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06	Consolidated PCSR update: - Sub-chapter 15.7 - Figure 7 and §3.3 updated consistent with Sub-chapter 15.4 (Sub-chapter 15.4.4.4 - Figure 7) - Clarification of text (§3.2 and Sub-chapter 15.7 - Figure 6)	21-11-2012

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SUB-CHAPTER 15.7 – PSA DISCUSSION AND CONCLUSIONS

1. INTRODUCTION

This sub-chapter presents the results and insights of the PSA, summarising the following analyses:

- The Level 1 PSA for internal events, as presented in Sub-chapter 15.1.
- The Internal and External Hazards analysis, presented in Sub-chapter 15.2.
- The accident in the fuel pool analysis, presented in Sub-chapter 15.3.
- The Level 2 PSA presented in Sub-chapter 15.4. This evaluates the nature, magnitude and frequency of radioactive releases to the environment as a result of the events analysed in the preceding sub-chapters.
- The Off-site Consequence Risk Assessment (Level 3 PSA) is presented in Sub-chapter 15.5, which assesses radiological impacts on persons off-site from these releases, in terms of individual and societal risk.
- The Seismic Margin Assessment (SMA), presented in Sub-chapter 15.6.

An iterative process to identify design improvements using PSA was implemented throughout the development of the EPR design, and the main examples of design changes made resulting from PSA studies are presented in this sub-chapter.

Unavailability due to repairs and preventive maintenance activities is included in the Level 1 PSA model. Quantification without preventive maintenance has been performed and the results are presented here. Other sensitivity analyses presented here consider the potential for ‘cliff edge’ effects from transients of more than 24 hours, initiating event frequencies, common cause failures, operator actions, certain reliability data and system design features.

The probabilistic studies performed for the UK EPR design during the GDA give assurance that the risk from accidents leading to release of radioactivity into the environment is reduced as low as reasonably practicable.

2. SUMMARY OF LEVEL 1 RESULTS

The overall **Core Damage Frequency (CDF) is calculated to be 7.1E-07/r.y**. This includes the contributions

- From internal events, internal hazards and external hazards (excluding seismic events),
- From all reactor states and,
- Including preventive maintenance in at-power and shutdown states.

The CDF without the inclusion of preventive maintenance is assessed in section 6.1 of this sub-chapter and is calculated to be 6.5E-07/r.y.

This level of risk is significantly below the safety objective of 1E-05/r.y.

Sufficient margins are expected to be available to include the additional core damage frequency contributions due to seismic events or additional external hazards (those not yet analysed) without exceeding the probabilistic target.

The evaluation of the consequences of a loss of cooling event or a draining event in the Spent Fuel Pool confirms that there is negligible risk of fuel damage in the Spent Fuel Pool. The global risk of **fuel damage is calculated to be 2.6E-09/r.y**, for which the draining events are the main contributors (90%).

The following sections of this sub-chapter discuss the PSA Level 1 results, conclusions and insights.

2.1. RISK RESULTS – CORE DAMAGE FREQUENCIES

The following table presents the main results (point estimates) for the CDF from internal events, internal hazards and external hazards with and without preventive maintenance (PM). The sensitivity analysis dealing with PM impact on CDF is presented in section 6.1 of this sub-chapter.

CDF with preventive maintenance:

	Internal Events	**Internal Hazards**	**External Hazards**	**TOTAL**
At-Power state and Hot Shutdown state (A&B)	4.61E-07	1.01E-07	7.43E-08	6.36E-07
Shutdown states (Ca, Cb, D, E)	7.08E-08	-	1.60E-09	7.24E-08
TOTAL	**5.31E-07**	**1.01E-07**	**7.59E-08**	**7.08E-07**

CDF without preventive maintenance:

	Internal Events	**Internal Hazards**	**External Hazards**	**TOTAL**
At-Power state and Hot Shutdown state (A&B)	4.22E-07	8.27E-08	7.34E-08	5.78E-07
Shutdown states (Ca, Cb, D, E)	6.72E-08	-	1.60E-09	6.88E-08
TOTAL	**4.89E-07**	**8.27E-08**	**7.50E-08**	**6.47E-07**

The additional core damage frequency due to internal hazards during shutdown states is considered to be negligible for the following reasons:

- Fire and Flooding would be detected with a higher probability because the personnel working on systems and components for tests and maintenance will detect the occurrence of such internal hazards quickly; and

- Longer grace periods during plant shutdown lead to more reliable measures to cope with internal fire or flooding.

The detailed discussion of the screening of internal hazards during shutdown states is presented in Sub-chapter 15.2.

2.2. RISK DISTRIBUTION

Note: The following results consider the CDF with preventive maintenance. The detailed impact of the preventive maintenance is analysed in section 6.1 of this sub-chapter.

2.2.1. Risk distribution between events and hazards

Internal events contribute 75% to the total CDF, internal hazards contribute 14% and external hazards contribute 11%. This is illustrated in Sub-chapter 15.7 - Figure 1.

2.2.2. Risk distribution between at-power and shutdown

Power operation contributes 90% (6.36E-07/r.y) to the overall CDF whereas shutdown states contribute 10% (7.2E-08/r.y). This is illustrated in Sub-chapter 15.7 - Figure 2.

The time spent in the shutdown states (C to E) represents around 4% of the year. Due to the improvement of the protection during shutdown states, there is no significant difference for the level of hourly risk between at-power and shutdown states. It should nevertheless be noted that the level of risk depends significantly on the state during shutdown.

2.2.3. Risk distribution between the initiating event groups

Sub-chapter 15.7 - Figure 1 shows the different initiating events and their contribution to the overall CDF.

As can be seen, the Loss of Cooling Chain (LOCC) initiating events dominate the CDF, accounting for about 17% to the total CDF.

The second highest contributor to the overall CDF is the LOCA initiating event group which contributes 15% to the total CDF. This relative contribution value is typical for pressurised water reactors.

The Loss of Off-site Power (LOOP) initiating events contribute 15% to the total CDF. This is mainly explained by the use of active components in the accident sequence. If consequential LOOP events following a reactor trip are included, the total LOOP contribution will increase to 21% and become the most dominant initiating event group.

The next contributor is the internal hazards, which contributes 14% to the total CDF. The dominant internal hazards are the fire events, which contribute about 13% to the total CDF. Within the fire events, a fire in the safeguard building and a fire in the switchgear building contribute 10.6% and 2.0% respectively to the CDF.

The primary transients contribute 12% to the total CDF. Within these transients, the homogeneous boron dilution, the uncontrolled drop of primary level and the spurious reactor trip contribute respectively 4.7%, 3.2% and 2.4% to the total CDF.

The external hazards (Loss of Ultimate Heat Sink initiating events) are also an important CDF contributor, accounting for close to 11% of the total CDF.

2.3. IMPORTANCE RANKING

Significant Structures, Systems and Components (SSC), I&C components, operator actions and common cause events are defined in Sub-chapter 15.1 for internal events and in Sub-chapter 15.2 for internal and external hazards.

The most important components, systems, operator actions and common cause failures according to the Level 1 PSA are discussed below.

2.3.1. Significant components

2.3.1.1. Fussel-Vesely Ranking

Sub-chapter 15.7 - Table 1 shows the forty most risk-significant component events ranked using the Fussel-Vesely (FV) importance measure (I&C, post-accident operator actions and common cause events are studied independently in sections 2.3.2 to 2.3.4 of this sub-chapter.) This parameter evaluates the weight of a basic event “A” in the total CDF as the ratio of the frequency of the minimal cutsets involving the event “A” to the overall frequency.

The highest ranked components are the main coolant pump shaft seal and the O-rings of the DEA [SSSS] exposed to RCP [RCS] pressure and temperature. These components provide the leak tightness of the main coolant pumps following the loss of both seal injection via the RCV [CVCS] and thermal barrier cooling by the RRI [CCWS]. The failure of the reactor coolant pump sealing system causes a small LOCA during loss of cooling chain transients (i.e. LOCC, LUHS) or during loss of off-site power transients (LOOP). Their importance is a result of the relatively high failure probability assumed in the PSA (see sensitivity analysis in section 6 of this sub-chapter).

The second highest ranked item is not explicitly a component, but the preventive maintenance on the cooling chain RIS [SIS]/RRI [CCWS]/SEC [ESWS] during power operation. During this preventive maintenance, one MHSI/LHSI train is unavailable. As the safety injection is needed in numerous transients and accidents, the reduction of RIS [SIS] redundancy makes a high impact on CDF.

The next most important components are the diesel generators: the four main emergency diesel generators LHP/Q/R/S (EDG) and the two station blackout diesels generators LJP/S (SBO-DG) which contribute to the mitigation of loss of offsite power events. Following the LOOP in at-power and shutdown states, which is one of the most dominant CDF contributors for the overall risk, the EDGs and SBO-DGs ensure the power supply of the safety systems. The high importance of the diesel is mainly due to the relatively low reliability of this equipment and the important contribution of the LOOP event to the overall CDF. The unavailability of EDG and SBO-DG due to preventive maintenance is ranked within the top 20.

The next most important components are the MHSI pumps, which contribute to the mitigation of numerous transients and accidents by performing the primary inventory control as well as the primary feed function. The failure of one MHSI pump reduces the system redundancy and, as a result, has an important impact on CDF.

The next most important component is the GCT [MSB]. The failure of GCT [MSB] impacts the partial cooldown function required during some LOCA situations; the reliability of GCT [MSB] is low, therefore, the loss of GCT [MSB] has a high ranking in terms of FV.

Note: As common cause failures are excluded from this part of the analysis, the highest contributing components do not include complete failure of all four trains of the 100% safety systems.

2.3.1.2. Risk Increase Factor Ranking

Sub-chapter 15.7 - Table 2 shows the forty most risk-significant component events (I&C and common cause events excluded) based on the Risk Increase Factor (RIF) importance measure. This parameter evaluates the factor by which the CDF is increased if the basic event occurs with a probability of 1.

The most important components using this measure are the reactor coolant pump breakers. These breakers trip the reactor coolant pumps following a loss of RCV [CVCS] seal injection and loss of thermal barrier cooling. If they do not operate correctly, a seal LOCA is assumed to occur.

The second most important component is the LHD busbar. This busbar (10kV) provides the power supply to train 4 of the RRI [CCWS], SEC [ESWS] and MHSI pump, as well as the RCV [CVCS] stand-by charging pump.

The next most important component is the heat exchanger of LHSI train 4. This failure makes the injection and heat removal functions impossible with the corresponding LHSI train.

2.3.2. Significant Systems

A ranking of systems based on their fractional contribution (based on Risk Decrease Factor) is illustrated in Sub-Chapter 15.7 - Figure 3. I&C is shown to be the most significant system (23.4%) which is expected for an active system design. The RCP [RCS] is the second most safety significant system (16.5%). This arises from its role in preventing LOCA events following the loss of reactor coolant pump seal cooling in the event of LOOP, LOCC and LUHS. The third most safety significant system is the RIS [SIS] (13.3%), which is needed to perform Safety injection or RHR for numerous transients and accident situations.

The importance of the support systems (10.6%) including cooling chains (RRI [CCWS] / SEC [ESWS] and DEL[SCWS]) with 2.0%, and electrical supply systems (including EDG: 5.8%, SBO: 2.5% and busbars: 0.2%) reflects the functional dependency between support and supported systems. The contribution of support systems is expected for an active system design. It should be noted that this contribution is underestimated as the failure of the cooling chain is also an initiating event during both at-power and shutdown states and the contribution of the initiating event to the CDF is not included in the present importance analysis.

The systems involved in the residual heat removal function (ASG [EFWS], ARE [MFWS], AAD [SSS], VDA [MSRT], VVP [MSSS] / MSSV, GCT [MSB]) are the next most important contributors (4.9%).

Preventive maintenance of the safety systems contributes to 3.9% to the core damage frequency.

2.3.3. Significant operator post-accident actions

Sub-chapter 15.7 - Table 3 shows the risk-significant operator post-accident actions based on the Fussel-Vesely (FV) importance measure discussed in sub-section 2.3.1.1 of this sub-chapter.

- The first most important operator post-accident action is the actuation of the fast secondary cooldown with a time window of 30 minutes. This manual action is required if the MHSI trains are unavailable following a small primary break.

- The second most important operator post-accident action based on FV is the manual start-up and control of ASG [EFWS] via the Non-Computerised Safety System (NCSS) with a time window of 60 minutes. This manual action is required in the case of a total loss of digital I&C in order to perform the secondary residual heat removal.

Sub-chapter 15.7 - Table 4 shows the risk-significant operator post-accident actions based on the Risk Increase Factor (RIF) importance measure discussed in sub-section 2.3.1.2 of this sub-chapter.

- The most important operator post-accident action based on RIF is the manual cross connection of the ASG [EFWS] tanks or the initiation of the ASG [EFWS] tank make-up if the inventory of one ASG [EFWS] tank is not sufficient. This is the case if the conditions for connection of the RIS [SIS] / RRA [RHRS] cannot be reached (when e.g. VDA [MSRT] are not available) or if the RIS [SIS] / RRA [RHRS] is unavailable (when e.g. a LUHS occurs). The high importance of this action is mainly due to the requirement for this function even if all four ASG [EFWS] trains are operating. However the reliability of this operator action is sufficiently high to avoid a major impact on the CDF.

- The second most important operator post-accident action based on RIF is the actuation of low head safety injection (RIS [SIS]) pumps 1 and 4, cooled by the safety chilled water system (DEL [SCWS]) with a time window of 120 minutes. The manual start of the LHSI with the dedicated cooling chain independent of the RRI [CCWS] / SEC [ESWS] following a total loss of cooling chain (TLOCC) or loss of ultimate heat sink (LUHS) in shutdown state D is required, because the loss of cooling results in a decrease in the primary coolant inventory due to boiling. The automatic makeup using the MHSI is unavailable as a consequence of the initiating event.

When looking at both rankings in combination (actions with FV>1% and RIF>2), the following operator post-accident actions are identified as most important:

ID	Failure Description	Nominal probability [per demand]	FV	RIF
OP_FSCD_30MN	Operator fails to initiate FSCD (30min)	4.3E-02	14.2%	4.2
OP_LHSI_IND_120MN	Operator fails to start LHSI independent of CCWS/ESWS (120min)	2.1E-03	3.6%	17.9
OP_BLEED_120MN	Operator fails to initiate Bleed (120min)	8.1E-03	3.3%	5.0
OP_FEED_TK	Operator fails the cross-connection of SG tank /Operator fails to re-feed SSS, MFWS or EFWS tank	1.0E-04	1.9%	190.4
OP_EFWS	Operator failure to start and control EFWS in case of PS failure	2.8E-03	1.7%	6.9
OP_SBODG2H	Operator fails to start SBO diesels or to close breakers within 2 hours	2.1E-03	1.6%	8.4
OP_SIS_INJ_80MN_NCSS	Operator fails to start SIS by MHSI/LHSI with NCSS (80min)	8.4E-03	1.3%	2.6

Three out of the seven identified operator post-accident actions have already been discussed above due to their high FV or high RIF. The four additional items identified are:

- Actuation of Feed and Bleed with a time window of 120 minutes. This action is required in case of the failure of secondary residual heat removal function, especially during primary and secondary transients.

- Ensure the start-up of the ASG [EFWS] and the control of the steam generator water level following the loss of the automatic start-up and control by the protection system.

- Starting the SBO diesel generators from the main control room within two hours. This action is required following the failure of the four emergency diesel generators in the case of LOOP.

- Ensure the start-up of MHSI or LHSI via NCSS. This action is required in case of an uncontrolled level drop in state Cb.

2.3.4. Significant Common Cause Failure events

The common cause failure (CCF) events are of high importance, as would be expected in a plant with four safety divisions.

Sub-chapter 15.7 - Table 5 shows the risk-significant common cause events based on the Risk Increase Factor (RIF) importance measure discussed in sub-section 2.3.1.2 of this sub-chapter.

- The most important common cause event based on RIF is the failure to open the first isolation check valves in the RIS [SIS] injection line. Failure to open this check valve causes the loss of an entire RIS [SIS] injection line (MHSI/accumulator/LHSI).

- The second most important common cause event is the total loss of SEC [ESWS] pumps in operation or the total loss of the RRI [CCWS] pumps in operation. The consequence of this failure is the total loss of the cooling chain supporting the MHSI pumps, LHSI pumps 2 and 3, RCV [CVCS] pumps, thermal barrier etc.

- The next most important common cause event is the CCF of the Emergency Feedwater System (ASG [EFWS]) valves for control of steam generator level and the CCF of the ASG [EFWS] pumps themselves. The consequence of the failure of the ASG system is the loss of residual heat and depressurisation with secondary side in most of the transients and LOCA.

Sub-chapter 15.7 - Table 6 shows the risk-significant common cause events based on the Fussel-Vesely (FV) importance measure discussed in sub-section 2.3.1.1 of this sub-chapter. The three most risk-significant common cause events based on FV are:

- The total loss of MHSI pumps.

- The total loss of emergency diesel generators.

- The total loss of ASG [EFWS] pumps.

2.3.5. Significant I&C events

Sub-chapter 15.7 - Table 7 shows the significant I&C events based on FV importance as discussed in sub-section 2.3.1.2 of this sub-chapter.

- The most important I&C event is the common cause event leading to the total failure of the SPPA-T2000 platform.

- The second important I&C event is the common cause event leading to the total failure of the TXS platform.

- The third most important I&C event is the common cause event leading to the total loss of the sub-system B of the RPR [PS]. The next most important I&C event is the common cause event leading to the total loss of the sub-system A of the RPR [PS].

- The next most important I&C event is is the common cause event leading to the total loss of the Non-Computerised Safety System (NCSS).

2.3.6. Summary of significant events

Sub-chapter 15.7 - Table 8 shows the significant events for components, I&C events, Common Cause Failure events and operator post-accident actions (both FV and RIF values are given).

These are discussed in more detail in sections 2.3.1 to 2.3.5 of this sub-chapter.

2.4. DOMINANT CORE DAMAGE SEQUENCES

The 50 most frequent Minimal Cutsets (MCS) are listed in Sub-chapter 15.7 - Table 9. This table lists the frequency of the MCS, the percentage contribution, cumulative frequency and cumulative percentage contribution to the overall CDF. A MCS description is also provided.

The results show that over 11000 MCS contribute to the first 95% of the overall CDF. This shows there are no significant outliers in the UK EPR overall CDF.

The 50 most frequent MCS can be grouped into the following main accident sequences, which contribute to 36% of the overall CDF:

Internal Event group sequences:

1. Among the 50 most frequent MCS, the dominant initiating events are the primary breaks, which contribute 9.6% of the overall CDF with a Core Damage Frequency of 6.8E-8. In this event group, the small break [2 – 45 cm^2] contributes to 5.46E-8. The significant contribution of the loss of coolant accident is mainly due to Common Cause Failures of MHSI pumps that perform Safety Injection. MHSI pumps are indeed markedly less reliable than other EPR pumps. [MCS # 2, 5, 6, 8, 25, 26, 30, 31, 41, 42 and 47]
2. Among the 50 most frequent MCS, the LOOP initiating events contribute 6.2% to the overall CDF, which is explained by the use of active components in the accident sequence. If consequential LOOP events following a reactor trip are included, the contribution will increase to 6.8%. [MCS #7, 10, 11, 27, 29, 32, 33, 35, 36, 37, 39 and 48]
3. Among the 50 most frequent MCS, the spurious reactor trip transients contribute 2.8% to the overall CDF, which is due to the high occurrence frequency of the initiating event. If the consequential LOOP sequences are excluded, the CDF contribution is 2.2%. [MCS #4 and 24]
4. Among the 50 most frequent MCS, the boron dilution transients, both homogeneous and heterogeneous, contribute 1.3% to the overall CDF with a frequency of 8.9E-09/r.y. [MCS #22 and 23]

Internal Hazard group sequences:

5. Internal Fire in the Safeguard Building followed by Reactor Coolant Pump seal LOCA due to failure of RCP shaft seals followed by the failure of the MHSI trains and the failure to perform fast secondary cooldown. Among the 50 most frequent MCS, this sequence contributes 3.7% of the overall CDF. [MCS #12, 13, 18 and 19]
6. Internal Fire in the Switchgear Building followed by the failure to perform secondary residual heat removal due to common cause failure to run the ASG [EFWS] pumps followed by the failure of the operator to actuate Feed & Bleed operation. Among the 50 most frequent MCS, this sequence contributes about 1% of the overall CDF. [MCS #14]

Loss of Cooling Chain group sequences (including LUHS event):

7. External hazard (mainly organic material blocking the flow path) leading to the loss of the ultimate heat sink during at-power operation followed by either the failure to perform secondary heat removal because of ASG [EFWS] failure or the failure of the reactor coolant pumps seal inducing a Loss of Coolant Accident. Among the 50 most frequent MCS, this sequence contributes 6.0% of the overall CDF. [MCS #3, 9, 15, 16, 17 and 38]

8. Among the 50 most frequent MCS, the LOCC initiating events contribute 5.0% of the overall CDF. The main risk is total loss of the cooling chain RRI [CCWS] / SEC [ESWS] due to mechanical failures during shutdown state D, when the RCP [RCS] level is low, followed by the failure of the operator to actuate the water make-up with the available LHSI train (1 or 4). This sequence contributes 3.4% of the overall CDF. [MCS #1, 20, 43, 44 and 45]

2.5. KEY ASSUMPTIONS

Most of the assumptions in the Level 1 PSA are described in sections 1 to 4 of Sub-chapter 15.1, and relate to aspects such as the definition of initiating events, CCF beta factors, I&C modelling, reliability data, operator modelling etc.

The main assumptions considered in the level 1 PSA are listed below:

1. The Human Reliability Analysis is performed with the assumptions that the operating procedures and guidelines will be well written and complete, as will be operator training. An analysis of the impact of operator actions, and of dependencies between operator actions, on the CDF is presented in section 6.5 of this sub-chapter.
2. The detailed Preventive Maintenance program is not implemented in the PSA model. Representative increases in CDF due to unavailability of safety systems linked to preventive maintenance activities are evaluated in the base case model by affecting the preventive maintenance one train and thus compare to the CDF without preventive maintenance in a specific sensitivity analysis (see section 6.1 of this sub-chapter).
3. The common cause failure parameters used in system modelling are based on the EUR [Ref-1]. The parameters (beta factors) are generic and not specific to the components. The use of beta factor is likely to be conservative compared to detailed CCF model.
4. In modelling the Boron dilution group of faults, core damage is assumed as soon as criticality is reached. This conservative assumption impacts both the functional safety requirements on front line systems and the time window for operator actions.
5. The recovery of the short duration loss of offsite power is assumed to be either 2 hours or 24 hours. (Possible shorter or longer recovery times could be credited by modifying the EDG running mission time.) A sensitivity analysis of longer term LOOP transients (and also of LUHS) is presented in section 6.2 of this sub-chapter.
6. The EDGs and SBO diesels are assigned to different common-cause groups. This assumption will be justified by providing diversity between the EDGs and the SBO diesels (different cooling systems, different starting systems and different fuel supplies). An analysis of the impact of this assumption on the level of risk is presented in section 6.4.2 of this sub-chapter.
7. When modelling the primary breaks group, the break is always assumed to occur on Train 4. This assumption impacts train-specific importance measures but has no impact on the PSA results.
8. For the four trains RRI [CCWS] / SEC [ESWS], train 1 and train 4 are assumed to be running in at-power states. This assumption affects train-specific importance measures.
9. When modelling small primary break events, if the MHSI system fails, it is assumed that the operators initiate a fast cooldown. However, if the partial cooldown function fails (therefore failing MHSI), it is assumed that operators will initiate feed and bleed. These modelling assumptions and the timing of these sequences will be analysed in more detail when the UK specific operating procedures are available.

10. When modelling small steam line break events, it is assumed that the VIVs [MSIVs] remain open. This conservative assumption impacts the reactivity control function by resulting in a greater secondary cooldown.
11. When modelling loss of main feedwater initiating events (LOMFW), the VIVs [MSIVs] are assumed to close. This conservative assumption affects the secondary residual heat removal function through the loss of the Main Steam Bypass (GCT [MSB]).
12. When modelling loss of main feedwater initiating events (LOMFW), a conditional probability of 0.1 is assumed for the failure of AAD [SSS] in order to take into account the inter-dependence. A sensitivity analysis on the conditional probability is presented in section 6.9 of this sub-chapter.
13. When modelling the Turbine Trip initiating event, the partial trip is assumed to fail with a probability of unity. This conservative assumption impacts the Turbine Trip frequency.
14. When modelling the Loss Of Off-site Power group, automatic switchover to house load operation is conservatively assumed to fail with a probability of unity. This conservative assumption impacts the LOOP frequency (see section 6.3.1 of this sub-chapter).
15. In modelling the Loss of Cooling Chain group, it is assumed that the common header is cooling the RCV [CVCS] charging pump in operation when it fails. This conservative assumption impacts the core damage frequency.
16. When modelling the Loss Of Off-site Power, the Loss Of Cooling Chain and the Loss of Utimate Heat Sink in state D, the RCP [RCS] inventory control is performed by LHSI trains 1 and 4, whose pump motors are cooled by the DEL [SCWS]. The grace period for the operator to initiate the LHSI injection on the onset of the loss of operating LHSI/RHR pumps is 1 hour in the thermal-hydraulic support studies [Ref-2]. However this time window is assumed to be 2 hours in the UK EPR PSA by considering:
    - The water level is assumed to be at 3/4-loop in state D. However the water inventory is actually between 3/4-loop and reactor pool flooded up to 19m. The mean time window for water make-up is much greater than the one for 3/4-loop used in the thermal-hydraulic analysis.
    - The safety injection signal will start the MHSI pumps even if the cooling chain is lost and they will inject a few cubic metres of water before failing. The volume of water injected by the MHSI will increase the time window for manual LHSI start-up.

Assumptions related to I&C modelling are presented in Sub-chapter 15.1. The key assumptions regarding NCSS modelling are presented hereafter:

17. The PSA model includes the automatic and manual actions described in the NCSS functional requirements. No thermo-hydraulic support studies are currently available to support the efficiency of the functions.
18. It is assumed that following the loss of SPPA-T2000 platform, switchover to NCSS panel is performed, even if the TXS platform is still available.
19. It is assumed that the RRC-A sensors are used by the NCSS. If no RRC-A sensors exists, the PS [RPR] sensors are assumed to be used.
20. In modelling the NCSS automatic and manual actions, it is assumed that all 4 trains are actuated by NCSS.

The impact of some of these assumptions is assessed in the sensitivity analyses presented in section 6 of this sub-chapter.

2.6. UNCERTAINTY ANALYSIS

In this section the uncertainty in the overall Core Damage Frequency (CDF), due to reliability data uncertainty and to truncation of the Minimal Cutsets (MCS) included in the calculation is addressed.

2.6.1. Reliability data uncertainty

Most of the PSA results presented in UK PSA documentation are based on point-estimate values (i.e. they consider only the mean and not the parameter uncertainty). The purpose of this section is to assess the impact of the reliability data uncertainty on the overall CDF and hence to assess the robustness of point estimated results.

Uncertainty on the Level 1 PSA results is quantified using the built-in uncertainty analysis capabilities of Risk Spectrum. This PSA uncertainty quantification evaluates parametric uncertainty. Uncertainty analyses are performed with 30,000 Monte Carlo simulations (30,000 is the maximum number of simulations that can be performed in Risk Spectrum).

Each parameter (probabilities, failure rates, frequencies, etc.) used in the level 1 PSA modelling is associated with a distribution (usually log-normal and beta). These distribution types and associated parameter values (error factor) are produced from the data base used to define the value of the parameters (EDF data, ZEDB, EG&G…). In a few cases, the distributions are not known. In these cases, a log-normal distribution with an error factor of 10 is assumed.

The results of the uncertainty analysis for the overall CDF are shown in Sub-chapter 15.7 – Figure 4.

The uncertainty results obtained are:

- Point estimate: 7.08E-7 /r.y
- Mean value: 6.41E-7 /r.y
- 5th percentile: 3.14E-7 /r.y
- Median: 5.41E-7 /r.y
- 95th percentile: 1.24E-6 /r.y

There is less than one decade between the 5th percentile and the 95th percentile. As discussed above, all parameters are subjected to uncertainty assessment. In addition, the distributions chosen, for example for unknown distributions, are considered to be conservative. Consequently the point estimate value is considered suitable to describe the level of risk.

2.6.2. Impact of the MCS Set truncation

Due to the size of a PSA model, it is not possible to generate all the minimal cut sets (MCS). Consequently, a truncation process of the MCS has to be used to reduce the amount of computation involved in the MCS generation. This truncation process is based on a probabilistic threshold. All MCS with an occurrence probability (or frequency) lower than the threshold are not included in the final results of the calculation.

The threshold value of 1E-15 avoids any significant under-estimation of the global CDF due to the MCS set truncation, as shown in Sub-chapter 15.7 - Figure 5.

2.7. SUMMARY OF MAIN LEVEL 1 PSA FINDINGS

Conservative assumptions are used in the Level 1 PSA to create margins to cover the impact of changes in the detailed design of the model and the completion of the hazards assessment. With these assumptions, the overall core damage frequency is low, relative to the safety objective of 1E-05/r.y at 7.1E-07/r.y including preventive maintenance.

The UK EPR is a Pressurised Water Reactor designed with active safety feature, thus the Loss of offsite power group is one of the most dominant initiating event groups with a CDF of 1.1E-07/r.y, which represents 15% of the overall CDF (when adding the LOOP induced by the reactor trip, the contribution increases to 21%). The loss of cooling chain group and the primary breaks group, which contribute a CDF of 1.2E-07/r.y and 1.1E-07/r.y respectively, represent 17% and 15% of the overall CDF. The absolute value of the CDF for these dominant groups remains low due to the high redundancy of safety trains and diversity in support systems.

The importance analyses show that single component failures do not contribute significantly to the overall CDF due to the redundancy and diversity of the safety systems. For example, several systems are able to provide the secondary residual heat removal function. These are as follows:

Steam generator feed function:

- ARE [MFWS],
- AAD [SSS],
- ASG [EFWS],

Steam release from steam generators:

- GCT [MSB],
- VDA [MSRT],
- VVP [MSSS] / MSSV.

The primary residual heat removal function can be performed by either two redundant lines (Primary Depressurisation lines) or three pressuriser safety valves.

The control rod drop is initiated by redundant and diversified actuators and signals which limit the frequency of ATWS scenarios.

The sensitivity analyses show that the overall CDF is sensitive to the human actions dependency modelling, to the grid reliability and to the common cause failure assumptions. The reliability of RCP [RCS] coolant pump seals is also a parameter that significantly affects the overall CDF.

The preventive maintenance during power operation does not represent an important part of the overall CDF of the plant.

3. SUMMARY OF LEVEL 2 RESULTS

3.1. INTRODUCTION

Sub-chapter 15.4 has presented the methodology and results of the UK EPR PSA Level 2 study for all plant states, internal events and internal and external hazards.

Regarding Total Release Frequency, the frequency of any severe accident (the core damage frequency – an input to level 2) is very low, at around 7.1E-07/r.y. The Level 2 results show that the EPR's strong containment and dedicated severe accident mitigation measures are very effective in minimising the frequency and magnitude of releases to the environment in case a core damage event does occur. These two points result in extremely low predicted absolute frequency of releases.

The use of the Level 2 results in assessing the off-site consequences of releases is discussed in Sub-chapters 15.4 and 15.5. Sub-chapter 15.5 presents the off-site consequence risk assessment in terms of individual doses and societal impact, the main results of which are given in section 4 of this sub-chapter.

In addition to the specific off-site consequence analyses, the Level 2 PSA results are interpreted in terms of the frequency of Large Releases. Large Releases are defined using a 100 TBq limit of Cs-137 – a smaller "target" than that use in many countries. The frequency with which a release of this magnitude could be exceeded is 7.7E-08/r.y. The Large Release frequency defined in this way represents 10.8% of the core damage frequency, and the fraction of large early release, with a frequency of 4.1E-08/r.y, represents 5.7% of CDF. Sub-chapter 15.7 - Figure 8 presents the contribution of initiating events to the overall LERF.

3.2. CONTRIBUTORS TO LARGE RELEASE FREQUENCY

Sub-chapter 15.7 - Figure 6 presents the frequencies of the different containment failure modes. Containment late failures are the dominant contributors to Large Release Frequency. A sensitivity analysis on release category 504, which is the main LRF contributor among the containment late failures, is presented in section 6.8 of this sub-chapter. Other contributors to Large Release include early containment failures, containment bypass sequences and containment isolation failures.

3.3. CONTRIBUTORS TO RELEASE RISK

Decomposition of the release risk (frequency x release magnitude) for Cs-137, presented in Sub-chapter 15.7 - Figure 7, reveals a dominant contribution from Spent Fuel Pool accidents. They indeed contribute about 86% to the Cs-137 release risk.

The second most contributing events to the Cs-137 release risk are the bypass events: interfacing system LOCAs (RC802) and steam generator tube ruptures (RC702). Containment bypass sequences contribute 9% of the release risk for Cs-137.

3.4. KEY ASSUMPTIONS

The main assumptions regarding Level 2 are detailed in Sub-chapter 15.4, in particular those regarding Core Damage End States in Sub-section 15.4.3.2 – Tables 2 and 4.

3.5. UNCERTAINTY ANALYSIS

The uncertainty analysis is presented in section 4.5 of Sub-chapter 15.4.

4. SUMMARY OF THE OFF-SITE CONSEQUENCE RISK ASSESSMENT (LEVEL 3 PSA) RESULTS

4.1. INTRODUCTION

A probabilistic assessment (a Level 3 PSA) of the UK EPR design has been performed to determine the off-site risk to the public due to postulated accidents. This sub-chapter summarises the process followed to perform this assessment and presents the results in terms of:

- Individual risk to any person off the site, i.e. frequency / consequence (dose band) couplets.
- Societal risk, in terms of the annual frequency of events which could potentially lead to more than 100 immediate or eventual fatalities in the wider population.

The dose band classification for radiological exposure is as follows:

DB1 : 0.1-1 mSv ; DB2 : 1-10 mSv ; DB3 : 10-100 mSv ; DB4 : 100-1000 mSv ; DB5 : > 1000 mSv.

The radiological consequence which is considered is the unmitigated effective dose to a child at 500 m downwind from the point of release during the first 7 days following the release, with standard weather condition “DF2” (see Chapters 12, 14 and 16 on parts concerning radiological consequences).

4.2. ASSESSMENT OF INDIVIDUAL RISK

The summated frequency of faults predicted to result in an off-site effective dose in excess of 0.1 mSv is 1.4E-03/r.y. Around 99% of this frequency is associated with very low off-site consequences in the lowest dose band, < 1mSv, and for this dose band the frequency is nearly one order of magnitude below the BSO. Only 2.3E-07/r.y of this frequency is associated with off-site consequences above 100 mSv.

In dose band DB1 (0.1 to 1 mSv), the dominant events are non core damage sequences (86%), mainly due to SGTR (affected SG isolated), a fuel handling accident in the fuel building with 1 fuel assembly partially damaged (all fuel rods along one edge) and filtration available and fuel assembly drop in the reactor building (14%).

In dose band DB2 (1 to 10 mSv), the dominant events are a fuel handling accident in the fuel building with 100% clad failure and filtration available (79%) and non core damage sequences, mainly due to SGTR (affected SG not isolated) (21%).

In dose band DB3 (10 to 100 mSv), the dominant events are a fuel handling accident in the fuel building with 100% clad failure and filtration not available (43%), core damage accidents with containment intact (annulus and building ventilation operational) (41%) and non core damage LOCA inside containment with 10 % clad failure, containment intact but failure of the ventilation systems (16%).

In dose band DB4 (100 to 1000 mSv), the dominant events are core damage accidents with containment intact (failure of annulus and building ventilation) (~100%). The contribution from non core damage LOCA inside containment with 1 % clad failure and containment bypass events is negligible (~0%).

In dose band DB5 (> 1000 mSv), the dominant events are core damage accidents with containment failure (95%) and loss of cooling or rapid drainage of the spent fuel pool (3%). The contribution from non core damage LOCA inside containment with 10% clad failure and containment bypass events is small (2%).

It is emphasised that the identification of the dominant events, as well as any analysis of risk balance, must be considered with care as modelling assumptions, especially where varying degrees of conservatism are introduced, lead to distortions in the risk breakdown and profile.

4.3. ASSESSMENT OF SOCIETAL RISK

For this part of the off-site consequence assessment, the events which are identified as major accidents, and are considered to have the potential to result in greater than 100 eventual fatalities are those leading to RCs that lie in dose band 5 of the individual risk assessment.

The summated frequency of the release categories which lead to more than 100 eventual fatalities, for a generic UK site, is 8.0E-08/r.y.

The contributions from events considered to have the potential to result in greater than 100 eventual fatalities are:

- Those Release Categories from the Level 2 PSA that fall within dose band 5 (95%). The main contributors are RC504 (41%) and RC303 (13%) (See Sub-chapter 15.4 for description).

- The fuel damages sequences following rapid water drainage in the spent fuel pool (3%) (See Sub-chapter 15.3).

- The non core damage sequences that fall within dose band 5 (2%). They correspond to non core damage LOCA inside containment with 10% of fuel clad failure and containment bypass.

5. PSA INSIGHTS – DESIGN IMPROVEMENTS

An iterative process to identify design improvements using PSA was implemented throughout the development of the EPR design. The GDA step 4 PSA presents the results of this process. During the detailed design, PSA insights covering both at-power and shutdown states will be used again to balance the UK EPR design and ensure that the ALARP principle is satisfied.

The following section presents the main examples of design improvements made as a result of insights from the probabilistic safety assessments:

- **Interfacing system LOCA:** The potential V-LOCA scenarios have been studied by considering the systems that interface with the RCP [RCS] to evaluate the levels of defence preventing a loss of the second barrier and a bypass of the third barrier. The probabilistic analysis (see section 5.2 of Sub-chapter 15.1) has resulted in design improvements and specifically the reinforcement of the isolation devices and their diversification to prevent interfacing LOCA. This has particularly been undertaken on the component cooling water (RRI [CCWS]) system, in the case of tube rupture(s) on the thermal barrier, and on the chemical and volume control system (RCV [CVCS]) with the implementation of the diversified No.1 Reactor Coolant Pump Seal Injection check valves.

- **ATWS**: Diversification of the reactor trip actuators and sensors to limit the frequency of ATWS transients (see section 5.10 of Sub-chapter 15.1).

- **Loss of main feedwater**: Reduction of the frequency of the loss of main feedwater event by the addition of a fourth main feedwater pump and the Start-up and Shutdown System (AAD [SSS]) (see section 5.6 of Sub-chapter 15.1).

- **SBO**: Addition of two station blackout diesel generators to reduce the contribution of the LOOP transients (see section 5.7 of Sub-chapter 15.1).

- **LOCC**: Alignment of the safety chilled water system (DEL [SCWS]) to provide cooling of the low head safety injection (LHSI) pump motors for train 1 and 4 if the RRI [CCWS] is lost. This reduces the LHSI dependence on the component cooling water (RRI [CCWS]) and essential service water (SEC [ESWS]) systems (see section 5.9 of Sub-chapter 15.1).

- **ULD**: Improvement of the reliability of the safety injection system (RIS [SIS]) automatic response at mid-loop conditions. This has involved adding diverse signals to auto-start medium head safety injection (MHSI) on low reactor coolant system (RCP [RCS]) loop level or low suction pressure to the residual heat removal (RHR) pumps (see section 5.8 of Sub-chapter 15.1).

- **Feed & Bleed**: In addition to the pressuriser safety valves, two bleed lines are installed to enhance the reliability of bleed operation for primary residual heat removal.

- **LUHS and Level 2 PSA**: Improved redundancy and reliability of the cooling system of the severe accident heat removal system (EVU [CHRS]) by providing two totally diversified cooling chains from the component cooling water (RRI [CCWS]) and essential service water (SEC [ESWS]) systems, each dedicated to the cooling of the associated EVU [CHRS] train.

- **Fuel damage in the spent fuel pool**: Addition of a third fuel pool cooling train cooled by the EVU [CHRS] cooling chain independent from the component cooling water (RRI [CCWS]) system (see Sub-chapter 15.3).

PSA has been used as a supplementary tool to the Design Basis Assessment (DBA) in safety assessment during the design phase. The contributions of these assessments include the following:

- Help with the design of safety systems, particularly in terms of redundancy and diversification.
- Verification of a balanced design by showing that no individual sequences have a dominant contribution to the frequency of core damage.
- Estimation of the deviations with respect to the safety requirements applied to operating reactors.
- Comparison of the level of safety of the future reactor with that of operating reactors or of other reactors under development.
- Help with the definition of operating conditions associated with multiple failures.
- Preliminary assessment of the safety improvement resulting from the severe accident mitigation measures.
- Demonstration that the consequences of sequences leading to offsite releases are acceptable when measured against the SAP numerical targets.

The PSA is also used to assess the impact of preventive maintenance on safety and will be used to prepare the detailed maintenance programmes.

The qualitative and quantitative analyses of the main PSA contributions will also be used during the detailed design to define operating and accident procedures and the training of operators, taking into account operator actions which, if they fail, may lead to a significant increase in the frequency of core damage

Further PSA applications can be foreseen, e.g. using PSA insights to optimise the Allowed Outage Times, Surveillance Test Intervals, test strategies, and procedures as indicated in Chapter 18 or using Internal Hazards PSA insights to support development of system design and hazard procedures.

6. SENSITIVITY ANALYSES

The sensitivity analyses described in this section are based on our experience and the conclusions of technical PSA reviews.

The sensitivity analysis on the Level 2 PSA assumptions is presented in section 4.5 of Sub-chapter 15.4.

6.1. PREVENTIVE MAINTENANCE

A preliminary assessment of the effect of the preventive maintenance (PM) is integrated in the model. However, the assessment remains preliminary as:

- The EPR is still in a design phase. Thus PM has not been defined in detail (scope, scheduling, duration).
- In the modelling with RiskSpectrum™, adding PM at an early stage of the design could bias the importance analysis.
- The modelling of PM with RiskSpectrum™ gives an average risk over one year and can hide hazardous plant configurations.
- The definition of electrical cross-connection during preventive maintenance operation is not defined.
- Unavailability due to corrective maintenance that requires operating experince shall also be considered.

Preventive maintenance as well as corrective maintenance is scheduled during both at-power and shutdown states.

In this first analysis, it has been considered that the provisions (duration) taken into account for the preventive maintenance covers the impact of the unavailability due to corrective maintenance.

The modelling of preventive maintenance in the UK EPR PSA is described in section 3.6.1 of Chapter 15.1 in the PCSR.

6.1.1. Results of Level 1 PSA (CDF)

In order to perform the analysis some maintenance groups have been defined by studying the functional dependencies between safety systems. A maintenance group gathers together the maintenance carried out on one or more systems. The groups are discussed in section 3.6 of Sub-chapter 15.1.

6.1.1.1. Analysis of Maintenance during Power Operation

The risk is compared to the CDF in Plant at-power and hot shutdown state (A&B) only, covering internal events, internal and external hazards. Plant state A&B is assumed to cover 8225 hours per calendar year (see Sub-chapter 15.1). The CDF contribution from plant state A&B without preventive maintenance is calculated to be 5.78E-07/r.y (the instantaneous CDF in state A&B itself is 7.03E-11 / hour).

PM Group		Duration (h)	CDF (/h)	∆CDF (/r.y)
no PM Case	w/o PM	w/o PM	7.03E-11	-
Group A	SEC [ESWS] / RRI [CCWS] / RIS [SIS] / PTR [FPCS]	672	7.31E-11	2.32E-08
Group B	RCV [CVCS]	No preventive maintenance		
Group C	ASG [EFWS]	672	7.11E-11	6.70E-09
Group D	AAD [SSS]	No preventive maintenance		
Group E	SRU [UCWS]/EVU [CHRS] / 3rd PTR [FPCS]	336	7.08E-11	4.50E-09

PM Group		Duration (h)	CDF (/h)	ΔCDF (/r.y)
Group F	LH-Diesels	672	7.15E-11	9.90E-09
Group G	LJ-Diesels	336	7.19E-11	1.33E-08
Group H	RBS [EBS]	No preventive maintenance		
Group A, C, F	SEC [ESWS] / RRI [CCWS] / RIS [SIS] / ASG [EFWS] / LH-Diesels	672	7.49E-11	3.80E-08
Group A, B, C, E, F, G, H	LH- switchboard	672	1.25E-10	4.46E-07

Three different cases have been analysed:

- Case 1: The PM is performed on the different groups sequentially. This case is the base case presented in the PCSR.
- Case 2: The PM of groups A, C and F is performed concurrently, and the PM of other groups E and G is performed sequentially. A sensitivity analysis is presented covering the duration of this maintenance.
- Case 3: The preventive maintenance is performed on one electrical division (LH-switchboard) without considering the potential cross-connections. It combines the groups A, B, C, E, F, G, and H. It should be noted that this case is unrealistic and goes against the conclusions drawn from the analysis presented above; especially the impact of the at-power maintenance on the RCV [CVCS] and the RBS [EBS] which is not foreseen has an important impact.

The results of the three different cases are presented below:

	Case 1	Case 2	Case 3
CDF without PM (state A&B) /r.y	**5.78E-07**	**5.78E-07**	**5.78E-07**
CDF with PM (state A&B) /r.y	**6.36E-07**	**6.34E-07**	**1.02E-06**
Risk Increase (/r.y)	*5.8E-08* *10%*	*5.6E-08* *9.7%*	*4.46E-07* *77%*

The analysis shows that at power maintenance results in an increase in risk of 10% to 77% in the core damage frequency in state A&B when the preventive maintenance is performed on train 2. However, the absolute value of the CDF remains low.

The sensitivity analysis of maintenance duration in Case 2 (or Case 3) shows that if the preventive maintenance for groups A, C and F (or groups A, B, C, E, F, G, and H) is extended, the increase of core damage frequency (per reactor-year) is 0.3% per day of preventive maintenance (or 1.6% for case 3). With this configuration, the maintenance work load is much higher than in case 1, because it is assumed that groups A, C and F (or groups A, B, C, E, F, G, and H for case 3) are maintained over 28 days while the maintenance on A, C and F separately would require 84 days (or maintenance on A, B, C, E, F, G and H separately would require 140 days).

The risk increase due to the at-power preventive maintenance should therefore be assumed to be between 10 and 77% and depends on the maintenance policy chosen by the utility. Case 1 (10% increase in CDF) is retained as the base case for the results presented in Level 1, Level 2 and offsite consequence assessments.

6.1.1.2. Analysis of Maintenance during Shutdown

During the plant shutdown some preventive maintenance is carried out while the core is still loaded in the reactor pressure vessel (Ca, Cb, D) or during unloading/reloading phases (E).

The preventive maintenance during state F has no impact on Level 1 PSA results. It is considered in the fuel pool accident assessment in section 6.1.4 of this sub-chapter.

The results for these two groups are summarised below:

		Group I		**Group J**		
		Steam Generators 3 and 4		SEC [ESWS] /RRI [CCWS] / RIS [SIS] / LH-		
State	**CDF Base (/r.y)**	**CDF w/o PM (/r.y)**	**PM Duration**	**CDF w/o PM (/r.y)**	**PM Duration**	**Risk increase (ΔCDF /r.y)**
Ca	1.02E-08	9.18E-09	All state	-	no PM	1.02E-09
Cb	2.76E-08	2.51E-08	All state	-	no PM	2.50E-09
D	3.42E-08	-	No SGs available	-	no PM	
E	3.24E-10	-	No SGs available	2.64E-10	All state	6.00E-11

The impact of the preventive maintenance during shutdown is very low, having a negligible effect on overall risk.

6.1.2. Results of Level 2 PSA (LRF and LERF)

The release categories considered in the Level 2 PSA for the UK EPR are described in Sub-chapter 15.4, section 3.5.1 and Sub-section 15.4.3.5 - Table 1. The sensitivity studies discussed here focus on the summated frequencies of those release categories that are considered Large Releases or Large Early Releases (see Sub-chapter 15.4, section 4.3).

The release categories corresponding to Large Release are: RC200, RC201, RC202, RC203, RC204, RC205, RC206, RC301, RC302, RC303, RC304, RC401, RC402, RC403, RC404, RC502, RC504, RC602, RC701, RC702, RC802, SFP (i.e. all except RC101, RC102, RC501 and RC503). The summated frequency of the release categories corresponding to Large Release is referred to as the Large Release Frequency (LRF).

The release categories corresponding to Large Early Release are: RC200, RC201, RC202, RC203, RC204, RC205, RC206, RC301, RC302, RC303, RC304, RC401, RC402, RC403, RC404, RC701, RC702, RC802 (i.e. all Large Releases except RC502, RC504, RC602 and SFP, which are late releases). The summated frequency of the releases categories resulting in Large Early Release is referred to as the Large Early Release Frequency (LERF).

The results of studying the maintenance groups described in sub-section 6.1.1.1 of this sub-chapter with their corresponding durations (base case) show that the contribution of the preventive maintenance in the base case is 16.2% in LRF and 5.7% in LERF. The 16.2% and 5.7% contribution represent the summated impacts of the increments in LRF and LERF resulting from the existence of each defined maintenance state for its corresponding duration. In absolute terms the changes in frequency between without and with preventive maintenance are as follows:

- LRF increases from 6.4E-8 /r.y to 7.7E-8 /r.y
- LERF increases from 3.8E-08 /r.y to 4.1E-8 /r.y.

The contributions of each maintenance group to the above changes in LRF and LERF are presented below:

PM group	Contribution to ΔLRF	Contribution to ΔLERF
Preventive Maintenance during At-power states		
Group A: SEC [ESWS] /RRI [CCWS] / RIS [SIS] / PTR [FPCS]	1.75E-09 (2.3% of base LRF))	1.38E-09 (3.4% of base LERF)
Group B: RCV [CVCS]	No preventive maintenance	
Group C: ASG [EFWS]	2.97E-10 (0.4% of base LRF)	2.27E-10 (0.6% of base LERF)
Group D: AAD [SSS]	No preventive maintenance	
Group E: SRU/EVU2 [CHRS2] / 3rd PTR [FPCS]	4.80E-09 (6.2% of base LRF)	1.09E-10 (0.3% of base LERF)
Group F: LH-Diesels	1.90E-09 (2.5% of base LRF)	2.03E-10 (0.5% of base LERF)
Group G: LJ-Diesels	3.54E-09 (4.6% of base LRF)	2.50E-10 (0.6% of base LERF)
Group H: RBS2 [EBS2]	No preventive maintenance	

PM group	Contribution to ΔLRF	Contribution to ΔLERF
Preventive Maintenance during shutdown states		
Group I: 2 steam generators (state C)	1.35E-10 (0.2% of base LRF)	8.71E-11 (0.2% of base LERF)
Group J: SEC [ESWS] /RRI [CCWS] / RIS [SIS] (state E)	6.08E-11 (0.1% of base LRF)	6.08E-11 (0.1% of base LERF)

6.1.3. Results of offsite consequence assessment (Target 8 and 9)

The sensitivity study described here concerns the impact of preventive maintenance on the Level 2 PSA sequences considered in the off-site consequence assessment. The impact of PM on fuel pool accidents is negligible (see section 6.1.4 of this sub-chapter). The impact of PM on non core damage sequences from Level 1 PSA has been integrated in the table below.

The case considered Case 1 of sub-section 6.1.1.1 of this sub-chapter, in which the PM is performed on different groups, one after the other.

The results for individual risk are presented below:

DB **(mSv)**	**DB1** 0.1 - 1	**DB2** 1 – 10	**DB3** 10 - 100	**DB4** 100 - 1000	**DB5** > 1000
Reference case frequency (/r.y)	1.41E-03	1.31E-05	1.17E-06	1.50E-07	7.97E-08
New frequency without PM (/r.y)	1.41E-03	1.30E-05	1.12E-06	1.27E-07	6.70E-08
Risk Decrease (/r.y)	**-**	**3.40E-08**	**4.94E-08**	**2.31E-08**	**1.27E-08**

The main Level 2 PSA sequences impacted by PM are those leading to DB3, i.e. RC102 with containment intact and credit taken for filtration in the annulus and the fuel and safeguard building ventilation systems.

Considering the impact of PM on societal risk (which is evaluated in terms of the frequency of accidents leading to doses in DB5, as described in section 4.3), the preventive maintenance results in an increase of 1.3E-8/r.y.

6.1.4. Analysis of the Maintenance of the Fuel Pool Cooling System

Preventive maintenance on the Fuel Pool Cooling System (PTR [FPCS]) and its support system during power operation has also been considered. Preventive maintenance on the PTR [FPCS] trains during refuelling operations is not considered.

The result of the sensitivity analysis case indicates that the impact of the preventive maintenance is about 4E-11. The main contributor is the preventive maintenance on the diverse cooling chain (SRU [UCWS]) which cools the third PTR [FPCS] train.

The maintenance of a PTR [FPCS] train during power operation has a negligible impact on the fuel pool accident, a benefit of the three-train design.

The preventive maintenance performed on the cooling chain of the Fuel Pool System is not independent of the maintenance performed on the cooling chain (RRI [CCWS] / SEC [ESWS]) for trains 1 and 2, or on the diversified cooling chain SRU [UCWS]/EVU [CHRS] for train 3. The assessment of the impact of maintenance of these systems on the core damage frequency is presented in sub-sections 6.1.1.1 and 6.1.1.2 of this sub-chapter.

6.2. LONG TERM ANALYSIS

6.2.1. Aims

The long-term probabilistic evaluation is undertaken using the EPR PSA model used for Level 1 PSA analysis as a basis.

The aim of this study is to confirm the absence of a cliff edge effect from transient periods of more than 24 hours. The core damage sequences following initiating events whose recovery time could be greater than this limit are studied.

The principles and specific data applied to this analysis are presented in the EDF/SEPTEN report [Ref-1].

6.2.2. Areas and scope of the study

The initiating events analysed are:

- Total loss of offsite power (LOOP).
- Loss of ultimate heat sink (LUHS).

The long-term studies undertaken are quantified assuming the following maximum periods considering experience feedback on similar plants [Ref-1]:

- 192 hours, for LOOP,
- 100 hours, for the total loss of the ultimate heat sink.

Those initiating events could be the consequence of external hazards affecting the grid or the pumping station.

6.2.3. Evaluation of long-term initiating event frequencies

6.2.3.1. Loss of Offsite Power

The following frequencies for very long term LOOP events in the different plant operating states are used. These values are derived from experience in France that gives a annual frequency of {CCI} [a] for a global unavailability period of 192 hours and are considered to be reasonable for application to UK sites.

Initiating event	Frequency [/r.y]
LOOP of 192 hours – At-Power states A, B	{CCI} [a]
LOOP of 192 hours – Shutdown state Ca	{CCI} [a]
LOOP of 192 hours – Shutdown state Cb	{CCI} [a]
LOOP of 192 hours – Shutdown state D	{CCI} [a]

6.2.3.2. Loss of Ultimate Heat Sink

The frequency of the loss of heat sink initiating event is predicted to be

- F_{LUHS} {CCI Removed} [a].

For the evaluation of the initiating event during the long-term phase, a 30 hour recovery time for the initiating event for a global unavailability period of 100 hours for the heat sink is considered. These 30 hours are included in the global unavailability period of 100 hours and represent an average recovery time for the initiating event LUHS. It is assumed this recovery time is constant.

For T > 24 hours, the initiating event frequency is calculated accordingly:

$$F_{LUHS\_LT}\ (t) = F_{LUHS} \times e^{(-t/30)}$$

For this PSA long term analysis, the annual frequency of a LUHS not recovered after 100 hours is:

- $F_{LUHS\_LT}\ (100h) = F_{LUHS\_24} \times e^{(-100/30)}$ {CCI Removed} [a]

Derived from this value, the following frequencies for very long term LUHS events in the different plant operating states are used.

Initiating event	Frequency [/r.y]
LUHS of 100 hours – At-Power states A and B	{CCI} [a]
LUHS of 100 hours – Shutdown state Ca	{CCI} [a]
LUHS of 100 hours – Shutdown state Cb	{CCI} [a]
LUHS of 100 hours – Shutdown state D	{CCI} [a]

6.2.4. Results

6.2.4.1. Very Long-Term Total Loss of Offsite Power group

In this group, the accident sequences initiated by a total loss of external electrical supplies (LOOP) are studied for an unavailability period of 192 hours.

In the event of very long term LOOP in an at-power state, and before complete draining of the ASG [EFWS] tanks, the operator shall make the connection of the ISBP [LHSI] in RRA [RHR] mode, to provide primary heat removal in the long-term phase of the transient. A LHSI mission has been added into the long term LOOP event trees.

For very long term LOOP in state C, if the SIS/RHR is not available to ensure residual heat removal, it is conservatively assumed that the storage capacity of EFWS tanks is not sufficient and the operator shall initiate the feed and bleed operating mode to maintain primary cooling.

The calculated frequency of core damage after a total loss of external electrical supplies (192 hours) is 9.34E-08/r.y. The following table gives the contribution in at-power and shutdown states

Initiating event	CDF [/r.y]	Contribution
LOOP of 192 hours at-power (A and B)	8.22E-08	88%
LOOP of 192 hours during shutdown (Ca, Cb, D)	1.12E-08	12%
Total	**9.34E-08**	**100%**

Analysis of the results

The 192 hour LOOP sequences in shutdown states represent 12% of the CDF for this group. These sequences are characterised by the loss of the LHSI function in RHR mode. The ASG [EFWS] unavailability is conservatively assumed in shutdown states Ca and Cb and the feed and bleed mode is required. The loss of these functions is due to the failure of the four main diesel generators, mainly caused by independent failures and common cause failures.

The 192 hour LOOP sequences whilst at-power represents 88% of the CDF for this group. The dominant sequence is associated with the loss of both Emergency Diesel Generators and Station Black-Out Diesel generators.

Another significant part of the risk is induced by failure of the RRI [CCWS] pumps, the SEC [ESWS] pumps or the RHR pumps whose failure rate is assumed to be constant during the transient. Hence, the core damage frequency for LOOP sequences whilst at-power is judged to be conservative.

6.2.4.2. LUHS long term

The accident sequences initiated by the loss of ultimate heat sink (LUHS) are studied for an unavailability period of 100 hours.

The main system mission affected by the failure of the long-term LUHS is the re-supply of the steam generators provided by the ASG [EFWS] in states A, B and C, due to the limited availability of the ASG [EFWS] tank.

The PSA model has been modified to include the dedicated ASG [EFWS] redundant motor-driven pumps that supply the ASG [EFWS] tanks, taking suction from the fire protection tank (JAC system). The ASG [EFWS] tanks of the ASG [EFWS] trains in operation can be replenished by using those motor-driven pumps. These pumps are required in LOOP transient conditions and are backed up by the emergency diesel generators and SBO diesel generators. They are required to operate for 100 hours. They are not maintained during reactor states when the ASG [EFWS] is required.

The calculated frequency of core damage for the group, 'Loss of ultimate heat sink' (LUHS) is 1.36E-08/r.y. The following table gives the contribution in at-power and shutdown states.

Initiating event	CDF [/r.y]	Contribution
LUHS of 100 hours at-power (A and B)	1.34E-08	98.8%
LUHS of 100 hours during a shutdown (Ca, Cb, D)	1.6E-10	1.2%
Total	**1.36E-08**	**100%**

Analysis of the results

The total loss of the heat sink whilst at power represents the major contribution to the core damage frequency for the LUHS group. The risk in shutdown states is very low and is due to LUHS of 100 hours in state D as the ASG [EFWS] is not available for residual heat removal in state D.

The dominant sequence is associated with the loss of the ultimate heat sink together with the failure of the RCP seals and failure of the operator to initiate fast secondary cooldown with a time window of 30 minutes (Fussel-Vesely=44.8%). This manual action is required since the MHSI trains are not available.

6.2.5. Conclusion

The total risk associated to the long term initiating events LOOP and LUHS is 9.34E-08/r.y and 1.36E-08/r.y respectively. This is judged to be acceptable.

The level of risk for the loss of the ultimate heat sink over a period of 100 hours is judged to be acceptable for state D without dedicated feature.

The success path for the total loss of heat sink for states A to C requires the refilling of the ASG [EFWS] tanks. This action corresponds to the implementation of an RRC-A device as described in Chapter 16.

The level of risk for the long-term total loss of offsite power over a period of 192 hours whilst at power or in shutdown is judged to be acceptable.

Note that the calculation of this risk value takes into account the repair of the diesel generators and the possibility of removing residual heat through the steam generators, if necessary during the repair period. This risk assessment also assumes a limited capability of water supply for the ASG [EFWS] tanks of 100 hours, assuming dedicated ASG [EFWS] pumps supplied by the EDG and SBO-diesels for re-supply to the ASG [EFWS] tanks.

6.3. SENSITIVITY TO INITIATING EVENT FREQUENCY

6.3.1. LOOP frequency

The sensitivity analysis assesses the importance of the LOOP initiating event frequencies and consequential LOOP conditional probability in the overall risk evaluation. The table below shows the effects of multiplying the frequencies and/or conditional probability of certain initiating events by a factor of 10:

Initiating Event	Base case CDF [/r.]	New CDF [/r.y]	Δ CDF
Without consequential LOOP	7.1E-07	6.7E-07	-6%
LOOP 2h frequency multiplied by 10		1.2E-06	+76%
LOOP 2h and LOOP 24h frequency multiplied by 10		1.7E-06	+134%
LOOP 2h and LOOP 24h frequency multiplied by 10 and conditional probability of Consequential LOOP multiplied by 10		2.0E-06	+188%

The above results show that the Core Damage Frequency result is sensitive to the LOOP initiating events frequency and to the consequential LOOP conditional probability. However, the current design with four Emergency Diesel Generators [LH-diesels] and two diverse SBO diesel generators [LJ-diesels] provides a sufficient level of protection as the overall family contribution including the induced LOOP is around 1.5E-7/r.y.

6.3.2. Medium LOCA frequency

The sensitivity study addresses the frequency of Medium LOCA, 1.6E-5/r.y, which is considered as realistic by EDF/AREVA experts. This more realistic frequency of 1.6E-5/r.y has replaced the frequency of medium LOCA presented in a previous PCSR edition. The frequency of 5.0E-4/r.y is used to assess the impact of this change on the Level 1 PSA results.

The results for Level 1 PSA are:

Medium LOCA frequency increased from 1.6E-5 to 5.0E-4/ ry	**Base case CDF [/r.y]**	**New CDF [/r.y]**	**Δ CDF**
	7.1E-07	9.9E-7	+39%

The Core Damage Frequency is seen to be sensitive to the medium LOCA frequency.

6.4. COMMON CAUSE FAILURES

6.4.1. Without CCF

The sensitivity analysis is performed to assess the importance of Common Cause Failure modelling.

CCF not considered	**Base case CDF [/r.y]**	**New CDF [/r.y]**	**Δ CDF**
	7.1E-07	3.2E-07	-55%

As expected for the EPR 4-safety train design, the Core Damage Frequency result is sensitive to the CCF modelling.

6.4.2. CCF between EDG and SBO

Common Cause Failures between the EDG and SBO diesel generators are not considered in the current PSA model, as explained in Sub-chapter 8.3. The justification of the CCF exclusion is based on measures which will be taken to minimize the likelihood of common cause failure between the Main diesels and the Ultimate Emergency diesels (SBO diesel). These measures are as follows:

- The main and SBO diesels will operate over different power ranges,
- The main diesels and SBO diesels will have a different rotation speed, thus requiring different mechanical design for each type of diesel,
- The main diesels drive 10kV generators and the SBO diesels drive 690V generators this is achieved using different excitation technologies,
- Diverse I&C equipment is used for the reloading of the diesels. Reloading of the main diesels is carried out automatically by the Protection System, however reloading of the SBO diesels is undertaken manually via the Safety Actuation System

In addition to the measures described above measures will also be taken in the design and operation of the diesels to reduce the likelihood of common cause failure as a result of environmental factors. These measures are detailed in Sub-chapter 8.3.

The sensitivity analysis assesses the importance of this modelling assumption on diesel diversity.

	Base case CDF [/r.y]	New CDF [/r.y]	Δ CDF
CCF assumed between EDG and SBO (β_6^6 = 0.125%)	7.1E-07	9.9E-07	+40%

If the CCF between the EDG and SBO were to be considered, an adequate β_6^6 value would have to be derived, taking into account the arrangements explained in Sub-chapter 8.3. A realistic β_6^6 value would not exceed 0.1% (expert judgment). With a β_6^6 value equal to 0.125%, the Core Damage Frequency significantly increases by 40%. The impact of this assumption is dependent on the assumed grid reliability. However the design is considered sufficiently diverse for CCF to not occur.

6.4.3. CCF on batteries

The CCF probability used for the 220 volt battery modelling contributes significantly to the overall risk arising from the LOOP group. The UK design provides two diverse types of battery. This sensitivity case assesses the benefit gained from this design improvement.

	Base case CDF (with diverse types of battery) [/r.y]	New CDF (without diverse type of battery) [/r.y]	Δ CDF
CCF for batteries	7.1E-07	7.5E-07	6%

The Core Damage Frequency result is not sensitive to this modelling assumption. It should be noted that the result of the sensitivity analysis depends on the reliability of the batteries.

6.4.4. Sensitivity to CCF parameters

In the UK EPR PSA, the generic parameters taken from European Utility Requirements (EUR) are applied to all CCF groups independent of either the component type or the failure mode.

The CCF group of MHSI pumps failure to run is the most important with respect to the FV criteria. A sensitivity study on the overall core damage frequency is performed by assigning the following NUREG parameters [Ref-1] to this CCF group (MGL parameters):

- Beta = 2.77E-2
- Gamma = 5.64E-1
- Delta = 2.68E-1

CCF to run MHSI pumps	Base case CDF [/r.y]	New CDF [/r.y]	Δ CDF
	7.1E-07	6.1E-07	-14%

Results show that the Core Damage Frequency is sensitive to this assumption, and that the current modelling is conservative with regard to Level 1 PSA results.

6.5. OPERATOR ACTION

6.5.1. Manual opening of ASG [EFWS] header before 6 hours

This operator action is the most important operator action modelled in the Level 1 PSA, based on the RIF measure (see section 2.3.3 of this sub-chapter). If RHR is performed by the secondary side because the RIS [SIS] / RRA [RHRS] is unavailable or the RIS [SIS] / RRA [RHRS] connection cannot be reached, the operator has to initiate the make-up of the ARE [MFWS] tank, to cross connect the ASG [EFWS] tanks or to initiate the make-up of the ASG [EFWS] tank before 6 hours have elapsed. The current assumed failure probability is 1E-04 per demand. The failure probability is multiplied by 10 as a sensitivity study to assess the importance of this local action.

Probability of operator failure multiplied by 10	Base case CDF [/r.y]	New CDF [/r.y]	Δ CDF
	7.1E-07	8.3E-07	+17%

The Core Damage Frequency result is considered to be sensitive to this modelling assumption. However the current failure probability (1E-04 per demand) is a reasonable value considering the time available to perform the action, the crisis team intervention and the controlled state of the plant (no break).

Emphasis should be put on the performance of this action in the EOP and operator training programmes.

6.5.2. Manual actuation of Fast Secondary Cooldown before 30 minutes

This operator action is the most important operator action modelled in the Level 1 PSA, based on the FV measure (section 2.3.3 of this sub-chapter). The time available for the operator to initiate fast secondary cooldown after the safety injection signal is considered to be 30min. The sensitivity case assesses the benefit of a less conservative assumption (i.e. 35 minutes [Ref-1]) for the time available to initiate fast secondary cooldown.

Time window [min]	Base case CDF [/r.y]	New CDF [/r.y]	Δ CDF
35	7.1E-7	6.7E-07	-6%

The Core Damage Frequency result is sensitive to a slight modification of the time available to initiate this operator action. This sensitivity shows the importance of the time available on human reliability when the time window is short. It also points out that consideration of the differences in the time available to inititate fast secondary cooldown between seal LOCA sequences and small break LOCA events [Ref-1] would have a favourable impact on the CDF calculation.

6.5.3. Dependency modelling

The PSA model takes into account the dependency between operator actions credited to mitigate the same accident sequence. Low, Medium and High dependencies are modelled according to the method presented in Sub-chapter 15.1. This sensitivity study investigates the sensitivity of the overall CDF to dependency modelling. Two cases are identified:

- All dependencies are assumed as high dependencies:

High dependencies assumed between operator actions	**Base case CDF [/r.y]**	**New CDF [/r.y]**	**Δ CDF**
	7.1E-7	8.6E-07	+22%

- All dependencies are assumed as total dependencies:

Total dependency assumed between operator actions	**Base case CDF [/r.y]**	**New CDF [/r.y]**	**Δ CDF**
	7.1E-7	1.0E-06	+45%

The overall Core Damage Frequency result is sensitive to the dependency modelling. The CDF increase is mainly linked to the dependency between the manual actuation and control of ASG [EFWS] before 60 minutes and the manual actuation to initiate Feed & Bleed before 120 minutes.

6.6. SEAL LOCA RELIABILITY MODEL

A conditional probability for the Main Coolant Pump shaft seal failure is used based on the results of mechanical tests and on expert judgment [Ref-1]:

- During the rundown of the Main Coolant Pump:
  - Failure probability of the shaft seal n°1: {CCI Removed} [a]
  - Conditional failure probability of the shaft seal n°2, following the failure of the shaft seal n°1: {CCI} [a]
  - Conditional failure probability of the shaft seal n°3, following the failure of the shaft seals n°1 and n°2: {CCI Removed} [a]

- During the shutdown of the Main Coolant Pump:
  - o Failure probability of the shaft seal n°1: {CCI Removed} [a]
  - o Conditional failure probability of the shaft seal n°2, following the failure of the shaft seal n°1: {CCI} [a]
  - o Conditional failure probability of the shaft seal n°3, following the failure of the shaft seals n°1 and n°2: {CCI Removed} [a]

Seal LOCA sequences, which may be caused by shaft seal failure, contribute 36% of the overall Core Damage Frequency. This sensitivity study assesses the impact of the Main Coolant Pump shaft seal reliability on the seal LOCA contribution to CDF, and on the overall CDF.

No failure of Reactor Coolant Pump shaft seals	**Base case CDF [/r.y] (seal LOCA contribution. %)**	**New CDF [/r.y] (seal LOCA contribution. %)**	**Δ CDF**
	7.1E-07 (36%)	4.5E-07 (0%)	-36%

Reactor Coolant Pump shaft seals not credited	**Base case CDF [/r.y] (seal LOCA contribution. %)**	**New CDF [/r.y] (seal LOCA contribution. %)**	**Δ CDF**
	7.1E-07 (36%)	3.8E-06 (88%)	+434%

The CDF is particularly sensitive to the increase/decrease of the pump shaft seal reliability. If the Reactor Coolant Pump shaft seals are not credited, the overall CDF will increase by 434%. The CDF is then dominated by LOOP sequences followed by consequential seal LOCA. On the contrary if no failure of Reactor Coolant Pump shaft seals is considered, the CDF decreases by 36%. However the current Main Coolant Pump shaft seal failure probability is supported by test results and expert judgment.

6.7. SENSITIVITY TO DESIGN FEATURES

6.7.1. Assumptions about EVU [CHRS] train sufficiency

The Containment Heat Removal System (EVU [CHRS]) is a 2 train-system in the current design. In the Level 1 PSA, EVU [CHRS] is claimed as a back-up of the Low Head Safety Injection in Residual Heat Removal mode [LHSI/RHR] to perform the IRWST cooling function. One out of two EVU [CHRS] train is required to operate following a fault during at-power states. Similarly, 1 out of 2 EVU [CHRS] trains is required to operate following a fault during shutdown states.

The following sensitivity study assesses the impact of the design change from 2 EVU [CHRS] trains to 1 EVU [CHRS] train. The base case CDF is the CDF calculated with 1 out of 2 EVU [CHRS] trains required during at-power and shutdown states. The new CDF is calculated assuming that only 1 EVU [CHRS] train exists in the EPR design. Considering only one EVU [CHRS] train in the EPR design, no preventive maintenance on EVU [CHRS] is modeled when core is loaded.

	Base case CDF [/r.y]	New CDF [/r.y]	Δ CDF
1 EVU [CHRS] train design instead of 2-train	7.1E-07	8.1E-07 [CHRS in Division 1 only]	+14%
		7.2E-07 [CHRS in Division 4 only]	+1%

The difference is due to the non-symetrical modelling of the fire events. It is assumed in the fire PSA in Sub-chapter 15.2 that the fire in the safeguard building affects the division 1. With that assumption, in the events of fire in safeguard building 1 and if the only EVU [CHRS] train is in the affected division, there is no EVU [CHRS] available. However, the probability of fire in a given safeguard building is one out of four. The CDF of an EPR with one EVU [CHRS] can be assessed as follow:

$$\frac{1 \times 8.1\mathrm{E}\text{-}07 + 3 \times 7.2\mathrm{E}\text{-}07}{4} = 7.4\mathrm{E}\text{-}07$$

This corresponds to an increase of 4% of the base case CDF.

The Core Damage Frequency result is not sensitive to this assumption. This is because the EVU [CHRS] is used as a back-up for the IRWST cooling function in the PSA level 1 and such sequences are not part of the main PSA level 1 sequences.

The sensitivity analysis of the EVU [CHRS] assumptions on the Level 2 result is discussed in section 4.5.2 of Sub-chapter 15.4. It shows that the LERF increases by 5% and LRF increases by 136%.

6.7.2. Assumption about RBS [EBS]

The extra-borating system (RBS [EBS]) is used to mitigate ATWS events in state A and boron dilution events in plant states B to D. The homogeneous boron dilution sequences, particularly in state B, are an important contributor to the overall CDF. This sensitivity study assesses the benefit of including an additional, third train (in stand by and ready to inject) in the design to mitigate the consequences of dilution events.

	Base case CDF [/r.y]	New CDF [/r.y]	Δ CDF
3 RBS [EBS] train design rather than 2-train	7.1E-07	6.8E-07	-4%

The Core Damage Frequency is not particularly sensitive to the 3 RBS [EBS] train design assumption. Moreover, the PSA does not claim the Medium Head Safety Injection to mitigate dilution as there are no automatic signals to start it. Consequently, the importance of both the homogeneous boron dilution events and of the RBS [EBS] system is likely to be overestimate.

The CDF decrease is not significant compared to the conservative assumptions used in the assessment. The addition of a third RBS [EBS] train is consequently not considered to provide a significant safety improvement.

6.8. SENSITIVITY TO MODELLING FEATURES

The release category 504 is the main contributor to the Assessment of Societal risk (41% see section 4.3 of this sub-chapter). This release is mainly caused by a station black-out (SBO): failure of all EDGs and SBO Diesel generators which causes the unavailability of the EVU [CHRS] due to the loss of power supply. Some conservative assumptions have been included in the current analysis of the SBO situation. In order to have a more accurate value for this release category (RC 504), the following sensitivity analysis is proposed:

a) in the current PSA (see Sub-Chapter 15.4), there is a total dependency between the failure to start the SBO diesel generators in the Level 1 PSA (before core damage, time window around 2 hours) and in the Level 2 PSA (before containment overpressure, time window greater than 24 hours). This conservatism is removed from the ALARP analysis and an overall failure probability of 1.0E-4 per demand is taken into account in the RC 504 for the operator failure to start at least one SBO diesel in less than 24 hours.

b) in the current PSA (see Sub-Chapter 15.1), no credit is taken for the potential recovery of failed equipment. The EDGs for EPR have been design with protection that stops them before they catastrophically fail. According to EDF experience, the mean time to repair the main diesel generators is 10 hours if these protections succeed in stopping the diesel before catastrophic failures.

Modelling of EDG recovery in 10 hours and realistic grace period for SBO start-up	**Base case RC504 [/r.y]**	**New RC504 [/r.y]**	**Δ RC504**
	3.29E-08	1.88E-08	-43%

The RC decrease is significant compared to the base case. An ALARP case is presented in section 3.2 of Sub-chapter 17.5, which details the impact of this assumption on EPR design.

6.9. SENSITIVITY TO AAD [SSS] / ARE [MFWS] DEPENDENCY

The PSA model takes into account the dependency between AAD [SSS] and ARE [MFWS], a conditional probability of 0.1 is considered for the failure of AAD [SSS] in case of ARE [MFWS] failure. The following table shows the sensitivity of the overall CDF to the conditional probability value.

Conditional probability	Base case CDF [/r.]	New CDF [/r.y]	∆ CDF
0 (no dependency)	7.1E-07	7.04E-07	-0.6%
0.5		7.24E-07	2.3%
1 (complete dependency)		7.44E-07	5.1%

The Core Damage Frequency is not particularly sensitive to the conditional probability of AAD [SSS] failure on ARE [MFWS] failure. The reason is that in the case of AAD [SSS] and ARE MFWS] failure, there are other means to ensure residual heat removal, such as secondary heat removal with SG feed by ASG [EFWS] or Feed and Bleed.

6.10. SENSITIVITY TO MEDIUM LOCA MODELLING

In the PSA model, the success criteria applied to the medium LOCA with a size between 45 and 100 cm² are based on thermal-hydraulic analyses carried out for 45, 60, 80 and 100cm² break size. In the absence of exhaustive analyses for the whole break range (e.g. analyses made for 45, 60, 80 but not systematically for 100 cm²), reasonably conservative assumptions are applied to derive the success criteria for the whole break range. The present sensitivity study assesses the impact of considering more conservative assumptions to derive these success criteria:

- No time is assumed available to perform fast secondary cooldown in the case of MHSI unavailability.
- No time is assumed available to perform feed&bleed in the case of failure of partial secondary cooldown.
- More than one MHSI train are assumed necessary to ensure safety injection.

Modelling of Medium LOCA (45-100cm²) with more conservative success criteria	Base case CDF [/r.y]	New CDF [/r.y]	∆ CDF
	7.08E-7	7.11E-7	0.5%

The results show that the overall core damage frequency is not sensitive to these success criteria. In addition, most of the increase in CDF is due to the assumption on the time available for initiating fast secondary cooldown which is known to be conservative through the analyses made for 45cm² and 80cm² break sizes.

SUB-CHAPTER 15.7 - TABLE 1

Component Ranking According To Fussell-Vesely (FV)

No.	ID	Description	Nominal probability [per demand]	FV
1	RCP_SEAL#1_RD	Failure of RCP shaft seals #1 during rundown phase	{CCI}[a]	2.66E-01
2	RCP_SEAL#2_RD	Conditional failure of RCP shaft seals #2 during rundown phase	{CCI}[a]	2.66E-01
3	RCP_SEAL#1_SD	Failure of RCP shaft seals #1 during shutdown phase	{CCI}[a]	9.75E-02
4	RCP_SEAL#2_SD	Conditional failure of RCP shaft seals #2 during shutdown phase	{CCI}[a]	9.75E-02
5	RCP-SSSS_ORING_52	Failure of the Orings exposed to RCS (P,T)	{CCI}[a]	9.57E-02
6	RCP-SSSS_ORING_53	Failure of the Orings exposed to RCS (P,T)	{CCI}[a]	9.57E-02
7	RCP-SSSS_ORING_51	Failure of the Orings exposed to RCS (P,T)	{CCI}[a]	7.50E-02
8	PM_GROUP_A_ST_A	Preventive Maintenance on the cooling chain (RIS/RRI/SEC) during power operation	{CCI}[a]	3.28E-02
9	LJP_____DFR	Station Blackout Diesel Generator fails to run	{CCI}[a]	3.11E-02
10	LHP_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.73E-02
11	LHS_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.70E-02
12	LHR_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.43E-02
13	LJS_____DFR	Station Blackout Diesel Generator fails to run	{CCI}[a]	2.19E-02
14	RIS3420POEFR	MHSI pump failure to run	{CCI}[a]	2.15E-02
15	RIS4420POEFR	MHSI pump failure to run	{CCI}[a]	2.07E-02
16	LHQ_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.03E-02
17	PM_GROUP_G_ST_A	Preventive Maintenance on the SBO-DG (LJ-) during state A	{CCI}[a]	1.87E-02
18	RIS1420POEFR	MHSI pump failure to run	{CCI}[a]	1.61E-02
19	PM_GROUP_F_ST_A	Preventive Maintenance on the EDG (LH-) during state A	{CCI}[a]	1.40E-02
20	GCT	By-pass Condenser Fails	{CCI}[a]	1.38E-02
21	RBS1220POEFR	failure to run - EBS pump (breaker & motor included)	{CCI}[a]	9.53E-03
22	RBS4220POEFR	failure to run - motor driven pump (breaker & motor included)	{CCI}[a]	9.53E-03
23	PM_GROUP_C_ST_A	Preventive Maintenance on the EFWS (ASG) during state A	{CCI}[a]	9.51E-03
24	RCP1299VZEFC	Failure to close - Motor Operated Valve	{CCI}[a]	9.14E-03
25	RCP2299VZEFC	Failure to close - Motor Operated Valve	{CCI}[a]	9.14E-03
26	RBS1220POEFS	failure to start - EBS pump (breaker & motor included)	{CCI}[a]	9.03E-03
27	RBS4220POEFS	failure to start - motor driven pump (breaker & motor included)	{CCI}[a]	9.03E-03
28	AAD1111VLMEC4	manual valve left in a wrong close position	{CCI}[a]	8.37E-03
29	RIS2420POEFR	MHSI pump failure to run	{CCI}[a]	8.25E-03
30	RCP3299VZEFC	Failure to close - Motor Operated Valve	{CCI}[a]	6.50E-03
31	RCP4299VZEFC	Failure to close - Motor Operated Valve	{CCI}[a]	6.50E-03
32	PM_GROUP_E_ST_A	Preventive Maintenance on the CHRS (SRU/EVU) during state A	{CCI}[a]	6.43E-03

SUB-CHAPTER 15.7 - TABLE 1

Component Ranking According To Fussell-Vesely (FV)

No.	ID	Description	Nominal probability [per demand]	FV
33	ASG1210POEFR	EFWS pump failure to run	{CCI}[a]	5.97E-03
34	AAD_DEP	Conditional probability of MFWS & SSS CCF	{CCI}[a]	5.67E-03
35	SGTR_SBS	Probability of SGTR in case or SBS	{CCI}[a]	5.57E-03
36	PM_GROUP_I_ST_C	Preventive Maintenance on the steam generator (inspection) during state C	{CCI}[a]	5.17E-03
37	ASG4210POEFR	EFWS pump failure to run	{CCI}[a]	3.43E-03
38	MOD_COEFF_UN	Moderator coefficient unfavourable	{CCI}[a]	3.05E-03
39	RCP11_SSSF	Mechanical failure of stand still seal of RCP 1	{CCI}[a]	3.05E-03
40	RCP21_SSSF	Mechanical failure of stand still seal of RCP 2	{CCI}[a]	3.05E-03

SUB-CHAPTER 15.7 - TABLE 2

Component Ranking According To Risk Increase Factor (RIF)

No.	ID	Description	Nominal probability per demand	RIF
1	RCP2110POBFO	Failure to open - indoor circuit breaker	{CCI}[a]	35.6
2	RCP1110POBFO	Failure to open - indoor circuit breaker	{CCI}[a]	27.2
3	LHD1101JBOFL_S	Failure (short) - busbar 10kV	{CCI}[a]	12.5
4	LHD1101JBOFL	Failure - busbar 10kV (long)	{CCI}[a]	12.5
5	RCP3110POBFO	Failure to open - indoor circuit breaker	{CCI}[a]	11.4
6	RCP4110POBFO	Failure to open - indoor circuit breaker	{CCI}[a]	11.4
7	RIS4250EXTLK	Leakage - tube - heat exchanger	{CCI}[a]	11.0
8	LAA1101BT_FS_ST	Fail of 220V-Batt. short term for start of the EDG	{CCI}[a]	9.5
9	LVA1101JBOFL	Failure - busbar (400V)	{CCI}[a]	9.4
10	LVD1101JBOFL	Failure - busbar (400V)	{CCI}[a]	8.9
11	LAD1101BT_FS_ST	Fail of 220V-Batt. short term for start of the EDG	{CCI}[a]	8.7
12	SGTR_SBS	Probability of SGTR in case or SBS	{CCI}[a]	6.6
13	LVA1101DLIFR	Failure to run - inverter	{CCI}[a]	4.8
14	RRI4210POMFR	CCWS pump motor failure to run	{CCI}[a]	4.7
15	RRI4210POEFR	CCWS pump failure to run	{CCI}[a]	4.7
16	SEC4110POEFR	ESWS pump failure to run	{CCI}[a]	4.7
17	SEC4110POMFR	ESWS pump motor failure to run	{CCI}[a]	4.6
18	LVD1101DLIFR	Failure to run - inverter	{CCI}[a]	4.6
19	SEC4110POBSO	Pump motor breaker - spurious operation	{CCI}[a]	4.5
20	RRI4210POBSO	Pump motor breaker - spurious operation	{CCI}[a]	4.5
21	RRI4313VNESO	Motor operated valve - spurious operation	{CCI}[a]	4.5
22	RRI4313VNBSO	Spurious operation - breaker	{CCI}[a]	4.5
23	LAA1101JBOFL	Failure - busbar (220V DC)	{CCI}[a]	4.4
24	RCP1299VZEFC	Failure to close - Motor Operated Valve	{CCI}[a]	4.4
25	RCP2299VZEFC	Failure to close - Motor Operated Valve	{CCI}[a]	4.4
26	RCP11_SSSF	Mechanical failure of stand still seal of RCP 1	{CCI}[a]	4.4
27	RCP21_SSSF	Mechanical failure of stand still seal of RCP 2	{CCI}[a]	4.4
28	RCP1285VZCFO	Failure to open - lift check valve	{CCI}[a]	4.4
29	RCP2285VZCFO	Failure to open - lift check valve	{CCI}[a]	4.4
30	RCP2284VZOFO	Failure to open - solenoid valve	{CCI}[a]	4.4
31	RCP1284VZOFO	Failure to open - solenoid valve	{CCI}[a]	4.4
32	RCP2299VZBFC	Failure to close - breaker	{CCI}[a]	4.4
33	RCP2284VZBFC	Solenoid valve contactor - failure to open	{CCI}[a]	4.4
34	RCP1284VZBFC	Solenoid valve contactor - failure to open	{CCI}[a]	4.4
35	RCP1299VZBFC	Failure to close - breaker	{CCI}[a]	4.4
36	RCP1238VPEFC	Failure to close - Motor Operated Valve	{CCI}[a]	4.3
37	RCP2238VPEFC	Failure to close - Motor Operated Valve	{CCI}[a]	4.3
38	RCP1240VPEFC	Failure to close - Motor Operated Valve	{CCI}[a]	4.3
39	RCP2240VPEFC	Failure to close - Motor Operated Valve	{CCI}[a]	4.3
40	RCP1260VPEFC	Failure to close - Motor Operated Valve	{CCI}[a]	4.3

SUB-CHAPTER 15.7 - TABLE 3

Operator action ranking according to Fussell-Vesely (FV)

No.	ID	Description	Delay [min]	Nominal probability per demand	FV
1	OP_FSCD_30MN	Operator fails to initiate FSCD (t<30min)	30	4.28E-02	1.42E-01
2	OP_EFWS_NCSS	Operator fails to start and control EFWS - NCSS	60	7.74E-02	5.36E-02
3	OP_BLEED_30MN_NCSS	Operator fails to initiate bleed in 30 min - NCSS	30	3.96E-01	4.90E-02
4	OP_FB_120M_MDEP_NCSS	Operator fails to initiate F&B (Tm=2h) with medium dependency - NCSS	120	1.50E-01	4.02E-02
5	OP_LHSI_IND_120MN	Operator fails to start LHSI indep. on CCWS/ESWS < 2h	120	2.13E-03	3.60E-02
6	OP_DIL_25MN	manual dilution isolation failure <25 min	25	1.45E-01	3.33E-02
7	OP_BLEED_120MN	Operator fails to initiate Bleed t<120mn	120	8.12E-03	3.28E-02
8	OP_EFW/MSRT_2H LOCAL	Operator fails SCD by the cross-connection of SGs and opening of MSRT before 2h in SBO condition -LOCAL	120	5.00E-02	2.01E-02
9	OP_FEED_TK	Operator fails the cross-connection of SG tank /Operator fails to re-feed SSS, MFWS or EFWS tank	240	1.00E-04	1.90E-02
10	OP_FB_120M_MDEP	Operator fails to initiate F&B (Tm= 2 h) with medium dependency	120	1.50E-01	1.76E-02
11	OPE_52_LOCAL	Operator fails to initiate IRWST cooling with CHRS (Grace period >4h) - local action	240	5.00E-02	1.72E-02
12	OP_BLEED_30MN MDEP	Operator Fails to Initiate Bleed in 30 min after failure of the PCD	30	2.30E-01	1.71E-02
13	OPE_PCD	Operator fails to start PCD before 15 mn	15	5.25E-01	1.69E-02
14	OP_EFWS	Operator failure to start and control EFWS in case of PS failure	60	2.84E-03	1.68E-02
15	OP_SBODG2H	Operator fails to start SBO diesels or to close breakers within 2 hours	120	2.13E-03	1.59E-02
16	OP_FSCD_30MN_IH	Operator fails to initiate FSCD (Tm=30mn) during internal hazard	30	1.01E-01	1.41E-02
17	OP_SIS_INJ_80MN_NCSS	Operator fails to start SIS by MHSI/LHSI (Tm=80min) - NCSS	80	8.44E-03	1.35E-02
18	OP_DIL_25MN_NCSS	Manual dilution isolation failure t<25min - NCSS	25	6.25E-01	7.37E-03
19	OP_SBODG30M	Operator fails to start SBO diesels or to close breakers within 30 minutes	30	4.28E-02	6.02E-03
20	OP_FSCD_15MN	Operator fails to initiate FSCD (Tm=15mn)	15	5.25E-01	4.19E-03
21	OP_LH/RHR_15MN	Operator fails to start LHSI train 4 for RHR (t<15min)	15	5.25E-01	4.03E-03
22	OP_SBODG_LOCAL	Operator fails to start the SBO Diesel by local action before 2h	120	5.00E-02	3.92E-03
23	OP_SCD 30MN	Operator fails to init second cooldown (Tm>30min)	30	4.28E-02	3.74E-03
24	OP_COMBI_240MN_LDEP	Operator fails to initiate primary bleed (Tm<4h) and LHSI for inj. with IRWST cooling + low dep	240	5.00E-02	3.21E-03
25	OPE_IH FL TH	Operator fails to prevent flooding (e.g. MCWS pump trip)	30	1.00E-01	2.84E-03

No.	ID	Description	Delay [min]	Nominal probability per demand	FV
26	OP_BLEED_30MN	Operator Fails to Initiate Bleed in 30 min	30	1.01E-01	2.64E-03
27	OP_SG_ISOL 1T	Operator fails to initiate SG isolation before Tm = 1h	60	2.84E-03	2.54E-03
28	OP_DIL_70MN	Manual dilution isolation failure < 70 min	70	2.44E-03	2.49E-03
29	OP_SIS_INJ_30MN	Operator fails to start SIS by MHSI/LHSI (Tm=30min)	30	4.28E-02	2.40E-03
30	OPE_IH FL SB	Operator fails to prevent flooding (e.g. ESWS pump trip)	30	1.00E-01	1.92E-03
31	OP_LHSI_IND_120_NCSS	Operator fails to start LHSI indep. on CCWS/ESWS < 2h - NCSS	120	8.27E-03	1.39E-03

SUB-CHAPTER 15.7 - TABLE 4

Operator action ranking according to Risk Increase Factor (RIF)

No.	ID	Description	Delay [min]	Nominal probability per demand	RIF
1	OP_FEED_TK	Operator fails the cross-connection of SG tank /Operator fails to re-feed SSS, MFWS or EFWS tank	240	1.00E-04	190.4
2	OP_LHSI_IND_120MN	Operator fails to start LHSI indep. on CCWS/ESWS < 2h	120	2.13E-03	17.9
3	OP_DIL_240MN	manual dilution isolation failure t> 240 min	240	1.00E-04	9.6
4	OP_SBODG2H	Operator fails to start SBO diesels or to close breakers within 2 hours	120	2.13E-03	8.4
5	OP_EFWS	Operator failure to start and control EFWS in case of PS failure	60	2.84E-03	6.9
6	OP_BLEED_120MN	Operator fails to initiate Bleed t<120mn	120	8.12E-03	5.0
7	OPE_SGTR	Operator fails to initiate the partial cooldown before IRWST drainage	240	1.00E-04	4.6
8	OP_FSCD_30MN	Operator fails to initiate FSCD (t<30min)	30	4.28E-02	4.2
9	OP_SIS_INJ_80MN_NCSS	Operator fails to start SIS by MHSI/LHSI (Tm=80min) - NCSS	80	8.44E-03	2.6
10	OPE_52	Operator fails to initiate IRWST cooling with CHRS (Grace period >4h)	240	1.00E-04	2.5
11	OP_DIL_70MN	Manual dilution isolation failure < 70 min	70	2.44E-03	2.0
12	OP_SG_ISOL 1T	Operator fails to initiate SG isolation before Tm = 1h	60	2.84E-03	1.9
13	OP_EFWS_NCSS	Operator fails to start and control EFWS - NCSS	60	7.74E-02	1.6
14	OP_EFW/MSRT_2H LOCAL	Operator fails SCD by the cross-connection of SGs and opening of MSRT before 2h in SBO condition -LOCAL	120	5.00E-02	1.4
15	OPE_52_LOCAL	Operator fails to initiate IRWST cooling with CHRS (Grace period >4h) - local action	240	5.00E-02	1.3
16	OP_FB_120M_MDEP_NCSS	Operator fails to initiate F&B (Tm=2h) with medium dependency - NCSS	120	1.50E-01	1.2
17	OP_DIL_25MN	manual dilution isolation failure <25 min	25	1.45E-01	1.2
18	OP_LHSI_IND_120_NCSS	Operator fails to start LHSI indep. on CCWS/ESWS < 2h - NCSS	120	8.27E-03	1.2
19	OPE_EBS 60MIN	Operator fails to initiate boration with EBS (1h)	60	2.84E-03	1.2
20	OP_SBODG30M	Operator fails to start SBO diesels or to close breakers within 30 minutes	30	4.28E-02	1.1
21	OP_SCD_120MN	Operator fails to init second cooldown (Tm=120mn)	120	2.13E-03	1.1
22	OPE_56	Operator fails to isolate letdown line (T>4h)	240	1.00E-04	1.1
23	OP_FSCD_30MN_IH	Operator fails to initiate FSCD (Tm=30mn) during internal hazard	30	1.01E-01	1.1
24	OP_BLEED_120MN_NCSS	Operator fails to initiate Bleed t<120mn - NCSS	120	8.27E-03	1.1
25	OP_FB_120M_MDEP	Operator fails to initiate F&B (Tm= 2 h) with medium dependency	120	1.50E-01	1.1

SUB-CHAPTER 15.7 - TABLE 5

Common Cause event ranking according to Risk Increase Factor (RIF)

No.	ID	Description	Nominal probability [per demand]	RIF
1	RIS1560VPCFO_D-ALL	CCF to open check valves (SIS first isolation valve)	{CCI}[a]	5156.0
2	RRI1210POMFR_D-ALL	CCF RRI pump motors	{CCI}[a]	2195.0
3	RRI1210POEFR_D-ALL	CCF to run CCWS pumps	{CCI}[a]	2195.0
4	SEC1110POEFR_D-ALL	CCF to run ESWS pumps	{CCI}[a]	2193.0
5	SEC1110POMFR_D-ALL	CCF SEC pump motors	{CCI}[a]	2192.0
6	RIS1560VPCFO_D-123	CCF to open check valves (SIS first isolation valve)	{CCI}[a]	1768.0
7	RRI1210POMFR_D-123	CCF RRI pump motors	{CCI}[a]	1664.0
8	RRI1210POEFR_D-123	CCF to run CCWS pumps	{CCI}[a]	1664.0
9	SEC1110POEFR_D-123	CCF to run ESWS pumps	{CCI}[a]	1663.0
10	SEC1110POMFR_D-123	CCF SEC pump motors	{CCI}[a]	1662.0
11	ASG1310VDEFO_D-ALL	CCF to open EFWS SG-Level control valves	{CCI}[a]	1541.0
12	ASG1212VDEFO_D-ALL	CCF to open EFWS pressure control valves	{CCI}[a]	1541.0
13	ASG1212VDBFC_D-ALL	CCF breaker	{CCI}[a]	1541.0
14	ASG1310VDBFC_D-ALL	CCF breaker	{CCI}[a]	1541.0
15	ASG1210POEFS_D-ALL	CCF to start EFWS pumps	{CCI}[a]	1539.0
16	9 STUCK RODS	At least 9 out of 89 stuck rods	{CCI}[a]	1530.0
17	ASG1411VDCFO_D-ALL	CCF to open EFWS checkvalves	{CCI}[a]	1501.0
18	ASG1211VDCFO_D-ALL	CCF to open EFWS check valves	{CCI}[a]	1501.0
19	ASG1210POEFR_D-ALL	CCF to run EFWS pumps	{CCI}[a]	790.4
20	RIS1560VPCFO_D-124	CCF to open check valves (SIS first isolation valve)	{CCI}[a]	533.1
21	RIS1420POEFR_D-ALL	CCF fail to run MHSI pump	{CCI}[a]	487.8
22	RIS1420POMFR_D-ALL	CCF RIS MP pump motors	{CCI}[a]	487.4
23	RIS1540VPCFO_D-ALL	CCF fail to open - check valves	{CCI}[a]	487.3
24	RIS1645VPCFO_D-ALL	CCF to open third SIS check valve inside containment	{CCI}[a]	487.3
25	RIS1420POBFC_D-ALL	CCF breaker	{CCI}[a]	486.9
26	RIS1420POEFS_D-ALL	CCF fail to start MHSI pumps	{CCI}[a]	485.6
27	RIS1420POMFS_D-ALL	CCF RIS MP pump motors	{CCI}[a]	481.1
28	RIS1480VPCFO_D-ALL	CCF to open check valves in MHSI Min Flow Line	{CCI}[a]	479.7
29	5 STUCK RODS	At least 5 out of 89 stuck rods	{CCI}[a]	459.2
30	RRI1210POMFR_D-234	CCF RRI pump motors	{CCI}[a]	442.5
31	RRI1210POEFR_D-234	CCF to run CCWS pumps	{CCI}[a]	442.4
32	SEC1110POEFR_D-124	CCF to run ESWS pumps	{CCI}[a]	440.6
33	SEC1110POMFR_D-234	CCF SEC pump motors	{CCI}[a]	439.4
34	ASG1212VDEFO_D-234	CCF to open EFWS pressure control valves	{CCI}[a]	273.3
35	ASG1310VDEFO_D-134	CCF to open EFWS SG-Level control valves	{CCI}[a]	273.3
36	ASG1212VDBFC_D-134	CCF breaker	{CCI}[a]	273.3
37	ASG1310VDBFC_D-134	CCF breaker	{CCI}[a]	273.3
38	ASG1210POEFS_D-234	CCF to start EFWS pumps	{CCI}[a]	271.7
39	RRI1210POMFR_D-23	CCF RRI pump motors	{CCI}[a]	270.4
40	RRI1210POEFR_D-23	CCF to run CCWS pumps	{CCI}[a]	270.3

SUB-CHAPTER 15.7 - TABLE 6

Common Cause event ranking according to Fussell-Vesely (FV)

No.	ID	Description	Nominal probability [per demand]	FV
1	RIS1420POEFR_D-ALL	CCF fail to run MHSI pump	{CCI}[a]	1.12E-01
2	LHP_____DFR_D-ALL	CCF to run emergency diesel generators	{CCI}[a]	5.11E-02
3	ASG1210POEFR_D-ALL	CCF to run EFWS pumps	{CCI}[a]	3.97E-02
4	RIS1420POEFR_D-124	CCF fail to run MHSI pump	{CCI}[a]	3.44E-02
5	RIS1420POEFR_D-234	CCF fail to run MHSI pump	{CCI}[a]	2.74E-02
6	RIS1420POEFR_D-123	CCF fail to run MHSI pump	{CCI}[a]	2.41E-02
7	LJP_____DFR_B-ALL	CCF to run SBO diesel generators	{CCI}[a]	1.70E-02
8	9 STUCK RODS	At least 9 out of 89 stuck rods	{CCI}[a]	1.58E-02
9	LHP_____DFR_D-134	CCF to run emergency diesel generators	{CCI}[a]	1.14E-02
10	LHP_____DFRB_D-ALL	CCF to run emergency diesel generators during 2 hours mission time	{CCI}[a]	8.36E-03
11	3 STUCK RODS	At least 3 out of 89 stuck rods	{CCI}[a]	8.11E-03
12	RIS1420POEFR_D-12	CCF fail to run MHSI pump	{CCI}[a]	7.94E-03
13	LHP_____DFR_D-124	CCF to run emergency diesel generators	{CCI}[a]	7.89E-03
14	RIS1420POEFR_D-24	CCF fail to run MHSI pump	{CCI}[a]	7.52E-03
15	ASG1210POEFR_D-234	CCF to run EFWS pumps	{CCI}[a]	7.13E-03
16	LHP_____DFR_D-234	CCF to run emergency diesel generators	{CCI}[a]	6.56E-03
17	LHP_____DFR_D-123	CCF to run emergency diesel generators	{CCI}[a]	6.43E-03
18	RBS1220POEFR_B-ALL	CCF to run EBS pumps	{CCI}[a]	6.06E-03
19	RBS1220POEFS_B-ALL	CCF to start EBS pumps	{CCI}[a]	5.74E-03
20	RRI1210POMFR_D-ALL	CCF RRI pump motors	{CCI}[a]	4.92E-03
21	RIS1420POEFR_D-134	CCF fail to run MHSI pump	{CCI}[a]	4.77E-03
22	5 STUCK RODS	At least 5 out of 89 stuck rods	{CCI}[a]	4.73E-03
23	RIS1420POMFR_D-ALL	CCF RIS MP pump motors	{CCI}[a]	4.45E-03
24	RRI1210POEFR_D-ALL	CCF to run CCWS pumps	{CCI}[a]	4.45E-03
25	RIS1540VPCFO_D-ALL	CCF fail to open - check valves	{CCI}[a]	4.19E-03
26	RCP6222VPRFO_C-ALL	CCF to open PZR safety valves	{CCI}[a]	3.86E-03
27	LHP_____DFR_D-14	CCF to run emergency diesel generators	{CCI}[a]	3.74E-03
28	RRI1210POMFR_D-123	CCF RRI pump motors	{CCI}[a]	3.73E-03
29	RIS1420POEFR_D-14	CCF fail to run MHSI pump	{CCI}[a]	3.70E-03
30	RRI1210POEFR_D-123	CCF to run CCWS pumps	{CCI}[a]	3.37E-03
31	LHP_____DFR_D-34	CCF to run emergency diesel generators	{CCI}[a]	3.20E-03
32	LHP_____DFR_D-13	CCF to run emergency diesel generators	{CCI}[a]	3.13E-03
33	ASG1210POEFR_D-123	CCF to run EFWS pumps	{CCI}[a]	3.07E-03
34	RIS1420POEFR_D-23	CCF fail to run MHSI pump	{CCI}[a]	2.79E-03
35	SEC1110POEFR_D-ALL	CCF to run ESWS pumps	{CCI}[a]	2.65E-03
36	LHP_____DFR_D-24	CCF to run emergency diesel generators	{CCI}[a]	2.37E-03
37	LHP_____DFR_D-12	CCF to run emergency diesel generators	{CCI}[a]	2.32E-03
38	VDA1110VVPFO_D-ALL	CCF fail to open MSR fluid valves	{CCI}[a]	2.32E-03
39	DEL1120GF_FR_B-ALL	CCF of Chiller units Tr1 and 4	{CCI}[a]	2.20E-03
40	LHP_____DFR_D-23	CCF to run emergency diesel generators	{CCI}[a]	2.04E-03

SUB-CHAPTER 15.7 - TABLE 7

I&C event ranking according to Fussell-Vesely (FV)

No.	ID	Description	Class	Nominal probability [per demand]	FV
1	SYS_OTHER_B_CC	Fail of common logic part (system B)	E1B/E2	{CCI}[a]	3.75E-01
2	SYS_PROTC_A_CC	E1A - Failure of common logic part	E1A	{CCI}[a]	1.66E-01
3	RPR_PS_DIV_B_A24SC	E1A, 2/4- Failure of specific logic part - PS diversity B	E1A	{CCI}[a]	2.61E-02
4	RPR_PS_DIV_A_A24SC	E1A, 2/4- Failure of specific logic part - PS diversity A	E1A	{CCI}[a]	2.27E-02
5	SYS_NCSS_FAIL	Failure of NCSS	E1B	{CCI}[a]	2.23E-02
6	RCP1223MT_AC	F1B, F2, NC sensor - Seal 1 cavity temperature RCP1 sensor 2	E1B/E2	{CCI}[a]	7.10E-03
7	RCP1222MT_AC	F1B, F2, NC sensor - Seal 1 cavity temperature RCP1 sensor 1	E1B/E2	{CCI}[a]	7.10E-03
8	RCP2223MT_AC	F1B, F2, NC sensor - Seal 1 cavity temperature RCP2 sensor 2	E1B/E2	{CCI}[a]	7.10E-03
9	RCP2222MT_AC	F1B, F2, NC sensor - Seal 1 cavity temperature RCP2 sensor 1	E1B/E2	{CCI}[a]	7.10E-03
10	RCPX861MN_AC_D-ALL	CCF between 4 hot leg loop sensors	E1A	{CCI}[a]	6.18E-03
11	RCPX861MN_AC_D-124	CCF between 4 hot leg loop sensors	E1A	{CCI}[a]	6.18E-03
12	RCPX861MN_AC_D-123	CCF between 4 hot leg loop sensors	E1A	{CCI}[a]	6.18E-03
13	RCPX861MN_AC_D-234	CCF between 4 hot leg loop sensors	E1A	{CCI}[a]	6.18E-03
14	RCPX861MN_AC_D-134	CCF between 4 hot leg loop sensors	E1A	{CCI}[a]	6.18E-03
15	ASGX221MD_AC_D-ALL	CCF to control EFWS pump flow due to flow measurement failure	E1A	{CCI}[a]	4.32E-03
16	RCP681YMP_AC_D-ALL	CCF between 4 pressuriser pressure sensors	E1A	{CCI}[a]	3.29E-03
17	RCP681YMP_AC_D-234	CCF between 4 pressuriser pressure sensors	E1A	{CCI}[a]	3.29E-03
18	RCP681YMP_AC_D-123	CCF between 4 pressuriser pressure sensors	E1A	{CCI}[a]	3.29E-03
19	RCP681YMP_AC_D-124	CCF between 4 pressuriser pressure sensors	E1A	{CCI}[a]	3.29E-03
20	RCP681YMP_AC_D-134	CCF between 4 pressuriser pressure sensors	E1A	{CCI}[a]	3.29E-03
21	RCP2343MT_AC	F1B, F2, NC sensor - upper pad temperature RCP2 motor sensor 1	E1B/E2	{CCI}[a]	2.81E-03
22	RCP2346MT_AC	F1B, F2, NC sensor - lower pad temperature RCP2 motor sensor 2	E1B/E2	{CCI}[a]	2.81E-03
23	RCP2344MT_AC	F1B, F2, NC sensor - upper pad temperature RCP2 motor sensor 2	E1B/E2	{CCI}[a]	2.81E-03
24	RCP2342MT_AC	F1B, F2, NC sensor - Upper bearing temperature RCP2 motor sensor 2	E1B/E2	{CCI}[a]	2.81E-03
25	RCP2345MT_AC	F1B, F2, NC sensor - lower pad temperature RCP2 motor sensor 1	E1B/E2	{CCI}[a]	2.81E-03
26	RCP2341MT_AC	F1B, F2, NC sensor - Upper bearing temperature RCP2 motor sensor 1	E1B/E2	{CCI}[a]	2.81E-03
27	RCP2311MT_AC	F1B, F2, NC sensor - lower bearing temperature RCP2 motor sensor 2	E1B/E2	{CCI}[a]	2.81E-03
28	RCP2313MT_AC	F1B, F2, NC sensor - lower bearing temperature RCP2 motor sensor 1	E1B/E2	{CCI}[a]	2.81E-03

No.	ID	Description	Class	Nominal probability [per demand]	FV
29	AREX83YMN_CCF_16	CCF on the 16 SG level wide range sensors	E1A	{CCI}[a]	2.61E-03
30	RCP1343MT_AC	F1B, F2, NC sensor - upper pad temperature RCP1 motor sensor 1	E1B/E2	{CCI}[a]	2.61E-03
31	RCP1344MT_AC	F1B, F2, NC sensor - upper pad temperature RCP1 motor sensor 2	E1B/E2	{CCI}[a]	2.61E-03
32	RCP1346MT_AC	F1B, F2, NC sensor - lower pad temperature RCP1 motor sensor 2	E1B/E2	{CCI}[a]	2.61E-03
33	RCP1313MT_AC	F1B, F2, NC sensor - lower bearing temperature RCP1 motor sensor 1	E1B/E2	{CCI}[a]	2.61E-03
34	RCP1345MT_AC	F1B, F2, NC sensor - lower pad temperature RCP1 motor sensor 1	E1B/E2	{CCI}[a]	2.61E-03
35	RCP1311MT_AC	F1B, F2, NC sensor - lower bearing temperature RCP1 motor sensor 2	E1B/E2	{CCI}[a]	2.61E-03
36	RCP1342MT_AC	F1B, F2, NC sensor - Upper bearing temperature RCP1 motor sensor 2	E1B/E2	{CCI}[a]	2.61E-03
37	RCP1341MT_AC	F1B, F2, NC sensor - Upper bearing temperature RCP1 motor sensor 1	E1B/E2	{CCI}[a]	2.61E-03
38	AAD_______AC	F1B, F2, NC sensor - On-Off pump sensor	E1B/E2	{CCI}[a]	1.90E-03
39	RCP3223MT_AC	F1B, F2, NC sensor - Seal 1 cavity temperature RCP3 sensor 2	E1B/E2	{CCI}[a]	1.89E-03
40	RCP4223MT_AC	F1B, F2, NC sensor - Seal 1 cavity temperature RCP4 sensor 2	E1B/E2	{CCI}[a]	1.89E-03

SUB-CHAPTER 15.7 - TABLE 8

Overall ranking according to FV (RIF given)

No.	ID	Description	Nominal probability [per demand]	FV	RIF
1	SYS_OTHER_B_CC	Fail of common logic part (system B)	{CCI}[a]	3.75E-01	38.1
2	RCP_SEAL#1_RD	Failure of RCP shaft seals #1 during rundown phase	{CCI}[a]	2.66E-01	2.5
3	RCP_SEAL#2_RD	Conditional failure of RCP shaft seals #2 during rundown phase	{CCI}[a]	2.66E-01	1.5
4	SYS_PROTC_A_CC	E1A - Failure of common logic part	{CCI}[a]	1.66E-01	1659.0
5	OP_FSCD_30MN	Operator fails to initiate FSCD (t<30min)	{CCI}[a]	1.42E-01	4.2
6	RIS1420POEFR_D-ALL	CCF fail to run MHSI pump	{CCI}[a]	1.12E-01	487.8
7	RCP_SEAL#1_SD	Failure of RCP shaft seals #1 during shutdown phase	{CCI}[a]	9.75E-02	1.7
8	RCP_SEAL#2_SD	Conditional failure of RCP shaft seals #2 during shutdown phase	{CCI}[a]	9.75E-02	1.2
9	RCP-SSSS_ORING_52	Failure of the Orings exposed to RCS (P,T)	{CCI}[a]	9.57E-02	1.4
10	RCP-SSSS_ORING_53	Failure of the Orings exposed to RCS (P,T)	{CCI}[a]	9.57E-02	1.4
11	RCP-SSSS_ORING_51	Failure of the Orings exposed to RCS (P,T)	{CCI}[a]	7.50E-02	1.5
12	OP_EFWS_NCSS	Operator fails to start and control EFWS - NCSS	{CCI}[a]	5.36E-02	1.6
13	LHP_____DFR_D-ALL	CCF to run emergency diesel generators	{CCI}[a]	5.11E-02	43.6
14	OP_BLEED_30MN_NCSS	Op. fails to initiate bleed in 30 min - NCSS	{CCI}[a]	4.90E-02	1.1
15	OP_FB_120M_MDEP_NCSS	Operator fails to initiate F&B (Tm=2h) with medium dependency - NCSS	{CCI}[a]	4.02E-02	1.2
16	ASG1210POEFR_D-ALL	CCF to run EFWS pumps	{CCI}[a]	3.97E-02	790.4
17	OP_LHSI_IND_120MN	Operator fails to start LHSI indep. on CCWS/ESWS < 2h	{CCI}[a]	3.60E-02	17.9
18	RIS1420POEFR_D-124	CCF fail to run MHSI pump	{CCI}[a]	3.44E-02	151.6
19	OP_DIL_25MN	manual dilution isolation failure <25 min	{CCI}[a]	3.33E-02	1.2
20	PM_GROUP_A_ST_A	Preventive Maintenance on the cooling chain (RIS/RRI/SEC) during power operation	{CCI}[a]	3.28E-02	1.4
21	OP_BLEED_120MN	Operator fails to initiate Bleed t<120mn	{CCI}[a]	3.28E-02	5.0
22	LJP_____DFR	Station Blackout Diesel Generator fails to run	{CCI}[a]	3.11E-02	1.7
23	RIS1420POEFR_D-234	CCF fail to run MHSI pump	{CCI}[a]	2.74E-02	120.4
24	LHP_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.73E-02	1.2
25	LHS_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.70E-02	1.2
26	RPR_PS_DIV_B_A24SC	E1A, 2/4- Failure of specific logic part - PS diversity B	{CCI}[a]	2.61E-02	261.9
27	LHR_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.43E-02	1.2
28	RIS1420POEFR_D-123	CCF fail to run MHSI pump	{CCI}[a]	2.41E-02	106.8

SUB-CHAPTER 15.7 - TABLE 8

Overall ranking according to FV (RIF given)

No.	ID	Description	Nominal probability [per demand]	FV	RIF
29	RPR_PS_DIV_A_A24 SC	E1A, 2/4- Failure of specific logic part - PS diversity A	{CCI}[a]	2.27E-02	227.9
30	SYS_NCSS_FAIL	Failure of NCSS	{CCI}[a]	2.23E-02	23.3
31	LJS_____DFR	Station Blackout Diesel Generator fails to run	{CCI}[a]	2.19E-02	1.5
32	RIS3420POEFR	MHSI pump failure to run	{CCI}[a]	2.15E-02	2.0
33	RIS4420POEFR	MHSI pump failure to run	{CCI}[a]	2.07E-02	2.0
34	LHQ_____DFR	Emergency Diesel Generator fails to run	{CCI}[a]	2.03E-02	1.2
35	OP_EFW/MSRT_2H LOCAL	Op. fails SCD by the cross-connection of SGs and opening of MSRT before 2h in SBO condition - LOCAL	{CCI}[a]	2.01E-02	1.4
36	OP_FEED_TK	Operator fails the cross-connection of SG tank /Operator fails to re-feed SSS, MFWS or EFWS tank	{CCI}[a]	1.90E-02	190.4
37	PM_GROUP_G_ST_A	Preventive Maintenance on the SBO-DG (LJ-) during state A	{CCI}[a]	1.87E-02	1.4
38	OP_FB_120M_MDEP	Operator fails to initiate F&B (Tm= 2 h) with medium dependency	{CCI}[a]	1.76E-02	1.1
39	OPE_52_LOCAL	Operator fails to initiate IRWST cooling with CHRS (Grace period >4h) - local action	{CCI}[a]	1.72E-02	1.3
40	OP_BLEED_30MN MDEP	Operator Fails to Initiate Bleed in 30 min after failure of the PCD	{CCI}[a]	1.71E-02	1.1

SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	Freq. [/r.y]	%	Freq. [r.y]	%								
1	2.41E-08	3.40	2.41E-08	3.40	D	Loss of Cooling Chain (total)	Operator fails to start LHSI independent of RRI [CCWS]/SEC[ESWS] (<120min)					
2	1.64E-08	2.32	4.05E-08	5.72	AB	Small break LOCA - Pressuriser break (2-45cm2)	Operator fails to initiate Fast Secondary Cooldow (<30min)	CCF to run all MHSI pump				
3	1.20E-08	1.69	5.25E-08	7.41	AB	Loss of Ultimate Heat Sink	Operator fails the cross-connection of SG tank / Operator fails to re-feed SSS, MFWS or EFWS tank					
4	1.16E-08	1.64	6.41E-08	9.05	A	Spurious Reactor Trip	Operator fails to start and control EFWS via NCSS	Operator fails to initiate F&B (120min) with medium dependency via NCSS	I&C failure of SPPA-T2000 platform common logic part	I&C failure of TXS platform common logic part		
5	1.15E-08	1.63	7.56E-08	10.68	AB	Small break LOCA (2-45cm2)	Operator fails to initiate Fast Secondary Cooldow (<30min)	CCF to run all MHSI pump				
6	1.15E-08	1.63	8.71E-08	12.31	AB	Small break LOCA (2-45cm2)	Operator fails to initiate Fast Secondary Cooldow (<30min)	CCF to run MHSI pump (train 1, 2, 3)				
7	1.10E-08	1.55	9.81E-08	13.86	AB	Short LOOP (<2h)	Failure of SSSS Orings 2	Failure of SSSS Orings 3	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	I&C failure of TXS platform common logic part	

SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	Freq. [/r.y]	%	Freq. [r.y]	%								
8	1.00E-08	1.41	1.08E-07	15.27	ALL	Reactor Pressure Vessel failure						
9	9.41E-09	1.33	1.18E-07	16.60	AB	Loss of Ultimate Heat Sink	Operator fails to initiate Fast Secondary Cooldow (<30min)	Failure of SSSS Orings 2	Failure of SSSS Orings 3	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	
10	7.33E-09	1.04	1.25E-07	17.64	AB	Short LOOP (<2h)	Failure of SSSS Orings 1	Failure of SSSS Orings 3	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	I&C failure of TXS platform common logic part	
11	7.33E-09	1.04	1.32E-07	18.68	AB	Short LOOP (<2h)	Failure of SSSS Orings 1	Failure of SSSS Orings 2	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	I&C failure of TXS platform common logic part	
12	7.10E-09	1.00	1.39E-07	19.68	AB	Fire in Safeguard Building 1	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	CCF to run all MHSI pump	I&C failure of SPPA-T2000 platform common logic part	Safety Train 1 unavailable after fire	
13	7.10E-09	1.00	1.46E-07	20.68	AB	Fire in Safeguard Building 1	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	CCF fail to run MHSI pump (train 2, 3, 4)	I&C failure of SPPA-T2000 platform common logic part	Safety Train 1 unavailable after fire	
14	6.94E-09	0.98	1.53E-07	21.66	AB	Fire in the Switchgear Building	CCF to run EFWS pumps	Operator fails to initiate primary bleed (120mn)				
15	6.28E-09	0.89	1.60E-07	22.55	AB	Loss of Ultimate Heat Sink	Operator fails to initiate Fast Secondary Cooldow (<30min)	Failure of SSSS Orings 1	Failure of SSSS Orings 2	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	

SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	**Freq. [/r.y]**	**%**	**Freq. [r.y]**	**%**								
16	6.28E-09	0.89	1.66E-07	23.44	AB	Loss of Ultimate Heat Sink	Operator fails to initiate Fast Secondary Cooldow (<30min)	Failure of SSSS Orings 1	Failure of SSSS Orings 3	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	
17	6.03E-09	0.85	1.72E-07	24.29	AB	Loss of Ultimate Heat Sink	CCF to run EFWS pumps					
18	5.88E-09	0.83	1.78E-07	25.12	AB	Fire in Safeguard Building 1	Failure of RCP shaft seals #1 during shutdown phase	Failure of RCP shaft seals #2 during shutdown phase	CCF to run MHSI pump (train 2, 3, 4)	I&C failure of SPPA-T2000 platform common logic part	Safety Train 1 unavailable after fire	
19	5.88E-09	0.83	1.84E-07	25.95	AB	Fire in Safeguard Building 1	Failure of RCP shaft seals #1 during shutdown phase	Failure of RCP shaft seals #2 during shutdown phase	CCF to run all MHSI pump	I&C failure of SPPA-T2000 platform common logic part	Safety Train 1 unavailable after fire	
20	5.34E-09	0.75	1.89E-07	26.70	AB	Loss of Cooling chain (patial: operating CCWS train)	Operator fails to start and control EFWS via NCSS	Operator fails to initiate F&B (120min) with medium dependency via NCSS	I&C failure of SPPA-T2000 platform common logic part	I&C failure of TXS platform common logic part		
21	4.56E-09	0.64	1.94E-07	27.34	AB	Small Secondary steam line break upstream MSIV	CCF to run all MHSI pump	I&C failure of SPPA-T2000 platform common logic part				
22	4.50E-09	0.64	1.98E-07	27.98	Ca	Heterogeneous Dilution						
23	4.38E-09	0.62	2.02E-07	28.60	B	Homogeneous Boron Dilution (100t/h)	Manual dilution isolation failure (<25min) via NCSS	I&C failure of SPPA-T2000 platform common logic part	I&C failure of TXS platform common logic part			

SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	Freq. [/r.y]	%	Freq. [r.y]	%								
24	3.87E-09	0.55	2.06E-07	29.15	A	Spurious Reactor Trip	Operator fails to initiate IRWST cooling with CHRS (240min) - local action	Operator fails to start and control EFWS via NCSS	I&C failure of SPPA-T2000 platform common logic part	I&C failure of TXS platform common logic part		
25	3.83E-09	0.54	2.10E-07	29.69	AB	Small break LOCA - Pressuriseur break (2-45cm2)	CCF to run all MHSI pump	I&C failure of SPPA-T2000 platform common logic part				
26	3.70E-09	0.52	2.14E-07	30.21	AB	Interfacing system LOCA						
27	3.38E-09	0.48	2.17E-07	30.69	AB	Long LOOP (<24h)	CCF to run all Emergency Diesel Generators	Operator fails secondary cooldown by the cross-connection of SGs and opening of MSRT before 2h in SBO condition – local action	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase		
28	2.90E-09	0.41	2.20E-07	31.10	A	Turbine trip	Operator fails to start and control EFWS via NCSS	Operator fails to initiate F&B (120min) with medium dependency via NCSS	I&C failure of SPPA-T2000 platform common logic part	I&C failure of TXS platform common logic part		
29	2.80E-09	0.40	2.23E-07	31.50	AB	Long LOOP (<24h)	CCF to run all Emergency Diesel Generators	CCF to run all SBO diesel generators				
30	2.69E-09	0.38	2.26E-07	31.88	AB	Small break LOCA (2-45cm2)	CCF to run all MHSI pump	I&C failure of SPPA-T2000 platform common logic part				

SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	Freq. [/r.y]	%	Freq. [r.y]	%								
31	2.69E-09	0.38	2.28E-07	32.26	AB	Small break LOCA (2-45cm2)	CCF to run MHSI pump (train 1, 2, 3)	I&C failure of SPPA-T2000 platform common logic part				
32	2.56E-09	0.36	2.31E-07	32.62	AB	Short LOOP (<2h)	Operator failure to start and control EFWS (60min) in case of PS failure	Operator fails to initiate F&B (120min) with medium dependency	I&C failure of TXS platform common logic part			
33	2.56E-09	0.36	2.33E-07	32.98	AB	Short LOOP (<2h)	Operator failure to start and control EFWS (60min) in case of PS failure	Operator fails to initiate F&B (120min) with medium dependency	I&C failure of PS sub-system A			
34	2.55E-09	0.36	2.36E-07	33.34	A	Total Loss Of Main FeedWater	Conditional probability AAD [SSS] failure in case of total loss of ARE [MFWS]	CCF to run EFWS pumps	Operator fails to initiate primary bleed (120mn)			
35	2.54E-09	0.36	2.38E-07	33.70	AB	Long LOOP (<24h)	CCF to run all Emergency Diesel Generators	Operator fails to start SBO diesels or to close breakers (120min)				
36	2.37E-09	0.33	2.41E-07	34.03	AB	Long LOOP (<24h)	CCF to run all Emergency Diesel Generators	Station Blackout Diesel Generator LJP fails to run	Station Blackout Diesel Generator LJS fails to run			

SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	Freq. [/r.y]	%	Freq. [r.y]	%								
37	2.25E-09	0.32	2.43E-07	34.35	A	Spurious Reactor Trip	Induced long LOOP (>2h) after Reactor Trip	CCF to run all Emergency Diesel Generators	Operator fails secondary cooldown by the cross-connection of SGs and opening of MSRT in SBO condition (120min) - local action	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	
38	2.20E-09	0.31	2.45E-07	34.66	AB	Loss of Ultimate Heat Sink	Failure of SSSS Orings 2	Failure of SSSS Orings 3	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	I&C failure of SPPA-T2000 platform common logic part	
39	2.18E-09	0.31	2.47E-07	34.97	AB	Long LOOP (<24h)	CCF to run all Emergency Diesel Generators	Station Blackout Diesel Generator LJP fails to run	Preventive Maintenance on the SBO diesel generators			
40	2.16E-09	0.31	2.50E-07	35.28	A	ATWS - Spurious Ppressuriser Spray	At least 5 out of 89 stuck rods	Moderator coefficient unfavourable				
41	2.03E-09	0.29	2.52E-07	35.57	AB	Small break LOCA - Pressuriser break (2-45cm2)	Failure of GCT [MSB]	Operator fails to start partial cooldown (15 min)	Operator fails to itnitiate primary bleed (30min) after operator failure of partial cooldown	I&C failure of TXS platform common logic part		
42	2.03E-09	0.29	2.54E-07	35.86	AB	Small break LOCA - Pressuriser break (2-45cm2)	Failure of GCT [MSB]	Operator fails to start partial cooldown (15 min)	Operator fails to tnitiate primary bleed (30min) after operator failure of partial cooldown	I&C failure of PS sub-system B		

SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	Freq. [/r.y]	%	Freq. [r.y]	%								
43	1.97E-09	0.28	2.56E-07	36.14	AB	Loss of Cooling chain (partial: 1 CCWS common user header)	Operator fails to start and control EFWS via NCSS	Operator fails to initiate F&B (120min) with medium dependency via NCSS	I&C failure of SPPA-T2000 platform common logic part	I&C failure of TXS platform common logic part		
44	1.92E-09	0.27	2.58E-07	36.41	AB	Loss of Cooling chain (partial: operating CCWS train)	Failure of SSSS Orings 2	Failure of SSSS Orings 3	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	CCF to run all MHSI pump	I&C failure of SPPA-T2000 platform common logic part
45	1.92E-09	0.27	2.60E-07	36.68	AB	Loss of Cooling chain (patial: operating CCWS train)	Failure of SSSS Orings 2	Failure of SSSS Orings 3	Failure of RCP shaft seals #1 during rundown phase	Failure of RCP shaft seals #2 during rundown phase	CCF to run MHSI pump (train 1, 2, 4)	I&C failure of SPPA-T2000 platform common logic part
46	1.91E-09	0.27	2.61E-07	36.95	AB	Steam Generator Tube Rupture (2 tubes)	Operator fails to initiate second cooldown (30min)	CCF to run all MHSI pump				
47	1.89E-09	0.27	2.63E-07	37.22	AB	Small break LOCA (2-45cm2)	Operator fails to initiate Fast Secondary Cooldow (<30min)	Preventive Maintenance on the cooling chaine (RIS/RRI/SEC)	CCF to run MHSI pump (train 1, 3)			
48	1.87E-09	0.26	2.65E-07	37.48	A	Spurious Reactor Trip	Induced long LOOP (>2h) after Reactor Trip	CCF to run all Emergency Diesel Generators	CCF to run all SBO diesel generators			
49	1.86E-09	0.26	2.67E-07	37.74	Cb	Uncontrolled Level Drop	Operator fails to start MHSI/LHSI (80min) via NCSS	CCF of RCP [RCS] loop level sensors	I&C failure of SPPA-T2000 platform common logic part			

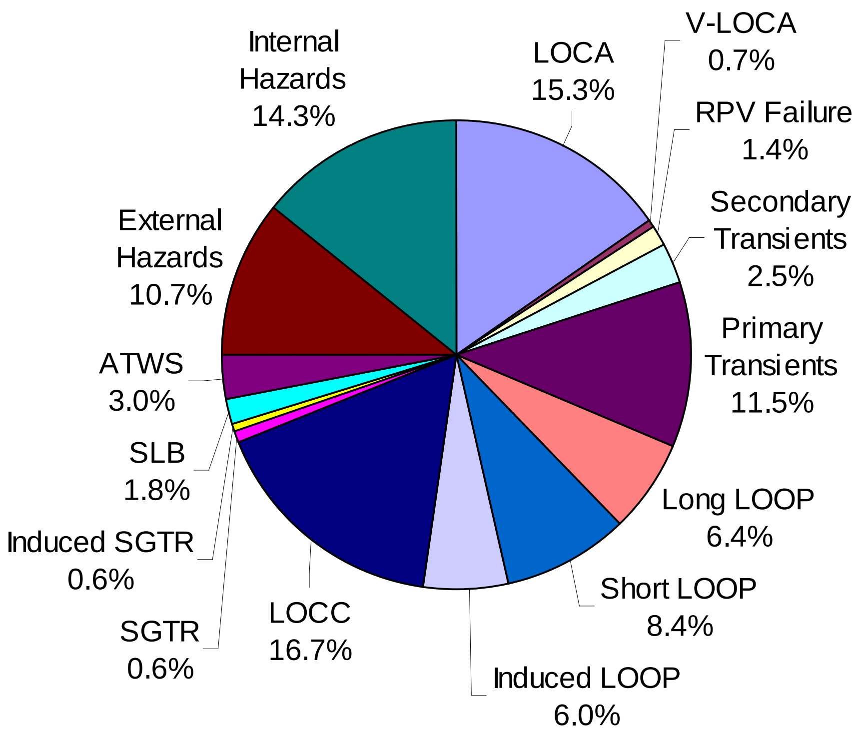
SUB-CHAPTER 15.7 - TABLE 9

Fifty most frequent minimal cutsets contributing to the overall CDF with preventive maintenance

Cutsets			Cumulative		Reactor state	Initiating Event	Failure 1	Failure 2	Failure 3	Failure 4	Failure 5	Failure 6
No.	Freq. [/r.y]	%	Freq. [r.y]	%								
50	1.86E-09	0.26	2.69E-07	38.00	Cb	Uncontrolled Level Drop	Operator fails to start MHSI/LHSI (80min) via NCSS	CCF of RCP [RCS] loop level sensors (3 out of 4)	I&C failure of SPPA-T2000 platform common logic part			

SUB-CHAPTER 15.7 - FIGURE 1

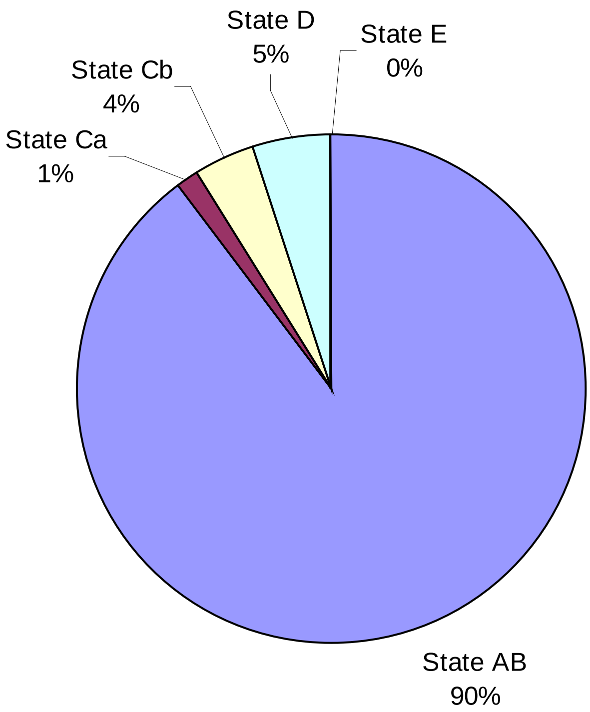
Contribution of the Initiating Events to the Overall CDF with preventive maintenance



Primary transients:	Boron dilution (homogeneous and heterogeneous) in all reactor states
	Loss of RIS [SIS]/RRA [RHRS] in shutdown states
	Reactor Trip in power operation
	Uncontrolled Level Drop during shutdown states Cb and D
Secondary transients:	Turbine trip
	Loss of condenser
	Loss of Main Feedwater System
	Loss of Start-up and Shutdown System

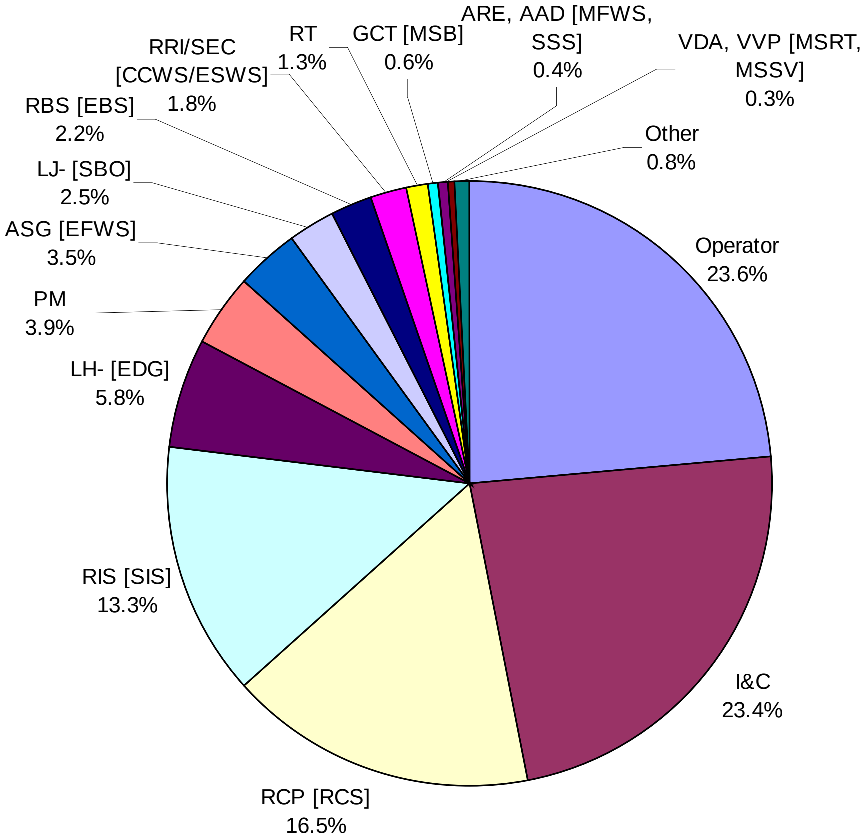
SUB-CHAPTER 15.7 - FIGURE 2

Contribution of the Plant Operating States to the Overall CDF with preventive maintenance



SUB-CHAPTER 15.7 - FIGURE 3

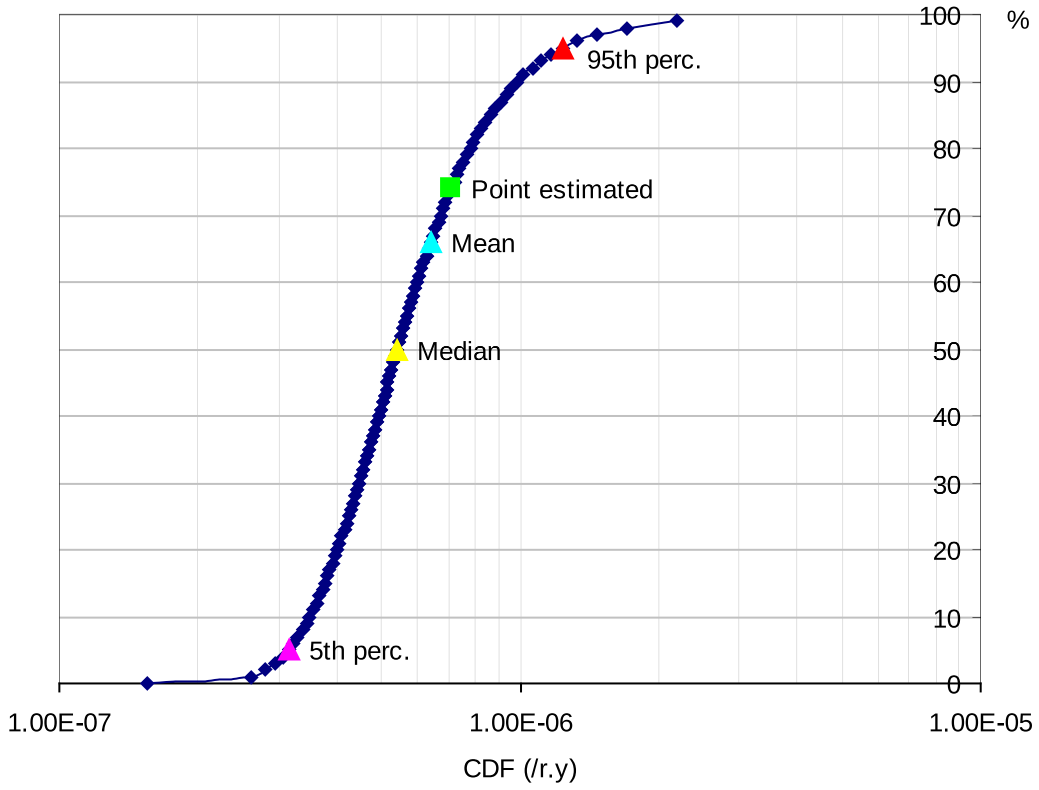
Systems Contribution to the Overall CDF



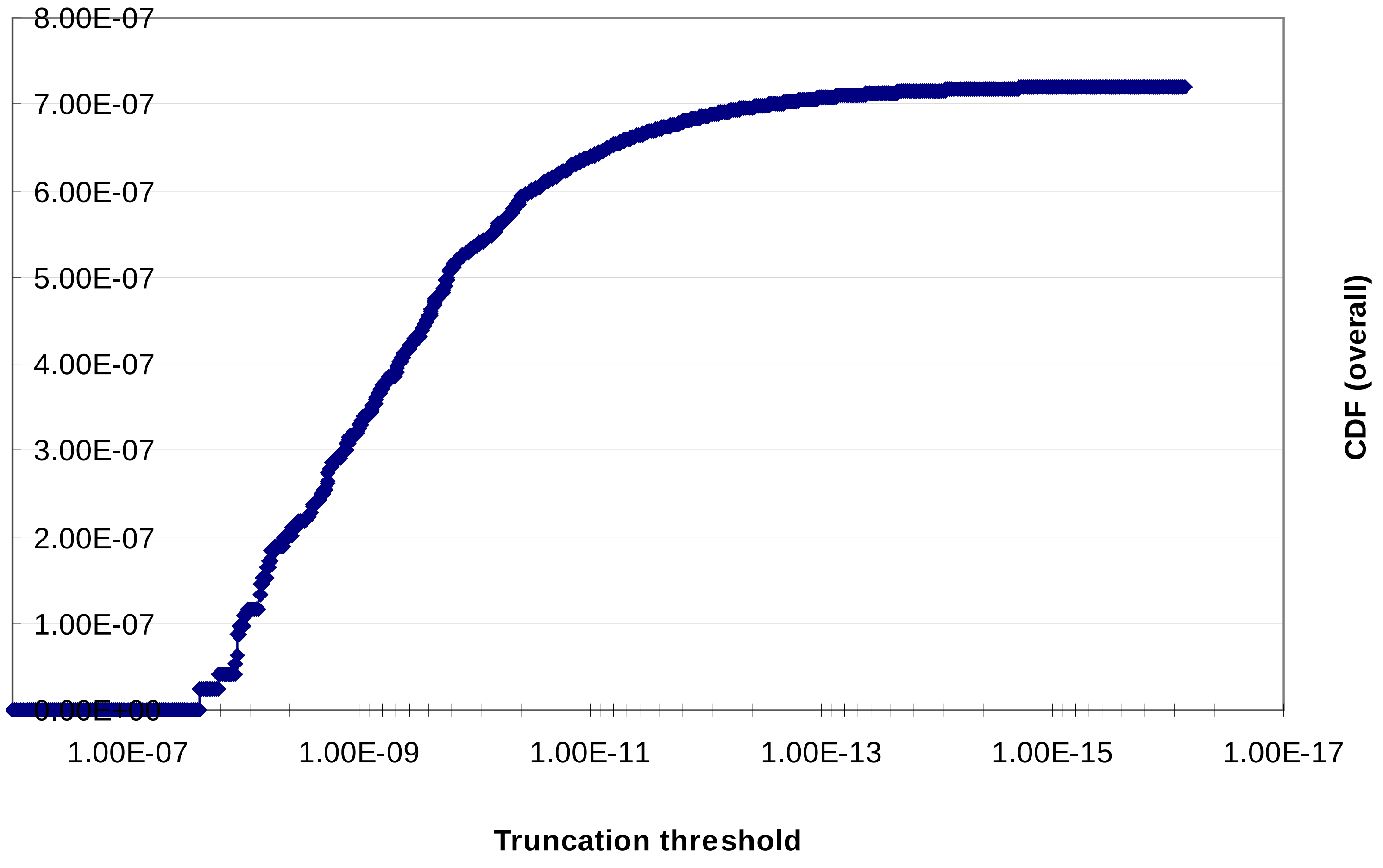
ID	Description
Operator	Operator Actions (including both pre-accident and post-accident actions)
I&C	Instrumentation & Control
RCP [RCS]	Reactor Coolant System (w/o Pressuriser)
RIS [SIS]	Safety Injection System
LH- [EDG]	Emergency Diesels Generators
PM	Preventive Maintenance
ASG [EFWS]	Emergency Feedwater System
LJ- [SBO]	Station Blackout Diesel Generators
RBS [EBS]	Extra Borating System
RRI/SEC [CCWS/ESWS]	Cooling Chain (CCWS, ESWS)
RT	Reactor Trip (blocage of rods)
GCT [MSB]	Main Steam Bypass
ARE, AAD [MFW, SSS]	Startup and Shutdown System, Main FeedWater System
VDA, VVP [MSRT,MSSV]	Main Steam Valves (MSSV, MSIV, MSRT)
Other	Other systems

SUB-CHAPTER 15.7 - FIGURE 4

Cumulative Distributions for the overall CDF with preventive maintenance

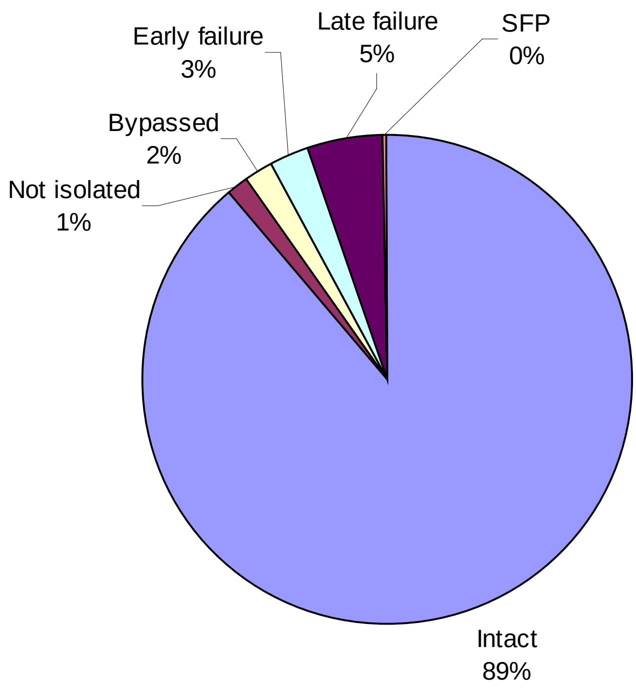


SUB-CHAPTER 15.7 - FIGURE 5



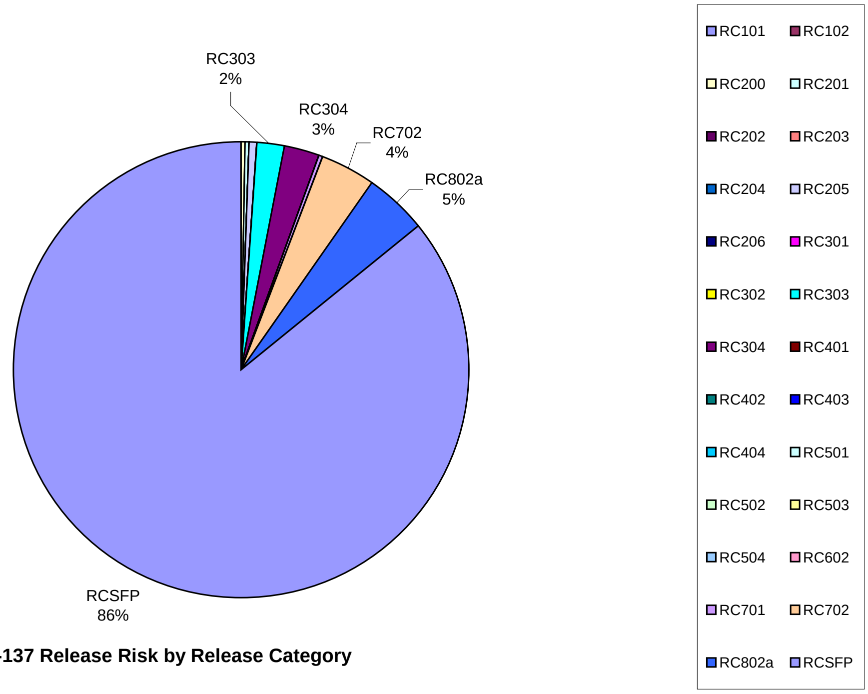
SUB-CHAPTER 15.7 - FIGURE 6

Containment Failure Modes Frequency (expressed in % of CDF)



SUB-CHAPTER 15.7 - FIGURE 7

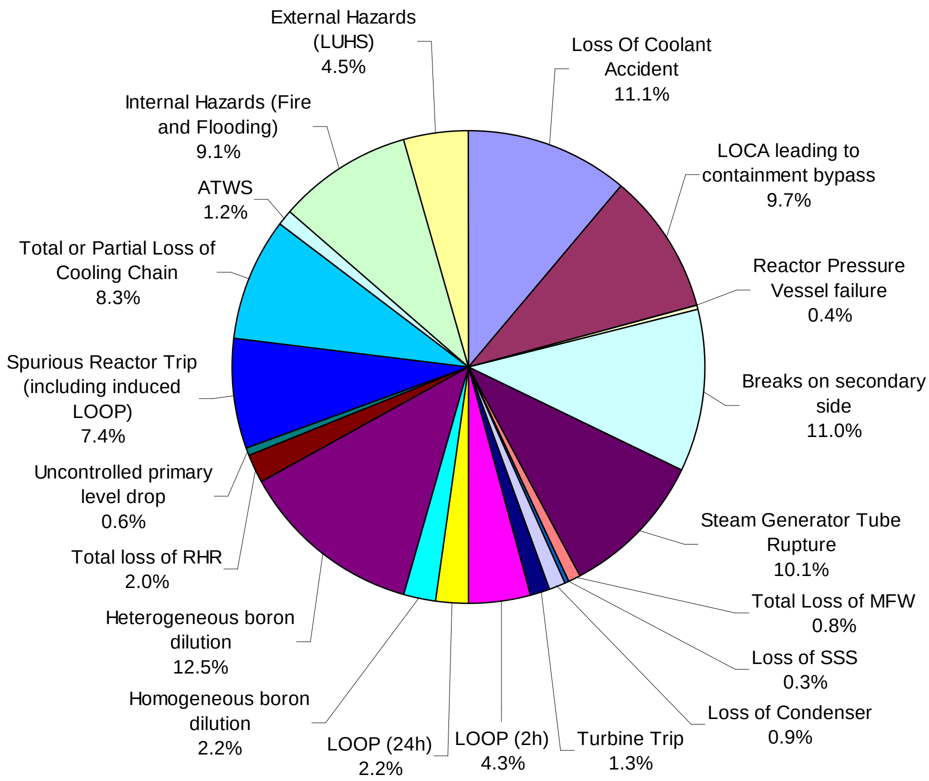
Relative Contributions to Cs-137 Release Risk by RC



Relative Contributions to Cs-137 Release Risk by Release Category

SUB-CHAPTER 15.7 - FIGURE 8

Relative Contributions of initiating events to LERF



SUB-CHAPTER 15.7 – 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.

2. SUMMARY OF LEVEL 1 RESULTS

2.5. KEY ASSUMPTIONS

[Ref-1] European Utility Requirements for LWR Nuclear Power Plants, Volume 2: Generic Nuclear Island Requirements, Chapter 17: PSA Methodology, Revision B.
EUR Document. November 1995. (E)

[Ref-2] A. Drevet. OL3 PSA Support Studies. NEPR-F DC 241 Revision B FIN.
AREVA/SIEMENS. June 2008. (E)

6. SENSITIVITY ANALYSES

6.2. LONG TERM ANALYSIS

6.2.1. Aims

[Ref-1] Long term probabilistic analysis of Loss Of Off-site Power (LOOP) and Loss of Ultimate Heat Sink (LUHS) situations.
ENFCFF040206 Revision C. EDF/SEPTEN. March 2006. (E)

6.2.2. Areas and scope of the study

[Ref-1] Long term probabilistic analysis of Loss Of Off-site Power (LOOP) and Loss of Ultimate Heat Sink (LUHS) situations.
ENFCFF040206 Revision C. EDF/SEPTEN. March 2006. (E)

6.4.4. Sensitivity to CCF parameters

[Ref-1] U.S. Nuclear Regulatory Commission, "CCF Parameter Estimations, 2007 Update", (http://nrcoe.inl.gov/results/CCF/ParamEst2007/ccfparamest.htm). September 2008. (E)

6.5. OPERATOR ACTION

6.5.2. Manual actuation of Feed & Bleed before 2 hours

[Ref-1] F. Godefroy. Human Reliability Analysis Notebook of the UK EPR Probabilistic Safety Assessment. NEPS-F DC 191 Revision A. AREVA. January 2010. (E)

6.6. SEAL LOCA RELIABILITY MODEL

[Ref-1] EPR – Failure of the Reactor Coolant Pump Seals. NEPS-F DC 383 Revision B. AREVA. 2009. (E)