-03	1.35E-01	2.36E-02	7.58E-02	1.78E-01	6.70E-02	9.07E-04	3.70E-03	1.15E-01	4.43E-06	3.67E-06
RC 802b	Large ISLOCA. unscrubbed. ventilation in building	Yes	No	1.00E+00	9.43E-04	1.50E-05	9.35E-04	1.43E-04	6.94E-04	9.40E-04	4.26E-04	6.30E-06	4.60E-05	9.49E-04	6.96E-08	2.59E-07
	Small ISLOCA. unscrubbed. ventilation in building	Yes	No	9.05E-01	9.17E-04	1.36E-05	7.91E-04	9.28E-05	5.73E-04	8.64E-04	3.92E-04	5.00E-06	2.33E-05	9.01E-04	2.75E-07	1.96E-08
RC SFP	Spent fuel pool accident: SFP accident inventory multiplier	N/A	N/A	1.00E+00	9.99E-01	1.50E-03	2.00E-02	2.00E-03	5.00E-02	1.00E+00	2.00E-03	1.00E-06	1.00E-06	1.00E-01	2.00E-02	2.00E-05

Note: For the RC 50* and RC 602 series, the PSA Level 3 is required to take account of a first phase of release due to containment leakage. It represents an important release for aerosols (except for CsI).

SUB-SECTION 15.4.4.3 - TABLE 2

{CCI removed}

{CCI removed}

a

SUB-SECTION 15.4.4.3 - TABLE 3

{CCI removed}

{CCI removed}

a

SUB-SECTION 15.4.4.3 - TABLE 4

UK EPR PSA2 – Fission Product Source Terms

				Release (Bq)												
Isotope	**Core Inventories (Bq)**	**SFP Inventories (Bq)**	**MAAP Fission Product Group**	**RC-101**	**RC-102**	**RC-200**	**RC-201**	**RC-202**	**RC-203**	**RC-204**	**RC-205**	**RC-206**	**RC-301**	**RC-302**	**RC-303**	**RC-304**
Kr-85	**{CCI}[a]**	{CCI}[a]	1	1.33E+14	4.45E+14	1.15E+16	1.72E+16	4.34E+16	4.62E+16	4.70E+16	4.74E+16	8.78E+15	4.61E+16	4.65E+16	4.73E+16	4.74E+16
Kr-85m	**{CCI}[a]**		1	3.86E+15	1.29E+16	3.33E+17	4.98E+17	1.26E+18	1.34E+18	1.36E+18	1.37E+18	2.54E+17	1.34E+18	1.35E+18	1.37E+18	1.37E+18
Kr-87	**{CCI}[a]**		1	7.56E+15	2.53E+16	6.52E+17	9.75E+17	2.46E+18	2.62E+18	2.67E+18	2.69E+18	4.98E+17	2.62E+18	2.64E+18	2.69E+18	2.69E+18
Kr-88	**{CCI}[a]**		1	1.07E+16	3.58E+16	9.24E+17	1.38E+18	3.49E+18	3.72E+18	3.78E+18	3.81E+18	7.06E+17	3.71E+18	3.74E+18	3.81E+18	3.81E+18
Xe-133	**{CCI}[a]**	{CCI}[a]	1	2.60E+16	8.70E+16	2.24E+18	3.35E+18	8.47E+18	9.02E+18	9.18E+18	9.25E+18	1.71E+18	9.01E+18	9.08E+18	9.24E+18	9.25E+18
Xe-135	**{CCI}[a]**		1	8.42E+15	2.82E+16	7.26E+17	1.09E+18	2.74E+18	2.92E+18	2.97E+18	3.00E+18	5.55E+17	2.92E+18	2.94E+18	2.99E+18	3.00E+18
I-131	**{CCI}[a]**	{CCI}[a]	2	2.53E+13	2.87E+12	1.43E+17	4.71E+17	2.44E+16	5.16E+17	6.68E+16	5.33E+17	2.68E+16	1.09E+17	2.39E+17	1.49E+17	3.19E+17
I-132	**{CCI}[a]**	{CCI}[a]	2	3.66E+13	4.15E+12	2.07E+17	6.82E+17	3.53E+16	7.47E+17	9.66E+16	7.71E+17	3.87E+16	1.58E+17	3.45E+17	2.16E+17	4.61E+17
I-133	**{CCI}[a]**		2	5.14E+13	5.83E+12	2.91E+17	9.58E+17	4.96E+16	1.05E+18	1.36E+17	1.08E+18	5.44E+16	2.22E+17	4.85E+17	3.03E+17	6.48E+17
I-134	**{CCI}[a]**		2	5.65E+13	6.42E+12	3.20E+17	1.05E+18	5.45E+16	1.15E+18	1.49E+17	1.19E+18	5.98E+16	2.44E+17	5.34E+17	3.34E+17	7.12E+17
I-135	**{CCI}[a]**		2	4.79E+13	5.44E+12	2.72E+17	8.93E+17	4.62E+16	9.78E+17	1.26E+17	1.01E+18	5.07E+16	2.07E+17	4.52E+17	2.83E+17	6.03E+17
Te-127	**{CCI}[a]**	{CCI}[a]	3	8.05E+11	3.12E+10	1.37E+15	3.87E+15	8.36E+14	1.91E+16	3.52E+15	1.93E+16	3.88E+15	1.08E+16	7.72E+15	4.71E+15	1.23E+16
Te-127m	**{CCI}[a]**	{CCI}[a]	3	1.08E+11	4.16E+09	1.83E+14	5.17E+14	1.12E+14	2.56E+15	4.70E+14	2.58E+15	5.18E+14	1.45E+15	1.03E+15	6.29E+14	1.64E+15
Te-129	**{CCI}[a]**	{CCI}[a]	3	2.34E+12	9.05E+10	3.98E+15	1.13E+16	2.43E+15	5.56E+16	1.02E+16	5.62E+16	1.13E+16	3.15E+16	2.24E+16	1.37E+16	3.56E+16
Te-129m	**{CCI}[a]**	{CCI}[a]	3	3.51E+11	1.36E+10	5.97E+14	1.69E+15	3.64E+14	8.34E+15	1.53E+15	8.42E+15	1.69E+15	4.72E+15	3.36E+15	2.05E+15	5.34E+15
Te-131m	**{CCI}[a]**		3	1.06E+12	4.10E+10	1.80E+15	5.09E+15	1.10E+15	2.51E+16	4.63E+15	2.54E+16	5.10E+15	1.42E+16	1.01E+16	6.19E+15	1.61E+16
Te-132	**{CCI}[a]**	{CCI}[a]	3	1.03E+13	3.98E+11	1.75E+16	4.95E+16	1.07E+16	2.44E+17	4.50E+16	2.47E+17	4.95E+16	1.38E+17	9.86E+16	6.02E+16	1.57E+17

				Release (Bq)												
Isotope	**Core Inventories (Bq)**	**SFP Inventories (Bq)**	**MAAP Fission Product Group**	**RC-101**	**RC-102**	**RC-200**	**RC-201**	**RC-202**	**RC-203**	**RC-204**	**RC-205**	**RC-206**	**RC-301**	**RC-302**	**RC-303**	**RC-304**
Sr-89	**{CCI}[a]**	{CCI}[a]	4	1.77E+12	5.29E+10	9.37E+13	4.09E+14	6.02E+15	6.10E+16	5.47E+15	4.56E+16	6.48E+15	7.50E+14	4.19E+14	2.07E+14	7.34E+14
Sr-90	**{CCI}[a]**	{CCI}[a]	4	1.27E+11	3.79E+09	6.72E+12	2.93E+13	4.31E+14	4.37E+15	3.92E+14	3.26E+15	4.65E+14	5.37E+13	3.00E+13	1.48E+13	5.26E+13
Sr-91	**{CCI}[a]**		4	2.13E+12	6.38E+10	1.13E+14	4.94E+14	7.25E+15	7.35E+16	6.60E+15	5.49E+16	7.82E+15	9.04E+14	5.05E+14	2.50E+14	8.85E+14
Sr-92	**{CCI}[a]**		4	2.26E+12	6.75E+10	1.20E+14	5.23E+14	7.68E+15	7.78E+16	6.98E+15	5.82E+16	8.28E+15	9.57E+14	5.34E+14	2.65E+14	9.37E+14
Mo-99	**{CCI}[a]**	{CCI}[a]	5	1.75E+13	4.50E+11	3.13E+15	8.94E+15	2.65E+16	1.52E+17	3.05E+16	1.80E+17	6.15E+16	7.10E+16	2.91E+16	1.53E+16	5.03E+16
Rh-105	**{CCI}[a]**		5	9.56E+12	2.46E+11	1.71E+15	4.88E+15	1.45E+16	8.30E+16	1.67E+16	9.80E+16	3.35E+16	3.87E+16	1.59E+16	8.35E+15	2.75E+16
Ru-103	**{CCI}[a]**		5	1.49E+13	3.83E+11	2.67E+15	7.61E+15	2.26E+16	1.29E+17	2.60E+16	1.53E+17	5.23E+16	6.04E+16	2.48E+16	1.30E+16	4.28E+16
Ru-105	**{CCI}[a]**		5	1.04E+13	2.67E+11	1.86E+15	5.31E+15	1.57E+16	9.03E+16	1.81E+16	1.07E+17	3.65E+16	4.22E+16	1.73E+16	9.09E+15	2.99E+16
Ru-106	**{CCI}[a]**		5	5.08E+12	1.31E+11	9.10E+14	2.60E+15	7.70E+15	4.41E+16	8.86E+15	5.21E+16	1.78E+16	2.06E+16	8.44E+15	4.44E+15	1.46E+16
Rb-86	**{CCI}[a]**	{CCI}[a]	6	9.77E+09	7.45E+08	2.39E+14	8.55E+14	2.16E+13	8.80E+14	6.28E+13	8.99E+14	4.46E+13	9.29E+13	2.74E+14	1.74E+14	3.69E+14
Cs-134	**{CCI}[a]**	{CCI}[a]	6	7.63E+11	5.82E+10	1.86E+16	6.68E+16	1.68E+15	6.87E+16	4.91E+15	7.02E+16	3.48E+15	7.25E+15	2.14E+16	1.36E+16	2.88E+16
Cs-136	**{CCI}[a]**	{CCI}[a]	6	2.76E+11	2.11E+10	6.75E+15	2.42E+16	6.10E+14	2.49E+16	1.78E+15	2.54E+16	1.26E+15	2.62E+15	7.74E+15	4.93E+15	1.04E+16
Cs-137	**{CCI}[a]**	{CCI}[a]	6	5.09E+11	3.88E+10	1.24E+16	4.45E+16	1.12E+15	4.58E+16	3.27E+15	4.68E+16	2.32E+15	4.83E+15	1.43E+16	9.08E+15	1.92E+16
Ba-139	**{CCI}[a]**		7	9.48E+12	2.41E+11	9.62E+14	3.46E+15	3.51E+16	2.10E+17	4.51E+16	1.64E+17	3.53E+16	1.57E+16	1.16E+16	4.24E+15	1.88E+16
Ba-140	**{CCI}[a]**	{CCI}[a]	7	9.14E+12	2.33E+11	9.28E+14	3.34E+15	3.39E+16	2.02E+17	4.35E+16	1.58E+17	3.40E+16	1.51E+16	1.12E+16	4.09E+15	1.81E+16
La-140	**{CCI}[a]**	{CCI}[a]	8	8.31E+10	2.80E+09	5.59E+12	2.78E+13	4.97E+14	4.15E+15	4.64E+14	3.12E+15	4.53E+14	1.14E+14	4.27E+13	4.07E+13	1.31E+14
La-141	**{CCI}[a]**		8	7.75E+10	2.62E+09	5.21E+12	2.59E+13	4.63E+14	3.87E+15	4.33E+14	2.91E+15	4.23E+14	1.06E+14	3.98E+13	3.79E+13	1.23E+14
La-142	**{CCI}[a]**		8	7.56E+10	2.55E+09	5.08E+12	2.53E+13	4.52E+14	3.78E+15	4.22E+14	2.84E+15	4.13E+14	1.03E+14	3.89E+13	3.70E+13	1.20E+14

Isotope	Core Inventories (Bq)	SFP Inventories (Bq)	MAAP Fission Product Group	Release (Bq)												
				RC-101	RC-102	RC-200	RC-201	RC-202	RC-203	RC-204	RC-205	RC-206	RC-301	RC-302	RC-303	RC-304
Nb-95	**{CCI}**[a]		8	8.38E+10	2.83E+09	5.64E+12	2.81E+13	5.01E+14	4.18E+15	4.68E+14	3.15E+15	4.57E+14	1.15E+14	4.31E+13	4.10E+13	1.33E+14
Nd-147	**{CCI}**[a]	{CCI}[a]	8	3.05E+10	1.03E+09	2.05E+12	1.02E+13	1.82E+14	1.52E+15	1.70E+14	1.15E+15	1.66E+14	4.17E+13	1.57E+13	1.49E+13	4.83E+13
Pr-143	**{CCI}**[a]	{CCI}[a]	8	7.34E+10	2.48E+09	4.93E+12	2.46E+13	4.39E+14	3.66E+15	4.10E+14	2.76E+15	4.00E+14	1.00E+14	3.77E+13	3.59E+13	1.16E+14
Y-90	**{CCI}**[a]	{CCI}[a]	8	3.91E+09	1.32E+08	2.63E+11	1.31E+12	2.34E+13	1.95E+14	2.18E+13	1.47E+14	2.13E+13	5.35E+12	2.01E+12	1.92E+12	6.19E+12
Y-91	**{CCI}**[a]	{CCI}[a]	8	6.63E+10	2.24E+09	4.46E+12	2.22E+13	3.97E+14	3.31E+15	3.70E+14	2.49E+15	3.62E+14	9.07E+13	3.41E+13	3.25E+13	1.05E+14
Y-92	**{CCI}**[a]		8	6.71E+10	2.26E+09	4.51E+12	2.25E+13	4.01E+14	3.35E+15	3.74E+14	2.52E+15	3.66E+14	9.17E+13	3.45E+13	3.28E+13	1.06E+14
Y-93	**{CCI}**[a]		8	7.56E+10	2.55E+09	5.08E+12	2.53E+13	4.52E+14	3.78E+15	4.22E+14	2.84E+15	4.13E+14	1.03E+14	3.89E+13	3.70E+13	1.20E+14
Zr-95	**{CCI}**[a]		8	8.27E+10	2.79E+09	5.56E+12	2.77E+13	4.95E+14	4.13E+15	4.62E+14	3.11E+15	4.51E+14	1.13E+14	4.25E+13	4.05E+13	1.31E+14
Zr-97	**{CCI}**[a]		8	7.79E+10	2.63E+09	5.23E+12	2.61E+13	4.66E+14	3.89E+15	4.35E+14	2.92E+15	4.25E+14	1.07E+14	4.00E+13	3.81E+13	1.23E+14
Ce-141	**{CCI}**[a]	{CCI}[a]	9	3.29E+11	6.89E+09	2.47E+13	1.35E+14	2.07E+15	9.59E+15	2.29E+15	6.28E+15	1.41E+15	7.94E+14	2.35E+14	6.17E+14	1.75E+15
Ce-143	**{CCI}**[a]		9	3.08E+11	6.44E+09	2.31E+13	1.26E+14	1.93E+15	8.95E+15	2.14E+15	5.86E+15	1.32E+15	7.42E+14	2.20E+14	5.76E+14	1.63E+15
Ce-144	**{CCI}**[a]	{CCI}[a]	9	2.56E+11	5.36E+09	1.92E+13	1.05E+14	1.61E+15	7.46E+15	1.78E+15	4.89E+15	1.10E+15	6.18E+14	1.83E+14	4.80E+14	1.36E+15
Np-239	**{CCI}**[a]	{CCI}[a]	9	4.12E+12	8.62E+10	3.09E+14	1.69E+15	2.58E+16	1.20E+17	2.86E+16	7.85E+16	1.76E+16	9.93E+15	2.94E+15	7.71E+15	2.18E+16
Pu-238	**{CCI}**[a]	{CCI}[a]	9	4.66E+08	9.75E+06	3.49E+10	1.91E+11	2.93E+12	1.36E+13	3.24E+12	8.89E+12	2.00E+12	1.12E+12	3.33E+11	8.73E+11	2.47E+12
Pu-239	**{CCI}**[a]	{CCI}[a]	9	7.26E+07	1.52E+06	5.44E+09	2.98E+10	4.55E+11	2.11E+12	5.04E+11	1.38E+12	3.11E+11	1.75E+11	5.18E+10	1.36E+11	3.85E+11
Pu-240	**{CCI}**[a]	{CCI}[a]	9	9.01E+07	1.89E+06	6.76E+09	3.70E+10	5.66E+11	2.62E+12	6.26E+11	1.72E+12	3.86E+11	2.17E+11	6.43E+10	1.69E+11	4.78E+11
Pu-241	**{CCI}**[a]	{CCI}[a]	9	2.72E+10	5.69E+08	2.04E+12	1.12E+13	1.71E+14	7.91E+14	1.89E+14	5.18E+14	1.16E+14	6.56E+13	1.94E+13	5.09E+13	1.44E+14
Sb-127	**{CCI}**[a]	{CCI}[a]	10	1.43E+12	3.76E+11	1.25E+15	4.89E+15	5.42E+15	2.20E+16	1.38E+16	5.15E+16	4.59E+15	4.06E+16	4.77E+16	3.74E+16	6.95E+16
Sb-129	{CCI}a		10	4.20E+12	1.10E+12	3.66E+15	1.43E+16	1.59E+16	6.46E+16	4.04E+16	1.51E+17	1.35E+16	1.19E+17	1.40E+17	1.10E+17	2.04E+17

				Release (Bq)												
Isotope	Core Inventories (Bq)	SFP Inventories (Bq)	MAAP Fission Product Group	RC-401	RC-402	RC-403	RC-404	RC-501	RC-502	RC-503	RC-504	RC-602	RC-701	RC-702	RC-802	RC SFP
Kr-85	**{CCI}[a]**	{CCI}[a]	1	4.63E+16	4.66E+16	4.74E+16	4.74E+16	3.70E+16	4.71E+16	4.74E+16	4.74E+16	4.71E+16	5.16E+15	5.16E+15	4.73E+16	1.25E+17
Kr-85m	**{CCI}[a]**		1	1.34E+18	1.35E+18	1.37E+18	1.37E+18	1.07E+18	1.37E+18	1.37E+18	1.37E+18	1.37E+18	1.49E+17	1.49E+17	1.37E+18	
Kr-87	**{CCI}[a]**		1	2.63E+18	2.65E+18	2.69E+18	2.69E+18	2.10E+18	2.68E+18	2.69E+18	2.69E+18	2.68E+18	2.93E+17	2.93E+17	2.69E+18	
Kr-88	**{CCI}[a]**		1	3.73E+18	3.75E+18	3.81E+18	3.81E+18	2.98E+18	3.79E+18	3.81E+18	3.81E+18	3.79E+18	4.15E+17	4.15E+17	3.81E+18	
Xe-133	**{CCI}[a]**	{CCI}[a]	1	9.05E+18	9.10E+18	9.25E+18	9.25E+18	7.23E+18	9.20E+18	9.25E+18	9.25E+18	9.20E+18	1.01E+18	1.01E+18	9.25E+18	3.08E+18
Xe-135	**{CCI}[a]**		1	2.93E+18	2.95E+18	3.00E+18	3.00E+18	2.34E+18	2.98E+18	3.00E+18	3.00E+18	2.98E+18	3.26E+17	3.26E+17	3.00E+18	
I-131	**{CCI}[a]**	{CCI}[a]	2	6.24E+16	7.35E+16	4.48E+16	1.19E+17	5.60E+15	1.03E+16	7.32E+15	8.68E+15	1.03E+16	1.95E+16	3.84E+17	4.30E+18	1.52E+18
I-132	**{CCI}[a]**	{CCI}[a]	2	9.03E+16	1.06E+17	6.49E+16	1.72E+17	8.10E+15	1.50E+16	1.06E+16	1.26E+16	1.50E+16	2.82E+16	5.55E+17	6.22E+18	2.20E+18
I-133	**{CCI}[a]**		2	1.27E+17	1.49E+17	9.11E+16	2.42E+17	1.14E+16	2.10E+16	1.49E+16	1.76E+16	2.10E+16	3.97E+16	7.80E+17	8.73E+18	
I-134	**{CCI}[a]**		2	1.39E+17	1.64E+17	1.00E+17	2.66E+17	1.25E+16	2.31E+16	1.64E+16	1.94E+16	2.31E+16	4.36E+16	8.57E+17	9.61E+18	
I-135	**{CCI}[a]**		2	1.18E+17	1.39E+17	8.49E+16	2.26E+17	1.06E+16	1.96E+16	1.39E+16	1.64E+16	1.96E+16	3.70E+16	7.27E+17	8.14E+18	
Te-127	**{CCI}[a]**	{CCI}[a]	3	1.33E+15	3.39E+15	1.22E+15	3.08E+15	1.87E+13	2.69E+13	2.01E+13	2.60E+13	2.69E+13	2.91E+15	5.82E+16	4.74E+17	3.42E+15
Te-127m	**{CCI}[a]**	{CCI}[a]	3	1.77E+14	4.53E+14	1.63E+14	4.11E+14	2.50E+12	3.59E+12	2.68E+12	3.47E+12	3.59E+12	3.89E+14	7.78E+15	6.33E+16	4.94E+14
Te-129	**{CCI}[a]**	{CCI}[a]	3	3.86E+15	9.85E+15	3.54E+15	8.93E+15	5.44E+13	7.80E+13	5.83E+13	7.55E+13	7.80E+13	8.46E+15	1.69E+17	1.38E+18	9.80E+15
Te-129m	**{CCI}[a]**	{CCI}[a]	3	5.78E+14	1.48E+15	5.31E+14	1.34E+15	8.15E+12	1.17E+13	8.74E+12	1.13E+13	1.17E+13	1.27E+15	2.54E+16	2.07E+17	1.47E+15
Te-131m	**{CCI}[a]**		3	1.74E+15	4.45E+15	1.60E+15	4.04E+15	2.46E+13	3.53E+13	2.64E+13	3.41E+13	3.53E+13	3.82E+15	7.65E+16	6.23E+17	
Te-132	**{CCI}[a]**	{CCI}[a]	3	1.70E+16	4.33E+16	1.56E+16	3.93E+16	2.39E+14	3.43E+14	2.56E+14	3.32E+14	3.43E+14	3.72E+16	7.44E+17	6.06E+18	4.31E+16
Sr-89	**{CCI}[a]**	{CCI}[a]	4	8.19E+14	1.47E+15	7.09E+14	6.34E+14	4.40E+13	3.85E+13	4.41E+13	4.41E+13	3.85E+13	3.13E+15	6.26E+16	7.48E+17	3.49E+15
Sr-90	**{CCI}[a]**	{CCI}[a]	4	5.86E+13	1.05E+14	5.08E+13	4.54E+13	3.15E+12	2.76E+12	3.16E+12	3.16E+12	2.76E+12	2.24E+14	4.48E+15	5.36E+16	4.78E+15
Sr-91	**{CCI}[a]**		4	9.87E+14	1.77E+15	8.55E+14	7.64E+14	5.31E+13	4.64E+13	5.32E+13	5.32E+13	4.64E+13	3.77E+15	7.54E+16	9.01E+17	

				Release (Bq)												
Isotope	**Core Inventories (Bq)**	**SFP Inventories (Bq)**	**MAAP Fission Product Group**	**RC-401**	**RC-402**	**RC-403**	**RC-404**	**RC-501**	**RC-502**	**RC-503**	**RC-504**	**RC-602**	**RC-701**	**RC-702**	**RC-802**	**RC SFP**
Sr-92	**{CCI}[a]**		4	1.05E+15	1.87E+15	9.05E+14	8.09E+14	5.62E+13	4.92E+13	5.63E+13	5.63E+13	4.92E+13	3.99E+15	7.99E+16	9.54E+17	
Mo-99	**{CCI}[a]**	{CCI}[a]	5	2.36E+16	3.27E+16	1.12E+16	2.93E+16	3.83E+14	3.69E+14	3.75E+14	3.75E+14	3.69E+14	4.07E+16	8.14E+17	5.88E+18	1.41E+17
Rh-105	**{CCI}[a]**		5	1.29E+16	1.78E+16	6.09E+15	1.60E+16	2.09E+14	2.02E+14	2.05E+14	2.05E+14	2.02E+14	2.22E+16	4.44E+17	3.21E+18	
Ru-103	**{CCI}[a]**		5	2.01E+16	2.78E+16	9.51E+15	2.50E+16	3.26E+14	3.15E+14	3.19E+14	3.19E+14	3.15E+14	3.46E+16	6.93E+17	5.01E+18	
Ru-105	**{CCI}[a]**		5	1.40E+16	1.94E+16	6.63E+15	1.74E+16	2.27E+14	2.19E+14	2.23E+14	2.23E+14	2.19E+14	2.42E+16	4.83E+17	3.49E+18	
Ru-106	**{CCI}[a]**		5	6.87E+15	9.49E+15	3.24E+15	8.52E+15	1.11E+14	1.07E+14	1.09E+14	1.09E+14	1.07E+14	1.18E+16	2.36E+17	1.71E+18	
Rb-86	**{CCI}[a]**	{CCI}[a]	6	5.59E+13	7.22E+13	3.01E+13	9.83E+13	2.41E+11	3.68E+12	2.43E+11	6.21E+11	3.68E+12	3.90E+13	7.79E+14	8.41E+15	2.98E+15
Cs-134	**{CCI}[a]**	{CCI}[a]	6	4.37E+15	5.64E+15	2.35E+15	7.68E+15	1.88E+13	2.87E+14	1.90E+13	4.85E+13	2.87E+14	3.04E+15	6.08E+16	6.57E+17	7.10E+17
Cs-136	**{CCI}[a]**	{CCI}[a]	6	1.58E+15	2.04E+15	8.51E+14	2.78E+15	6.80E+12	1.04E+14	6.87E+12	1.76E+13	1.04E+14	1.10E+15	2.20E+16	2.38E+17	8.43E+16
Cs-137	**{CCI}[a]**	{CCI}[a]	6	2.91E+15	3.76E+15	1.57E+15	5.12E+15	1.25E+13	1.91E+14	1.26E+13	3.23E+13	1.91E+14	2.03E+15	4.06E+16	4.38E+17	1.78E+18
Ba-139	**{CCI}[a]**		7	1.28E+16	3.26E+16	2.22E+16	1.65E+16	2.24E+14	2.03E+14	2.01E+14	2.01E+14	2.03E+14	2.29E+16	4.58E+17	3.58E+18	
Ba-140	**{CCI}[a]**	{CCI}[a]	7	1.23E+16	3.14E+16	2.14E+16	1.60E+16	2.16E+14	1.96E+14	1.94E+14	1.94E+14	1.96E+14	2.21E+16	4.41E+17	3.46E+18	5.40E+15
La-140	**{CCI}[a]**	{CCI}[a]	8	1.34E+14	3.02E+14	1.20E+14	1.23E+14	2.38E+12	1.78E+12	2.34E+12	2.34E+12	1.78E+12	1.85E+14	3.71E+15	5.20E+16	2.75E+12
La-141	**{CCI}[a]**		8	1.25E+14	2.82E+14	1.12E+14	1.15E+14	2.22E+12	1.66E+12	2.18E+12	2.18E+12	1.66E+12	1.73E+14	3.46E+15	4.85E+16	
La-142	**{CCI}[a]**		8	1.22E+14	2.75E+14	1.10E+14	1.12E+14	2.16E+12	1.62E+12	2.13E+12	2.13E+12	1.62E+12	1.69E+14	3.37E+15	4.73E+16	
Nb-95	**{CCI}[a]**		8	1.35E+14	3.05E+14	1.22E+14	1.24E+14	2.40E+12	1.80E+12	2.36E+12	2.36E+12	1.80E+12	1.87E+14	3.74E+15	5.25E+16	
Nd-147	**{CCI}[a]**	{CCI}[a]	8	4.92E+13	1.11E+14	4.42E+13	4.52E+13	8.73E+11	6.55E+11	8.58E+11	8.58E+11	6.55E+11	6.81E+13	1.36E+15	1.91E+16	1.01E+12
Pr-143	**{CCI}[a]**	{CCI}[a]	8	1.18E+14	2.67E+14	1.06E+14	1.09E+14	2.10E+12	1.58E+12	2.06E+12	2.06E+12	1.58E+12	1.64E+14	3.28E+15	4.59E+16	2.43E+12
Y-90	**{CCI}[a]**	{CCI}[a]	8	6.31E+12	1.42E+13	5.67E+12	5.80E+12	1.12E+11	8.40E+10	1.10E+11	1.10E+11	8.40E+10	8.73E+12	1.75E+14	2.45E+15	1.27E+12

				Release (Bq)												
Isotope	**Core Inventories (Bq)**	**SFP Inventories (Bq)**	**MAAP Fission Product Group**	**RC-401**	**RC-402**	**RC-403**	**RC-404**	**RC-501**	**RC-502**	**RC-503**	**RC-504**	**RC-602**	**RC-701**	**RC-702**	**RC-802**	**RC SFP**
Y-91	**{CCI}**[a]	{CCI}[a]	8	1.07E+14	2.41E+14	9.61E+13	9.83E+13	1.90E+12	1.42E+12	1.87E+12	1.87E+12	1.42E+12	1.48E+14	2.96E+15	4.15E+16	2.21E+12
Y-92	**{CCI}**[a]		8	1.08E+14	2.44E+14	9.72E+13	9.94E+13	1.92E+12	1.44E+12	1.89E+12	1.89E+12	1.44E+12	1.50E+14	2.99E+15	4.20E+16	
Y-93	**{CCI}**[a]		8	1.22E+14	2.75E+14	1.10E+14	1.12E+14	2.16E+12	1.62E+12	2.13E+12	2.13E+12	1.62E+12	1.69E+14	3.37E+15	4.73E+16	
Zr-95	**{CCI}**[a]		8	1.33E+14	3.01E+14	1.20E+14	1.23E+14	2.37E+12	1.78E+12	2.33E+12	2.33E+12	1.78E+12	1.85E+14	3.69E+15	5.18E+16	
Zr-97	**{CCI}**[a]		8	1.26E+14	2.83E+14	1.13E+14	1.15E+14	2.23E+12	1.67E+12	2.19E+12	2.19E+12	1.67E+12	1.74E+14	3.47E+15	4.87E+16	
Ce-141	**{CCI}**[a]	{CCI}[a]	9	8.35E+14	1.56E+15	4.98E+14	1.77E+15	6.61E+12	5.52E+12	5.74E+12	5.74E+12	5.52E+12	8.79E+14	1.76E+16	3.61E+17	2.61E+12
Ce-143	**{CCI}**[a]		9	7.80E+14	1.46E+15	4.65E+14	1.66E+15	6.18E+12	5.15E+12	5.36E+12	5.36E+12	5.15E+12	8.21E+14	1.64E+16	3.37E+17	
Ce-144	**{CCI}**[a]	{CCI}[a]	9	6.50E+14	1.22E+15	3.87E+14	1.38E+15	5.15E+12	4.29E+12	4.47E+12	4.47E+12	4.29E+12	6.84E+14	1.37E+16	2.81E+17	2.99E+12
Np-239	**{CCI}**[a]	{CCI}[a]	9	1.04E+16	1.95E+16	6.22E+15	2.22E+16	8.27E+13	6.89E+13	7.18E+13	7.18E+13	6.89E+13	1.10E+16	2.20E+17	4.51E+18	3.27E+13
Pu-238	**{CCI}**[a]	{CCI}[a]	9	1.18E+12	2.21E+12	7.04E+11	2.51E+12	9.36E+09	7.80E+09	8.13E+09	8.13E+09	7.80E+09	1.24E+12	2.49E+13	5.10E+14	4.10E+10
Pu-239	**{CCI}**[a]	{CCI}[a]	9	1.84E+11	3.44E+11	1.10E+11	3.91E+11	1.46E+09	1.21E+09	1.27E+09	1.27E+09	1.21E+09	1.94E+11	3.87E+12	7.94E+13	7.89E+09
Pu-240	**{CCI}**[a]	{CCI}[a]	9	2.29E+11	4.27E+11	1.36E+11	4.85E+11	1.81E+09	1.51E+09	1.57E+09	1.57E+09	1.51E+09	2.40E+11	4.81E+12	9.86E+13	1.14E+10
Pu-241	**{CCI}**[a]	{CCI}[a]	9	6.90E+13	1.29E+14	4.11E+13	1.46E+14	5.46E+11	4.55E+11	4.74E+11	4.74E+11	4.55E+11	7.25E+13	1.45E+15	2.98E+16	2.02E+12
Sb-127	**{CCI}**[a]	{CCI}[a]	10	3.88E+16	9.27E+16	1.38E+16	5.68E+16	5.32E+14	8.84E+15	1.22E+14	3.13E+14	8.84E+15	3.54E+15	7.09E+16	4.84E+17	1.70E+16
Sb-129	**{CCI}**[a]		10	1.14E+17	2.72E+17	4.06E+16	1.67E+17	1.56E+15	2.59E+16	3.57E+14	9.19E+14	2.59E+16	1.04E+16	2.08E+17	1.42E+18	

SUB-SECTION 15.4.4.3 - TABLE 5

UK EPR PSA2 – RELEASE ENERGIES AND LOCATIONS

Release Category	Containment Failure Mechanism	MAAP Run	Alarm Time (core uncovery) (hr)	Release Start Time (hr)	Release End Time (hr)	Release Duration (h)r	Release Duration (sec)	Release Energy Rate W (J/s)	Release Energy (total) (J)	Junction for Energy Release	Release Height (m)
RC 101	None - deposition in annulus & building	ST1.10g	2.4	4.5	19.0	14.5	52200	1.77E+04	9.24E+08	Junction 187	34.75
RC 102	None - annulus and building ventilation	ST1.10	2.4	3.6	9.5	5.9	21240	9.44E+03	2.00E+08	Junction 187	60.5
RC 200	Isolation failure - in-vessel recovery with sprays	ST1.11	2.4	3.3	3.6	0.3	1080	5.86E+08	6.33E+11	Junction 153	0.83
RC 201	Isolation failure - in-vessel recovery without sprays	ST1.11	2.4	3.3	3.6	0.3	1080	5.86E+08	6.33E+11	Junction 153	0.83
RC 202	Isolation failure	ST1.8g	2.4	4.6	8.3	3.7	13320	4.39E+07	5.85E+11	Junction 153	0.83
RC 203	Isolation failure	ST1.8h	2.4	3.4	8.8	5.4	19440	4.57E+08	8.88E+12	Junction 153	0.83
RC 204	Isolation failure	ST1.8i	2.4	4.6	6.8	2.2	7920	3.69E+07	2.92E+11	Junction 153	0.83
RC 205	Isolation failure	ST1.8j	2.4	3.5	10.0	6.5	23400	1.40E+08	3.27E+12	Junction 153	0.83
RC 206	All small isolation failures (<2")	ST1.8f	2.4	3.4	10.0	6.6	23760	1.57E+06	3.73E+10	Junction 153	0.83
RC 301	Early failure	ST4.1	2.4	3.5	7.2	3.7	13320	1.85E+08	2.46E+12	Junction 153	35.7
RC 302	Early failure	ST4.2	2.4	3.6	9.7	6.1	21960	3.73E+08	8.20E+12	Junction 153	35.7
RC 303	Early failure	ST4.3	2.4	3.6	7.5	3.9	14040	3.05E+08	4.28E+12	Junction 153	35.7
RC 304	Early failure	ST4.4	2.4	3.7	7.6	3.9	14040	3.43E+08	4.81E+12	Junction 153	35.7
RC 401	Intermediate failure	ST4.5	2.4	7.6	12.0	4.4	15840	5.64E+07	8.93E+11	Junction 153	35.7
RC 402	Intermediate failure	ST4.6	2.4	7.0	10.0	3.0	10800	5.16E+06	5.58E+10	Junction 153	35.7
RC 403	Intermediate failure	ST4.7	2.4	7.5	11.9	4.4	15840	2.76E+08	4.37E+12	Junction 153	35.7
RC 404	Intermediate failure	ST4.8	2.4	7.8	20.5	12.7	45720	3.43E+08	1.57E+13	Junction 153	35.7

SUB-SECTION 15.4.4.3 - TABLE 5 (CONT'D}

UK EPR PSA2 – RELEASE ENERGIES AND LOCATIONS

Release Category	Containment Failure Mechanism	MAAP Run	Alarm Time (core uncovery) (hr)	Release Start Time (hr)	Release End Time (hr)	Release Duration (h)r	Release Duration (sec)	Release Energy Rate W (J/s)	Release Energy (total) (J)	Junction for Energy Release	Release Height (m)
RC 50*	Late failure (phase 1)	ST1.10	2.4	3.8	9.0	5.2	18720	9.57E+03	1.79E+08	Junction 187	60.5
RC 501	Late failure	ST1.10e	2.4	60.0	70.0	10.0	36000	5.01E+08	1.80E+13	Junction 153	35.7
RC 502	Late failure	ST1.10a	2.4	60.0	70.0	10.0	36000	5.01E+08	1.80E+13	Junction 153	35.7
RC 503	Late failure	ST1.10d	2.4	85.0	125.0	40.0	144000	2.41E+09	34.7E+13	Junction 153	35.7
RC 504	Late failure	ST1.10	2.4	85.0	125.0	40.0	144000	7.10E+07	1.02E+13	Junction 153	35.7
RC 602	Basemat failure	ST1.10a	2.4	216.0	222.0	6.0	21600	5.01E+08	1.08E+13	Junction 153	0
RC 701	SGTR scrubbed	ST2.3	1.2	3.3	7.5	4.2	14976	1.40E+07	2.10E+11	Calculated	24.75
RC 702	SGTR unscrubbed	ST2.3	1.2	3.3	7.5	4.2	14976	1.40E+07	2.10E+11	Calculated	24.75
RC 801	ISLOCA scrubbed	ST3.1a	1.3	1.8	2.9	1.1	3960	1.97E+07	7.80E+10	Junction 195	10
RC 802	Large ISLOCA unscrubbed, no deposition	ST3.1a	1.3	1.8	2.9	1.1	3960	1.97E+07	7.80E+10	Junction 195	10
	Small ISLOCA unscrubbed, no deposition	ST3.2a	6.4	7.4	8.9	1.5	5400	1.25E+07	6.77E+10	Junction 195	10
RC 802a	Large ISLOCA, unscrubbed, deposition in building	ST3.1c	1.4	1.9	4.3	2.4	8640	6.37E+07	5.50E+11	Junction 195	10
	Small ISLOCA, unscrubbed, deposition in building	ST3.2c	6.5	7.8	10.9	3.1	11160	2.33E+07	2.59E+11	Junction 195	10
RC 802b	Large ISLOCA, unscrubbed, ventilation in building	ST3.1a	1.3	1.8	2.9	1.1	3960	1.97E+07	7.80E+10	Junction 195	10
	Small ISLOCA, unscrubbed, ventilation in building	ST3.2a	6.4	7.4	8.9	1.5	5400	1.25E+07	6.77E+10	Junction 195	10
SFP	Spent fuel pool accident:	ST3.2c	6.5	7.8	10.9	3.1	11160	2.33E+07	2.59E+11	Junction 195	10

Note: For the RC 50* and RC 602 series, the PSA Level 3 is required to take account of a first phase of release due to containment leakage. It represents an important release for aerosols (except for CsI).

4.4. LEVEL 2 PSA RISK INTEGRATION AND INTERFACE WITH THE LEVEL 3 PSA

4.4.1. Introduction

This section assembles and presents the results of the UK EPR Level 2 PSA study, in terms of frequencies, releases and release risk. It also discusses the interface with the Level 3 PSA and concepts of “large release”.

4.4.2. Interface with Level 3 PSA

Level 3 PSA analysis needs three types of input from the results of Level 2 PSA:

- The frequency of each Release Category
- The source term associated with each Release Category
- Additional information associated with the releases, such as release energy, height and timing.

Release Category Frequencies

Release category frequencies were presented in section 4.1 (plant states A and B) and section 4.2 (plant states C, D and E). For convenience, the definitions of each release category and the calculated *total* frequencies (at power states) are shown in Sub-section 15.4.4.4 – Table 1 and Sub-section 15.4.4.4 - Figure 1. These results are discussed in later sub-sections.

Source Terms

Although the source term analysis (section 4.3) provides time dependent release information for each of the 12 fission product isotope groups modelled in MAAP, for use in the Level 3 PSA, these results are presented as the final integrated value of the release, together with a start time and duration. MAAP analysis yields fission product release fractions (which is the fraction of the initial core inventory for the associated isotope). To obtain the release for a given isotope in Bq, the release fraction is multiplied by the core inventory of that isotope. Release fractions, releases in Bq and fission product inventories for both accidents in the core and in the spent fuel pool were presented in section 4.3. These results are discussed in later sub-sections.

Additional Information

The additional information required by the Level 3 PSA analysts is, for each source term:

- The height of the release,
- the energy of the release,
- the time of start of the release, and
- the duration of the release.

This information has been presented in section 4.3.

4.4.3. "Large Release Frequency"

4.4.3.1. Release Targets in the UK

UK targets in T/AST/030 [Ref-1] and SAPs [Ref-2] are expressed in terms of doses to persons on- and off-site, and mortality risk. Both of these metrics are the result of a Level 3 PSA analysis, and compliance with them is demonstrated elsewhere in this report. However, it is noted that the EPR core damage frequency reported here ($<10^{-6}$ /ry) is such that regardless of release or calculated dose, the frequency/doseband targets will be met for events involving core damage (see for example Sub-section 15.4.4.4 - Figure 2). Further discussion of compliance with UK targets is provided in section 4.4.3.5 of this report.

Although there is no release target as such, it can be instructive (for example in order to define and perform sensitivity studies) to refer to a Large Release Frequency (LRF) or a "Large Early Release Frequency" (LERF). The LRF would be the sum of the frequencies of release categories exceeding some release threshold. Sometimes the LRF is expressed in terms of a fraction of core damage frequency.

The definition of "large" varies considerably amongst countries in which release targets exist, as discussed in sections 4.4.3.2 to 4.4.3.4 below.

4.4.3.2. Large Release Targets from the IAEA

The IAEA INSAG-12 report [Ref-1] states:

"27. The target for existing nuclear power plants consistent with the technical safety objective is a frequency of occurrence of severe core damage that is below about 10^{-4} events per plant operating year. Severe accident management and mitigation measures could reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response. Application of all safety principles and the objectives of paragraph.25 to future plants could lead to the achievement of an improved goal of not more than 10^{-5} severe core damage events per plant operating year. Another objective for these future plants is the practical elimination of accident sequences that could lead to large early radioactive releases, whereas severe accidents that could imply late containment failure would be considered in the design process with realistic assumptions and best estimate analyses so that their consequences would necessitate only protective measures limited in area and in time."

This description implies that "large" refers to a release large enough to require emergency counter-measures off-site, and that "early" refers to the need for those measures to be performed "short term". By implication, for new plants, if the core damage frequency target is 1E-5, then the LERF target is implied to be 1E-6/ry.

Results presented in this report for the EPR show a core damage frequency of 7.08E-7/ry < 1E-6/ry for all states and a fuel damage frequency in the spent fuel pool of 2.55E-09/ry (total is therefore 7.11E-7/ry); thus this IAEA target is clearly met, regardless of the release profile.

4.4.3.3. Large Release Targets in the US

NUREG/CR-6595 [Ref-1] provides guidance on defining and assessing LERF. The basis for the guidance is taken from the US NRC Safety Goal Policy document, [Ref-2] and is: “The early fatality QHO [Quantitative Health Objective] defined in the NRC Safety Goal Policy is: "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.””

NUREG/CR-6595 [Ref-1] presents analysis which relates this objective to release fractions. It concludes that:

“Three types of assumptions have been utilised in analysing the above information in the IPE database for exploring a possible definition of LERF:

(1) LERF consists of the total frequency of all release classes that occur under the early containment failure or containment bypass categories of the containment failure mode matrix.

(2) LERF consists of the frequency of release classes associated with the early failure and bypass containment failure modes which have release fractions of the volatile/semi-volatile fission products (Iodine, Caesium, Tellurium) equal to or greater than about 2.5 to 3% (based on the insights of the Large Release Study discussed above).

(3) A third alternative, based on a memorandum prepared for the ACRS [Ref-3], is that LERF is the frequency of early failure and bypass containment failure modes that have a release fraction of iodine equal to or greater than about 10%, based on calculations performed by Kaiser [Ref-4].

If the second (2) definition is used, and applied to the results presented in this report (and making allowance for the increased inventory of an EPR compared with an existing reactor), it would be concluded (based on Cs release, MAAP FP group 6), that the following release categories would be considered as “LRF”: 200, 201, 203, 205, 30x, 404, 702, 802, and SFP. On this basis, the large release frequency (LRF) would be 3.13E-08/ry or 4.4% of CDF. Note that the large *early* release frequency “LERF” would be defined in the same way, but excluding the SFP release category, which is 2.55E-9/ry and would not therefore impact the result.

Note that RC50* and RC60* are late (not early) releases. However, these are not large enough to contribute to "large release". Therefore the difference between LRF and LERF on definition 2 is not very important in practice.

4.4.3.4. Large Release Targets in Scandinavia

In both Finland and Sweden, release targets for severe accidents exist. For example, in Finland, according to Decision of the [Finnish] Council of State [Ref-1].

“Limit for a severe accident

The limit for the release of radioactive materials arising from a severe accident is a release which causes neither acute harmful health effects to the population in the vicinity of the nuclear power plant, nor any long-term restrictions on the use of extensive areas of land and water. For satisfying the requirement applied to long-term effects, the limit for an atmospheric release of cesium-137 is 100 TBq. Regarding the long term (starting three months after the accident), the combined fall-out consisting of nuclides other than Caesium-isotopes shall not cause a hazard greater than would arise from a Caesium release corresponding to the above-mentioned limit. The possibility that, as the result of a severe accident, the above mentioned requirement is not met shall be extremely small. “

This 100 TBq limit is also used similarly in Sweden.

For an EPR reactor of 4500 MWth, and based on the fission product inventories presented earlier, the fraction of initial core inventory of Cs-137 corresponding to 100 TBq is 2.1E-04, or 0.021%. Applying this definition of “large release” would lead to the inclusion of all release categories except RC101 and 102 (intact containment) and RC501 and 503 (late failure with sprays available). The “large release” frequency (LRF) so defined would be then 7.69E-8 /ry, or 10.8% of CDF.

Release categories 50x and 60x and SFP represent very late containment failures; thus a “large early release” frequency (LERF) based on this release definition is be obtained by considering the frequency of all release categories except RC10x, RC50x, RC60x and SFP. This leads to a LERF value of 4.07E-8/ry, or 5.7% of CDF.

The frequency of exceedance is shown as a function of release magnitude (in Bq) and release fraction in Sub-section 15.4.4.4 - Figure 3 and Sub-section 15.4.4.4 - Figure 4 respectively.

4.4.3.5. Release Targets for the UK EPR Level 2 PSA Study

As noted above, no specific large release target is used for the UK licensing process. Compliance with risk and dose targets is demonstrated elsewhere in this report, since they are based on the Level 3 PSA calculations. However, for the purposes of discussing “large release” in terms of Level 2 PSA results, the target adopted in a number of European countries of 100TBq of Cs-137 release has been used in this study.

Assessing whether a release should be considered “early” or not involves evaluating the time needed to initiate and perform off-site counter-measures. In this study it is conservatively assumed that any large release which occurs up to and including the time of vessel failure should be considered as “early”.

With these definitions, the following frequencies are obtained in this study:

- Large release frequency (LRF) (all RC except RC101, 102, 501 and 503):

  7.69E-8 /ry, or 10.8% of CDF.

- Large early release frequency (LERF) (all RC except RC10x, 50x, 60x, SFP):

  4.07E-8 /ry, or 5.7% of CDF.

The frequency of exceedance is shown as a function of release magnitude (in Bq) and release fraction in Sub-section 15.4.4.4 - Figure 3 and Sub-section 15.4.4.4 - Figure 4 respectively.

4.4.4. Release Risk

Since risk involves a combination of frequency and consequence it is instructive to present the Level 2 PSA results in terms of “release risk”, which is the frequency of a given release multiplied by its magnitude, as well as in terms of the large release frequencies as discussed in the previous section.

For the purpose of presenting release risk predictions for the EPR in a manageable way, three isotopes which are known to be important for consequences are considered. These are Cs-137, I-131 and Sr-90. (The release risk for all other isotopes considered in the modelling can of course be presented also.)

Sub-section 15.4.4.4 - Figure 5 shows the release risk (calculated per the above definition, for the specified three isotopes) for each of the release categories considered in the Level 2 PSA.

Sub-section 15.4.4.4 - Figures 6, 7 and 8 show the relative contribution of the different release categories to the release risk for I-131, Cs-137 and Sr-90 respectively.

Regarding I-131 and the Cs-137 release risk, the largest contributors are interfacing system LOCA sequences, containment isolation, SGTR and some early containment rupture.

The spent fuel pool accidents include although a large contribution of Cs-137 releases.

The specific release categories contributing to the release risk as presented in the figures are discussed in more detail in the next sub-section.

4.4.5. Discussion of Key Release Risk Contributing Release Categories

Decomposition of the release risk, as presented in Figures 5 to 8 reveals a dominant contribution from interfacing system LOCA without fission product scrubbing: RC802a contributing to 18% of I-131 release risk, 5% of Cs-137 release risk and 36% of Sr-90 release risk.

The early containment failure without MCCI and with melt flooded ex-vessel has a significant impact on the I-131 release risk (11% for RC303 and 17% for RC304).

The containment isolation sequences are also significant contributors to the Sr-90 release risk, except for the sequences with MCCI (RC202 and RC203): RC204, RC205 and RC206 contributing to about 6% of this risk. However, it is important to note that shutdown sequences occurring from plant states where the containment is open are also classified under RC20x, meaning that these RCs contain a mixture of isolation failure and containment open contributions.

Other significant contributors to release risk are the steam generator tube ruptures without fission product scrubbing: RC702 contributing to about 14% of I-131 release risk, 4% of Cs-137 release risk and about 35% of Sr-90 release risk.

Spent fuel pool accidents contribute significantly to the I-131 release risk (29%), to the Cs-137 release risk (up to 86%) and to the Sr-90 release risk (19%). For Cs-137 this is the major risk.

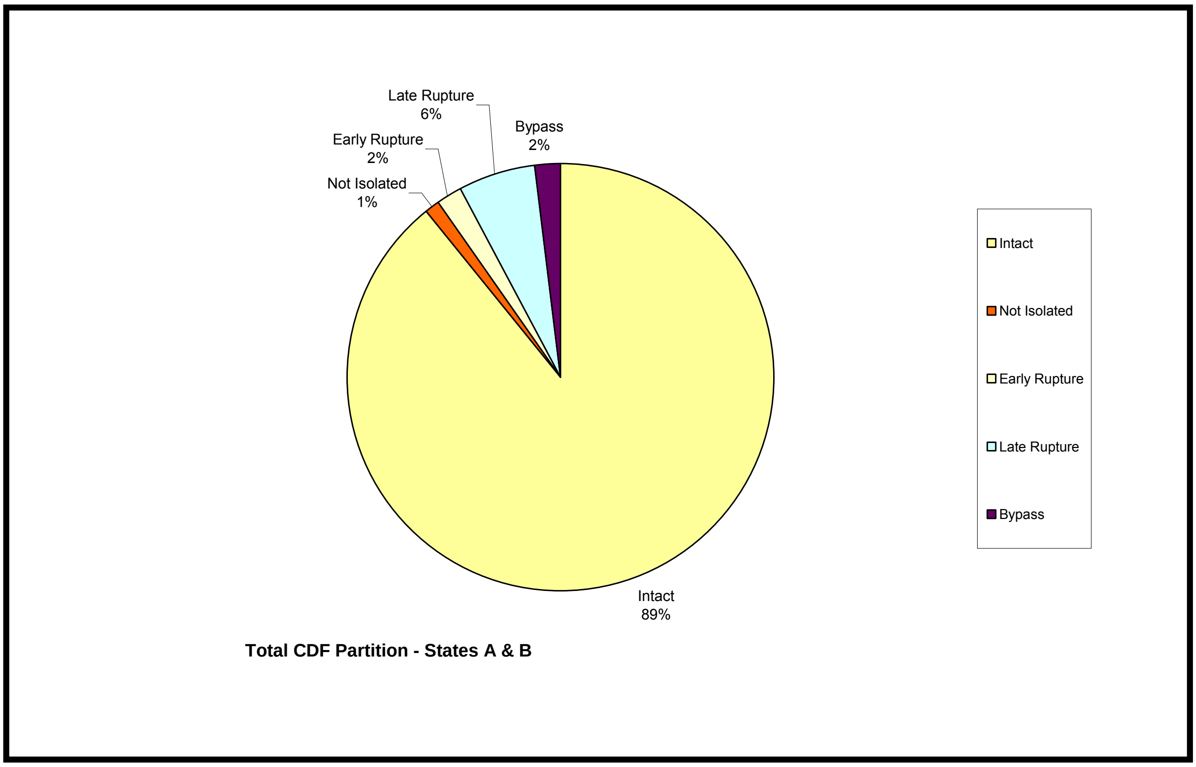
SUB-SECTION 15.4.4.4 - TABLE 1

Characterisation and Frequency of Release Categories (all plant states)

RC	Containment Failure Mode	Debris flood	Spray	Freq (/yr)	%CDF without SFP	%CDF with SFP
RC 101	none - deposition in annulus and building			1.49E-07	21.02%	20.94%
RC 102	none - annulus and building ventilation			4.84E-07	68.30%	68.06%
RC 200	isolation failure - in-vessel recovery	yes	Yes	9.02E-10	0.13%	0.13%
RC 201	isolation failure - in-vessel recovery	yes	No	3.01E-10	0.04%	0.04%
RC 202	isolation failure	no	Yes	2.60E-12	0.00%	0.00%
RC 203	isolation failure	no	No	3.04E-13	0.00%	0.00%
RC 204	isolation failure	yes	Yes	1.95E-09	0.28%	0.27%
RC 205	isolation failure	yes	No	4.51E-10	0.06%	0.06%
RC 206	all small isolation failures (< 2")			4.61E-09	0.65%	0.65%
RC 301	early	no	Yes	8.06E-12	0.00%	0.00%
RC 302	early	no	No	5.84E-12	0.00%	0.00%
RC 303	early	yes	Yes	1.02E-08	1.44%	1.43%
RC 304	early	yes	No	6.98E-09	0.99%	0.98%
RC 401	intermediate	no	Yes	2.67E-11	0.00%	0.00%
RC 402	intermediate	no	no	8.37E-12	0.00%	0.00%
RC 403	intermediate	yes	yes	1.23E-09	0.17%	0.17%
RC 404	intermediate	yes	no	1.09E-09	0.15%	0.15%
RC 501	late	no	yes	6.51E-13	0.00%	0.00%
RC 502	late	no	no	3.96E-11	0.01%	0.01%
RC 503	late	yes	yes	1.27E-09	0.18%	0.18%
RC 504	late	yes	no	3.29E-08	4.65%	4.63%
RC 602	basemat		no	6.57E-10	0.09%	0.09%
RC 701	SGTR scrubbed			4.14E-09	0.58%	0.58%
RC 702	SGTR unscrubbed			5.01E-09	0.71%	0.70%
RC 802	large ISLOCA, unscrubbed, deposition in building			3.83E-09	0.54%	0.54%
SFP	spent fuel pool			2.55E-09		0.36%
TOTAL CDF without SFP				7.08E-07	100.00%	
TOTAL CDF with SFP				7.11E-07		100.00%

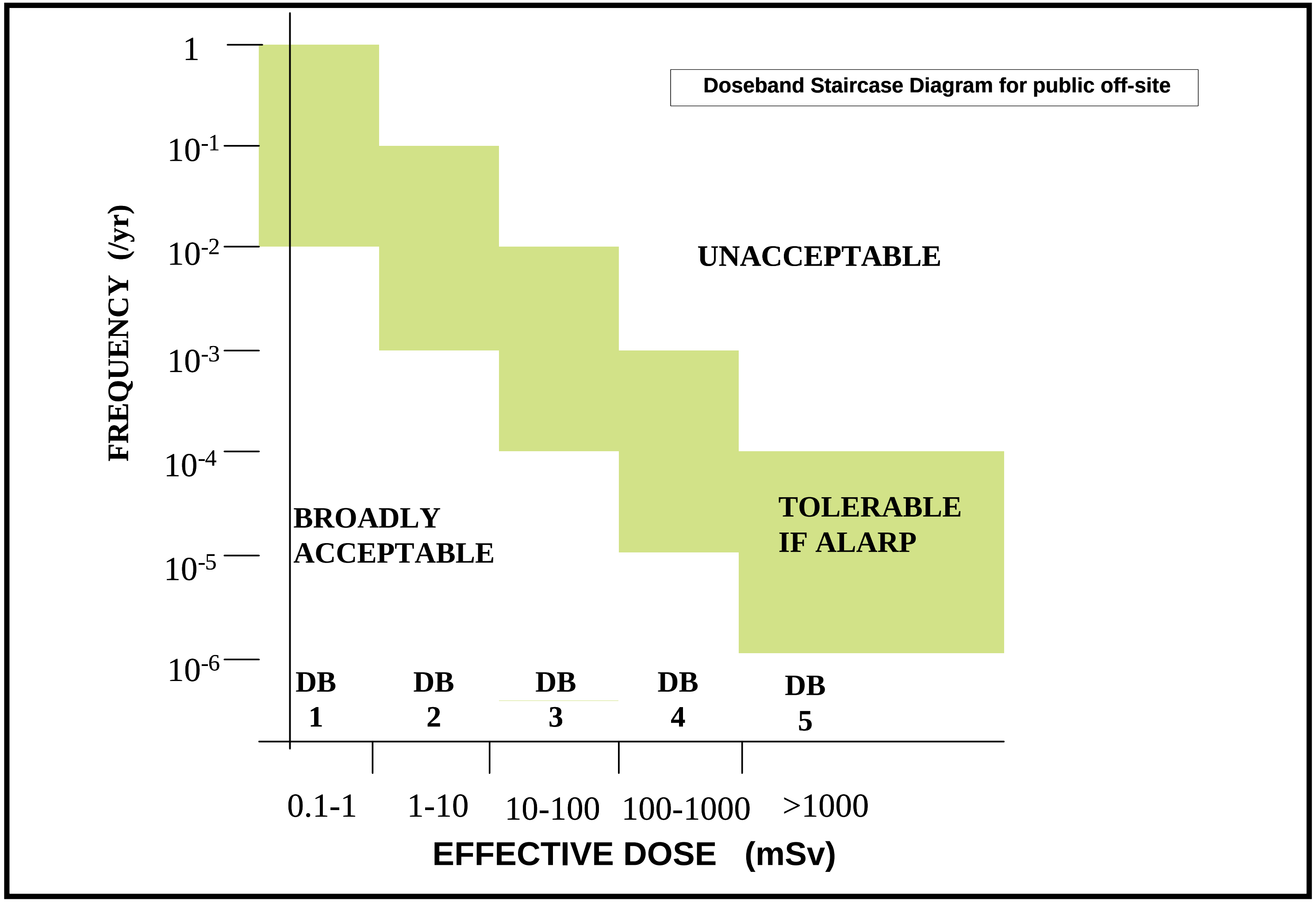
SUB-SECTION 15.4.4.4 - FIGURE 1

Frequency of Different Containment Failure Modes at Power



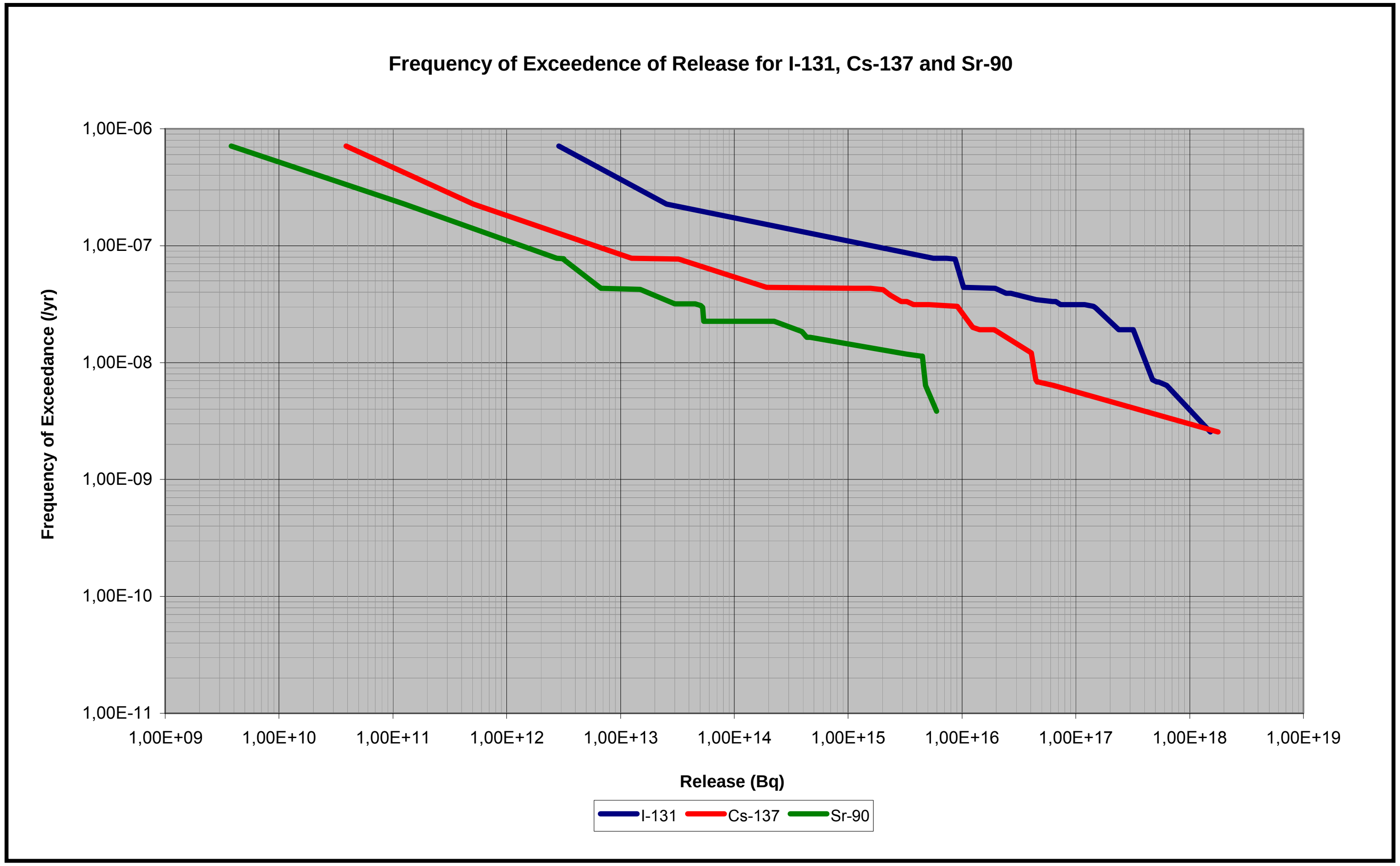
Failure Mode	Frequency /ry	%CDF
Intact	5.68E-07	89.2%
Not isolated	6.60E-09	1.0%
Bypassed	1.29E-08	2.0%
Early failure	1.25E-08	2.0%
Late failure	3.65E-08	5.7%
Total CDF	6.36E-07	100.0%

SUB-SECTION 15.4.4.4 - FIGURE 2



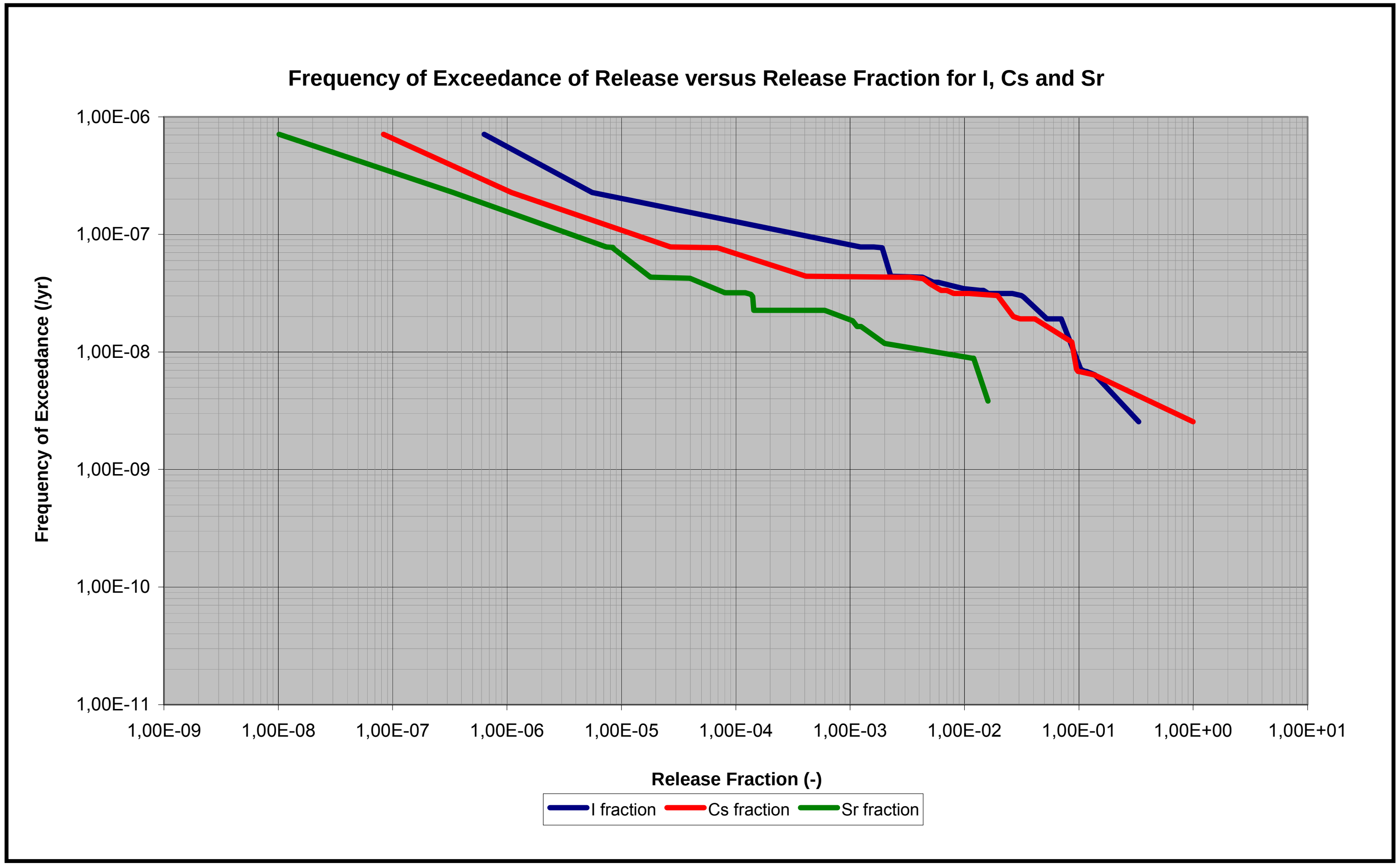
SUB-SECTION 15.4.4.4 - FIGURE 3

Frequency of Exceedance CCDF for Cs-137, I-131 and Sr-90 Releases (in Bq)



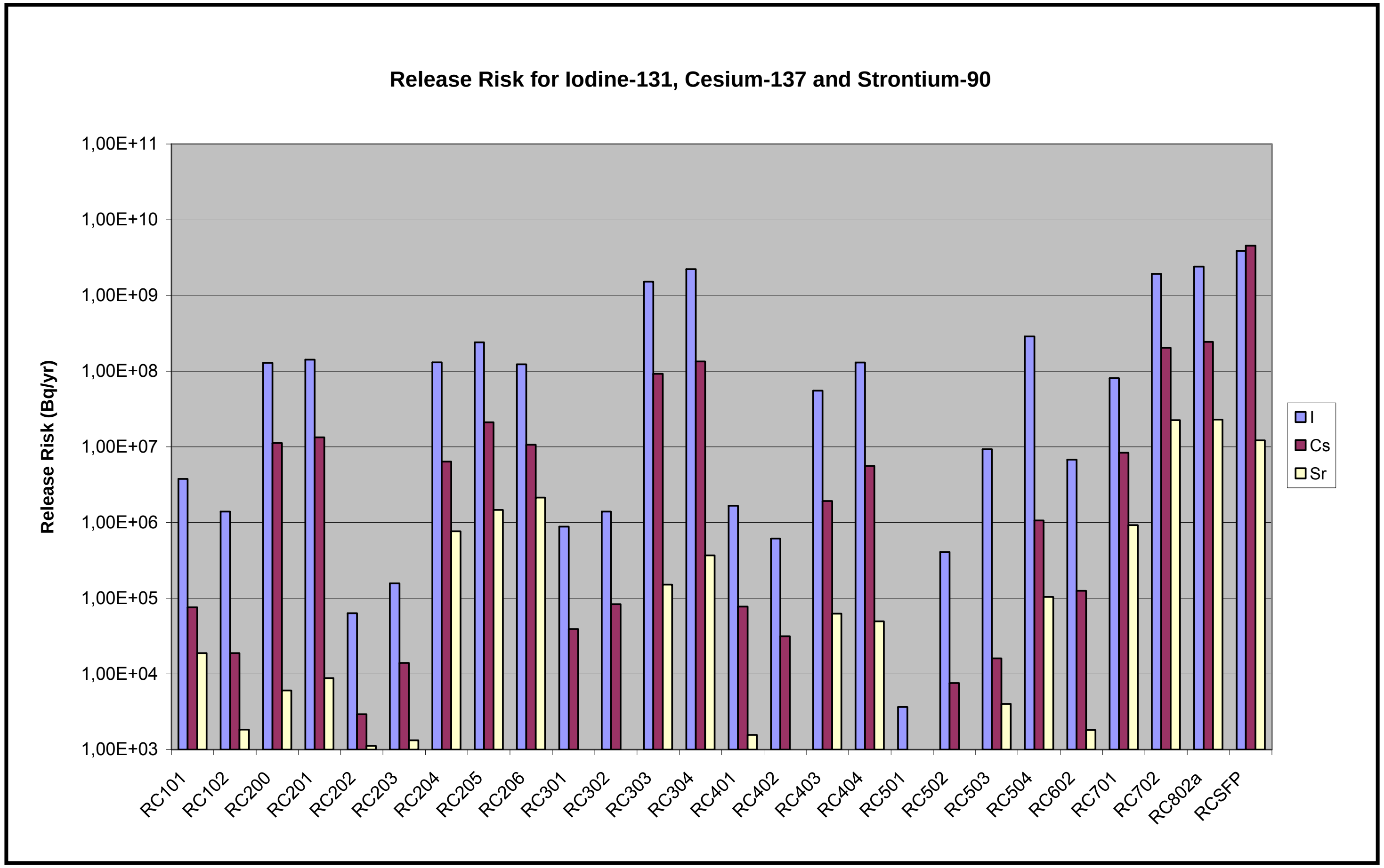
SUB-SECTION 15.4.4.4 - FIGURE 4

Frequency of Exceedance CCDF for Cs-137, I-131 and Sr-90 Releases (in fraction of initial core inventory)

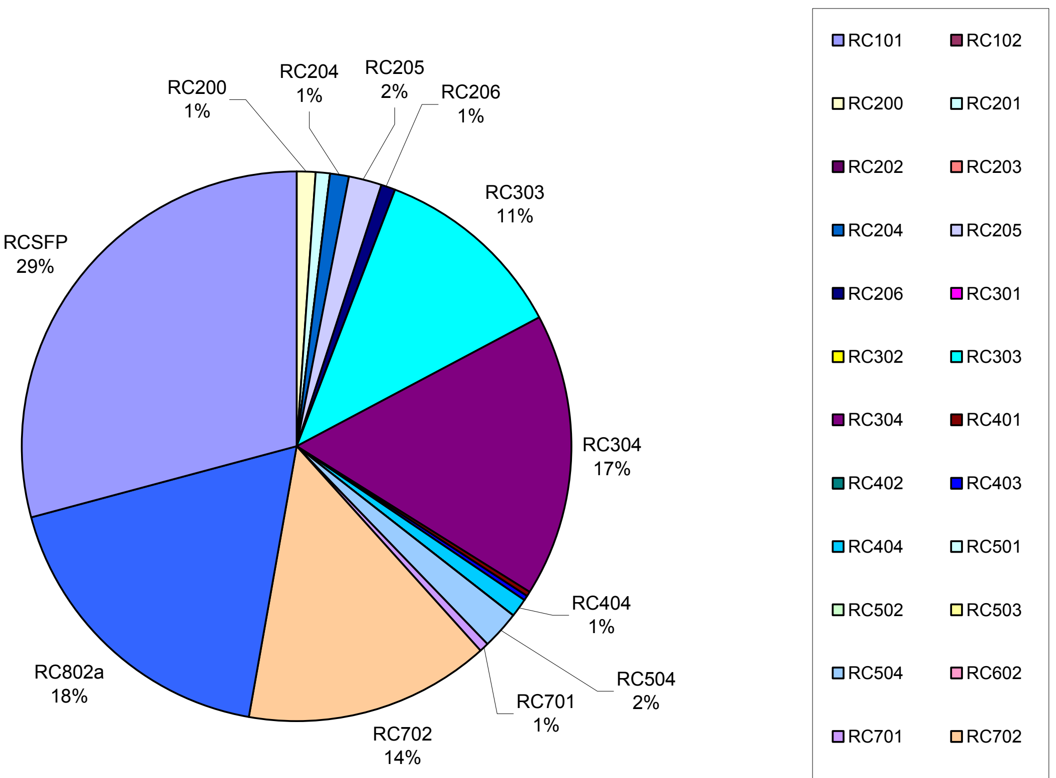


SUB-SECTION 15.4.4.4 - FIGURE 5

Relative Release Risk for Cs-137, I-131 and Sr-90 Releases by Release Category (Bq/y)

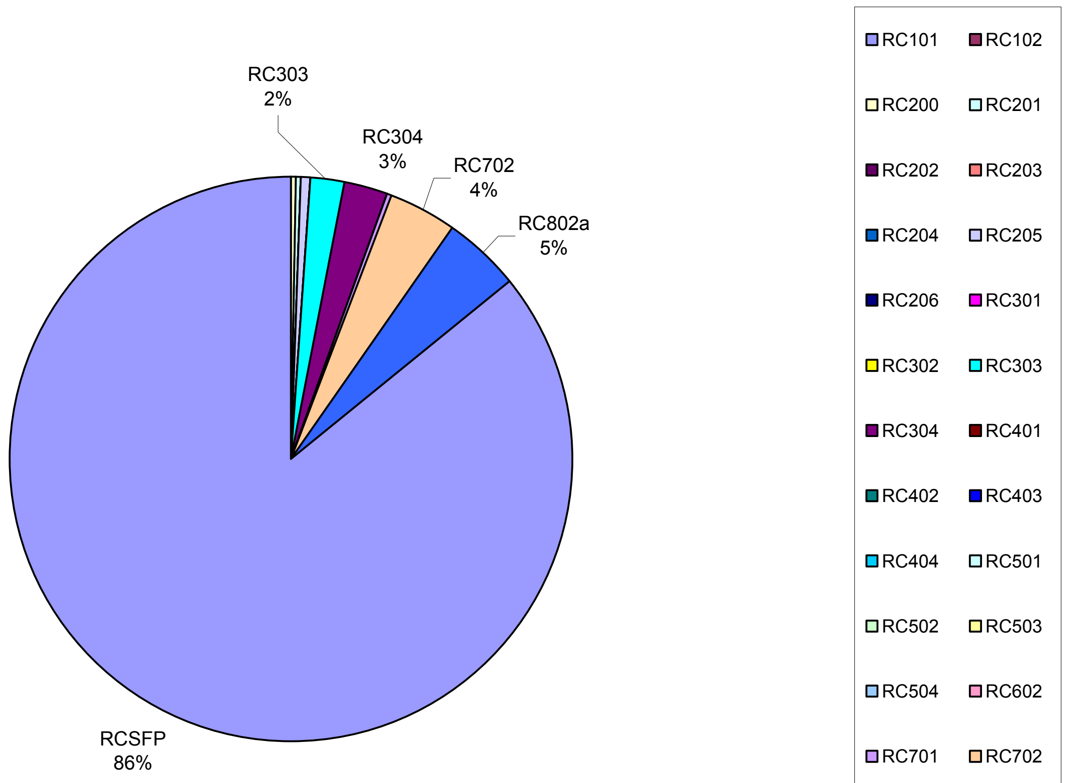


SUB-SECTION 15.4.4.4 - FIGURE 6



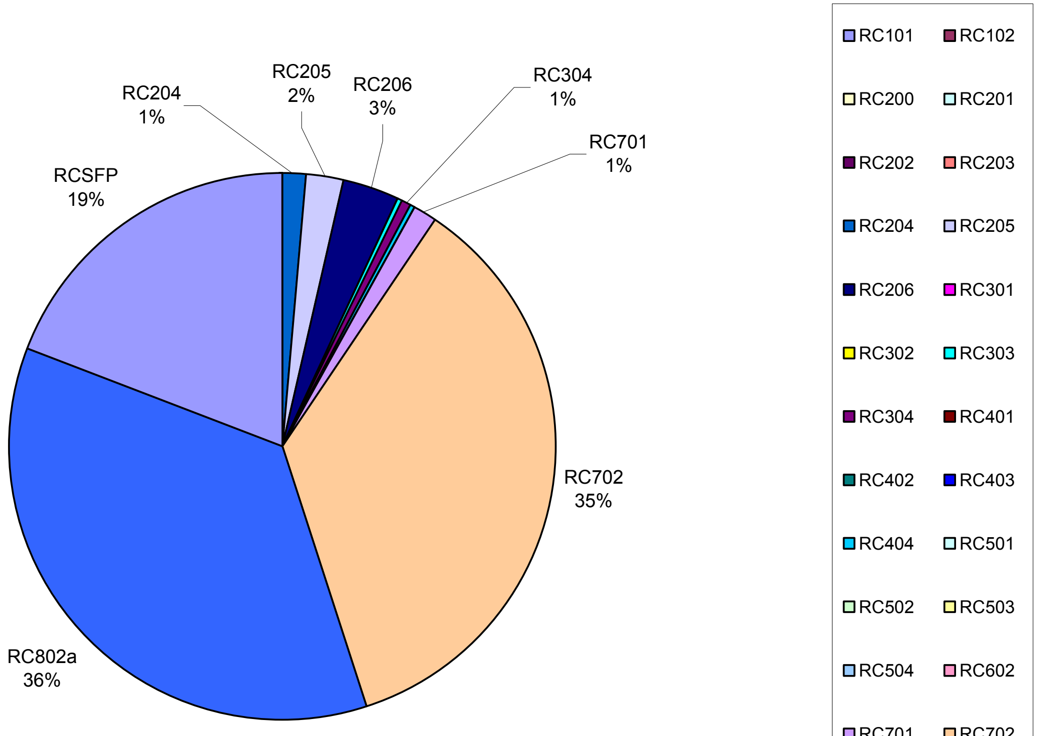
Relative Contribution to I-131 Release Risk by Release Category

SUB-SECTION 15.4.4.4 - FIGURE 7



Relative Contributions to Cs-137 Release Risk by Release Category

SUB-SECTION 15.4.4.4 - FIGURE 8



Relative Contribution to Sr-90 Release Risk by Release Category

4.5. SENSITIVITY AND UNCERTAINTY ANALYSES

4.5.1. Uncertainty analysis results for all states

Uncertainty analysis was performed for the Large Release Frequency (LRF) and the Large Early Release Frequency (LERF), both as defined in section 4.3 of this sub-chapter. A sample size of 30,000 was used for this analysis. The following results were obtained:

- For the the 5th percentile was 1.84E-8/ry, the median was 3.94E-8/ry, and the 95th percentile was 1.41E-7/ry.

- For the, the 5th percentile was 7.22E-9/ry, the median was 1.81E-8 /ry, and the 95th percentile was 7.6E-8 /ry.

Sub-section 15.4.4.5 - Figure 1 shows the cumulative probability and probability density curves for LRF. Sub-section 15.4.4.5 - Figure 2 is the corresponding curves for LERF.

Note that for both the LRF and the LERF some discrepancy is seen between the sampled mean frequencies and the point-estimate means generated by multiplication and summation of basic event values and cutset frequencies.

For the LRF the sampled mean is approximately 16.3% lower than the point-estimate value obtained from the corresponding MCS Analysis case (run to generate the merged cutsets for LRF and LERF respectively for uncertainty analysis). This point-estimate value is in turn a further 4.3% lower than the point-estimate values generated by summation of the individual RC frequencies (as presented in section 4.1 of this sub-chapter).

For the LERF the sampled mean is approximately 13.5% lower than the point-estimate value obtained from the corresponding MCS Analysis case (run to generate the merged cutsets for LRF and LERF respectively for uncertainty analysis). This point-estimate value is in turn a further 0.2% lower than the point-estimate values generated by summation of the individual RC frequencies (as presented in section 4.1 of this sub-chapter).

The reasons for the 13.5 to 16.3% discrepancy have not been investigated in detail. On the other hand, the reasons for the 0.2 to 4.3% discrepancy between the merged MCS quantification and the summated RC frequencies are well understood. It is well known that the numerical summation process only approximates the correct LRF/LERF values due to the presence of cutsets in the individual RC frequency cutset lists which are absorbed (non-minimal cutsets) when a Boolean combination of the individual cutset lists is performed.

Finally, for clarity, note that only the numerical results from the most conservative case (summated RC frequencies) are presented throughout Sub-chapter 15.4.

4.5.2. Sensitivity analysis results for all states

4.5.2.1. Sensitivity to phenomenological events

As discussed in section 3.6 of this sub-chapter, sensitivity to phenomenological events is assessed by evaluating the impact on the Large Early Release Frequency (LERF) and the Large Release Frequency (LRF) for two cases. The first case evaluated is where the phenomenological event is assumed never to occur (probability = 0). The second case evaluated is where the phenomenological event is assumed always to occur (probability=1). As discussed in section 3.6 of this sub-chapter, this approach is judged appropriate for phenomenological events since they generally represent events whose outcome is deterministic but unknown, rather than stochastic. The results of the sensitivity studies with the probabilities of these events set equal to zero and one are summarised below.

The study of the impact on LERF and LRF of setting phenomenological events equal to zero did not identify any important sensitivities (change in LRF or LERF by a factor of 2 or greater). The largest impact for any one event was a reduction in LERF of 26.8%, corresponding to the event L2PH VECF-FA(H), which represents very early containment failure due to hydrogen flame acceleration loads in a high pressure core damage sequence. Similar sensitivity results are seen for LRF; in the case of LRF the reduction corresponding to setting L2PH VECF-FA(H) = 0 is 14.7%.

The study of the impact on LERF and LRF of setting phenomenological events equal to one identified several cases where the increase in LERF or LRF exceeded a factor of two when the basic event probability was changed. For LERF the following important sensitivities were observed:

- Setting the event L2PH VECF-FA(H), which represents very early containment failure due to flame acceleration loads in a high pressure core damage sequence, to a probability of 1.0 results in a factor of 17.5 increase in LERF.

- Setting the event L2PH VECF-FA(HL), which represents very early containment failure due to flame acceleration loads in a high pressure core damage sequence with hot leg rupture, to a probability of 1.0 results in a factor of 17.3 increase in LERF. The order of magnitude is close to that from event L2PH VECF-FA(H), as both basic events impact the same function event “containment failure before vessel breach”.

- Setting the event L2PH STM EXP INV LP, which represents failure of the containment due to an in-vessel steam explosion in a depressurised core damage sequence, to a probability of 1.0 results in a factor of 17 increase in LERF. The phenomenological evaluation of this event resulted in a very low base probability indicating that the event occurrence has a very low credibility. Increasing its base probability to 1.0 leads to an increase in the RC302 frequency and therefore the LERF.

- Setting the event L2PH VECF-H2DEF(HL), which represents very early containment failure due to hydrogen deflagration loads in a high pressure core damage sequence with hot leg failure, to a probability of 1.0 results in a factor of 17 increase in LERF.

- Setting the event L2PH VECF-H2DEF(H), which represents very early containment failure due to hydrogen deflagration loads in a high pressure core damage sequence after pressuriser valve cycling phase, to a probability of 1.0 results in a factor of 13.7 increase in LERF.

For LRF the following important sensitivities were observed:

- Setting the event L2PH VECF-FA(H), which represents very early containment failure due to flame acceleration loads in a high pressure core damage sequence, to a probability of 1.0 results in a factor of 10.1 increase in LRF.

- Setting the event L2PH VECF-FA(HL), which represents very early containment failure due to flame acceleration loads in a high pressure core damage sequence with hot leg rupture, to a probability of 1.0 results in a factor of 10 increase in LRF. The order of magnitude is close to that from event L2PH VECF-FA(H), as both basic events impact the same function event “containment failure before vessel breach”.

- Setting the event L2PH STM EXP LH LP, which represents a steam explosion causing failure of the lower head in a depressurised core damage sequence, to a probability of 1.0 results in a factor of 9.82 increase in LRF.

- Setting the event L2PH STM EXP INV LP, which represents failure of the containment due to an in-vessel steam explosion in a depressurised core damage sequence, to a probability of 1.0 results in a factor of 9.75 increase in LRF. The sensitivity to this event is seen because it applies to a large number of accident sequences, due to the natural and engineered depressurisation mechanisms. The phenomenological evaluation of this event resulted in a very low base probability indicating that the event occurrence has a very low credibility.

- Setting the event L2PH VECF-H2DEF(HL) which represents very early containment failure due to hydrogen deflagration loads in a high pressure core damage sequence with hot leg rupture, to a probability of 1.0 results in a factor of 9.72 increase in LRF.

- Setting the event L2PH VECF-H2DEF(H) which represents very early containment failure due to hydrogen deflagration loads in a high pressure core damage sequence during the in-vessel phase with the pressuriser safety valves cycling, to a probability of 1.0 results in a factor of 7.91 increase in LRF.

- Setting the event L2PH CCI, which represents significant MCCI under conditions of a flooded pit, to a probability of 1.0 results in a factor of 6.5 increase in LRF. The sensitivity to this event is seen because RC602 (basemat penetration) is classed as large release.

- Setting the event L2PH STM EXP EXV, which represents damage to the reactor pit following an ex-vessel steam explosion, to a probability of 1.0 results in a factor of 6.04 times the increase in LRF. The sensitivity to this event is seen for the same reasons as the sensitivity to L2PH STM EXP LH LP; RC602 (basemat penetration) is classified as a large release and there is a conservative modelling assumption that any steam explosion induced damage to the reactor pit will impact the melt retention and transfer from the pit in such a way that it leads to failure of ex-vessel melt stabilisation with a probability of 1.0.

4.5.2.2. Contribution of operator actions

Importance analysis results were reviewed to identify operator actions contributing more than 5% to LRF or to LERF.

The following operator actions contribute more than 5% to LRF:

- OPD-L2-ENTEROSSAM, which is the dependent operator failure to enter the OSSA guidelines, contributes up to 8.83% of LRF.
- OP_FSCD_30MN, which is a Level 1 PSA operator error to perform primary fast cooldown, contributes up to 7.45% of LRF.
- OPD-L2-ENTEROSSA-28H, which is the dependent operator failure to enter the OSSA guidelines in the long term, after failing to enter the OSSA early, contributes up to 7.17% of LRF.
- OP_SBODG2H, which is a Level 1 PSA operator error to start SBO, contributes up to 6.17% of LRF. No subsequent SBO start is considered in Level 2 PSA
- OP_EFWS_NCSS, which is a Level 1 PSA operator error to start and control ASG [EFWS] using the NCSS, contributes up to 5.88% of LRF.
- OP_BLEED_30MN_NCSS, which is a Level 1 PSA operator error to initiate bleed in 30 minutes using the NCSS, contributes up to 5.11% of LRF.

The following operator actions contribute to more than 5% to LERF:

- OPD-L2-CIH NCSS, which is an operator failure to close containment isolation valves using the NCSS, contributes up to 8.95% of LERF.
- OP-EFWS_NCSS, which is an Level 1 PSA operator error to start and control ASG [EFWS] using the NCSS, contributes up to 8.24% of LERF.
- OP_FSCD_30MN, which is a Level 1 PSA operator error to perform primary fast cooldown within 30 minutes, contributes up to 8.24% of LERF.
- OP_FB_120M_MDEP_NCSS, which is a Level 1 PSA operator error to initiate feed and bleed within 120 minutes, contributes up to 7.4% of LERF.
- OP_BLEED_30MN NCSS, which is a Level 1 PSA operator error to perform feed and bleed within 30 minutes from the NCSS, contributes up to 7.34% of LERF.
- OP_SCD 30MN, which is a Level 1 PSA operator error to initiate secondary cooldown within 30 minutes used in SGTR sequences, contributes up to 6.54% of LERF.
- OPD-L2-CIH, which is an operator failure to close containment isolation, contributes up to 5.41% of LERF.

4.5.2.3. Sensitivity to operator error, function events, system trains and modelling assumptions

A series of sensitivity studies were performed identified as S1 to S6 to assess the impact of sets of operator actions, CET functional events, system trains and modelling assumptions used during the development of the CET models and supporting fault trees. The set of sensitivity studies performed was as follows, based on an initial LERF of 4.07E-08/ry and an initial LRF of 7.69E-08/ry:

Case	Description	New LERF (frequency /yr)	New LRF (frequency /yr)	Change in LERF (%)	Change in LRF (%)
S1	Human actions for containment isolation failed.	**6.35E-08**	**9.69E-08**	**+56%**	**+32%**
S2	One EVU [CHRS] train failed	**4.26E-08**	**1.74E-07**	**+5%**	**+136%**
S3	No primary depressurisation (potential for creep rupture or HPME)	**8.90E-08**	**1.29E-07**	**+119%**	**+75%**
S4	All operator errors set to zero (equivalent to force the success of all operator actions)	**3.01E-08**	**5.76E-08**	**-26%**	**-22%**
S5	EVU [CHRS] sprays not available for long term heat removal or for ST mitigation (not containment isolation failure case)	**4.26E-08**	**2.04E-07**	**+5%**	**+177%**

In case S1, where no credit is taken for human action to verify the containment isolation and perform the manual back-up of the containment isolation signal, the LERF increases by 56% and the LRF increases by 32%.

In case S2, where no credit is taken for one of the EVU [CHRS] trains, the LERF increases by 5% and the LRF increases by 136%. This sensitivity case highlights the importance of the EVU [CHRS] to fulfil the required containment depressurisation to avoid containment failure in the long term. The calculated increase in the LRF is an average calculated value between the increase in LRF for the train 1 failed and for the train 2 failed. Internal hazard modelling is asymmetric; only division 1 may be lost due to a fire or flooding event. When the EVU [CHRS] train 2 is set failed, the EVU [CHRS] train 1 may also be partially unavailable due to a fire in division 1. In such a case both trains are unavailable and the impact on the LRF is higher.

In case S3, where no credit is taken for the RCP [RCS] depressurisation, the LERF increases by 119% and LRF increases by 75%. As part of the EPR concept the RCP [RCS] depressurisation system is dedicated to avoid the high pressure accident sequences, and the linked SGTR and HPME.

In case S4, where no account is taken of the potential operator errors, the LERF decreases by 26% and LRF decreases by 22%. This is representative of the impact of the human actions on both the LRF and the LERF.

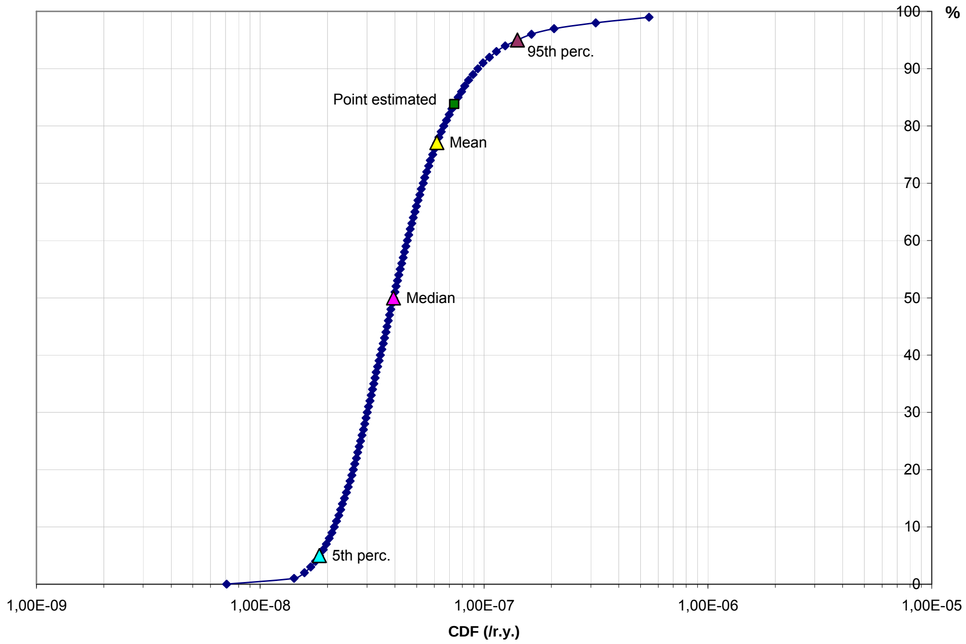
In case S5, where no credit is taken for both trains of the EVU [CHRS] in terms of long term containment depressurisation capacity or source term reduction usage, LERF increases by 5% and LRF increases by 177%.

4.5.2.4. Sensitivity to maintenance events

Sensitivity studies addressing the impact of maintenance on LERF and LRF are presented in Sub-chapter 15.7.

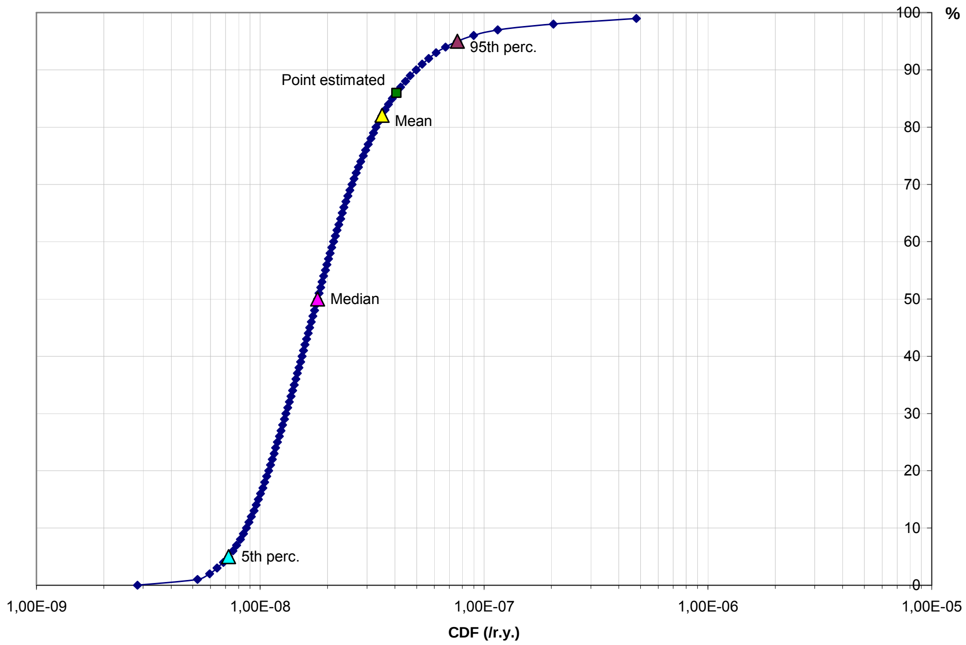
SUB-SECTION 15.4.4.5 - FIGURE 1

Cumulative Probability Distribution for Large Release Frequency (LRF) for all states



SUB-SECTION 15.4.4.5 - FIGURE 2

Cumulative Probability Distribution for Large Early Release Frequency (LERF) for all states



5. CONCLUSIONS AND INSIGHTS

This sub-chapter of the PCSR has presented the methodology and results of the UK EPR PSA Level 2 PSA study.

Total Release Frequency

- The severe accident frequency (the core damage frequency – an input to the Level 2 PSA) is 6.36E-7/ry for states A and B. This increases to 7.08E-7 when states C, D and E are included, and to 7.11E-7 when states C, D, E and the spent fuel pool are included.
- The Level 2 PSA results show that the strong containment and dedicated severe accident mitigation measures of the EPR plant are efficient in reducing the frequency and magnitude of releases to the environment the case of a severe core damage event.
- These two points result in the absolute frequency of a large radioactivity release to the environment being predicted to be extremely low.

Magnitude of Releases

- A release of 100TBq of Cs-137 is used as a guide to define “large release”. This is a lower “target” than that applied in many countries.
- The frequency with which a release of this magnitude could be exceeded is calculated as 6.73E-8 /ry for states A and B, and 7.69E-8 /ry when states C, D, E and the spent fuel pool are included.
- The overall large release frequency defined in this way, including all states and the spent fuel pool, represents 10.8% of the core damage frequency.
- Comparison with the UK dose and risk targets is presented as part of the Level 3 PSA analysis in Sub-chapter 15.5 of the PCSR.

Contributors to Large Release Frequency

- Sequences initiated in states A and B contribute 87.5% to the total LRF, sequences initiated in state C contribute 8.3%, sequences initiated in states D and E contribute respectively 0.5% of the LRF and spent fuel pool events contribute 3.3%.
- At power, the main sequences included in the LRF are severe accident sequences with long term containment failure during and after debris quench due to rupture, without MCCI, with debris flooding, but with no containment spray.
- The heterogenous dilution leading to early containment failure (i.e. before vessel failure) without the containment spray working is the dominant mode of failure leading to large releases in state C.
- The containment isolation failures are not the dominant contributors to the LRF in shutdown states, following the modelling of the containment hatch reclosure in numerous sequences.

Contributors to Large Early Release Frequency

- Sequences initiated in states A and B contribute 84.1% to the total LERF, sequences initiated in state C contribute 14.9%, sequences initiated in state D contribute 0.2%, and sequences initiated in state E contribute 0.8%.
- At power, release category RC504, which contains all the long term containment failure without containment sprays, is considered in the LRF but not the LERF. The difference between the LRF and the LERF is therefore mainly due to the containment spray failure.

Contributors to Release Risk

- Decomposition of the release risk in terms of frequency reveals a dominant contribution from the late failure of the containment.
- In terms of Cs-137, I-131 and Sr-90, the major contributors to the risk are the spent fuel and the containment bypass severe accident sequences, including both ISLOCA and SGTR.
- Spent fuel pool accidents contribute significantly to the Cs-137 release risk.

Uncertainty analysis results

Uncertainty analyses indicate that the median large release frequency is 3.94E-8 /ry with the 95th percentile frequency being 1.41E-7 /ry. These results increase confidence that the large release frequency for the EPR is small.

Model sensitivity

Investigation of the sensitivity of results for large release frequency generated with the Level 2 PSA model indicate a significant sensitivity to flame acceleration loads in a high pressure core damage sequence and in-vessel steam explosion.

The in-vessel steam explosion events referred to above include those which directly cause containment failure and those which damage the reactor pit and disrupt the ex-vessel melt stabilisation process. The ex-vessel events studied in these sensitivities also impact the release frequency via disruption of the melt stabilisation process. It is to be noted that this sensitivity to steam explosion events which indirectly affect melt stabilisation arises because of conservative assumptions regarding (i) basemat melt-through, which has been classified as a large release, and (ii) the impact of reactor pit damage, which is assumed to cause failure of the melt stabilisation process with a probability of 1.0.

The sensitivity study of the human actions to close the containment isolation valves, to initiate secondary cooldown, to perform feed and bleed and fast cooldown in all cases (using the NCSS or not) shows its importance to the LERF. The sensitivity study of the human actions to enter the OSSA guidelines, to perform primary fast cooldown, to start SBO, to start and control ASG [EFWS] using the NCSS shows its importance to the LRF.

When no account is taken of the potential operator errors, the LERF decreases by 26% and LRF decreases by 22%. This is representative of the impact of the human actions on both the LRF and the LERF.

SUB-CHAPTER 15.4 – 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.

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