UK EPR	Title: PCSR – Sub-chapter 14.4 – Analyses of the PCC-3 events
	UKEPR-0002-144 Issue 08

Total number of pages: 180	Page No.: I / VII
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REVISION HISTORY

Issue	Description	Date
00	First issue for INSA review.	04.01.2008
01	Second issue for INSA review.	04.02.2008
02	Integration of co-applicant and INSA review comments.	30.04.2008
03	Addition of section relative to SGTR (1 tube)	29.06.2008
04	Remarking with RESTRICTED classification	16.01.2009
05	June 2009 update including: – Text clarifications – Addition of references – Technical update to account for December 2008 Design Freeze notably new permissive signal for URBWZ (section 6), new PSRV design and pressuriser spray classification and partial cooldown rate (sections 3, 5 and 7) and fuel pool and FPCS/FPPS characteristics including available volumes and RRI parameters (sections 14, 15 and 16).	28-06-2009
06	Removal of RESTRICTED classification	10-06-2010
07	Consolidated Step 4 PCSR update: - Minor editorial changes - Update of references - Update of steam generator tube rupture (1 tube) fault analysis (§6), - Update of the spent fuel pool fault analyses (§14, §15 and §16)	27-03-2011

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REVISION HISTORY (Cont'd)

Issue	Description	Date
08	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes - Update of references (English translations) - Addition of a remark about inherent boron dilution in case of small break LOCA and addition of an associated reference (§5) - Update of Steam Generator Tube Rupture (1 Tube) (§6) - Removal of text regarding breaks located upstream of the two isolation valves not being taken into account (§16)	26-11-2012

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SUB-CHAPTER 14.4 - ANALYSES OF THE PCC-3 EVENTS

1. SMALL STEAM OR FEEDWATER SYSTEM PIPING FAILURE (DN < 50), INCLUDING BREAK OF CONNECTING LINES TO STEAM GENERATOR (DN < 50) [STATES A, B]

1.1. ACCIDENT DESCRIPTION

Minor piping failures (DN < 50mm) located in the steam or feedwater system, or in the Steam Generators (SGs) connecting lines are defined as PCC-3 events related to steam or feedwater breaks.

The consequences of these PCC-3 events are bounded by the PCC-4 events ‘Main steam line break’ or ‘Main feedwater line break’, which are analysed in sections 2 and 3 of Sub-chapter 14.5. For the smaller PCC-3 breaks, the steam or water break flow is much lower than in the larger PCC-4 breaks. This results in a less severe uncontrolled RCP [RCS] overcooling or overheating rate. For instance, at the hot shutdown pressure of 90 bar the highest operating pressure:

- The energy removed through a DN50 break (20 cm^2) with saturated steam flow conditions is less than 2% NP (NP = nominal power of 4500 MWth)

- The feedwater flow lost through a DN50 break (20 cm^2) with sub-cooled liquid flow conditions is in the range of 10% NF (NF= nominal flow of app. 2580 kg/s in four SGs)

When compared to the PCC-4 transients, the resulting low secondary side depressurisation rate may not be sufficient to activate the F1 signals dedicated to the fast PCC-4 transients. These are the RT and turbine trip and VIV [MSIV] closing on "high steam pressure drop MAX1", and the complete feedwater isolation in the affected SG on "high steam pressure drop MAX2". However, the lack of activation of those signals does not invalidate the bounding aspect of the PCC-4 analyses, as:

- During power operation, the core is protected by the dedicated protection signals "high core power level" and "low DNBR", independent of the steam pressure signals,

- After reactor shutdown:

    - o For the steam release, if the "high steam pressure drop" F1 signal is not activated, the VIV [MSIV] is closed automatically by the "low steam pressure MIN" F1 signal at 50 bar. Complete feedwater isolation in the affected SG is initiated by the "second low steam pressure MIN2" F1 signal at 40 bar. Above 50 bar, on the secondary side, the core remains subcritical, even without boration. The shutdown margin from the control and shutdown rods is sufficient to compensate for the reactivity insertion down to a core temperature of 260°C the saturation temperature for 47 bar. This assessment assumes the highest worth rod stuck in its upper position. Below 50 bar, in the affected SG, core criticality may occur, but the PCC-3 break cooling capability for DN < 50, i.e. < 20 cm^2 is much lower than the PCC-4 case analysed in section 2 of Sub-chapter 14.5. That case considers a 2A steam line break of area 1300 cm^2 at the SG outlet. The power excursion calculated in the PCC-4 analysis bounds the one associated with the smaller PCC-3 breaks by a significant margin.

- For the feed release, the ‘Main feedwater line break’ analysis performed in section 3 of Sub-chapter 14.5 does not claim the "steam pressure" signals, "high steam pressure drop" or "low steam pressure". The analysis only relies on the "low SG-level MIN" in the unaffected SG to initiate the F1 protection and mitigation actions. This decoupling assumption ensures that the maximum loss of feedwater is used in the PCC-4 analysis. It therefore covers any break size within the whole feedwater break size spectrum, including the smaller PCC-3 breaks.

1.2. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

2. LONG-TERM LOSS OF OFFSITE POWER (>2 HOURS)

2.1. INTRODUCTION

The loss of off-site power or loss of non-emergency AC power to the station auxiliaries is caused by either a complete loss of off-site power coincident with a turbine trip, or a loss of onsite power. The loss of power results in an immediate reactor coolant pump coastdown and loss of main feedwater flow. A reactor trip will occur on low pump speed as a result of the loss of power. Diesel generators are started and provide electric power to essential loads. The sensible and decay heat loads are removed by the steam relief and ASG [EFWS].

Two types of LOOP are identified, depending on their probability of occurrence:

- Short-term LOOP with an off-site power recovery time of two hours. This is classified as a PCC-2 event.
- Long-term LOOP with an off-site power recovery time of 24 hours. This is classified as a PCC-3 event.

The short-term LOOP is analysed in section 6 of Sub-chapter 14.3; the long-term LOOP is analysed below.

2.2. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

Information on the identification of the causes and the accident description are provided in section 6 of Sub-chapter 14.3. The description of the short-term LOOP also applies to the long-term LOOP of more than two hours.

The steady-state plant condition that is reached a few minutes after the LOOP occurs, using only F1A functions, also applies to cases lasting up to 24 hours. The main characteristics of the safe plant conditions are:

- The emergency diesel generators (EDGs) are in operation. These are started about 10 seconds after the LOOP occurs. These supply all essential systems needed in all circumstances for boration, heat removal and activity containment. The systems supported by the diesel generators are the ASG [EFWS], RRI [CCWS] and SEC [ESWS], RBS [EBS], RCV [CVCS] and RIS/RRA [SIS/RHRS].
- The reactor is tripped with the RCCAs inserted with sufficient shutdown margin at the hot standby temperature.
- The reactor coolant pumps are shutdown and the heat removal from the core takes place thanks to natural circulation in all loops.
- The secondary side SG pressure control is performed by the F1A classified VDA [MSRT] at a pressure of 95.5 bar. The main condenser is unavailable following LOOP.

- The primary coolant temperatures are consistent with the secondary pressure at about 306°C in the cold leg. The hot leg temperature is a few degrees higher, depending on the decay heat level.

- The primary system pressure is in the normal operating range around 155 bar and is controlled by the EDG powered auxiliary spray and pressuriser heaters (576 kW).

- The pressuriser water inventory is between the HZP and HFP nominal level. The actual level depends on control being performed by the EDG-powered RCV [CVCS] pumps and the letdown control.

- The SG water inventory is at its nominal level of 15.7 m and is controlled by the EDG-powered ASG [EFWS]. The ASG [EFWS] pumps are actuated automatically by the signal “SG level < MIN2” or earlier.

2.3. FURTHER EVENT SEQUENCE UP TO 24 HOURS

There are two main possibilities for long-term LOOP: remaining in a controlled state at hot shutdown, or cooling the RCP [RCS] until RIS/RRA [SIS/RHRS] connecting conditions are reached, the safe shutdown state.

2.3.1. Staying at Hot Conditions

The hot shutdown state corresponding to a secondary side pressure at the VDA [MSRT] as described above can be safely maintained for 24 hours. This is achieved using the following functions and systems that are powered by the EDG or by battery supplies:

- RCP [RCS] integrity, i.e., prevention of leaks such as leaks at RCP [RCS] pumps seals. This is provided by the seal water injection from the RCV [CVCS] charging pumps or the thermal barrier cooled by the component cooling water system (RRI [CCWS]).

- RCP [RCS] inventory and pressuriser level is controlled by the RCV [CVCS] charging and letdown.

- RCP [RCS] pressure control is performed by the pressuriser heaters (576 kW).

- The secondary side heat removal is provided via the VDA [MSRT]. These are battery powered using batteries that can be charged by the EDGs.

- The SG inventory is maintained by the ASG [EFWS]. The total water inventory in the tanks is sufficient for heat removal over the 24 hour period. The tanks can be refilled from other sources for use beyond 24 hours.

The functions mentioned above are F1A classified, with the exception of the RCV [CVCS] and the pressuriser heaters. If the RCV [CVCS] is assumed to be unavailable, the RBS [EBS] can be used for boration and inventory control. Under extreme conditions, the MHSI could be used for inventory control after a partial cooldown has been completed using the VDA [MSRT]. The sources to refill the ASG [EFWS] tanks are non classified.

Single Failure (SF) and Preventive Maintenance (PM)

Single failure and preventive maintenance assumptions do not affect the ability to maintain the plant in hot shutdown condition. All the above-mentioned safety and operational systems are designed to meet the single failure criterion, with the exception of the RCV [CVCS]. In addition, the loss of one or two safety system trains, such as the ASG [EFWS], can be compensated for by opening the normally passive headers. This allows all four loops to be used for heat removal. The limiting event for the heat removal capacity of the ASG [EFWS] is the feedwater line break described in section 3 of Sub-chapter 14.5. Thus there is additional margin in the system for this event.

2.3.2. Cooldown of the Plant to RIS/RRA [SIS/RHRS] (Safe Shutdown State)

The plant can be cooled to the RIS/RRA [SIS/RHRS] connecting conditions by means of the EDG-powered safety systems alone. The connection conditions are a RCP [RCS] pressure less than 30 bar and a hot leg temperature of 180°C. For plant cooldown, the RBS [EBS] is available for boration and to compensate for the reduction in coolant volume during the cooldown. The VDA [MSRT] are available for steam removal to the atmosphere, and the ASG [EFWS] for water supply to the SGs.

The justifications for the adequacy of the cooling capacity and the functions required to reach the safe shutdown state are provided in other sections as identified in Section 14.4.2 - Table 1.

2.4. CONCLUSION

The long-term LOOP is safely mitigated by means of the functions and systems powered by the EDGs. The RCP [RCS] and secondary side safely maintain their integrity and only steam is released to the atmosphere. Therefore, all the acceptance and safety criteria for a PCC-3 event are satisfied.

Both the controlled state and the safe shutdown state are safely reached and maintained.

2.5. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

SECTION 14.4.2 - TABLE 1

Cooldown of the Plant to RIS/RRA [SIS/RHRS] connecting conditions (Safe Shutdown)

Criteria	Reference Case	Remark/Reason
Subcriticality	Section 13 of Sub-chapter 14.3 (Uncontrolled boron dilution)	Achieved with one stuck rod
Activity release	Section 6 of this sub-chapter (SG tube rupture)	Tube rupture is the limiting PCC-3 event
Heat removal	Section 3 of Sub-chapter 14.5 (Feedwater line break)	In the reference case only one train is available for cooldown.

3. INADVERTENT OPENING OF A PRESSURISER SAFETY VALVE (STATE A)

3.1. EVENT DESCRIPTION

The inadvertent opening of a pressuriser safety valve event is defined as the spurious opening of a pressuriser safety valve (PSV) which is normally closed during plant operation.

During normal power operation, the opening or closing demand for a PSV is hydraulic and valve-specific. Thus, a single failure can impact only one PSV. An inadvertent opening of a pressuriser safety valve is similar to a small break loss of coolant accident (SB-LOCA) on the hot side of the reactor coolant system (RCP [RCS]).

The inadvertent opening of a pressuriser safety valve is a PCC-3 event and is bounded by the analyses performed for another PCC-3 event, ‘SB LOCA DN < 50’ in section 5 of this sub-chapter because:

- The steam relief capacity of a PSV of 300 te/h at 176 bar corresponds to an equivalent leak area of about 30 cm^2.
- The PCC-3 SB-LOCA break analysed in section 5 of this sub-chapter is the largest PCC-3 SB-LOCA event, with an equivalent break area of 20 cm^2, corresponding to an equivalent diameter of 50 mm.
- The PSV leak of 30 cm^2 located in the RCP [RCS] hot side is bounded, for the loss of RCP [RCS] coolant, by the SB-LOCA cold leg break of 20 cm^2. This is because of a higher injection/leak ratio and a lower leak flow rate:
    - For hot leg breaks, the minimum effective injection capacity is 100 % more than for cold leg breaks once the effect of a single failure and preventive maintenance are considered. Injection into the broken loop of the RIS [SIS] can be claimed for the PSV leak. Therefore, two MHSI are available versus one MHSI for the SB-LOCA cold leg break.
    - Leak size is 50% larger (PSV leak: 30 cm^2, SB-LOCA cold leg: 20 cm^2).
    - Even with the same injection/leak ratio, breaks located in the hot side of the RCP [RCS] are less onerous than breaks in the cold side. This is because of the higher break flow quality. The presence of steam at the break reduces the leak mass flow rate and increases the RCP [RCS] depressurisation.

Based on these statements and the SB-LOCA results of section 5 of this sub-chapter, it can be concluded without additional transient analysis, that core uncovery does not occur during the PCC-3 event ‘Inadvertent opening of a pressuriser safety valve’.

3.2. INFLUENCE OF THE PRESSURISER SAFETY VALVES NEW DESIGN

The accident was analysed assuming the SEBIM model for the Pressuriser Safety Valves (PSV) whilst the SEMPELL model valve will be implemented on the EPR pressuriser.

As the steam relief capacity for a SEMPELL pressuriser safety valve is the same as that for a SEBIM pressuriser safety valve (300 te/h at 176 bar), the results and thus conclusions resulting from the analysis of the inadvertent opening of a SEBIM type pressuriser safety valve are the same with the PSV SEMPELL type valve.

3.3. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

4. INADVERTENT OPENING OF AN SG RELIEF TRAIN OR A SAFETY VALVE (STATE A)

The most severe core conditions following an accidental depressurisation of the main steam system due to a spurious or inadvertent opening of a main steam valve result from:

- The spurious opening of a main steam relief train (VDA [MSRT]).
- The spurious opening of a main steam safety valve (MSSV)

These are both classified as PCC-3 events.

During power operation, core protection is provided by a reactor trip initiated by the Reactor Protection System (RPR [PS]), which includes the DNBR protection channel. The automatic actuation of the reactor trip prevents any core damage prior to completion of the plant shutdown.

After plant shutdown, the core overcooling transient continues for as long as the main steam system depressurisation continues. There is the potential for a return to core criticality. The severity of the event depends on this potential return to core criticality after the reactor trip.

The classification of each event and the corresponding severity in terms of the uncontrolled RCP [RCS] cooldown is discussed below.

4.1. FAILURE OF ONE MAIN STEAM RELIEF CONTROL VALVE

The maximum discharge capacity of each VDA [MSRT] is 50% of the full load steam generation of the associated SG [Ref-1]. This corresponds to approximately 14% of the plant full load steam flow. Each VDA [MSRT] consists of an MSRIV, which is normally closed and an MSRCV, which is normally open, in series. The MSRCV is normally open and the relief train is kept closed by the normally closed MSRIV. Therefore, the failure of the MSRCV will not affect normal plant operation.

4.2. SPURIOUS OPENING OF A MAIN STEAM RELIEF TRAIN

The spurious opening of a VDA [MSRT] is caused by the spurious opening of the corresponding main steam relief isolation valve (MSRIV). This occurs via the spurious opening of the two solenoid-driven pilot valves in series. The spurious opening of a VDA [MSRT] is consequently classified as a PCC-3 event.

On actuation of the "SG Pressure Drop > MAX1" signal, all the VIV [MSIV] are automatically closed. Once the main steam lines isolate, only the affected steam generator continues to depressurise. When main steam pressure reaches the MIN3 setpoint, the VDA [MSRT] is automatically isolated by closing both the MSRCV and MSRIV.

Therefore, the limiting single failure for this scenario is a failure of the associated MSRCV to close. This will result in a continued un-isolatable steam discharge from the affected SG.

The potential cooldown resulting from the Inadvertent Opening of a MSRIV event is bounded by that resulting from the Main Steam Line Break event. The Excessive Increase in Secondary Steam Flow event discussed in section 3 of Sub-chapter 14.3 and the Main Steam Line Break event discussed in section 2 of Sub-chapter 14.5 represent a far greater increase in steam load. Consequently they result in a bounding rate of reactivity increase relative to this event.

Therefore, the spurious opening of a VDA [MSRT] event is bounded by the consequences of other increase in steam flow events.

4.3. SPURIOUS OPENING OF ONE MAIN STEAM SAFETY VALVE (MSSV)

The main steam safety valve (MSSV) is a spring-loaded valve. The spurious opening of a MSSV is therefore classified as a PCC-3 event.

The capacity of each MSSV is 25% of the full load steam generation of the assigned SG [Ref-1]. This corresponds to roughly 6% of the plant full load steam flow. The increase in steam load following a single MSSV failing open is small and is bounded by the consequences of the Main Steam Line Break (MSLB) in State A. This is classified as a PCC-4 event and is analysed in section 2 of Sub-chapter 14.5, “Main Steam Line Break (States A, B).

The MSLB analysis shows that fuel damage does not occur during the subsequent uncontrolled core cooling transient. It is confirmed that no core DNB occurs in this transient.

Therefore, the Inadvertent Opening of a MSSV event is bounded by the consequences of other increases in steam flow.

NOTE: *Failure to close one SG safety valve after a demand to open*

In the PCC analyses, the SG safety valve is only actuated after failure of the VDA [MSRT] to open, in cases where the main steam header is isolated. Consequently, a stuck-open SG safety valve need not to be included in the PCC analyses, since it would result from a double failure. This would be the failure of one VDA [MSRT] isolation valve to open, followed by the failure of one SG safety valve to close after us).

4.4. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

5. SMALL BREAK LOCA (DN < 50) INCLUDING A BREAK IN THE RBS [EBS] INJECTION LINE (STATES A AND B)

A small break is defined as a break of equivalent diameter smaller than or equal to 50 mm (~20 cm^2).

5.1. SMALL BREAK LOCA IN STATE A

5.1.1. Identification of Causes and Accident Description

5.1.1.1. General Comments

The following small breaks are not studied in this sub-chapter:

- Steam generator tube ruptures for which the physical phenomena are different. These transients are described in section 6 of this sub-chapter for 'Steam Generator Tube Rupture (one tube)' and section 10 of Sub-chapter 14.5 for 'Steam Generator Tube Rupture (two tubes in 1 SG)'.

- Leaks in the RCP [RCS] that are compensated for by the RCV [CVCS].

A loss of coolant accident causes:

- Loss of reactor coolant inventory and a possible core heat-up

- Containment loads due to overpressure from a mass and energy release. This is discussed in section 1 of Sub-chapter 6.2

- Mechanical loads on RCP [RCS] components and associated supports and structures as discussed in Sub-chapter 3.4

- Mechanical loads on RPV internals as discussed in Sub-chapter 3.4.

The radiological consequences of a LOCA are analysed in the section related to 'Small Break LOCA' in Sub-chapter 14.6.

The risk of inherent boron dilution following a LOCA may lead to the formation and accumulation of a plug of clear water (low boron concentration) in a Steam Generator, after which this then starts to flow towards the reactor vessel. A complete description of the calculation procedure used to assess the inherent boron dilution following LOCA is presented in [Ref-1].

A LOCA is classified as a PCC-3 or a PCC-4 event according to the break size and the plant initial state. This section only deals with the small break LOCA in state A, classified as a PCC-3 event as discussed in Sub-chapter 14.0.

5.1.1.2. Typical sequence of events

5.1.1.2.1. *From the initiating event to the controlled state*

The break results in a loss of reactor coolant inventory that cannot be compensated for by the RCV [CVCS]. The loss of primary coolant causes a decrease in the primary system pressure and the pressuriser level.

A reactor trip (RT) occurs following a “low pressuriser pressure < MIN2” signal. The RT signal automatically trips the turbine and closes the ARE [MFWS] high-load lines.

As the secondary side pressure increases, the GCT [MSB] valves open allowing steam to dump to the condenser. If the condenser is unavailable for steam dump, for example if Loss of Off-site Power (LOOP) occurs coincident with the turbine trip, the VDA [MSRT] open allowing the steam to be dumped to the atmosphere.

The SGs are fed by the ARE [MFWS] through the low-load lines. If the ARE [MFWS] is unavailable, the start-up and shutdown pump AAD [SSS] starts and feeds the SG through the low-load lines. If the AAD [SSS] is unavailable, for example if LOOP occurs coincident with the turbine trip, the ASG [EFWS] is actuated following a “low SG level < MIN2” signal.

A Safety Injection (SI) signal is actuated following a “pressuriser pressure < MIN3” signal. The SI signal automatically starts the MHSI and LHSI pumps and initiates a partial cooldown of the secondary system. The partial cooldown cools the primary system and reduces the RCP [RCS] pressure.

During the partial cooldown, the RCP [RCS] pressure decreases sufficiently to allow MHSI injection into the cold legs. The partial cooldown is performed by all SGs using the steam dump to condenser or to the atmosphere for cases with LOOP. This is performed by automatically decreasing the respective relief valve setpoints for a constant cooling rate of -100°C/h. This decrease continues to a fixed pressure value which is low enough to allow the necessary MHSI injection but high enough to prevent core re-criticality.

For these break sizes, the volume of the flow through the break is less than the volume being added by the MHSI and steam production in the core due to the decay heat. Depressurisation of the RCP [RCS] therefore stops at the end of the partial cooldown. This continues until the energy removed at the break becomes sufficient to remove the decay heat. The RCP [RCS] inventory continues to decrease whilst MHSI injection is insufficient to match the break flow. During this phase the break flow is sub-cooled until it eventually reaches saturation conditions.

The break flow decreases as the void fraction in the cold legs increases. Eventually, the break flow changes to single phase steam. The RCP [RCS] inventory depletion stops once sufficient MHSI flow is available to compensate for the break flow.

Prior to the time at which the MHSI flow matches the break flow, the core may become uncovered. In these circumstances, the fuel clad temperature increases above the saturation level in the uncovered part of the core. The cladding temperature increase is directly proportional to the depth and duration of the core uncovery. Therefore, dedicated criteria discussed in Sub-chapter 14.1, must be met to prevent any unacceptable core damage and to limit the radiological consequences to the environment.

The controlled state is reached when the following conditions are met:

- The core is subcritical

- The core power is removed
- The reactor coolant inventory is stable or increasing.

5.1.1.2.2. *From Controlled State to Safe Shutdown State*

During the controlled state, the core cooling is provided by the RCP [RCS] water supply from the RIS [SIS] and the RCP [RCS] heat removal via the break, and the SG if needed. This state can not last for a long period of time for the following reasons:

- The ASG [EFWS] tanks empty
- The containment pressure and temperature increase.

Thus a transfer to the safe shutdown state is necessary. Following a SB-LOCA, the safe shutdown state is reached when the following conditions are met:

- The core is subcritical even after xenon depletion
- The break flow is matched by RIS [SIS] flow
- The decay heat is removed by the cooling chain LHSI/RRI/SEC [LHSI/CCWS/ESWS] with the LHSI used in RHR mode and partially by the break flow
- The activity release is within the PCC-3 limits.

A transfer to the RIS/RRA [SIS/RHRS] connection conditions is required to reach the safe shutdown state. The sequence of actions to be performed is as follows:

- Boration phase: The RCP [RCS] must be borated to keep the core subcritical throughout the transient during the transition to the safe shutdown state.
  - For smaller break sizes[1] MHSI boration is not sufficient, due to a low injection flow rate. For these breaks the RCP [RCS] boration is performed during the RCP [RCS] cooldown by the RCV [CVCS], if available, via the charging line. If it is not available it is performed by the F1A classified Extra Boration System (RBS [EBS]), with injection of boric acid of 7000 ppm enriched boron. The RCP [RCS] cooldown rate is either 25°C/h if one RBS [EBS] pump is in operation, or 50°C/h if two RBS [EBS] pumps are in operation. The RBS [EBS] is designed such that the injection of boron fully compensates for the reactivity insertion resulting from the RCP [RCS] cooldown rates.
  - For larger break sizes[2], MHSI boration is sufficient.

[1] Break sizes typically lower than 1 cm^2 (∅ 10 mm), with RCP [RCS] highly sub-cooled.
[2] Break sizes typically higher than 1 cm^2 (∅ 10 mm).

- RCP [RCS] cooldown: The RCP [RCS] cooldown is manually initiated via the secondary side. It is performed by decreasing the GCT [MSB] setpoint, if available, or by decreasing the VDA [MSRT] setpoints. This manual cooldown is F1B classified. The minimum cooldown rate claimed in the EPR design is 25°C/h, provided it is not limited by the VDA [MSRT] or GCT [MSB] capacity. As detailed previously, the cooldown rate depends on the number of RBS [EBS] trains in operation.

- RCP [RCS] depressurisation: The depressurisation is performed by shutting down the MHSI pumps once the RCP [RCS] hot leg temperature falls below 200°C. This temperature is sufficiently low for LHSI injection to provide the required injection. However, this action is only undertaken if the LHSI pumps are already in operation and the RCP [RCS] water inventory is sufficient. The Reactor Pressure Vessel Level (RPVL) and ΔT_{sat}[3] measurements, both F1B classified, provide the required information on RCP [RCS] water inventory, as required by the emergency operating procedures.

Connection of the LHSI/RHR trains is possible when the following RCP [RCS] conditions are met:

- RCP [RCS] hot leg pressure below 30 bar

- RCP [RCS] hot leg temperature below 180°C

- ΔT_{sat} and RPVL consistent with LHSI/RHR suction from the hot leg.

The safe shutdown state is then maintained by controlling the RCP [RCS] water inventory with one LHSI operating in SI mode, and by controlling the RCP [RCS] temperature with the other LHSI operating in RHR mode. The LHSI injection can be complemented by the MHSI if required to maintain the RCP [RCS] water inventory. It operates in its appropriate configuration with the large mini-flow line open before LHSI/RHR injects. This limits the MHSI delivery pressure to around 40 bar. IRWST cooling is performed by the LHSI mini-flow. Switch-over of the LHSI injection from the cold legs to the hot legs is not necessary.

5.1.2. Safety Criteria

The safety criteria to be met are the dose equivalent limits for PCC-3 events described in Sub-chapter 14.6.

For LOCA analysis, the following criteria shall be met as defined in Sub-chapter 14.0:

- The peak cladding temperature shall remain below 1200°C.

- The maximum cladding oxidation shall remain below 17% of the total cladding thickness.

- The maximum hydrogen generation shall remain below 1% of the amount that would be generated if all the active part of the cladding were to react.

- The core geometry shall remain coolable. Calculated changes in core geometry shall be such that the core remains capable of being cooled.

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[3] $\Delta T_{sat} = T_{sat}$ (Hot Leg pressure) - T_{co}, with T_{co} = core outlet temperature.

- Long-term cooling shall be maintained. The calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed.

A demonstration must show that the following two safe states are reached with the application of the safety analysis rules defined in ‘PCC Accident Analysis Rules’ discussed in Sub-chapter 14.0:

- The controlled state, using only F1A means.
- The safe shutdown state, using only F1A and F1B means.

5.1.3. Methods and Assumptions

5.1.3.1. Method

The CATHARE computer code described in Appendix 14A is used.

The CATHARE thermal-hydraulic code has been developed jointly by CEA, EDF, and AREVA NP. A rigorous assessment methodology has been implemented. The code assessment matrix relies on numerous separate effect tests (SET) and integral effect tests (IET):

- The SET and component test matrix contains about 300 tests selected from various experiments.
- The IET matrix contains a selection of 27 tests performed on the BETHSY, LOBI, LOFT, LSTF, PACTEL, PMK, and SPES facilities.

This realistic deterministic methodology is characterised by two main features:

- Key code models for SB- and IB-LOCAs: The dominant phenomena are realistic, though conservatively oriented, and bound the experimental results without excessive conservatism.
- Initial and boundary conditions are conservatively selected.

The basic steps of the realistic deterministic methodology consist of:

- A phenomenological analysis of the LOCA scenario and the identification of key phenomena.
- A judgement on the adequacy of the code to calculate the LOCA scenario, based on physical understanding, the experimental database, and code assessment examination, supplemented where necessary by sensitivity studies.
- An evaluation of calculation uncertainty with emphasis on dominant parameters, through sensitivity studies, or the use of the code assessment matrix to demonstrate the code conservatism for the key SB-LOCA phenomena.
- Introduction, where necessary, of conservative biases as close as possible to the uncertainty on the key phenomena. The biases are introduced either in a code model, in a nodalisation scheme, or in a boundary condition.
- Use of conservative assumptions for initial and boundary conditions.

The realistic, deterministic methodology provides a conservative result that can be directly compared to the acceptance criteria.

The CATHARE code provides a detailed representation of the primary and secondary systems. For the PCC LOCA analysis, the broken loop is explicitly modelled; two intact loops are either, lumped into a single loop, weighted accordingly, or modelled separately. The fourth loop is modelled with the pressuriser connected to the hot leg.

The break is conservatively located at the lower part of the cold leg.

The CATHARE point kinetic model is not utilised. The residual fission power (term A) is provided as an input to the CATHARE code. The term A is calculated in a decoupled conservative RT simulation described in Section 14.4.5 - Table 1.

The transient analyses are performed using the conservative PCC analysis rules defined in ‘PCC Accident Analysis Rules’ in Sub-chapter 14.0.

A first CATHARE calculation is performed which models the entire RCP [RCS] and the relevant boundary conditions. In this calculation, the ‘system calculation’, only the average core assembly is modelled.

A second CATHARE calculation is then performed, the ‘hot assembly calculation’. This only models the hot assembly and the hot rod within that assembly. It utilises the core boundary conditions from the ‘system calculation’.

5.1.3.2. Main Assumptions

5.1.3.2.1. *Accident Definition*

The break size analysed is 20 cm^2, an equivalent diameter of 50 mm.

The analysis assesses the emergency core cooling capability. The purpose is to demonstrate that the relevant acceptance criteria are met.

5.1.3.2.2. *Protection and mitigation actions*

In a LOCA event, automatic protection actions using F1A classified systems, trip the reactor, remove the residual heat, and match the inventory loss to allow the plant to reach the controlled state.

The rules defined for safety analyses in the section ‘PCC Accident Analysis Rules’ of Sub-chapter 14.0 require the controlled state to be reached using only F1A classified systems. The safe shutdown state is reached using F1A and F1B classified systems only.

The F1A I&C signals considered in this analysis are:

- Reactor Trip on “pressuriser pressure < MIN2”
- Turbine Trip on RT signal
- Safety Injection on “pressuriser pressure < MIN3”

- Reactor coolant pumps trip on “Reactor coolant pumps pressure drop < MIN1”. This is not modelled in the calculation because of the assumed LOOP
- VDA [MSRT] opening and pressure control on “SG pressure > MAX1”
- Partial cooldown on SI signal
- ASG [EFWS] injection and APG [SGBS] isolation on “SG level < MIN2”

5.1.3.2.3. *Operator actions*

No operator action is claimed until 30 minutes after the reactor trip. After 30 minutes, operator actions are required to reach the safe shutdown state C.

The operator diagnoses the RCP [RCS] state based on information provided by two measurements, Reactor Pressure Vessel Level (RPVL) and ΔT_{sat}. This process is controlled by the emergency operating procedures discussed in section 2 of Sub-chapter 18.3. The operator then has to perform the following actions:

- Criticality Control

  The sequence of operator actions to perform RCP [RCS] boration depends on the RCP [RCS] state revealed by the RPVL and ΔT_{sat} measurements. In a standard LOCA transient, the RCP [RCS] boration will be performed in parallel with the RCP [RCS] cooldown. It will be performed by the RCV [CVCS], which is not F1 classified and/or the RBS [EBS] which is F1A classified. Both systems are actuated by the operator. If a high flux is detected by the intermediate range channels, F1B classified, or the MHSI pumps are unavailable, the operator is instructed to immediately actuate the RBS [EBS], and the RCV [CVCS] if available.

- Water Inventory Control

  The MHSI pump(s) can be stopped provided that:

  - The core outlet temperature is below 200°C
  - The LHSI/RHR required number of pumps are available and in operation
  - The core outlet is not superheated
  - The RPVL measurement indicates that the core remains covered.

The accumulators are isolated before the MHSI cut-off criterion is reached, at a time dependent on the RCP [RCS] sub-cooling.

If more severe depletion of the RCP [RCS] coolant inventory occurs, further operator actions are performed. These actions are described in ‘Intermediate and Large Break LOCA (up to the Surge Line Break in States A and B)’, section 6 of Sub-chapter 14.5.

- RCP [RCS] temperature and pressure control

In the RCP [RCS] state described above, the operator initiates the RCP [RCS] cooldown either at -50°C/h or at -25°C/h.

The list of F1B operator actions, with indication of the main F1B information required, are:

- Manual MHSI cut-off
  - o ΔT_{sat} , core outlet temperature (T_{co}), RPV level (RPVL)
- Manual accumulator isolation
  - o ΔT_{sat} , core outlet temperature (Tco), RPV level (RPVL)
- Manual RBS [EBS] actuation
  - o High neutron flux from ex-core detectors, low MHSI flow rate or indirect information as ΔT_{sat}, RCP [RCS] pressure
- Manual VDA [MSRT] opening/closing
  - o SG pressure, RCP [RCS] temperature
- Manual ASG [EFWS] control if no F1B automatic control of SG level
  - o SG level, ASG [EFWS] flow rate
- Manual opening of ASG [EFWS] headers at pump suction/discharge
  - o ASG [EFWS] tank level, ASG [EFWS] flow rate
- Manual VIV [MSIV] by-pass opening
  - o SG pressure
- Manual connection of LHSI/RHR trains
  - o RCP [RCS] hot leg pressure, RCP [RCS] hot leg temperature, ΔT_{sat}, RPVL

5.1.4. Definition of cases analysed.

The cases studied in sub-section 5.1 of this sub-chapter correspond to a 20 cm^2 break equivalent diameter 50 mm, the largest SB-LOCA size. The break is assumed to be located in a cold leg pump discharge pipe. The cold leg location produces the worst consequences for the core when compared to any other location in the RCP [RCS]. This includes breaks in the hot leg or in the pressuriser, and breaks in the pressuriser steam space. These cases must consider:

- The limiting single failure
- The most onerous preventive maintenance activity
- A LOOP coincident with the event.

Two CATHARE calculations are performed:

- The first CATHARE calculation ends when the controlled state is reached. Its objective is to demonstrate that the acceptance criteria related to the controlled state are met. Therefore, excessive fuel clad heat-up must be prevented, using only of F1A systems as described in sub-section 5.1.5 of this sub-chapter.

- The second CATHARE calculation ends when the safe shutdown state is reached. Its objective is to demonstrate that the acceptance criteria for the safe shutdown state of long-term core cooling are met. These must be met with the use of only F1A and F1B systems. In addition it must be shown that the transfer to the safe shutdown state can be achieved. The CATHARE code is coupled with the CONPATE containment code for this purpose. This methodology is described in section 1.6 of Sub-chapter 14.5.

5.1.5. Description of Cases Analysed, from Initiating Event to the Controlled State

The safety analysis is performed using conservative assumptions.

5.1.5.1. Choice of Single Failure and Preventive Maintenance

The worst active single failure (SF) is the loss of one Emergency Diesel Generator (EDG) following LOOP. As a consequence of the diesel loss, one RIS [SIS] train of one MHSI pump and one LHSI/RHR pump and one ASG [EFWS] pump are unavailable.

The preventive maintenance (PM) of one diesel is the worst assumption as one RIS [SIS] train of one MHSI pump and one LHSI/RHR pump and one ASG [EFWS] pump are consequently unavailable.

The assumption of LOOP is limiting for the SB-LOCA analysis when the SF and PM assumptions are applied.

5.1.5.2. Initial State

The initial state conditions, given in Section 14.4.5 - Table 2 are chosen to maximise the time required to actuate the RT and SI signal. This is the most conservative assumption for SB-LOCA mitigation.

The initial water temperature within the RPV dome is maximised. It is assumed to operate at the hot leg temperature. This is the most conservative assumption for the SB-LOCA transient assessing core cooling.

The axial power shape for the average rod in the average assembly is identical to the one used in the PCC-4 analyses and shown in Section 14.5.6 - Figure 1. This power shape is a top skewed power shape having the following characteristics:

- An enthalpy rise factor of 1.00

- A peaking factor of 1.57

- An axial offset of 18% (15% + 3% uncertainty)

The axial power shape chosen for the hot rod in the hot assembly has no effect on the cladding heat-up as the core remains covered throughout the transient.

A top-skewed axial power shape is used in the analysis because it represents a distribution with power concentrated in the upper region of the core. This distribution is limiting for the analysis of the SB-LOCA because it minimises coolant level swell, while maximising vapour superheating and fuel rod heat generation at the uncovered elevation. Should the core be uncovered, the cladding in the upper part of the core would heat up. The higher the linear power, the higher the clad temperature excursion. The lower part of the core remains close to the saturation temperature as it remains covered.

The initial fuel temperature is to the same as the one used in the PCC-4 analyses shown in Section 14.5.6 - Figure 3. This temperature is averaged over a section of fuel and is given as a boundary condition limit. This initial temperature has a negligible impact on the maximum clad temperature reached, because the core heats up some time after RT in a typical SB-LOCA sequence The initial stored energy is removed in the period following reactor trip and before the heat-up occurs.

5.1.5.3. Specific Assumptions

- Neutronic data and decay heat

The core power is set at 102% of full power until reactor trip. After RT, the residual fission power (term A) is defined by a conservative decoupling curve given in Section 14.4.5 – Table 1. The decay heat is the maximum curve described in the relevant section of Sub-chapter 14.1 and represents term B+C ‘MAX 2σ’.

- Assumptions related to non-F1 systems

The non-safety-grade systems, control systems, are not claimed when they perform a mitigating function.

The pressuriser pressure control system using the pressuriser heaters is modelled as it delays the RT signal. A total heating power of 2592 kW is assumed until the pressuriser is totally empty.

The flow to the turbine is assumed to remain constant until the turbine trip.

The ARE [MFW] flow is assumed to remain constant until reactor trip.

The GCT [MSB] and AAD [SSS] are not operating.

The RCV [CVCS] is not modelled. The letdown line is normally isolated on low pressuriser level, not F1A, and on the SI signal, F1A, while the RCV [CVCS] charging line is not isolated on the SI signal. This modelling is conservative for the RCP [RCS] water inventory balance.

- Assumptions related to F1 systems

Reactor Trip (F1A): The RT signal is actuated following a “pressuriser pressure < MIN2” signal. At a setpoint of 135 bar - 3 bar. The uncertainty on the setpoint includes the effects of the degraded containment conditions [Ref-1].

The specific conservative delays that are assumed are listed below:

  - RT signal 0.9 seconds after "pressuriser pressure < MIN2"
  - Beginning of rod insertion 0.3 seconds after RT signal

- 5 seconds for complete rod insertion assuming seismic condition)
- Total main feed water isolation 0.3 seconds after RT signal
- Turbine trip 0.3 seconds after RT signal.

Safety Injection (F1A): The SI signal is actuated following a “pressuriser pressure < MIN3” signal with a setpoint of 115 bar – 3 bar. The uncertainty on the setpoint includes the effect of the degraded containment conditions. This is included to delay the start of the RIS [SIS] pumps and the beginning of the partial cooldown [Ref-1].

The specific conservative delays that are assumed are listed below:

- SI signal 0.9 seconds after the "pressuriser pressure < MIN3" signal
- Partial cooldown signal simultaneous with SI signal. The VDA [MSRT] setpoint is decreased at a rate equivalent to -100°C/h
- 2 seconds delay for VDA [MSRT] opening
- 40 seconds delay for RIS [SIS] pump start-up. This delay includes the diesel unloading and reloading sequence and the RIS [SIS] pump start time
- Minimum characteristics for RIS [SIS] pumps as defined in the relevant section of Sub-chapter 14.1
- 50°C for initial IRWST temperature and injection flow temperature. This temperature is held constant for the transient calculation. This is justified by the short time duration of the transient during which the temperature increase of the IRWST is low.

The following assumptions relating to the accumulators are defined to minimise the accumulator injection:

- 47 m^3 total volume
- 35 m^3 water volume
- 45 bar initial pressure
- 2500 m^{-4} discharge line resistance.

The water temperature in each accumulator is assumed to be 50°C which corresponds to the maximum temperature used in the analysis.

The RIS [SIS] train injecting to the broken loop is assumed to spill directly to the containment. It is assumed to make no contribution to the RCP [RCS] injection[4]. The RIS [SIS] components that are available to perform the cold leg injection taking into account the assumed single failure and preventive maintenance are:

- One MHSI pump

[4] As no core heat-up is expected during the SB-LOCA transient, this assumption has no major impact on the acceptance criteria (see section 5.1.2 of this sub-chapter).

- Three accumulators
- One LHSI pump.

Based on the reactor modelling in sub-section 5.1.3.1 of this sub-chapter, the following RIS [SIS] modelling is adopted:

- One RIS [SIS] train injects into a single intact loop
- Three accumulators inject into the 3 intact loops.

VIV [MSIV] (F1A): There is no main steam line isolation.

VDA [MSRT] (F1A): VDA [MSRT] setpoints are the nominal values plus a 1.5 bar uncertainty. There is no impact from degraded containment conditions as the pressure sensors are located outside of containment. This maximises the RCP [RCS] pressure. The setpoints are as follows:

- 95.5 + 1.5 bar before the start of the partial cooldown
- 60.0 + 1.5 bar after the end of the partial cooldown.

All VDA [MSRT] are available (one per SG).

ASG [EFWS] (F1A): ASG [EFWS] is actuated train-by-train following a “very low SG level < MIN2” signal. The associated setpoint is: 40% - 5% uncertainty on the wide range channel.

The uncertainty on the setpoint includes the effect of the degraded containment conditions. This delays the start-up of the ASG [EFWS] pumps.

Two ASG [EFWS] trains are available. They are modelled injecting into the SG associated with the loop with the pressuriser, loop 2 in CATHARE, and the broken loop, loop R in CATHARE.

The specific conservative delays that are assumed are as follows:

- ASG [EFWS] actuation signal 1.5 seconds after the "SG level < MIN2" signal
- 50 seconds delay for the start-up of the ASG [EFWS] pump. This delay includes the diesel unloading and reloading sequence and the ASG [EFWS] pump start time.
- Minimum characteristic for the ASG [EFWS] pump. A constant injection flow rate of 90 te/h is assumed
- 50°C for the injection flow temperature.

Reactor coolant pump trip (F1A). In the EPR design, an automatic reactor coolant pump trip is provided for the LOCA event. This is based on a "ΔP over reactor coolant pump" measurement with a setpoint of 80% $\pm$ 5% of nominal ΔP as defined in Sub-chapter 14.1 - Table 9.

The analysis assumes the reactor coolant pumps are tripped when the reactor is tripped. This is the most conservative reactor coolant pumps trip time between the reactor trip and the reactor coolant pumps trip signal time. For the smaller break areas below 20 cm^2, the early reactor coolant pump trip contributes to an additional loss of RCP [RCS] water inventory. It keeps the primary pressure high at the beginning of the transient while the RCP [RCS] is still in the liquid phase.

In addition, a reactor coolant pump coast down at RT/TT is consistent with the LOOP assumption.

- Other assumptions

The Loss Of Off-site Power is included in the sequence and occurs at the same time as the turbine trip.

5.1.5.4. Results

The sequence of events is given in Section 14.4.5 - Table 3 [Ref-1].

The most representative parameters are presented in the following figures:

- Section 14.4.5 - Figure 2: RCP [RCS] and secondary side water inventories

  RCP [RCS] and secondary side pressures

- Section 14.4.5 - Figure 3: Total and steam break flow rates

  Total break and RIS [SIS] flow rates

- Section 14.4.5 - Figure 4: Hot spot clad temperature

  Core swelled level

The SB-LOCA analysis to the controlled state has not been performed for the PCSR. The capability to reach the controlled state is demonstrated for the EPR_{4500} by satisfying the acceptance criteria with a large margin for the EPR_{4250} and by comparison of pertinent features between the EPR_{4500} and the EPR_{4250}.

SB-LOCA Analysis for EPR_{4250}

The RCP [RCS] water inventory stops decreasing, with a minimum water content of approximately 83 te at around 2375 seconds. The decay heat is fully removed, partly by the break, mostly by the SG via the ASG [EFWS] and the VDA [MSRT]. The core is subcritical with no credit taken for the RCP [RCS] boration, see sub-section 5.1.6.3 of this sub-chapter. The controlled state is reached.

The following conclusions can be drawn for the acceptance criteria:

- There is no core uncovery and thus no core heat-up; the peak clad temperature remains below the acceptance criterion limit of 1200°C

- There is no clad oxidation

- There is no clad rupture

- The integrity of the core geometry is maintained
- The long-term cooling is addressed in sub-section 5.1.6 of this sub-chapter.

In conclusion, the F1A systems the RIS [SIS], in conjunction with VDA [MSRT] and ASG [EFWS], are sufficient to reach the controlled state.

Extrapolation of the Results from the EPR_{4250} to the EPR_{4500}

The two EPRs share the same systems for RCP [RCS] injection and cooling:

- The flow capacity of the MHSI pumps is identical for the two power levels
- The capacity of the ASG [EFW] tank is unchanged for the higher power
- The capability of ASG [EFW] and VDA [MSRT] is identical for the two power levels
- The capacity of the IRWST is identical for the two power levels
- The SG capability is the same for the two power levels with an identical partial cooldown.

The flow via the break is determined by the primary pressure and the break quality. For a break of 20 cm^2, the SG are required to remove the decay heat and the stored heat. Because the initial RCP [RCS] power is slightly higher, about 6%, the RCP [RCS] will depressurise more slowly. Comparing the EPR_{4250} results and the BDR-99 results at 4900 MW in Appendix 14B, the pressuriser pressure of 112 bars is reached at 296 seconds for the EPR_{4250} versus 300 seconds for the EPR_{4900}. This shows that the difference in power has a minimal impact on the initial depressurisation. After this phase, the inventory at the two power levels will change insignificantly. The primary pressure will remain slightly below the secondary pressure during and after the partial cooldown. As the pressure for the partial cooldown is identical for both cases, the break flow will also be similar.

The break flow is mostly liquid until 2400 seconds, and the quality at the break depends mostly on the initial RCP [RCS] mass inventory. This phase of the transient is almost identical in the EPR_{4250} and the EPR_{4500}.

After 2400 seconds, the flow at the break is mostly vapour. The steam flow from the core is larger for the EPR_{4500} due to the higher power. The quantity of liquid reaching the break is also slightly higher. The primary pressure stays at 61.5 bar, the setpoint for the end of the partial cooldown. The SI flow at this pressure is identical for the two EPR. The SI flow does not match the break flow for the EPR_{4250}, and falls further behind for the EPR_{4500} at 2400 seconds. However, the results for the EPR_{4900} show that the primary pressure is subsequently decoupled from the SG pressure. Once this occurs, the SI flow equals and exceeds the break flow, the RCP [RCS] starts to refill, and the risk of core uncovery is eliminated. The SI will exceed the break flow earlier for the EPR_{4500} than for the EPR_{4900}.

5.1.6. Description of cases studied from the controlled state to the safe shutdown state

The safety analysis is performed assuming the same level of conservatism as used for the transition to the controlled state discussed in sub-section 5.1.5 of this sub-chapter.

In particular, the maximum decay heat curve and the minimum capacity of F1 systems considered in sub-section 5.1.5 of this sub-chapter are also assumed in sub-section 5.1.6 of this sub-chapter.

5.1.6.1. Choice of Single Failure and Preventive Maintenance

The worst single failure (SF) is the loss of one EDG following the LOOP assumed to occur at the time of reactor trip:

- As a consequence, one RIS [SIS] train, of one MHSI pump and one LHSI pump, and one ASG [EFWS] pump are unavailable.
- The remote control of the VDA [MSRT] assigned to this division fails two hours after the LOOP, because the battery is discharged. This assumes that the power supply from the neighbouring division via the dedicated DC cross-connection is unavailable because of preventive maintenance on the associated diesel. The battery discharge does not affect the availability of the VDA [MSRT], as it can be controlled locally. Thus the consequence of the single failure and preventive maintenance is limited to the loss of two MHSI and LHSI and ASG [EFWS] pumps.

The preventive maintenance (PM) on one emergency diesel is the most conservative assumption as:

- One RIS [SIS] train, of one MHSI pump and one LHSI pump, and one ASG [EFWS] pump are unavailable.
- The VDA [MSRT] assigned to this division is lost two hours after the LOOP because of the battery discharge. This assumes the power supply from the neighbouring division via the dedicated DC cross-connection is unavailable because of a single failure on the associated diesel.

The assumption of LOOP is conservative for the SB-LOCA analysis when the SF and PM principles are applied.

5.1.6.2. Specific Assumptions

- RCP [RCS] cooldown phase

The controlled state, corresponding to the hot shutdown state at the end of the partial cooldown, is maintained in the CATHARE calculation until two hours after reactor trip. The RCP [RCS] and SG pressures are held in the range of 60 bar without a requirement for operator action.

The four reactor coolant pumps were tripped at RT.

At RT plus two hours, the RPVL is stabilised between the Top of the Hot Leg and the Bottom of the Hot Leg. The ΔT_{sat} indicates saturated conditions at the core outlet.

At that time in the CATHARE calculation, the operator starts the RCP [RCS] cooldown to the safe shutdown state. This is undertaken using three out of four VDA [MSRT]. The unavailability of one VDA [MSRT] is assumed to cover the single failure of a VDA [MSRT]. The failure of one valve to open is combined in the calculation with the single failure of one diesel. This combination of the SF of one VDA [MSRT] with the SF of one EDG is used only to minimise the number of calculations.

In the CATHARE accident analysis, the minimum RCP [RCS] cooldown rate of -25°C/h is assumed to maximise the consumption of ASG [EFWS] water.

- Assumptions concerning the secondary side

The application of the single failure criterion and preventive maintenance results in the availability of only two ASG [EFWS] delivering to two SG. The flow rate per pump is that used in the analysis to the controlled state in sub-section 5.1.5 of this sub-chapter. The SG level is controlled to prevent the SG from overfilling.

The set point of the available VDA [MSRT] is decreased to 5 bar, the saturation pressure for 150°C at a rate consistent with the RCP [RCS] cooldown phase described above.

The MSH is conservatively isolated at the beginning of the RCP [RCS] cooldown.

- Assumptions concerning the RIS [SIS]

The MHSI and LHSI water injection temperatures are maximised, including the IRWST temperature increase arising from the degraded containment conditions caused by the break mass and energy releases. This temperature is obtained from a calculation with coupled CATHARE-CONPATE codes as described in Appendix 14A.

The IRWST temperature variation is calculated conservatively:

- The initial IRWST temperature is at its maximum value of 50°C
- The IRWST water volume is at its minimum value of 1450 m^3
- The minimum capacity of LHSI/RRI [LHSI/CCWS] heat exchanger is used, resulting in a minimum LHSI heat removal capability.

The MHSI operating in the analysis is conservatively assumed to be stopped as soon as the core outlet temperature falls below 200°C. The LHSI continues to deliver to the RCP [RCS] until the RIS/RRA [SIS/RHRS] connection conditions are reached.

The accumulators are conservatively assumed to be isolated at the beginning of the RCP [RCS] cooldown phase.

- Assumptions about the LHSI/RHR connection conditions

The RIS/RRA [SIS/RHRS] connection conditions are reached when the following conditions are met:

- Hot leg temperature below or equal to 180°C
- Hot leg pressure below or equal to 30 bar
- ΔT_{sat} and RPVL consistent with LHSI/RHR suction from the hot leg.

In the CATHARE calculation, it is assumed that the RIS/RRA [SIS/RHRS] connection conditions are reached when the RPVL is above the top of hot leg and ΔT_{sat} is greater than 10°C.

5.1.6.3. Results

The SB-LOCA transient from the controlled state to the safe shutdown state has not been recalculated in the PSCR. The capability to reach the safe shutdown state while satisfying the acceptance criteria is derived from the results of calculations performed in BDR-99 for the EPR_{4900} as discussed in Appendix 14B. These calculations show that the criteria are met with large margin. These calculations are combined with a comparison of the relevant characteristics of the EPR_{4500} and the EPR_{4900}. This comparison shows that the EPR_{4500} is more favourable than the EPR_{4900}.

5.1.6.3.1. *BDR-99 Status*

Results of the BDR-99 analysis are provided in section 2.2.1 of Appendix 14B. They are summarised below.

The core remains covered throughout transient, with the RPV level above the bottom of the cold/hot legs.

The RIS/RRA [SIS/RHRS] connection conditions are reached seven hours after the reactor trip. This includes a delay of two hours without any manual action, and five hours of RCP [RCS] cooldown at -25°C/h. At the end of the cooldown the RCP [RCS] conditions are as follows:

- Hot leg temperature is about 160°C
- Hot leg pressure is about 10 bar
- The sub-cooling margin ΔT_{sat} is positive, higher than 10°C
- The RPVL is above the top of the hot leg.

When the LHSI/RHR is to be connected, the MHSI pump is shut down, one LHSI train operates in SI mode with injection into one cold leg, and the remaining LHSI train operates in RHR mode. The flow rate injected by the LHSI pump operating in SI mode is sufficient to maintain a RCP [RCS] loop level in the hot legs providing suitable operating conditions for the LHSI pump in RHR mode.

No further RCP [RCS] water inventory reduction occurs after the transfer to the safe shutdown state.

The MHSI pump, which has previously been shut down under the emergency operating guidelines, remains available for injection in addition to the LHSI pump in injection mode. If it were started again, its delivery head would be limited to around 40 bar, as described in sub-section 5.1.1.2.2 of this sub-chapter.

The transfer to the safe shutdown state occurs without emptying the ASG [EFWS] tanks. About 600 te of ASG [EFWS] water is used out of the initial total ASG [EFWS] tanks content of 1500 tons.

Core subcriticality is maintained throughout the transient for the 20 cm^2 break size analysed, by the boron injected by the MHSI pump, and after LHSI/RHR connection by the LHSI pump operating in SI mode. No extra boration is needed. For very small breaks, where SI injection flow may not be sufficient to achieve the required boration, the required boration would be provided by the F1A classified RBS [EBS]. Sub-section 5.1.6.3.3 below, ‘Reactivity control in the event of a very small LOCA located in one SI CL-injection line’, provides more information.

5.1.6.3.2. *Extrapolation of BDR-99 Results to the EPR_{4500}*

The results of the BDR-99 calculation for EPR_{4900} demonstrate that the F1-classified systems are sufficient to reach the safe shutdown RCP [RCS] boron concentration and to cool and depressurise the RCP [RCS] down to RIS/RRA [SIS/RHRS] connection conditions. This can be performed within a time period consistent with the ASG [EFWS] tanks capacity, while keeping the core covered for the entire transient. This is achieved despite the assumption of the worst single failure and the most onerous preventive maintenance.

The F1-classified systems used to achieve the safe shutdown state are:

- MHSI and LHSI for RCP [RCS] injection
- MHSI, and RBS [EBS] if needed, for RCP [RCS] boration
- ASG [EFWS] and VDA [MSRT] for RCP [RCS] cooling
- LHSI/RHR for long term RCP [RCS] and IRWST heat removal.

When compared to the EPR_{4900} characteristics, the EPR_{4500} is similar, or better, for RCP [RCS] injection, the RCP [RCS] cooling, and the RCP [RCS] boration capabilities:

- The EPR_{4500} power level is 9% lower when compared with the EPR_{4900}
- MHSI injection flow at high pressure ($P > 50$ bar) is increased in the EPR_{4500} compared with the EPR_{4900} [5]
- MHSI injection flow at low pressure ($P < 50$ bar), and LHSI injection flow, are unchanged in the EPR_{4500} compared to the EPR_{4900}
- The same RBS [EBS] boration capacity for the EPR_{4500} and the EPR_{4900}
- The ASG [EFWS] tanks water inventory unchanged in the EPR_{4500} compared with the EPR_{4900}, providing higher heat removal capacity relative to the power level
- Similar ASG [EFWS] and VDA [MSRT] flow capacities between the EPR_{4500} and the EPR_{4900}, in terms of percent of nominal power
- Same IRWST water content between the EPR_{4500} and the EPR_{4900}, providing higher heat sink reserve relative to the power level.

It can be concluded without need for dedicated calculations that achieving the Safe Shutdown State is demonstrated for the EPR_{4500}. This conclusion is based on the results of the BDR-99 calculations, and on the similar or beneficial aspects of the EPR_{4500} compared with the EPR_{4900} for RCP [RCS] injection, RCP [RCS] cooling, and RCP [RCS] boration,

The following section provides additional information about the RCP [RCS] boration in the event of a very small LOCA. In these circumstances MHSI boration only is not sufficient and RBS [EBS] boration is required to reach the safe shutdown state. Data used are for the EPR_{4500}.

[5] see Appendix 14B 15.0.2 – Tab.13/25 and Sub-chapter 14.1 – Table 13

5.1.6.3.3. *Reactivity control in the event of a very small LOCA located in one SI CL-injection line*

In a SB-LOCA, two F1 boration systems are available to provide boration for core subcriticality after RT, the MHSI and the RBS [EBS].

- For a SB-LOCA of sufficient size, the MHSI is able to perform the required boration alone,
- For very small LOCA, the MHSI flow injected into RCP [RCS] is too small to perform the required boration alone. In this case RBS [EBS] boration is required.
- If the very small LOCA is not located in one SI cold leg injection line, at least one RBS [EBS] train is fully available for boration, including PM and SF, as for any non-LOCA event. This RBS [EBS] train is able to inject more boron than necessary, as the RBS [EBS] is designed to cope with the more limiting boron dilution leading to core criticality after RT.
- If the very small LOCA is located in one SI cold leg injection line, the corresponding RBS [EBS] train is not fully available for boration. A part of the RBS [EBS] flow is assumed to be directly lost to the break without entering the RCP [RCS]. Each RBS [EBS] pump delivers into two SI CL lines via an RBS [EBS] header.

This paragraph considers this last case, to show that the RBS [EBS] flow injected into the RCP [RCS] in this case is sufficient to maintain core subcriticality after RT. This is maintained up to and including reaching the RIS/RRA [SIS/RHRS] connection conditions of 30 bar and 180°C in the RCP [RCS] hot leg.

The non F1-classified RCV [CVCS] is not considered as a source of boration.

Period from RT to the End of Partial Cooldown

The reactor trip takes the core subcritical. The negative reactivity worth of the rods is designed to keep the core subcritical to the end of Partial Cooldown (PC):

- Core subcriticality is maintained by N-1 rods to:
  - 260°C, using conservative assumptions.
- This design condition bounds the ‘end of PC’ conditions of:
  - 275 ± 1.5°C with PC performed by VDA [MSRT] down to 60 ± 1.5 bar.
  - 270 ± 1.5°C with PC performed by GCT [MSB] down to 55 ± 1.5 bar.

Consequently, RCP [RCS] boration is not required in SB-LOCA until after the ‘end of PC’.

Period from ‘End of PC’ to RIS/RRA [SIS/RHRS] Connection Conditions

During the transfer from the ‘end of PC’ to the RIS/RRA [SIS/RHRS] connection conditions, RCP [RCS] boration is required to maintain the core subcritical. The negative reactivity introduced by the control rods is not sufficient. The Emergency Operating Guidelines (EOG) will instruct the operator to actuate the RBS [EBS] before transferring the plant to cold shutdown in any PCC except a large LOCA.

The worst configuration for RCP [RCS] boration is shown in the upper part of Section 14.4.5 - Figure 5:

- Single failure of one EDG causes the unavailability of one RIS [SIS] train and one RBS [EBS] train.
- Preventive maintenance on another EDG causes unavailability of one RIS [SIS] train.
- SB-LOCA located in one SI cold leg injection line, between the RIS [SIS] nozzle and the first RCP [RCS] isolation check-valve. This is assumed to result in the complete loss of RIS [SIS] and RBS [EBS] flows delivered via this line.
- Only one MHSI-flow plus half of the RBS [EBS] train flow is available for boron injection into the RCP [RCS].

The worst case for RCP [RCS] boration is shown in the bottom part of Section 14.4.5 - Figure 5:

- The SB-LOCA is assumed to be small enough for the pressure to remain above the MHSI pump delivery pressure. Thus MHSI injection will not occur via the intact SI line (RCP [RCS]).
- Only half the RBS [EBS] flow is available for boron injection into the RCP [RCS]. In these circumstances the intact SI line and the affected SI line have the same back pressure, equal to the RCP [RCS] cold leg back pressure. This is a consequence of the very large ratio between the SB-LOCA size and the SI line section as shown by the CATHARE results.

The RCP [RCS] average boron concentration against time is given by the following equation:

$$M_{RCS}\frac{dC_{RCS}}{dt} = C_{in}\,W_{in} - C_{RCS}\,W_{out} \quad \text{with} \quad \frac{dM_{RCS}}{dt} = W_{in} - W_{out}$$

M_{RCS}: RCP [RCS] water mass

C_{RCS}: RCP [RCS] average boron concentration

C_{in}: Boron concentration of flow rate W_{in}

W_{in}: Flow rate entering the RCP [RCS], called 'injection flow'

W_{out}: Flow rate leaving the RCP [RCS] via the SB-LOCA

This equation is used to calculate the following parameters, using the conservative assumptions detailed below:

- Only the RBS [EBS] injection flow is needed to reach the critical boron concentration for RIS/RRA [SIS/RHRS] connection conditions. No MHSI injection is assumed
- Only the MHSI injection flow is needed to reach the critical boron concentration at RIS/RRA [SIS/RHRS] connection conditions. No RBS [EBS] injection is assumed
- The minimum average RCP [RCS] boron concentration is reached at RIS/RRA [SIS/RHRS] connection conditions, in the worst case, with only half the RBS [EBS] flow entering the RCP [RCS].

The RCP [RCS] critical boron concentration is defined as the concentration limit which maintains core subcriticality at RIS/RRA [SIS/RHRS] connection conditions. A lower concentration would lead to core criticality.

- To perform these calculations, the following set of conservative assumptions are made as discussed in the boration phase section of Sub-chapter 14.1
  - o The RIS/RRA [SIS/RHRS] connection conditions are defined as 30 bar and 150°C. 150°C is the lowest temperature consistent with a HL temperature of 180°C if a Reactor Coolant Pump is not operating.
  - o There is assumed to be no boration of the RCP [RCS] before starting the transfer to cold shutdown. No credit is taken for MHSI or RBS [EBS] injection during the PC prior to the operator decision to initiate the transfer to cold shutdown. This is a very conservative assumption.
  - o The boration of the RCP [RCS] is performed during the transfer to cold shutdown. This period lasts five hours, starting at the end of the PC with a temperature of 275°C, stopping at the RIS/RRA [SIS/RHRS] connection conditions with a temperature of 150°C. The transfer is performed at a cooling rate of -25°C/h consistent with the availability of one out of two RBS [EBS] trains.
- Boron concentrations (all boron concentrations refer to natural boron):
  - o The most onerous fuel cycle and point within that cycle are chosen, for UO_2 and MOX. This provides the maximum difference in boron concentration between the initial RCP [RCS] boron concentration at 100% FP and the critical boron concentration at 150°C. EOL operation at 100% FP is the limiting case. The following values of initial and critical boron concentrations refer to this limiting case.
  - o Minimum initial boron concentration in the RCP [RCS] of 10 ppm for UO_2 and 10 ppm for MOX.
  - o Maximum critical boron concentration at a temperature of 150°C of 480 ppm for UO_2 and 530 ppm for MOX,. These are calculated assuming all rods inserted. The SF is applied to one EDG. The calculation assumes an allowance of 200 ppm for uncertainties the presence of a xenon-peak at the time of RT, and no-xenon at 150°C. The xenon approach is over conservative, as the boration lasts five hours, compared to a period of about two days between the xenon-peak and no-xenon.
  - o Minimum boron concentration of the RIS [SIS] flow of 2450 ppm for UO_2 and 2700 ppm for MOX.
  - o Minimum boron concentration of the RBS [EBS] flow of 11200 ppm for UO_2 and 11825 ppm for MOX.
- RCP [RCS] water inventory and in/out flow rates:

- o A maximum RCP [RCS] water inventory at the beginning of boration is conservative. A value of 300 tons is assumed, being the RCP [RCS] initial inventory at 100% FP. This bounds the RCP [RCS] water inventory at the beginning of boration with no credit taken for RCP [RCS] water depletion between 100% FP and the beginning of the injection into the RCP [RCS]).
- o Minimum RBS [EBS] pump flow rate of 10 te/h or 2.8 kg/s.
- o The flow out of the RCP [RCS] through the break is ignored. This is conservative for the calculation of the RCP [RCS] boration.

Results

The worst initial operating condition is:

- 100% NP EOL with MOX
- Initial RCP [RCS] boron concentration at 100% FP, xenon-peak of 10 ppm, natural boron
- Critical boron concentration at 150°C, all rods in, no-xenon of 530 ppm, natural boron.

Without considering RBS [EBS] injection: the minimum MHSI injection flow rate that just maintains core subcriticality at the RIS/RRA [SIS/RHRS] connection conditions of 30 bar and 180°C at hot leg is approximately 3.2 kg/s or 12 te/h. This assumed the MHSI alone, without the RBS [EBS] providing boration. This MHSI injection flow rate is achieved when the RCP [RCS] pressure drops 3 bar below the MHSI shutoff head:

Relevant data:

M_{RCS} at beginning of boration	:	300 te
C_{RCS} at beginning of boration	:	10 ppm
C_{in} (MHSI alone)	:	2700 ppm
W_{in}	:	calculated
W_{out}	:	0 kg/s

This MHSI injection flow rate, matches the break flow for a SB-LOCA of 1 cm^2. This is very small when compared to the inner area of the SI CL line of 390 cm^2. Consequently, the break has no significant impact on the back pressure balance between the affected SI line and the intact SI lines with the pressure at break being the RCP [RCS] cold leg pressure:

Break flow rate:		Break size:
3.2 kg/s at 85 bar/275°C (RCP [RCS] side)	→	break size approximately 0.46 cm^2
3.2 kg/s at 85 bar/10°C (IRWST side)	→	break size approximately 0.25 cm^2

Without considering MHSI injection: the minimum RBS [EBS] injection flow rate which just maintains core subcriticality at the RIS/RRA [SIS/RHRS] connection conditions of 30 bar and 150°C is approximately 0.7 kg/s or 2.6 te/h. This representing 26% of the minimum capacity of one RBS [EBS] train:

Relevant data:

M_{RCS} at beginning of boration	:	300 te
C_{RCS} at beginning of boration	:	10 ppm
C_{in} (RBS [EBS] alone)	:	11825 ppm
W_{in}	:	calculated

W_{out} : 0 kg/s

The minimum RCP [RCS] boron concentration when the LHSI/RHR is connected, using the set of conservative assumptions defined above has been calculated for the limiting case. For this case, with the flow entering the RCP [RCS] with no MHSI, and only half the RBS [EBS], the flow is sufficient to maintain a significant margin to the critical concentration:

M_{RCS} at beginning of boration : 300 te
C_{RCS} at beginning of boration : 10 ppm
C_{in} (RBS [EBS]) : 11825 ppm
W_{in}(½ RBS [EBS]) : 1.4 kg/s
W_{out} : 0 kg/s

Limiting Fuel Management	MOX at EOL
Initial RCP [RCS] boron concentration, 100% FP, operation at Xe peak	10 ppm (natural boron)
Average RCP [RCS] boron concentration at 150°C, after 5 hours boration by half RBS [EBS]	1000 ppm (natural boron)
Critical boron concentration at 150°C, all rods in, xenon = 0	530 ppm (natural boron)
Conclusion	Core subcriticality maintained by approximately 470 ppm

This evaluation shows that whatever the SB-LOCA break size, the RBS [EBS] and MHSI injection is sufficient to provide the required RCP [RCS] boration to maintain core subcriticality from RT, up to and including the Safe Shutdown State. The safe shutdown state is defined as the RIS/RRA [SIS/RHRS] connection conditions. Subcriticality can be maintained despite the worst PM and SF, This is true for the limiting case of a very SB-LOCA located in one SI CL line, with a break size small enough to avoid MHSI injection into RCP [RCS].

5.1.6.4. Conclusion

The analysis of the SB-LOCA shows that despite the worst single failure and the limiting preventive maintenance:

- The controlled state can be reached meeting all safety acceptance criteria as discussed in sub-section 5.1.5 above.

- The safe shutdown state can be reached meeting all safety acceptance criteria, and consistent with the following F1 capabilities:

  During the transfer from the controlled state to the safe shutdown state,

  - The VDA [MSRT] and ASG [EFWS] pump and tank capacities for core heat removal.

  - The LHSI/RHR heat exchange capacity for IRWST heat removal.

  - The MHSI injection capacity from the IRWST for RCP [RCS] water inventory.

- o The MHSI, and RBS [EBS] for smaller breaks only, boron injection capacities for core subcriticality.

At the RIS/RRA [SIS/RHRS] connection conditions corresponding to the safe shutdown state,

- o The LHSI/RHR heat exchange capacity for core heat removal.
- o The LHSI/RHR heat exchange capacity for IRWST heat removal.
- o The LHSI/RHR injection capacity for RCP [RCS] water inventory.
- o The LHSI/RHR boron injection capacity for core sub-criticality.

5.2. SMALL BREAK LOCA ≤ DN 50 IN STATE B

5.2.1. Accident Definition

The initiating event is a non isolable break or leak in the RCP [RCS].

It is defined as a break size of equivalent diameter below 50 mm (~20 cm^2). This small break LOCA is classified as a PCC-3 event in state B, intermediate shutdown state as discussed in section 0 of Sub-chapter 14.1.

State B covers all shutdown states during normal plant operation, where primary heat is removed by the SG, and where some F1A classified I&C signals have been changed compared to power operation (state A). It extends from 130 bar when the deactivation of some F1A signals occurs to 30 bar and 120°C, the RCP [RCS] conditions for connection of the RIS/RRA [SIS/RHRS].

The consequences of such an event are analysed with the same objectives as those for reactor state A.

The SB-LOCA analysis in state B addresses the core cooling capability. The objective is to demonstrate that the relevant safety and acceptance criteria are met, including the capability of the F1 systems to maintain the safe shutdown state.

Compared to state A, state B introduces the following differences in terms of F1 mitigation:

- The source of the SI signal
- The accumulators are unavailable below a RCP [RCS] pressure of 70 bar.

As mentioned in section 5.1.1.1, a complete description of the calculation procedure used to assess the inherent boron dilution following LOCA is presented in [Ref-1].

5.2.2. Typical sequence of events

5.2.2.1. From the initiating event to the controlled state

In reactor state B, the RCP [RCS] break causes a loss of RCP [RCS] water inventory. Three levels of mitigation are provided to compensate for the loss of RCP [RCS] water inventory:

- The pressuriser level control, which is not F1 classified, will operate as the first level of mitigation. This will use the allowable imbalance between the RCV [CVCS] charging flow and letdown flow.

- Isolation of the RCV [CVCS] letdown line following a low pressuriser level signal is the second level of mitigation. This isolation, which is not F1 classified limits the loss of coolant.

- The third level of mitigation is the SI signal, which is F1A classified. This signal actuates the MHSI and LHSI pumps with the LHSI operating on its mini-flow. It also initiates the RCP [RCS] / containment isolation including isolation of the RCV [CVCS] letdown line. The SI signal on "low pressuriser pressure" is inhibited in state B. This avoids spurious actuation in reduced RCP [RCS] pressure conditions. Therefore, another SI signal is implemented. This SI signal uses the ΔP_{sat}[6] measurement which is F1A classified. It is calculated from the hot leg temperature measurement to define the corresponding saturation pressure and the hot leg pressure measurement. This signal actuates the LHSI/RHR trains while the RCP [RCS] hot legs are still under sub-cooled conditions.

Break flow compensation: Following the automatic RIS/RRA [SIS/RHRS] start-up, the RCP [RCS] water inventory loss is quickly matched by the injection flow. For all SB-LOCAs up to the largest break size of 20 cm^2, the loss of RCP [RCS] inventory at the break location is matched by the MHSI injection, as described in sub-section 5.2.5.5.1 of this sub-chapter.

Core heat removal: The break does not remove all the decay heat. Therefore secondary side heat removal is required, using the ASG [EFWS] and VDA [MSRT], which is F1A classified. The ARE [MFWS], AAD [SSS], and GCT [MSB] are used if available, but are not F1 classified.

Containment heat removal: The break flow temperature is higher than 100°C Therefore some flashing to steam occurs at the break. This results in a steam release inside the containment which leads to an IRWST temperature increase. IRWST heat removal via the LHSI/RHR heat exchangers limits this IRWST temperature increase.

The controlled state is defined as a state, where:

- The reactivity is under control

- The core power is removed via the SG, which were in standby prior to the accident

- The reactor coolant inventory is stable or increases again, i.e., the safety injection flow matches the break flow.

5.2.2.2. From the controlled state to the safe shutdown state

The safe shutdown state is defined as in reactor state A.

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[6] $\Delta P_{sat} = P_{hot\ Leg} - P_{sat}\ (T_{hot\ leg})$

The transient to reach the safe shutdown state from the controlled state is similar to that in state A, and described in that section.

5.2.3. Safety Criteria

The safety criteria and acceptance criteria to be met are the same as those described for the SB-LOCA in state A.

5.2.4. Definition of Studied Cases

As for state A, the limiting break location is the cold leg. The small break LOCA studied in this section corresponds to the largest PCC-3 break size of equivalent diameter 50 mm, equivalent to a break area of 20 cm^2.

This case is analysed assuming:

- The worst single failure
- The limiting equipment unavailability due to preventive maintenance.

In state B, LOOP does not need to be combined with the event.

There is no need for a specific code calculation of SB-LOCA in state B conditions where the accumulators are available, since this scenario is covered by the more limiting situation in state A.

A CATHARE (version V1.3L) calculation is performed for the range of state B conditions where the accumulators are isolated. It will be demonstrated that the acceptance criteria related to the controlled state are met using only F1A functions.

Reference can be made to the corresponding demonstration for state A conditions for successfully achieving the safe shutdown state, which clearly bound state B conditions.

5.2.5. Description of studied cases from the initiating event to the controlled state

The safety demonstration is performed considering the same conservative assumptions as in state A.

5.2.5.1. Choice of Single Failure and Preventive Maintenance

The worst single failure is the loss of one MHSI pump when the RIS/RRA [SIS/RHRS] is actuated. Thus one MHSI pump is unavailable.

Preventive maintenance on one division, affecting one MHSI, one LHSI, and one ASG [EFWS] pump is the limiting configuration. Thus, one MHSI pump, one LHSI pump, and one ASG [EFWS] pump are also assumed to be unavailable.

5.2.5.2. Initial State

Reactor state B ‘Intermediate shutdown state’ is defined as follows:

It extends below the hot shutdown state, where some F1A I&C signals are deactivated compared with power operation It corresponds to a RCP [RCS] pressure below 130 bar and a RCP [RCS] temperature below 303°C. It extends, to the cold shutdown state which is reached with the LHSI/RHR connected. This occurs at a RCP [RCS] pressure below 30 bar and a RCP [RCS] temperature below 120°C. It is noted that state B is assumed to start five hours after RT.

Two sub-states, B1 and B2, are considered in the safety assessment of an SB-LOCA in state B, based on the availability of the accumulators:

- State B1: all accumulators are available (not isolated)
    - RCP [RCS] pressure ranges between 70 bar[7] and 130 bar
    - RCP [RCS] temperature ranges between 245°C and 303°C.
- State B2: all accumulators are isolated
    - RCP [RCS] pressure ranges between 30 bar and 70 bar.
    - RCP [RCS] temperature ranges between 120°C and 245°C.

When considering the RCP [RCS] water inventory and core cooling aspects, the higher the RCP [RCS] pressure and the RCP [RCS] temperature when the break occurs, the more severe the consequences:

- A high RCP [RCS] pressure increases the initial break flow and minimises the immediate SI capacity.
- A high RCP [RCS] temperature increases the break flow rate once RCP [RCS] saturation is reached due to the higher RCP [RCS] saturation pressure. Similarly, it minimises the available SI capacity due to a lower SI flow rate at the higher RCP [RCS] pressure.

Consequently, the two following conservative initial states have to be considered in the safety assessment:

- State B1: most conservative initial condition:
    - RCP [RCS] pressure: 132.5 bar, including 2.5 bar uncertainty
    - RCP [RCS] temperature: 308°C, including 5°C for uncertainty and dead band.
- State B2: most conservative initial condition:
    - RCP [RCS] pressure: 72.5 bar, including 2.5 bar uncertainty
    - RCP [RCS] temperature: 250°C, including 5°C for uncertainty and dead band.

[7] During the normal RCP [RCS] cooldown to cold shutdown, the accumulators are isolated at a RCP [RCS] pressure of 70 bar. This is above their operating pressure range of 45 to 50 bar They are de-isolated at 70 bar during the normal RCP [RCS] heat-up to hot shutdown. At a RCP [RCS] pressure of 70 bar, the maximum RCP [RCS] temperature is 245°C.

However, state B1 is clearly covered by the analysis in state A. The same equipment and capacities are available as in state A, but the core power is much lower in state B. Therefore, a safety analysis is only performed for state B2.

5.2.5.3. Specific Assumptions

a) Assumptions related to non-F1 systems

As in state A, the non-classified systems are not claimed when they perform a mitigating function.

b) Assumptions related to F1 systems

Safety Injection (F1A): SI signal is automatically actuated on a “low RCP [RCS] sub-cooling margin ΔP_{sat}[8] “ signal with a setpoint in the range of 10 to 5 bar. To cover the uncertainty on the setpoint and to take account of the degraded containment conditions, the setpoint is conservatively assumed to be 0. This results in a maximum delay to start the LHSI/RHR pumps, and to the beginning of the partial cooldown.

The conservative assumptions for the resulting actions are listed below:

- 20 seconds maximum delay for LHSI/RHR pumps start-up including signal delay
- Minimum characteristics for LHSI/RHR pumps
- 50°C for initial IRWST temperature and injection flow temperature.

The LHSI/RHR train injecting into the broken loop is assumed to spill directly into the containment with LHSI/RHR components available to perform the cold leg injection are the following, taking the single failure and preventive maintenance assumptions into account,:

- One MHSI pump.
- Three accumulators when available (state B1).
- Two LHSI pumps.

VDA [MSRT] (F1A): VDA [MSRT] setpoints correspond to the saturated conditions of the RCP [RCS], plus 1.5 bar uncertainty to model the pressure measurement uncertainty in a conservative manner.

All VDA [MSRT], one per SG, are available.

The SI signal initiates the Partial Cooldown as in state A. However, the automatic decrease of SG pressure at a rate of -100°C/h down to 60 bar (± 1.5 bar uncertainty) is only effective in the high RCP [RCS] pressure range of state B1 where the initial SG pressure is higher than 60 bar.

ASG [EFWS] (F1A): ASG [EFWS] is used to feed the SG.

Three trains are available.

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[8] $\Delta P_{sat} = P_{hot\ Leg} - P_{sat}\ (T_{hot\ Leg})$

The conservative assumptions for the ASG [EFWS] pump characteristics and the ASG [EFWS] water temperature are similar to those described in sub-section 4.5.1.5.3 of Sub-chapter 14.1.

Reactor coolant pumps trip (F1A): the reactor coolant pumps trip is automatically actuated on a reduced reactor coolant pumps pressure drop signal, normally at 80% of the reactor coolant pumps nominal pressure drop. However, in the analysis a setpoint of 50% is assumed to include some margin. This results in a later reactor coolant pumps trip and a consequent later SI signal.

5.2.5.4. Other Assumptions

The pressuriser water level/volume of 19.6 m^3 corresponds to the minimum water level for zero power, including conservative uncertainties of 5% MR.

The initial secondary pressure corresponding to the primary temperature of 250°C is 40 bar. For the progression of the event it is assumed that the plant is in cooldown mode with a gradient of 50°C/h. The secondary and primary pressure decreases in line with this cooldown rate.

5.2.5.5. Results

The safety assessment utilises a qualitative argument for a SB-LOCA in state B1, but the SB-LOCA scenario for state B2 is explicitly analysed.

5.2.5.5.1. *State B1: Accumulators Available (Not Isolated)*

The effects of the 20 cm^2 SB-LOCA on the RCP [RCS] water inventory and core cooling in state B1 are less severe than those calculated by the CATHARE code for the accident analysis in state A, since:

- All F1A mitigation measures claimed in state A are available in state B1
    - o MHSI, LHSI and accumulators for SI injection, VDA [MSRT], and ASG [EFWS] for heat removal.
- There are further F1A systems available in state B1
    - o Single failure applies to only one item of equipment instead of one electrical division. This is a consequence of the continued availability of off-site power in this case. There is, therefore, no loss of one division power supply as a result of a failure of an emergency diesel generator (EDG).
    - o The SI signal occurs before RCP [RCS] saturation is reached. The RCP [RCS] is still full of sub-cooled water, except for the pressuriser. The MHSI pumps are operable earlier in state B1 as there is no requirement to include a delay from a diesel reloading sequence in the absence of a LOOP.
- Initial SG pressure[9] is lower in state B1 than in state A, resulting in:
    - o Lower break flow rate as a lower RCP [RCS] pressure is reached when the RCP [RCS] saturates.

[9] Maximum SG pressure 90+1.5 bar in state B1, instead of 95.5+1.5 bar in state A.

- o MHSI injection occurs earlier. The Partial Cooldown[10] is more efficient. The initial SG pressure is lower and leads to the MHSI delivery pressure of 85 bar/300°C being reached earlier. Similarly, the completion of the Partial Cooldown at 60 bar/275°C also occurs earlier.
- o The SI injection flow is higher due to the lower RCP [RCS] pressure.

- Core decay heat is lower in state B1, allowing a faster RCP [RCS] depressurisation.

As a result, due to the lower RCP [RCS] break flow and the earlier actuation and greater effectiveness of the MHSI in state B1, the RCP [RCS] water inventory is higher compared with state A for the same size break.

Therefore, the CATHARE analysis results of the limiting PCC-3 SB-LOCA in state A for a break of 50 mm cross section are conservative for a PCC-3 SB-LOCA in state B1:

- The controlled state is reached without core uncovery, and consequently without core heat-up, using only F1A systems.
- The prevention of core uncovery is achieved without using the accumulators, and LHSI as one MHSI is sufficient.

The consequences of a small break LOCA transient in state B1 to the controlled state are bounded by the consequences obtained in state A.

5.2.5.5.2. *State B2: Accumulators Unavailable (Isolated)*

The initial plant state covered by this CATHARE calculation is given in Section 14.4.5 - Table 4. Although state B2 is only reached at more than 5 hours after RT, where the decay heat is 42.4 MW, a more conservative value of 50 MW is assumed in the calculation. Thus, the analysis results include some additional margins.

The sequence of events is given in Section 14.4.5 - Table 5 [Ref-1].

The most important parameters are presented in the following figures:

- Section 14.4.5 - Figure 6: Reactor and SG power, RCP [RCS] and SG secondary side pressures.
- Section 14.4.5 - Figure 7: Break mass flow (steam, liquid and total), break flow steam and liquid.
- Section 14.4.5 - Figure 8: Swell levels in RPV, core void fraction.
- Section 14.4.5 - Figure 9: Coolant liquid and vapour temperatures in RPV and core, clad temperatures
- Section 14.4.5 - Figure 10: Integrated flow rates of RIS [SIS], coolant inventory on primary and secondary side

[10] With SG initial pressure of 90 bar in state B1 (saturation 303°C), instead of 95.5 bar in state A (saturation 307.5°C), the MHSI injection into RCS (RCS saturation 85 bar/300°C) can start a few minutes earlier in state B1 compared with state A assuming the same partial cooldown rate of: -100°C/h).

- Section 14.4.5 - Figure 11: Total RIS [SIS] and break flows, integrated RIS [SIS] and break flows

A detailed explanation of these parameters is provided in Section 14.4.5 - Table 6.

The main automatic safety measure is starting the MHSI and LHSI pumps following a SI signal, generated by the ΔP_{sat} low setpoint being reached. However the LHSI pumps do not deliver to the RCP [RCS]. The break flow is matched early in the transient. The lack of accumulator injection is not significant to the mitigation of the event. About 2500 seconds, or some 40 minutes after the break occurs, the RCP [RCS] inventory slowly starts increasing as a result of the operation of only one MHSI pump.

As the core is covered by at least a two-phase mixture throughout the transient, there is no core heat-up and the acceptance criterion of 1200°C in the fuel rod is met.

After about 3000 seconds (50 minutes) the controlled state is reached.

5.2.6. Description of Cases Studied from the Controlled State to the Safe Shutdown state

The plant state at the controlled state is better in state B than in state A.

All F1A and F1B mitigation systems used in state A to transfer the plant from the controlled state to the safe shutdown state are also available in state B.

Consequently, the safety assessment performed for the transfer to the safe shutdown state in a PCC-3 SB-LOCA from state A bounds the PCC-3 SB-LOCA from state B.

It can be concluded that, despite of the worst single failure and limiting preventive maintenance, the safe shutdown state is reached with all safety and acceptance criteria fulfilled, using only F1A and F1B systems.

5.2.7. Conclusion

A SB-LOCA in state B has the following main differences when compared to a SB-LOCA in state A:

- Changes in F1A mitigation means
  - SI signal "ΔP_{sat} low" in state B, instead of "pressuriser pressure low" in state A
  - Accumulators isolated below RCP [RCS] pressure of 70 bar in state B
- More favourable initial plant conditions.

The conclusions reached for the PCC-3 SB-LOCA in reactor state A for a break smaller than 50 mm diameter, or area of 20 cm^2), also apply to the PCC-3 SB-LOCA in reactor state B in the same size range, as discussed below.

All the safety criteria are met despite the worst single failure and preventive maintenance assumptions:

- There is no core uncovery and thus no core heat-up. The peak clad temperature remains below the acceptance criterion of 1200°C
- There is no clad oxidation
- There is no clad rupture
- Integrity of the core geometry is maintained
- The long-term core cooling is maintained.

The controlled state is reached, using F1A classified systems. The LHSI/RHR in conjunction with the VDA [MSRT] and the ASG [EFWS] only are used.

The safe shutdown is reached, using only F1A and F1B classified systems:

- During the transfer from the controlled state to the safe shutdown state:
  - The VDA [MSRT] and ASG [EFWS] pump and tank capacities for core heat removal
  - The LHSI/RHR heat exchange capacity for IRWST heat removal
  - The MHSI capacity for RCP [RCS] water inventory
  - The MHSI and RBS [EBS] boron injection capacities for core subcriticality The RBS [EBS] is only required for smaller breaks, and if the RCP [RCS] boron concentration related to cold shutdown was not achieved at initial state B.
- At RIS/RRA [SIS/RHRS] connection conditions corresponding to the safe shutdown state:
  - The LHSI/RHR heat exchange capacity for RCP [RCS] heat removal
  - The LHSI/RHR heat exchange capacity for IRWST heat removal
  - The LHSI capacity for RCP [RCS] water inventory
  - The LHSI boron injection capacity for core subcriticality.

5.3. CONSEQUENCES OF THE MODIFICATION OF THE PARTIAL COOLDOWN RATE

Following the SI signal actuated either on a “pressuriser pressure < MIN3” signal in state A or on a “low RCP [RCS] sub-cooling margin ΔP_{sat}” signal in state B, the partial cooldown is automatically actuated in order to remove the energy of the primary side via the secondary side. Consequently the primary pressure is reduced which allows the MHSI to inject borated water.

The cooldown rate increase from -100°C/h to -250°C/h will have a beneficial effect on the mitigation of the small break LOCA accident. As a consequence of the change, the MHSI delivery pressure of 85 bar will be reached earlier as the primary pressure will decrease faster.

At the end of the partial cooldown, the final primary pressure will remain the same (61.5 bar) and the break flow will be similar.

It can thus be concluded that all the decoupling criteria will be met.

5.4. SYSTEM SIZING

5.4.1. Residual Heat Removal System (RIS [SIS])

This transient places requirements on the MHSI flow.

The required MHSI flow rate is governed by the following requirements:

- For SB-LOCA of 5 cm^2, equivalent diameter of 25 mm, in state A limited RCP [RCS] draining. This transient places a requirement on the MHSI flow rate at approximately 70 bar.

- Allowance for reactor coolant pumps running during the SB-LOCA of 5 cm^2 equivalent diameter of 25 mm. This transient places a requirement on the MHSI flow rate with the large mini-flow line open at 25 bar

- No core uncovery for break size below 20 cm^2, equivalent diameter of 50 mm, in power state A (PCC-3) imposes a required MHSI flow rate

SECTION 14.4.5 - TABLE 1

Fission Power (Term A)

The residual fission power (term A) results from a MANTA decoupled Reactor Trip (RT) simulation.

Point-kinetic neutronic model is used with a conservative set of neutronic data, bounding all UO_2 and MOX fuel managements:

- Maximum moderator temperature coefficient - 0.515 ΔK/K per g/cm^3(constant value)
- Minimum Doppler temperature coefficient - 4.03 pcm/°C (constant value)
- Minimum Doppler power coefficient - according to 14.1 Table 4
- Minimum rod worth - 5100 pcm (N-1 rods, UO_2 fuel)
- Minimum rod worth insertion versus - according to 14.1 Table 8
- Rod drop time
- Maximum rod drop time - 5 seconds (with earthquake)

RCP [RCS] and SG thermal hydraulic calculation is based on conservative boundary conditions:

- Minimum decrease of ARE [MFW] flow after RT:

From RT to RT+15 s	100% to 30% (linear decrease)
From RT+15 seconds to 0% SG-level setpoint	30% (constant value), afterwards flow controlled to keep SG-level

- Maximum decrease of ARE [MFW] temperature after RT:

At RT	from 230°C down to 190°C in one step

- Minimum GCT [MSB] opening time after RT

GCT [MSB] valves opening time	0 s
GCT [MSB] I&C-processing delay	0 s

TIME (s) from beginning of rod drop	FISSION POWER % of initial core power
0	92.4
1	90.5
2	90.2
3	82.1
4	28.8
5	10.9
10	7.8
50	2.8
100	1.1
200	0.4
300	0.0

SECTION 14.4.5 - TABLE 2

Initial Conditions
20 cm^2 (∅ 50 mm) Cold Leg Break (State A)

Parameters	Limiting Values
Reactor Coolant System	
• Initial reactor power (% of nominal power)	100 + 2 = 102.
• Initial average RCP [RCS] temperature (°C)	312.8 + 2.5 = 315.3
• Initial pressuriser pressure (bar)	155 + 2.5 = 157.5
• Reactor coolant flow (kg/s)	22225 (T/H flow rate)
• Pressuriser water volume / level (m^3 / m)	43.4 / 7.4 (nominal+ 5%MR)
• Dome liquid temperature (°C)	332 (hot leg temperature)
Steam Generators	
• Initial steam pressure (bar)	80.1(Consistent with RCP [RCS] T)
• Initial SG level (m)	15.7 (nominal)
• Tube plugging Level	10%
Feedwater	
• Main feedwater flow (% of nominal flow)	100 + 2 = 102
• Initial ARE [MFWS] temperature (°C)	232

SECTION 14.4.5 - TABLE 3

Sequence of Events Related to Controlled State
Typical Results for 20 cm^2 (∅ 50 mm) Cold Leg Break (State A)

TIME (s)	EVENT
0.0	Break opening
81.6	Pressuriser pressure < MIN2 (132 bar)
82.5	Reactor trip signal
82.8	RT (start of rod drop), TT, Reactor coolant pumps trip, loss of ARE [MFW] flow
230	Pressuriser level = 0% MR
296	Pressuriser pressure < MIN3 (112 bar)
297	SI and PC signal
300	VDA [MSRT] opening
337	Starting MHSI, LHSI pumps
752	Start of MHSI injection in loop 2 (RCP [RCS] pressure < 85 bar)
1380	ASG [EFWS] injection in loops 2 and R
5000	End of calculation

SG1: related to the unaffected SG in loop1	intact loop (with PM or SF)
SG2: related to the unaffected SG in loop2	intact loop with pressuriser
SG3: related to the unaffected SG in loop3	intact loop (with SF or PM)
SGR: related to the affected SG in loop R	broken loop

SECTION 14.4.5 - TABLE 4

Initial Conditions
20 cm^2 (∅ 50 mm) Cold Leg Break, State B2

Parameters	Values Used
Reactor Coolant System	
Initial reactor power (% of nominal power)	1.17% or 50 MW
Initial average RCP [RCS] temperature (°C)	245 + 5 = 250
Initial Pressuriser pressure (bar)	70 + 2.5 = 72.5
RCP [RCS] loop flow rate (m^3/hr)	27180
Pressuriser water volume (m^3)	24.5
Steam Generators	
Initial steam pressure (bar)	40 (for RCP [RCS] at 250°C)
Initial water temperature (°C)	130
Initial water level (m)	15.69

SECTION 14.4.5 - TABLE 5

Sequence of Events Related to Controlled State (Global Time Values)
20 cm^2 (∅ 50 mm) Cold Leg Break, State B2

TIME (s)	EVENT
0	Break opening
200	Pressuriser empty
250	RCP [RCS] pressure stabilises some 5 bar above secondary pressure (around 45 bar)
1100	Setpoint for the RCP [RCS] trip criterion (Δp across Reactor coolant pumps < 50% of nominal) reached
1130	Setpoint for the SI signal on ΔP_{sat} = 0 reached
1150	Start of safety injection by MHSI, RCP [RCS] pressure at about 35 bar
2500	Minimum RCP [RCS] inventory, minimum level in RCP [RCS] well above loop level MHSI flow is higher than break flow, continuous filling up of RCP [RCS]
3000	End of calculation

SECTION 14.4.5 - TABLE 6

Explanation of Parameters for Figures 6 through 11
20 cm^2 (Ø 50 mm) Cold Leg Break – State B

Figure 6: **Core Power and Total Heat Exchange in Steam Generator**

(CORE: core power, SGPOWER: heat exchange in steam generator)

Primary and Secondary System Pressure

(UPPL: upper plenum, COLDL1: cold leg loop 1, SECSG1 sec. pressure, PMIN1: reactor signal, PMIN2: safety injection signal)

Figure 7: **Vapour and Liquid Mass Flow at the Leak**

(MVPLEAK: vapour flow, MLQLEAK: liquid flow, LEAKTOT: total flow)

Vapour and Liquid Velocity at the Leak

(VVPLEAK: vapour velocity, VLQLEAK: liquid velocity)

Figure 8: **Swell Level in Reactor Pressure Vessel**

(HDOME: vessel head, LEPLENSU: upper plenum, HLMID: hot leg middle)

Void fraction in the core

(VOIDCM1: 0.105m, VOIDCM2: 0.525m, VOIDCM8: 3.465m, VOIDCM9: 3.990m)

Figure 9: **Liquid and Vapour Temperature in Reactor Pressure Vessel**

(TEMLCM8: liquid at 3.465m, TEMLCM9: liquid at 3.990m,
TEMGCM9: vapour at 3.990m, TGPLENSU: vapour in upper plenum)

Cladding Temperature of the Average Rods

(CLADTA1: 0.105m, CLADTA5: 1.995m, CLADTA8: 3.465m, CLADTA9: 3.990m)

Figure 10: **Integral Safety Injection Rate**

(LHSIINT: low head safety pump, MHSIINT medium head safety injection, RIS [SIS] INT: total)

Water Inventory in the Primary and Secondary System

(PMASS: primary system, SMASS: secondary system)

Figure 11: **Safety Injection and Leak Discharged Rate**

(RIS [SIS] TOT: total safety injection rate, LEAKTOT: total leak discharge rate)

Integral Safety Injection and Leak Discharged Rate

(RIS [SIS] INT: safety injection, LEAKINT: leak discharged rate)

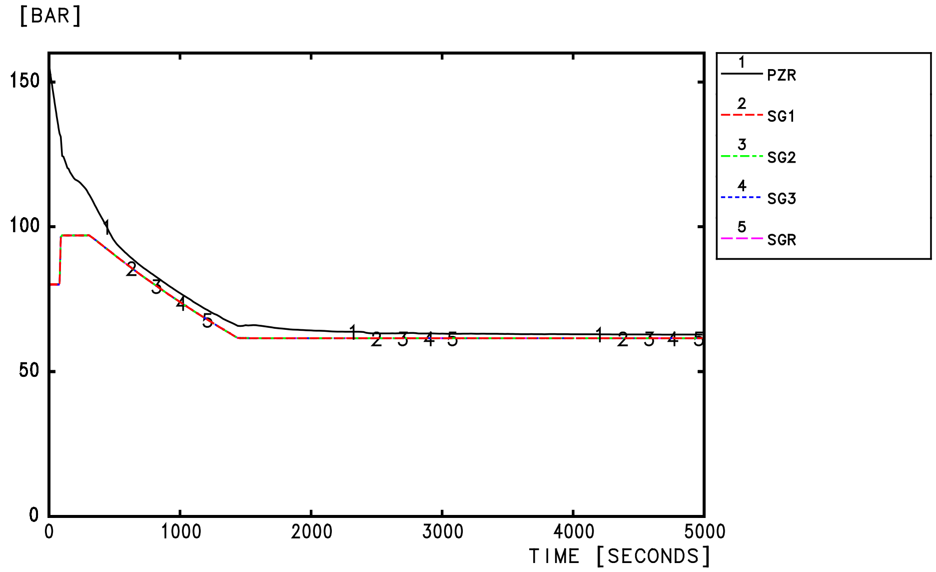
SECTION 14.4.5 - FIGURE 1

Typical (P,T) Reactor Operating Range

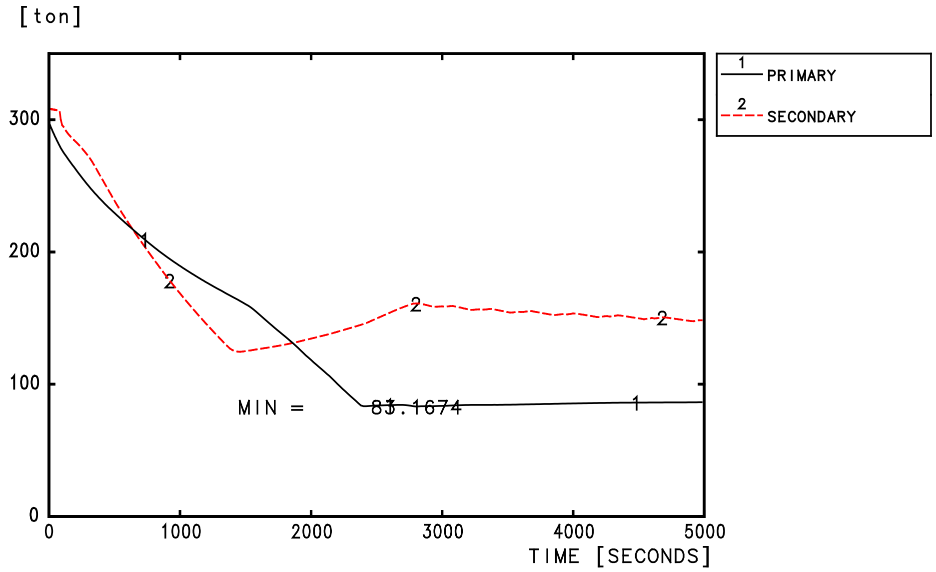
Note:
This figure is provided
in Section 7 of Sub-chapter 14.5,
SB-LOCA in Shutdown State.

SECTION 14.4.5 - FIGURE 2

Secondary Side Water Inventories
RCP [RCS] and Secondary Side Pressures
Typical Results for 20 cm² (∅ 50 mm) Cold Leg Break (State A, PCC-3)



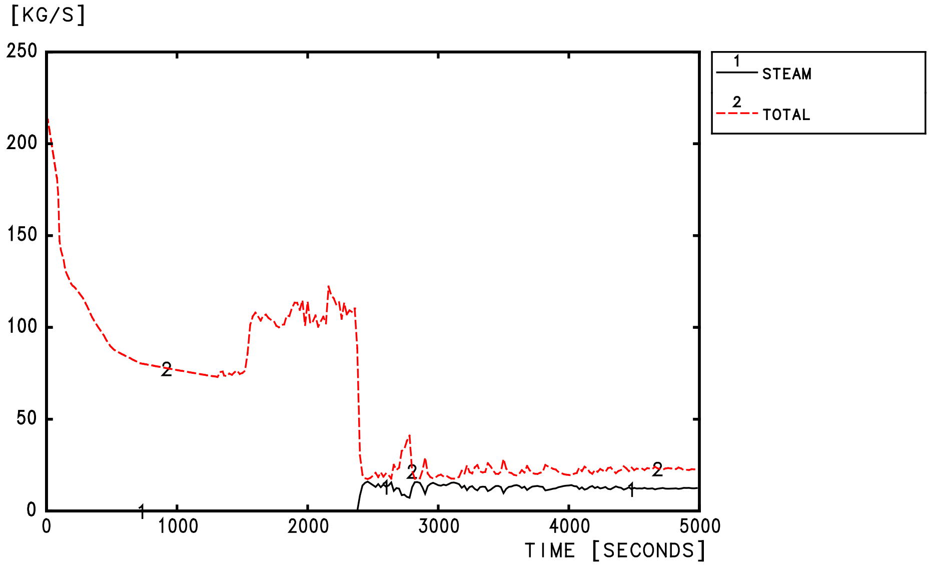
PRIMARY AND SECONDARY PRESSURES



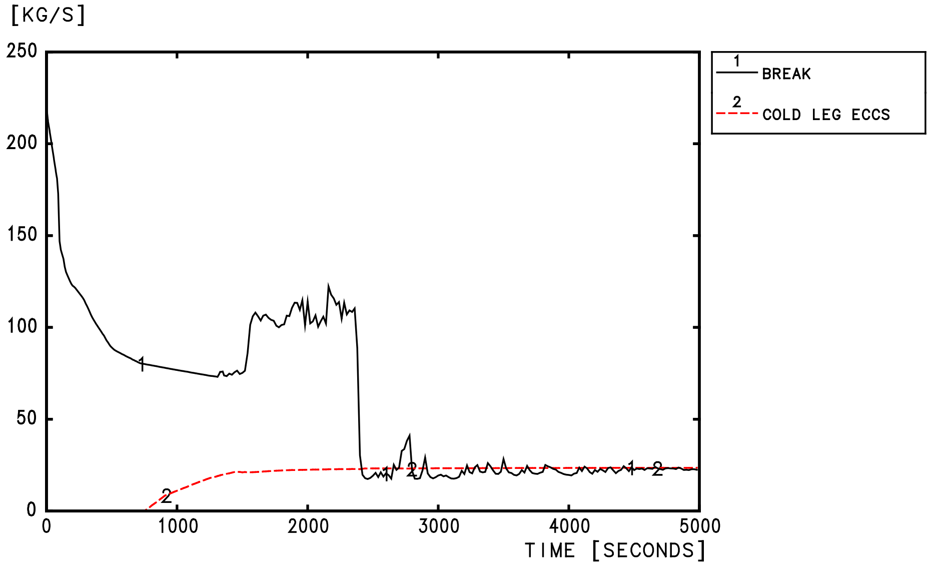
PRIMARY AND SECONDARY MASSES

SECTION 14.4.5 - FIGURE 3

Total Break and RIS [SIS] Flow Rates
Total Break and Steam Flow Rates
Typical Results for 20 cm² (∅ 50 mm) Cold Leg Break (State A, PCC-3)



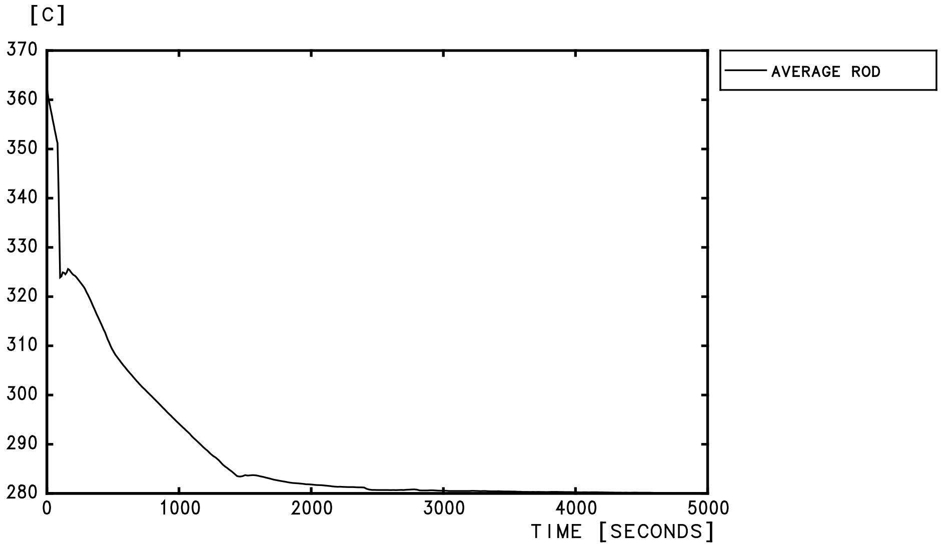
BREAK MASS FLOW RATE



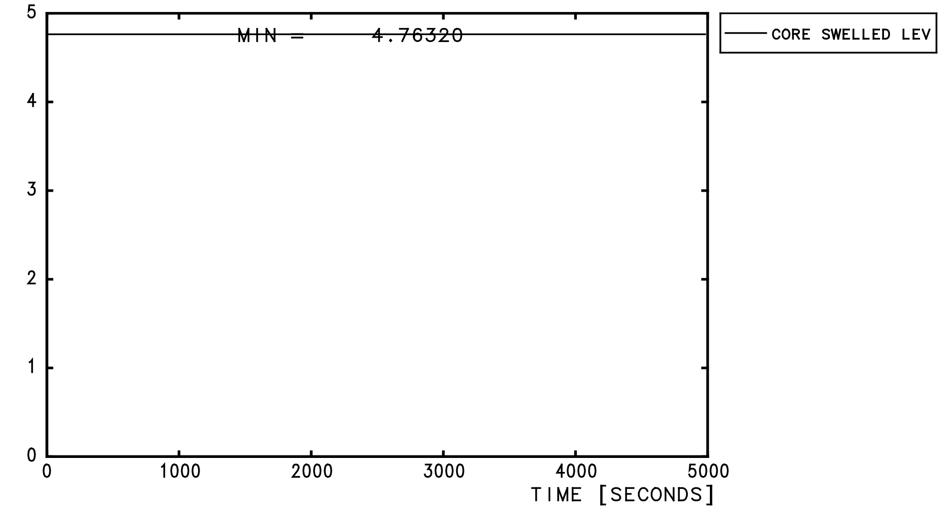
BREAK MASS FLOWRATE AND TOTAL ECCS INJECTION RATE

SECTION 14.4.5 - FIGURE 4

Maximum Clad Temperature
Core Two-Phase Level
Typical Results for 20 cm2 (∅ 50 mm) Cold Leg Break (State A, PCC-3)



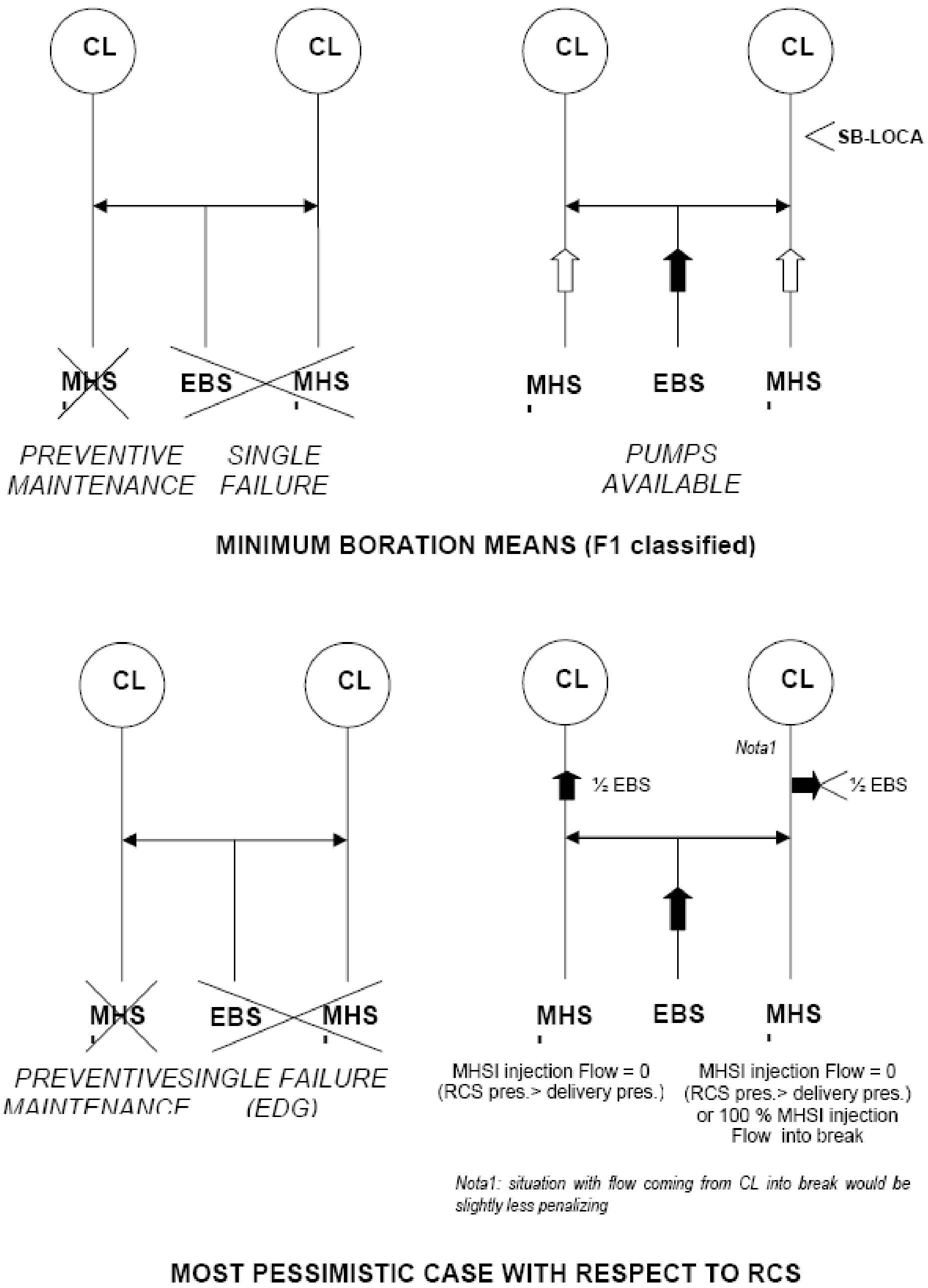
MAXIMUM CLAD TEMPERATURE



CORE LEVEL

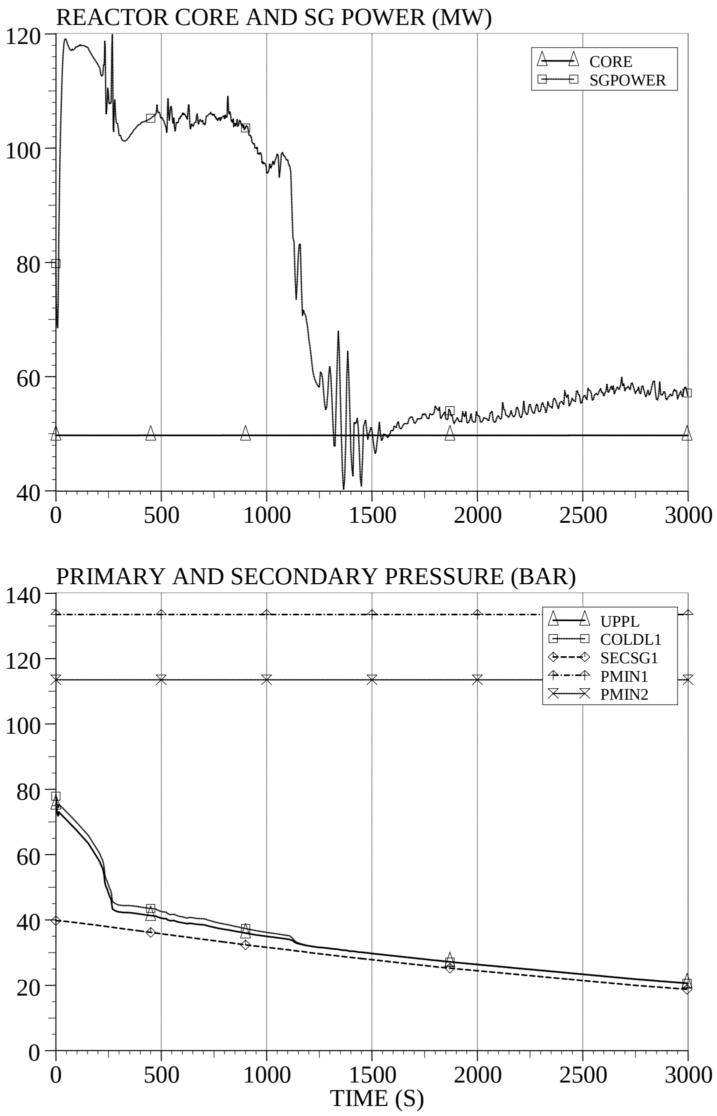
SECTION 14.4.5 - FIGURE 5

RCP [RCS] Boration in SB-LOCA



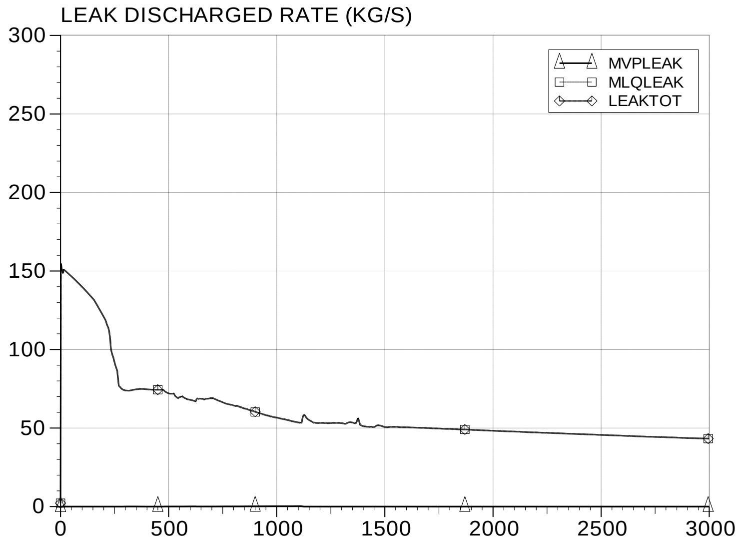
SECTION 14.4.5 - FIGURE 6

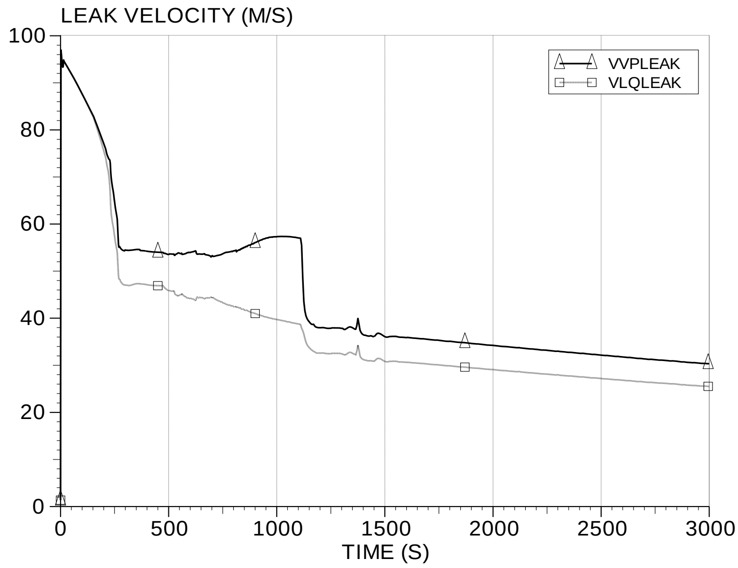
20 cm² (Ø 50 mm) Cold Leg Break – State B2
Core Power and Total Heat Exchange in Steam Generator
Primary and Secondary System Pressure



SECTION 14.4.5 - FIGURE 7

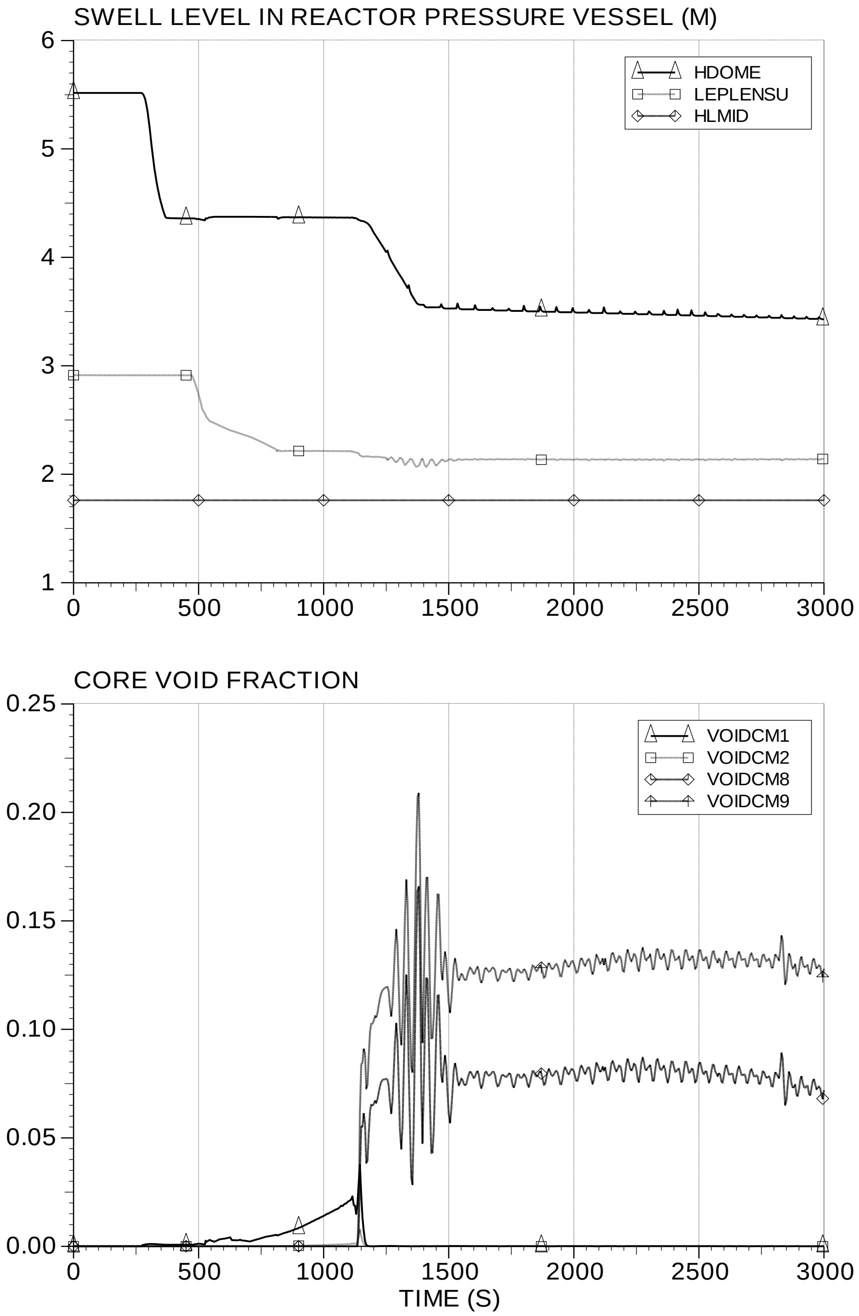
20 cm² (Ø 50 mm) Cold Leg Break – State B2
Vapour and Liquid Mass Flow at the Leak
Vapour and Liquid Velocity at the Leak





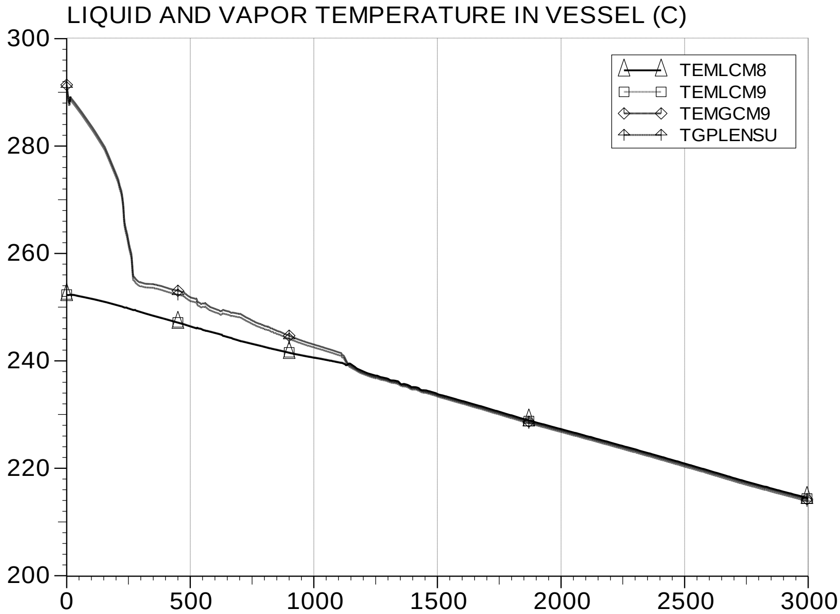
SECTION 14.4.5 - FIGURE 8

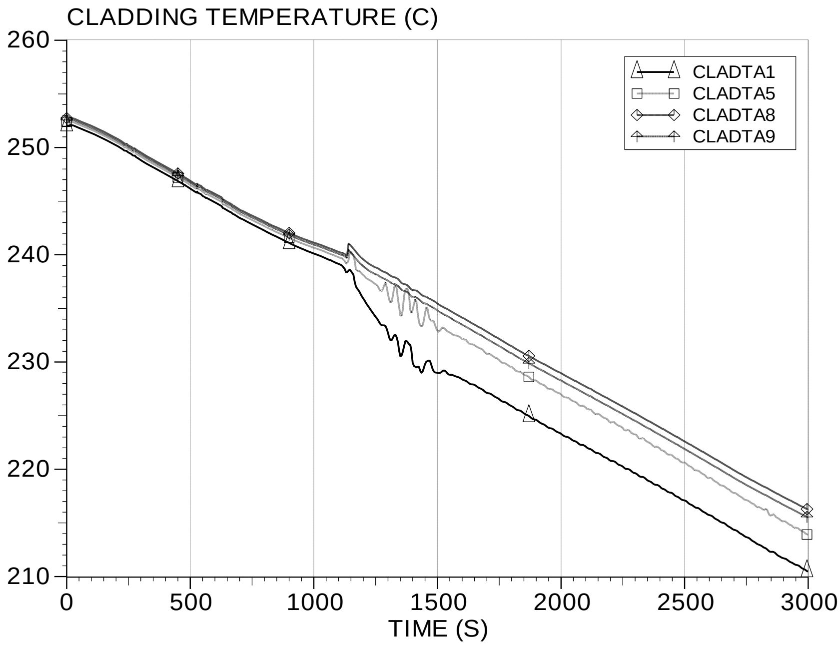
20 cm^2 (Ø 50 mm) Cold Leg Break – State B2
Swell Level in Reactor Pressure Vessel
Void Fraction in the Core



SECTION 14.4.5 - FIGURE 9

20 cm^2 (Ø 50 mm) Cold Leg Break – State B2
Liquid and Vapour Temperature in Reactor Pressure Vessel
Cladding Temperature of the Average Rods

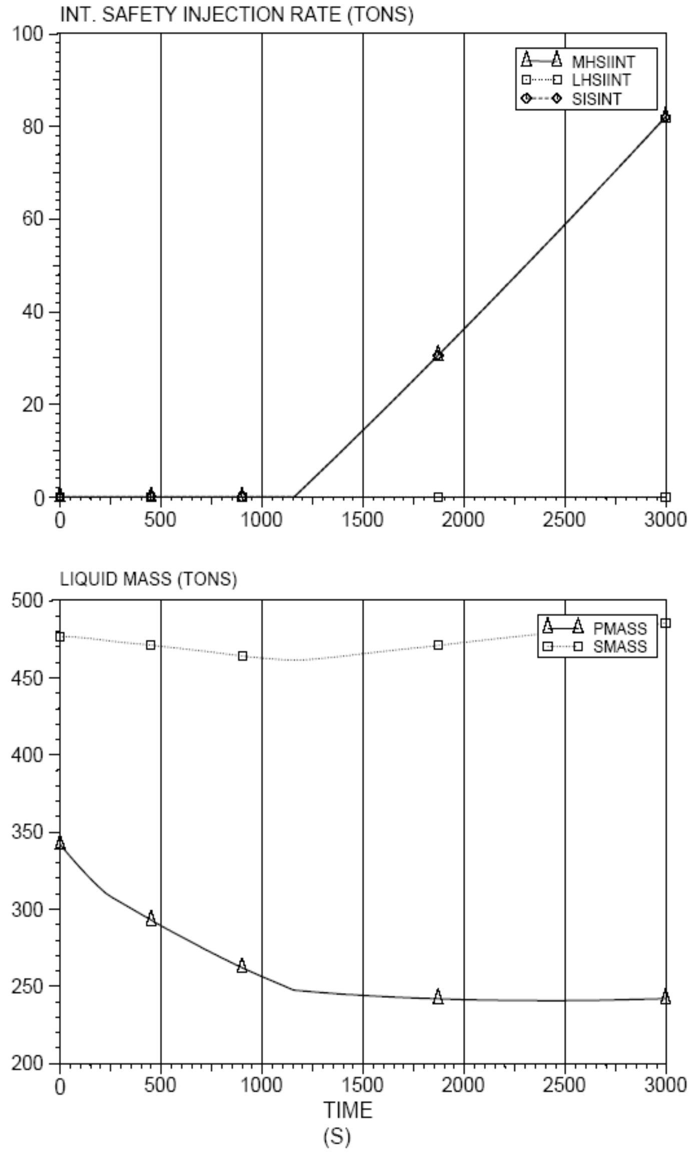




SECTION 14.4.5 - FIGURE 10

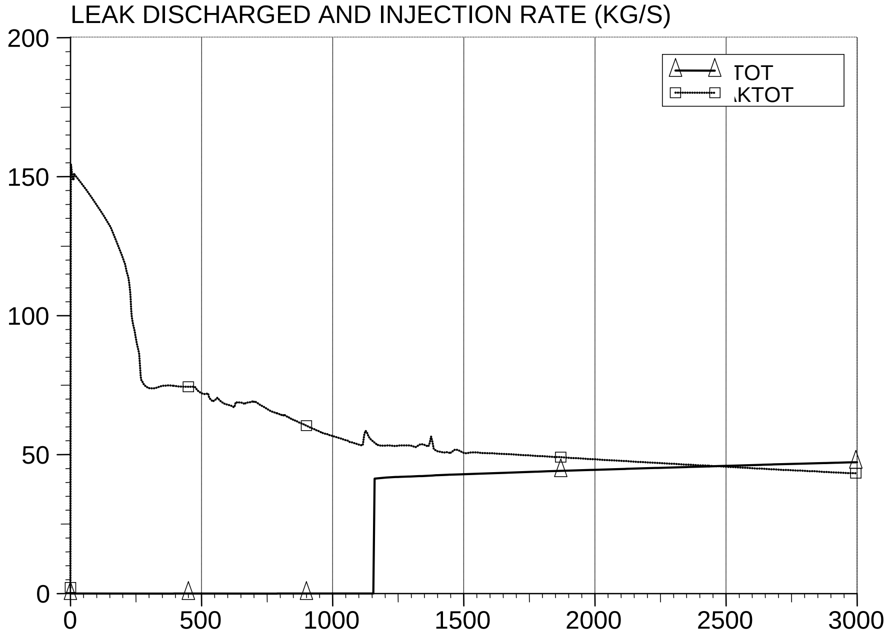
20 cm² (Ø 50 mm) Cold Leg Break – State B2

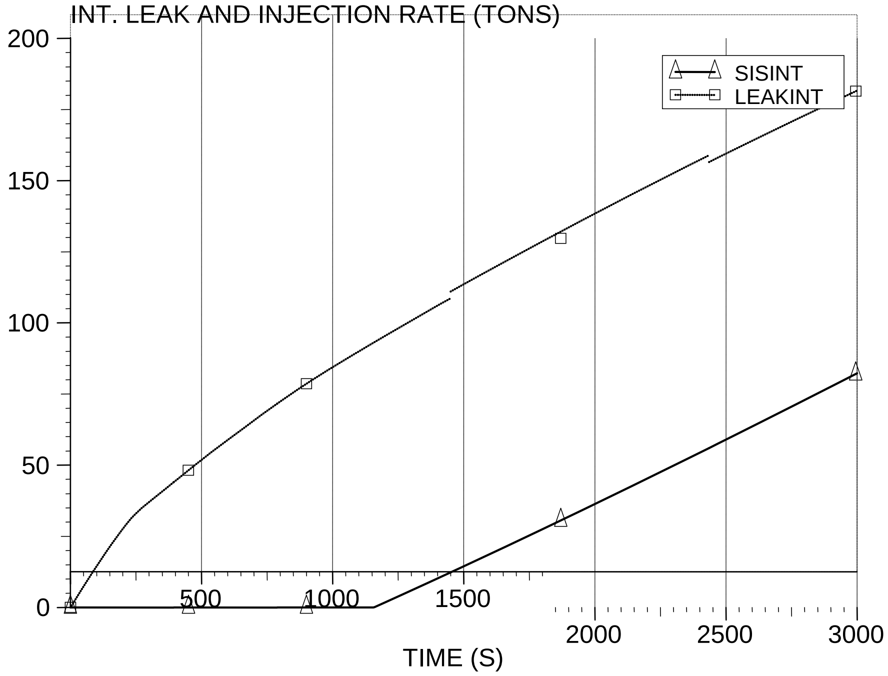
Integral Safety Injection Rate
Water Inventory in the Primary and Secondary System



SECTION 14.4.5 - FIGURE 11

20 cm² (Ø 50 mm) Cold Leg Break – State B2
Safety Injection and Leak Discharged Rate
Integral Safety Injection and Leak Discharge





6. STEAM GENERATOR TUBE RUPTURE (1 TUBE)

The steam generator tube rupture of one tube (2A-SGTR) occurring in state A is classified as a PCC-3 event.

6.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

6.1.1. General Concern

The steam generator tube rupture (SGTR) event is defined as the double-ended rupture of a single steam generator (SG) tube.

The main consequences of this accident are associated with the secondary side contamination and possible discharge of radioactive products to the atmosphere. The contamination comes from the rupture which connects the reactor coolant system (RCP [RCS]) to the secondary side. The primary side coolant is assumed to be radioactive due to corrosion and fission products associated with continuous operation with a limited amount of defective fuel rods. This risk of contamination is primarily in the affected SG (SGa). The possible discharge of radioactive products to the atmosphere can occur either in the steam or liquid phase via the main steam relief trains (VDA [MSRT]) or the main steam safety valves, if challenged.

The description of the transient is subdivided into short term and long term to clearly separate the phases of radioactivity release to the atmosphere. The short term phase is defined as up to the point of leak termination. This includes the controlled state in which the leak is matched by the RCP [RCS] injection. In the long term phase, the plant is transferred to the safe shutdown state conditions with a possible additional activity release if depressurisation of the affected SG by the VDA [MSRT] is required.

6.1.2. Typical Sequence of Events

The typical sequence of events in the case of a 2A-SGTR, apart from additional events that could occur because of the single failure assumptions is as follows:

6.1.2.1. From rupture initiation to leak termination (short term)

a) From rupture initiation to the controlled state

The controlled state is defined as a state where the RIS [SIS] flow and/or RCV [CVCS] flow if available, is able to compensate for the flow through the tube rupture, with decay heat removed from the RCP [RCS] by the SG.

The loss of primary coolant causes a decrease in the primary pressure and contamination of the secondary side. The reactor trip can occur either following a “pressuriser pressure < MIN2” or a “SG level > MAX1” signal generated in the affected SG, or via an operator initiated shutdown at 30 minutes after activity detection via a safety class 1 system. The latter corresponds to ‘Design Option 1’ of the SGTR mitigation strategy [Ref-1]. Which signal occurs depends on the initial state and operating conditions of the plant. The following discussion of a typical sequence of events, within this sub-section, assumes the break is of sufficient magnitude to lead to an automatic reactor trip prior to the operator initiating a controlled shutdown.

At Power Operation

At power operation, the SGTR flow takes part in the production of steam. The SGa level is not significantly increased. The RT on “pressuriser pressure < MIN2” signal is therefore generated prior to the one on “SG level > MAX1” signal.

Zero Power Operation

At hot shutdown initial conditions, the primary heat transferred to the secondary side is insufficient to boil all the SGTR flow. This causes the affected SG level to increase. A RT occurs and the affected SG is isolated on the feed side following a “SG level > MAX1” signal. The ARE [MFWS], or ASG [EFWS] if already actuated are isolated.

A reactor trip signal automatically trips the turbine and the steam generator pressure rapidly increases. The GCT [MSB] is assumed to be unavailable as it is not F1-classified. It would also be unavailable following a Loss of Offsite Power (LOOP) assumed at turbine trip. When the VDA [MSRT] pressure setpoint is reached, the VDA [MSRT] isolation valve opens and contaminated steam is discharged to the atmosphere from the affected SG.

The continuous loss of RCP [RCS] coolant inventory causes the pressuriser to empty. This results in a depressurisation of the primary side because the RCV [CVCS] is not able to match the break flow.

Partial cooldown is initiated either following a Safety Injection (SI) signal on “pressuriser pressure < MIN3” or following a “SG level > MAX2” signal from the affected SG. Partial cooldown lowers the pressure in all four SG to nominally 60 bar by reducing each VDA [MSRT] pressure setpoint at a rate corresponding to a given RCP [RCS] cooldown.

Following the SI signal the MHSI pumps are actuated. However they do not inject as the RCP [RCS] pressure is above their shutoff head.

The controlled state is reached when MHSI injection and RCV [CVCS], if available, are able to match the SGTR flow rate. However, since the leak has not been halted at this point, the affected SG continues to fill with contaminated water and activity release to the atmosphere continues.

b) From the controlled state to leak termination

The affected SG is identified and isolated following the combination of “SG level > MAX2” and “end of partial cooldown” signals or operator action. The RCV [CVCS] charging lines are isolated automatically. The isolation of the affected SG is performed by raising the setpoint of the VDA [MSRT] to above the MHSI shutoff head but below the MSSV pressure setpoint and closing the VIV [MSIV] if this has not already occurred. This prevents any further release of radioactive products to the atmosphere. This setpoint increase can be performed automatically or manually.

The isolation of the affected SG, including the closure of the affected SG VDA [MSRT] and VIV [MSIV] causes the flow via the break to increase the pressure in the affected SG. Once the primary side and secondary side of the affected SG pressures equalise, the flow via the break is finally terminated. This corresponds to the end of the short term phase. Significant margin to overfilling of the affected SG is maintained. Consequently only steam is discharged during this phase.

The short term phase, until SGTR leak termination, uses only automatic F1 signals and systems.

6.1.2.2. From leak termination to the safe shutdown state (long term phase)

The safe shutdown state is defined as a state where the affected SG is isolated and at least one RIS/RRA [SIS/RHRS] train is connected to the RCP [RCS]. One out of four LHSI in RHR mode trains is sufficient to provide the required heat removal. The connection conditions are:

- RCP [RCS] hot leg pressure < 30 bar and,
- RCP [RCS] hot leg temperature < 180°C and,
- ΔTsat[1] and Reactor Pressure Vessel Level (RPVL) consistent with LHSI in RHR mode suction conditions from the hot leg.

The sequence of actions to be performed by the operator to reach the safe shutdown can be divided into 2 successive phases: boration and RCP [RCS] cooldown, and final depressurisation.

Boration and RCP [RCS] Cooldown

The boration and RCP [RCS] cooldown actions are performed by the operator 30 minutes after reactor trip. The RBS [EBS] delivers a constant boration flow rate to the RCP [RCS], providing the negative reactivity required to reach the safe shutdown state. The allowed cooldown rate depends on the number of available RBS [EBS] trains:

- 25°C/h with 1 RBS [EBS] train in operation,
- 50°C/h with 2 RBS [EBS] trains in operation.

The RCP [RCS] cooldown is performed using the unaffected steam generators. This cooldown occurs with the MHSI operating to prevent perturbing the pressure balance between primary side and the affected SG.

Final Depressurisation

At the end of the RCP [RCS] cooldown phase, the RCP [RCS] pressure is close to the MHSI shutoff head with the MHSI on. This pressure is higher than the LHSI in RHR mode maximum connecting pressure of 30 bar. If the affected SG level is too high, the operator first opens the transfer line between SGa and its partner SG to limit risks of water hammer in the SGa steam line. This also prevents overfilling of the affected SG and large activity release to the atmosphere. Prior to this transfer, the partner SG is prepared to accept the additional volume from the affected SG by:

- lowering the value for level control slightly above the SG level MIN2 setpoint,
- ASG [EFWS] is stopped, VIV [MSIV] is closed, and the setpoint of the VDA [MSRT] increased

Once the level in the affected SG falls below MAX2 (NR), the affected SG VDA [MSRT] is opened. This allows the depressurisation to 30 bar.

[1] $\Delta T_{sat} = T_{sat}$ (hot leg pressure) - T_{co} , with T_{co} = core outlet temperature

6.1.2.3. Radiological consequences

Contaminated steam flows through the turbine before reactor trip and is condensed in the steam dump system. Gaseous and insoluble radioactive products are evacuated to the atmosphere through air ejectors and are detected by continuous activity control and periodic measurements.

After turbine trip, if the condenser is no longer available, the opening of the VDA [MSRT] cannot be prevented. Steam is then released to the atmosphere.

The radiological consequences are assessed in Sub-chapter 14.6.

6.1.2.4. Precautions limiting the event occurrence

The probability of SGTR event is reduced through the following precautions:

- the SG tube material is highly ductile,
- the blowdown system is located at the bottom of the SG tube bundle and is designed to prevent solid deposits on the tube plate,
- the secondary water is conditioned chemically, thereby protecting the SG tubes from corrosion phenomena,
- the steam generators are designed to prevent any projectile coming from the main feedwater to directly strike one or several tubes,
- SG support plates maintaining tube bundle are designed (type of material, geometry of borings) to prevent denting in the tubes and, in the case of a double-ended guillotine rupture, to prevent pipe whip which could cause the rupture of neighbouring tubes,
- activity control of the secondary side SG water and SG steam assumes continuous monitoring of the compliance within defined limits.

6.2. SAFETY CRITERIA

The safety criteria to be complied with are the radiological limits for PCC-3 events discussed in Sub-chapter 14.0.

The safety criteria to be met are the dose equivalent limits in the case of release to the atmosphere.

To meet these criteria, the following decoupling criteria are to be met:

- no core damage (fuel cladding integrity),
- no MSSV challenge to prevent any possibility of MSSV failure in the open position,
- possible return to LHSI in RHR mode conditions of boration, depressurisation and heat removal with achievement of safe shutdown conditions using only F1 systems.

The prevention of core damage, to keep the core coolable and to prevent increased activity concentration on primary side, is demonstrated by the LOCA studies.

The SGTR mitigation strategy, involving automatic and manual actions, has been produced to meet the following two objectives:

- prevention of SG overfilling, to avoid increased activity release by liquid discharge to the atmosphere,
- minimisation of the SGTR leak backflow, to avoid problems with low borated water slugs in the primary circuit when the reactor coolant pumps are off (e.g. following a loss of offsite power).

6.3. METHODS AND ASSUMPTIONS

6.3.1. Methods of analysis

Case 1 – without LOOP

The Case 1 SGTR analysis, to assess the maximum steam release, is performed with the CATHARE code using a realistic deterministic methodology.

The realistic deterministic methodology is characterised by two main features:

- the key code models are realistic though conservatively oriented, bounding the experimental results without excessive conservatism, and
- the initial and boundary conditions are conservatively selected.

The basic steps of the realistic deterministic methodology consist of:

- the phenomenological analysis of the accident scenario, and the identification of the key phenomena,
- the judgement of adequacy of the code to calculate the accident scenario, based on physical understanding, experimental data base, code assessment examination, and supplemented when necessary by sensitivity studies,
- the evaluation of calculation uncertainty with emphasis on dominant parameters (through sensitivity studies as necessary), or the check of a bounding conservative approach of key phenomena by the code, relying on the assessment matrix of the code,
- the introduction, when necessary, of conservative biases as close as possible to the uncertainty on the key phenomena. They are introduced either in a code model, in a nodalisation scheme, or in a boundary condition, and
- the use of conservative assumptions for initial and boundary conditions.

The dominant phenomena of the SGTR transient are:

- the SGTR flow rate and the resultant SG level increase,

- the moderate RCP [RCS] draining (pressuriser) and depressurisation until the equilibrium with the affected SG pressure is reached,
- the asymmetric RCP [RCS] heat removal via the unaffected SG in subcooled RCP [RCS] conditions,

All these phenomena are within the applicable range of the CATHARE code, for which validation is based on [Ref-1]:

- the qualification of correlations and physical laws on separate effect tests (SET) or component tests, for example:
    - CATHARE SGTR operator for accurate SGTR break flow prediction,
- the validation of the axial SG model, with an economiser of the N4 SG-type from MEGEVE small scale model tests,
- the overall verification of the code by simulation of integral effect tests (IET), covering a wide range of representative PWR transients on small-scale facilities, for example:
    - BETHSY 4.3b ‘6 SGTR’ Test in which CATHARE accurately predicted the mass discharged at the faulted SG relief valve, the faulted SG mass inventory, the RCP [RCS] mass inventory and distribution. It accurately predicted the formation of the large steam space inside the faulted SG tubes, the slow RCP [RCS] depressurisation during the draining of the faulted SG, the restart of the loop circulation after the reactor coolant pump start in the affected loop, and the consequent fast depressurisation due to the condensation and collapse of the faulted SG tubes steam space.

Cases 2 and 3 – with LOOP

The S-RELAP5 code coupled with the I&C routines of the NLOOP code is used for Cases 2 and 3 [Ref-2], which assess the potential for overfilling of and water relief from the SG and address the long term cooldown capability following LOOP. The family of transients in the S-RELAP5 model is SGTR.

Primary side phenomena:

- Small break LOCA behaviour as primary pressure and pressuriser level decrease until the initiation of the MHSI.
- Steam-side isolation of the affected SG and secondary side cooldown with the intact SGs can lead to a very low circulation ratio within the affected SG.

Secondary side phenomena:

- Main steam pressure increases following closing of the turbine / main steam bypass valves.
- Fast opening of the main steam relief train and reclosing to maintain a constant main steam pressure.

- Filling of the affected SG above the top edge of the separators after reactor trip.
- Overfill of the affected SG may occur.

Generally, for S-RELAP5, a wide range of validation at different test facilities for the apparent phenomena are performed as integral or separate tests. These tests are applied for verification of all primary side phenomena. Explicitly on the basis of tests at the PKL and THTF facilities, a wide range of validation for smaller leaks is performed. This also covers thermal-hydraulic phenomena in transients including the phenomena on the secondary side. Additionally, an SGTR incident occurring at the Doel2 PWR was recalculated successfully using S-RELAP5. [Ref-3]

Both the CATHARE and S-RELAP5 transient analyses rely on the application of the conservative PCC analysis rules defined in Sub-chapter 14.0. The rules include the deterministic penalisation of all relevant boundary conditions relative to the decoupling criteria under consideration. This penalisation addresses at least:

- the characterisation of the initiating event,
- the plant initial conditions (control dead band limits, maximum measurement uncertainties) and,
- the efficiency of the protection and mitigation actions with maximum uncertainty on each I&C measurement and signal delay, and on each system response time and capacity.

This analysis methodology provides conservative results that can be used directly for the assessment of the decoupling criteria.

6.3.2. Main Assumptions

6.3.2.1. Accident Definition

The cases studied in section 6 of this sub-chapter correspond to the double-ended guillotine rupture of 1 tube in a steam generator, which allows unimpeded blowdown from both ends of the severed tube.

The tube rupture is located at the bottom of SG-tubes bundle, on the cold side. This location maximises the SGTR leak flow rate (higher fluid density).

For accident analyses, a loss of off-site power (LOOP) is combined with the accident, if this is more conservative.

6.3.2.2. Protection and Mitigation Actions

The automatic protection systems and operator actions (by means of F1-classified systems) following a SGTR event, aim at tripping the reactor, removing the residual heat, terminating the primary to secondary leak flow, limiting contaminated SG liquid mass release to the atmosphere, and, finally, taking the reactor to a safe shutdown state.

The different automatic protections and alarms, which could occur during the recovery from the SGTR event, are linked either to the reactor coolant depressurisation or to the level increase in the affected SG.

The reactor trip is initiated via one of the following F1A signals:

- “pressuriser pressure < MIN2”,
- “SG level in affected SG > MAX1”,

If the leak rate is insufficient to reach either of the F1A actions listed above, a manual reactor trip may be assumed after the first significant indication is received in the control room.

The other F1A automatic protection includes the following:

- Turbine trip: the turbine trip is actuated on RT check-back signal,
- Safety injection signal and partial cooldown: the RIS [SIS] is actuated on the "pressuriser pressure < MIN3" signal. A partial cooldown using all SGs, including the affected one (unless it has been isolated by operator action), is initiated if it has not already been actuated,
- Partial cooldown can also be actuated on the “SG level > MAX2" signal[2] if partial cooldown has not already been actuated. The partial cooldown is performed by all SGs, including the affected one (unless it has been isolated by operator action),
- Affected SG isolation (feed side): The affected SG is isolated on a “SG level > MAX1" signal. This initiates isolation of full load ARE [MFWS] (MAX1 narrow range) and ASG [EFWS] line if the ASG [EFWS] was previously actuated (MAX1 wide range). This signal is SG related,
- Affected SG isolation (steam side): The affected SG is isolated on the “SG level > MAX2" and “end of partial cooldown” signals. This is performed by closure of the VIV [MSIV], increasing the VDA [MSRT] pressure setpoint to above the MHSI shutoff head but below the MSSV pressure setpoint,
- ASG [EFWS] actuation: The ASG [EFWS] train is actuated on “SG level < MIN2" in the corresponding SG. This signal is SG specific. In the event of a LOOP, the ASG [EFWS] is actuated on an RIS [SIS] signal. The time delay between the setpoint being reached and the effective ASG [EFWS] flow delivery is defined in accordance with the assumption of LOOP or no LOOP,
- VDA [MSRT] actuation: The VDA [MSRT] opens and performs the heat removal with pressure control when the SG pressure reaches the VDA [MSRT] setpoint, “SG pressure > MAX1",
- VIV [MSIV] isolation: The VIV [MSIV] closure is initiated on a “SG pressure < MIN1" or “SG pressure drop > MAX1" signal. In this case all the MS lines are isolated as the signal is not SG specific,
- VDA [MSRT] isolation: The VDA [MSRT] of the corresponding SG is isolated when the SG pressure reaches the VDA [MSRT] setpoint, “SG pressure < MIN3”. This is performed by closure of the associated MSRIV. This signal is SG specific.

[2] The MAX2 level can only be reached in the affected SG because all other SG feedwater delivery is isolated on SG level > MAX1.

For long term mitigation of the event, the operator actions aim to transfer the plant to the safe shutdown state. The operator performs the boration and cooldown using the unaffected SGs prior to the final depressurisation of the RCP [RCS] and the affected SG to the LHSI in RHR mode connecting conditions.

6.3.2.3. Operator Actions

No operator action is claimed until 50 minutes after the first significant indication (detection of increased level of activity within the affected SG).

The 50 minute delay takes into account an initial operator action at 30 minutes after the activity detection and then a delay of 20 minutes for a power decrease. When local operator action is needed, the delay is extended to 1 hour.

The actions presented below are extracted from the Emergency Operating Procedures (EOP) available at the time of the study.

The F1B operator actions related to the affected SG isolation are:

- manual ASG [EFWS] isolation, if not done (SG level, secondary side activity)[3]
- manual ARE [MFWS] isolation, if not done (SG level, secondary side activity)
- manual VIV [MSIV] isolation, if not done (SG pressure)
- manual VDA [MSRT] setpoint increase, if not done (SG pressure)

The F1B operator actions related to the non-affected SGs are:

- manual ASG [EFWS] actuation
- manual ARE [MFWS] Low Load isolation

The F1B operator actions to cooldown the plant to RHR connection conditions are:

- manual actuation of cooldown, if not done (SG level, secondary side activity)
- reactor coolant boration:
    - o The boration is automatically performed by the Chemical and Volume Control System (RCV [CVCS]). In the event that this system is unavailable, the operator actuates the Extra Boration System (RBS [EBS]).

If the reactor is tripped manually, all of the above actions will be performed by the operator at the time of reactor trip, with the exception of initiating the manual cooldown. The RBS [EBS] will be started and manual cooldown initiated after the completion of the automatic partial cooldown.

[3] The parenthesis indicates potential parameters used to take actions.

6.4. DEFINITION OF CASES STUDIED

6.4.1. Short Term Cases

The short term phase, as defined in the SGTR study, is the time period between SGTR initiation and leak termination. This phase includes reaching the controlled state, which corresponds to the state where the core is subcritical and the SI flow rate or, if working properly, RCV [CVCS] flow rate compensates for the SGTR leak rate.

For the short term phase, the purpose of the study is to evaluate the maximum amount of fluid released to the atmosphere from the affected SG prior to termination of the leak. This is done in two separate steps:

- To evaluate the maximum amount of activity release to the atmosphere, it is conservative to assume maximum power to be removed as prior to the isolation of the affected SG there is only steam release. Therefore, the event is initiated at 102% of full power. This results in the maximum decay heat. In addition, no LOOP is assumed, requiring the reactor coolant pump power to be removed in addition to the decay heat.

- To verify that no SG overfilling occurs, and thus no liquid is released to the atmosphere prior to leak termination, a minimum power is conservative. This assumption minimises the steam releases and hence maximises the liquid content of the SG. Therefore the event is initiated at 2% of full power and LOOP is assumed.

Thus, two cases with different assumptions are analysed to address each of these two objectives.

<u>Case 1: Without LOOP</u>

To maximise the steam release from the affected SG it is necessary to maximise the heat to be removed by the SGs. Therefore, the worst case is:

- maximum initial power of 102% FP

- no LOOP

It is assumed that the reactor coolant pumps remain running throughout the transient.

The other specific assumptions for this case are described in sub-section 6.5.1.3 of this sub-chapter.

<u>Case 2: With LOOP</u>

To maximise the liquid content in the affected SG the initial liquid inventory should be maximised. This occurs at a low power level. In addition, the heat to be removed by the SGs should be minimised to give the minimum steam release.

Therefore, the worst case is:

- low initial power of 2% FP,

- LOOP coincident with a turbine trip signal.

The other specific assumptions related to this case are described in sub-section 6.5.2.3 of this sub-chapter.

6.4.2. Long Term Case

The long term phase covers the time period between leak termination and the safe shutdown state with LHSI in RHR mode connection conditions being reached. This phase includes the phases of boration and simultaneous cooldown of the RCP [RCS] using the unaffected SGs and the final depressurisation of the RCP [RCS] and the affected SG.

For the long term phase, the objective of the study is:

- To confirm the ability to reach safe shutdown conditions, LHSI in RHR mode connection conditions using only F1 systems.
    - The adequacy of ASG [EFWS] tank capacity is demonstrated
    - The adequacy of the RBS [EBS] is demonstrated
- To determine the total amount of activity released to the atmosphere, mainly during the final depressurisation phase for the RCP [RCS] and the affected SG.

Only one case is studied for this phase.

<u>Case 3: With LOOP</u>

This case, with the inclusion of LOOP on turbine trip, demonstrates the capacity of the F1 systems to sufficiently borate the affected loop and to depressurise the RCP [RCS] and the affected SG without significant leak backflow, when the reactor coolant pumps are tripped.

The maximum radiological releases to the atmosphere are calculated during this phase.

Note that a long term case with a maximum initial power and without LOOP would be relevant for the demonstration of the capacity of F1 systems, especially of the ASG [EFWS] tank, to cool down the plant to LHSI in RHR mode conditions with a maximum power to remove. However, this case is analysed in the 2-tube SGTR case of section 10 of Sub-chapter 14.5. The amount of contaminated steam released by the SGa during the final depressurisation with a single SGTR is bounded by the results of the analysis performed for the 2 SGTR case in section 10 of Sub-chapter 14.5.

The other specific assumptions related to this case are described in sub-section 6.6.1.3 of this sub-chapter.

6.5. SHORT TERM STUDY

6.5.1. Case 1: Without LOOP

6.5.1.1. Choice of Single Failure (SF) and Preventive Maintenance (PM)

The single failure on the Main Steam Relief Control Valve (MSRCV) of the affected SG is assumed. The valve is assumed to remain stuck in its initial, fully-open position. Post-trip, SG pressure increases in the steam generator until the VDA [MSRT] isolation valve opens. When this occurs, the affected SG will begin an uncontrolled blowdown. The blowdown will continue until the MSRIV is shut on low SG pressure (< MIN3).

This case is studied to confirm that the SG does not overfill or dry out, and that the steam releases remain below an acceptable value.

It is assumed that no preventive maintenance is limiting for this case.

Operator actions will be further optimised during the Nuclear Site Licensing phase, to ensure that affected SG dry out is avoided.

6.5.1.2. Initial State

The initial state conditions, given in Section 14.4.6 – Table 1, are chosen to maximise the heat to be removed and the pressure difference between the RCP [RCS] and affected SG.

Note that the impact of an increase of the initial pressuriser level is assessed in sub-section 6.5.3.1 of this sub-chapter.

6.5.1.3. Specific Assumptions

a) Neutronic data

The core power is assumed constant at 102% (4590 MWth) of full power until reactor trip. Following RT, the time-dependent A term with 1.645 σ uncertainties on B+C term is used as described in Sub-chapter 14.1.

b) Assumptions related to control systems

Turbine: Turbine control maintains the turbine flow rate at 102% until turbine trip on reactor trip.

GCT [MSB]: Not considered

ARE [MFWS]: SG level control is in operation when the accident occurs and works properly until feed isolation on reactor trip. This is consistent with the analysis rules presented in Sub-chapter 14.0. This maximises the activity transferred to the affected SG. High load main feedwater flow is isolated on reactor trip with no delay. Low load main feedwater is manually isolated by the operator.

RCV [CVCS]: To maximise the pressure difference between the RCP [RCS] and the affected SG, the maximum charging flow rate with 2 charging pumps in operation at the start of the transient is assumed. Letdown is isolated on “pressuriser level < MIN3” (12%). RCV [CVCS] charging is isolated automatically following the combination of "SG level > MAX2” and the “end of partial cooldown” signals.

The injection temperature is modelled as 270°C with letdown in operation. The injection temperature is reduced to 50°C when the letdown is isolated.

Pressuriser heaters and pressuriser sprays: To maximise the pressure difference between the RCP [RCS] and the affected SG, all pressuriser heaters are initially actuated. This maximises the RCP [RCS] pressure, while the spray flow rate is not considered. The heaters are shut off following a “pressuriser level < MIN3” signal.

c) Assumptions related to F1 systems

The relevant setpoints are given in Section 14.4.6 - Table 3.

Reactor Trip: With the RCV [CVCS] system in operation, the makeup flow is sufficient to maintain the primary circuit pressure above the reactor trip setpoint on low pressuriser pressure for the first 3000 seconds of the transient. The high activity signal is detected immediately following the initiating event; a time delay of 3000 seconds is taken for the operator action. Therefore, a manual reactor trip is assumed 3000 seconds after the break is opened.

VDA [MSRT]: A minimum setpoint for the VDA [MSRT] on the affected SG is assumed and a maximum setpoint on the unaffected SGs. This maximises the steam release from the affected SG.

After the manual reactor trip, the pressure increases in the SG, until the VDA [MSRT] opens. It is assumed that the VDA [MSRT] in the affected SG opens first (at 94 bar abs). The associated MSRCV is assumed to have failed in the fully open position, which leads to an uncontrolled blowdown.

The blowdown of the affected SG results in a “low-low primary circuit pressure” signal, which initiates a partial cooldown. The VDA [MSRT] setpoints are then decreased from 97.0 bar to 61.5 bar by the end of the partial cooldown, while in the affected SG, the VDA [MSRT] setpoint is raised to 99 bar abs.

VIV [MSIV]: The VIV [MSIV] of all the SGs are closed on the low SG pressure signal (“SG pressure < MIN1”) due to the depressurisation caused by blowdown of the affected SG. In the event that this does not occur, the operator will perform this manually after the partial cooldown is complete.

MHSI: SI is actuated following a "pressuriser pressure < MIN3" signal with a setpoint of 115 + 1.5 bar. This signal initiates a partial cool down. The MHSI injects with a maximum flow rate, which begins injection at a pressuriser pressure below 97 bar. This maximises the pressure difference between the RCP [RCS] and the affected SG at the end of partial cooldown.

ASG [EFWS]: EFW flow is started to the intact SGs at the minimum flow rate and maximum temperature by the operator following the manual reactor trip. EFW to the affected SG is isolated at this time.

MSSV: To confirm opening of the MSSV is not demanded during the transient, the MSSV is assumed to have a minimum setpoint of 103.5 bar (105 - 1.5 bar).

RBS [EBS]: The RBS [EBS] pump(s) are started when the operator initiates the manual cooldown (not modelled in the calculation). The cooldown rate is determined by the number of RBS [EBS] trains available. If two trains are available, the cooldown proceeds at -50°C/hr. If only one train is available, the cooldown is limited to -25°C/hr.

6.5.2. Case 2: With LOOP

6.5.2.1. Choice of Single Failure and Preventive Maintenance

This case is studied to confirm that no SG overfilling, and thus no liquid release, occurs before leak termination occurs.

The worst case assumes neither a single failure nor a preventive maintenance. Each loss of a F1 system reduces the filling rate of the affected SG, which is the main concern for this case.

6.5.2.2. Initial State

The initial state conditions, given in Section 14.4.6 – Table 2, are chosen to maximise the initial liquid content of the affected SG, to maximise the pressure difference between the primary and secondary side, and to minimise the heat to be removed by the affected SG.

Note that the impact of an increase of the initial pressuriser level is assessed in sub-section 6.5.3.2 of this sub-chapter.

6.5.2.3. Specific Assumptions

a) Neutronic data

The core power is assumed to start at 2% full power, decreasing following the realistic decay heat curve is presented in Sub-chapter 14.1.

b) Assumptions related to control systems

Turbine: The turbine is isolated at the beginning of the transient to simulate initial hot standby conditions.

GCT [MSB]: GCT [MSB] is assumed to be lost when the fault occurs to reduce the steam release from the affected SG.

ARE [MFWS]: No main feedwater supply either by ARE [MFWS] or by AAD [SSS] is assumed when the fault occurs. This increases the amount of subcooled ASG [EFWS] injection and, thus, the liquid content of the SG.

RCV [CVCS]: The charging flow rate is maximised with 2 charging pumps in operation to maximise the pressure difference between the RCP [RCS] and the affected SG. Letdown is isolated following a “pressuriser level < MIN3” signal. The RCV [CVCS] is isolated automatically following a "SG Level > MAX2 (NR)” and “end of partial cooldown” signals.

Pressuriser heaters and sprays: Pressuriser pressure control is in operation. After the emergency power mode is reached, 576 kW of emergency power is assumed to be supplied to the pressuriser heaters. The pressuriser heaters are shut off when the pressuriser empties.

c) Assumptions related to F1 systems

Reactor Trip**:** The RT occurs following a “SG pressure > MAX1” signal, with a minimum delay to shorten the time to reach the emergency power mode, which is assumed to occur coincident with the reactor trip/ turbine trip signal.

VDA [MSRT]: A minimum VDA [MSRT] setpoint on the affected SG is assumed at 95.5 - 1.5 bar. This maximises the pressure difference between the RCP [RCS] and the affected SG. A calculation with an increased actuation pressure for the affected loop and reduced actuation pressure for the other loops, to reduce steam removal from the affected SG shows an opposite effect with a slightly reduced maximum liquid content in the SG.

After reaching the MAX2 setpoint in the affected SG, a partial cooldown at 100°C/h is performed automatically to 58.5 bar in the affected SG and 61.5 bar in the unaffected SGs. When the partial cooldown is finished, the set value for the relief train of the affected loop is manually set to 100 bar. The VDA [MSRT] set pressure increase is automatically actuated following the combination of “SG level > MAX2 (NR) and “end of partial cooldown” signals.

VIV [MSIV]: The VIV [MSIV] of the affected SG is manually closed by the operator at the end of the PC. The VIV [MSIV] would automatically close following a "SG Level MAX2 + PC finished" signal.

MHSI: The safety injection signal is actuated following a "pressuriser pressure < MIN3" signal at a setpoint of 115 + 1.5 bar. When the RCP [RCS] pressure drops below 97 bar, the MHSI injects with a maximum flow rate to maximise the pressure difference between the RCP [RCS] and the affected SG.

ASG [EFWS]: ASG [EFWS] is manually actuated immediately an emergency power mode occurs. The ASG [EFWS] control is not considered, which leads to maximum emergency feedwater injection and an automatic isolation of the system following a “SG liquid level above MAX1” signal.

MSSV: To confirm the MSSV is not demanded during the transient, the MSSV setpoint is assumed at its minimum of 105 - 1.5 bar.

6.5.3. Results

6.5.3.1. Case 1

6.5.3.1.1. From the initiating event to the controlled state

The sequence of events for Case 1 is given in Section 14.4.6 – Table 4[4].

The transient progress is shown graphically in Section 14.4.6 – Figure 1 to Section 14.4.6 – Figure 11. The results are discussed below.

[4] Case 1 is a new calculation performed in the frame of the GDA. All related data given (sequence of events, figures,…) come from new analysis results.

The break is opened at the start of the transient with an initial flow of about 28 kg/s. Pressuriser pressure decreases slowly and pressuriser level decreases steadily until the letdown flow is isolated on low pressuriser level at 1095 seconds. After letdown is isolated, the pressuriser level decreases at a slower rate. The primary circuit pressure continues to slowly decrease, which decreases the leakage flow rate and increases the RCV [CVCS] charging flow. This maintains the primary circuit pressure above the reactor trip setpoint and the system approaches a new steady state condition at a lower pressure as the RCV [CVCS] charging flow increases to match the leakage flow.

At 50 minutes after the break is opened, the reactor is tripped manually. In addition to the reactor trip, the operator performs the following actions:

- Stops MFW flow to all SGs
- Disables EFW flow to the affected SG
- Starts EFW flow to the intact SGs

Rod insertion begins 0.4 seconds after RT and the turbine is isolated 2.5 seconds after RT. Following the turbine trip, the secondary pressure increases. It is assumed that the VDA [MSRT] opens in the affected SG (at 94 bar). The MSRCV and MSRIV open. The MSRCV on the affected SG is assumed to have failed in the fully open position. This results in a rapid depressurisation of the affected SG. The depressurisation of the SG results in a cooldown of the primary circuit and a drop in primary circuit pressure. The low pressuriser pressure setpoint is reached about 50 seconds after the VDA [MSRT] in the affected SG opens and the low-low pressuriser pressure setpoint is reached about 45 seconds after that. The pressure in the intact SGs also decreases.

The decrease in primary circuit pressure results in an increase in RCV [CVCS] charging flow and a decrease in break flow. After the reactor trip, the charging flow exceeds the break flow.

The “low-low pressuriser pressure” signal activates the safety injection system and initiates an automatic partial cooldown at -250°C/hr. The partial cooldown signal is ineffective because the failed-open MSRCV in the affected SG results in a cooldown slightly in excess of -250°C/hr. The depressurisation continues until the pressure in the SGs reaches the MIN1 setpoint at 3595 seconds.

The end of partial cooldown is detected after the VIV [MSIV] isolation on “SG pressure < MIN1”, once the partial cooldown setpoint finally reaches 61.5 bar (at 3666 seconds). Pressure continues to decrease in the affected SG. Finally, the MSRIV on the affected SG is shut, isolating the affected SG.

Because of the RCV [CVCS] charging flow, the level as well as the pressure in the affected SG increases. The affected VDA [MSRT] opens again (at 8382 seconds), despite the setpoint increase. The MSRCV being still stuck open, the pressure decreases in the affected SGs. The MSRT is finally closed on the MIN3 setpoint (at 8818 seconds).

At 8418 seconds, the level in the affected SG reaches the MAX2 setpoint. An automatic signal is generated to stop the RCV [CVCS] charging flow on “Partial Cooldown Complete and SG Level > MAX2” and the RCV [CVCS] charging flow is isolated after a 41.5 second delay. After the RCV [CVCS] charging is isolated, the primary circuit pressure begins to decrease. At 14244 seconds, the differential pressure across the break is calculated to be less than one bar and the leak is considered to be halted. The transient is terminated 500 seconds later.

During this transient, where few operator actions are modelled, 218 tons of steam flows through the VDA [MSRT] of the affected SG (steam coming from the 4 SGs). About 450 kg of liquid is discharged from the VDA [MSRT] of the affected SG in the form of entrained moisture. The minimum liquid mass contained in the affected SG is 26 tons.

6.5.3.1.2. From the controlled state and safe shutdown state

The emergency operating procedures will be followed by the operator immediately following the manual reactor trip so as to reach the safe shutdown state. The safe shutdown state is defined as a state where the affected SG is isolated and at least one RIS/RRA [SIS/RHRS] train is connected to the RCP [RCS]. One out of four LHSI trains in RHR mode is sufficient to provide the required heat removal. The connection conditions are:

- RCP [RCS] hot leg pressure < 32 bar and,
- RCP [RCS] hot leg temperature < 180°C and,
- ΔTsat[5] and Reactor Pressure Vessel Level (RPVL) consistent with LHSI in RHR mode suction conditions from the hot leg.

The sequence of actions to be performed by the operator to reach the safe shutdown can be divided into two successive phases: boration and RCP [RCS] cooldown, and final depressurisation. While performing these operations, the operator is required to monitor key safety functions (e.g. saturation margin) and some plant parameters, (e.g. pressuriser level).

Boration and RCP [RCS] Cooldown

The boration and RCP [RCS] cooldown actions are performed by the operator after reactor trip. The RBS [EBS] delivers a constant boration flow rate to the RCP [RCS], providing the negative reactivity required to reach the safe shutdown state. The allowed cooldown rate depends on the number of available RBS [EBS] trains:

- 25°C/h with one RBS [EBS] train in operation,
- 50°C/h with two RBS [EBS] trains in operation.

The RCP [RCS] cooldown is performed using the unaffected steam generators. This cooldown occurs with the MHSI operating to prevent perturbing the pressure balance between primary side and the affected SG.

Final Depressurisation

At the end of the RCP [RCS] cooldown phase, the RCP [RCS] pressure is higher than the LHSI in RHR mode maximum connecting pressure of 32 bar. If the affected SG level is too high, the operator first opens the transfer line between SGa and its partner SG to limit risks of water hammer in the SGa steam line. This also prevents overfilling of the affected SG and large activity release to the atmosphere.

Once the level in the affected SG falls below MAX2 (NR), the affected SG VDA [MSRT] is opened. This allows the depressurisation to 32 bar.

[5] $\Delta T_{sat} = T_{sat}$ (hot leg pressure) - T_{co} , with T_{co} = core outlet temperature

Safety Function monitoring

During the operations performed to reach the safe shutdown, the operator is still required to monitor some key plant parameter such as saturation margin and pressuriser level. The operator could then be requested to operate the pressuriser normal spray, shut the MHSI one-by-one, etc. in order to reach the RHR connection conditions These operations are aimed at avoiding opening of the second VDA [MSRT].

6.5.3.2. Case 2

Case 2 is not recalculated in the PCSR. The capacity to reach the controlled state and to terminate the SGTR leak while meeting the relevant criteria is derived from the results of the Case 2 analysis performed in BDR-99, discussed in Appendix 14B. The criteria covered are steam relief to the atmosphere such that radioactive releases are significantly below the regulatory limits, no MSSV demand, and no SG overfilling. This assessment relies on the fact that differences between the EPR_{4500} characteristics for the PCSR and EPR_{4900} characteristics of BDR-99 are very similar when considering SGTR mitigation. There are no differences which could significantly modify the conclusions of the BDR-99 analysis.

6.5.3.2.1. Case 2: EPR_{4900} Accident Analysis of BDR-99 (1 SGTR, 2%FP, with LOOP, short term phase up to leak cancellation)

The results of the BDR-99 case 2 analysis, performed with the S-RELAP5 code, are provided in section 2.17.5.3 of Appendix 14B. They are summarised as follows:

- The amount of steam discharged from the affected SG to the atmosphere is approximately 35 te.
- Overfeeding of the affected SG is prevented with a significant margin, by automatic F1A actions and systems.
- The RCV [CVCS] charging flow must be cut off by the operator after 30 minutes, to prevent SG overfilling.

The sequence of events for Case 2 is given in Section 14.4.6 – Table 5.

The EPR_{4500} is quite similar to the EPR_{4900} regarding SGTR relevant parameters as shown in Section 14.4.6 – Table 6.

EPR_{4500} provides some improvement compared to EPR_{4900} by having automatic isolation of the RCV [CVCS] charging line following a “SG level > MAX2” signal. This is performed manually for EPR_{4900}. This allows isolation before the operator grace period of 30 minutes has expired and increases the reliability of this countermeasure, which is needed in the relatively short term.

The two notable differences between EPR_{4500} and EPR_{4900} are:

- the power level, 8% less on EPR_{4500}, reduces the steam release. The power level difference has no significant impact on the overfilling prevention as the major mitigation countermeasures are based on SG level setpoints. This mainly results in different timings for the progression of the transient.

- The MHSI injection has a delivery pressure 5 bar higher for the EPR_{4500} compared to EPR_{4900}[6]. As SG pressure at the end of partial cooldown is unchanged, the time delay to terminate the SGTR leak is slightly longer in EPR_{4500} when compared to EPR_{4900}. This slightly increases the SG level increase in the affected SG at the time the SGTR flow is terminated.

It can be concluded, without the need for a dedicated calculation, that the controlled state and the decoupling criteria associated with the short term phase of the SGTR transient are achieved in EPR_{4500}. This is based upon the results of the Case 2 analysis in BDR-99 and on the similar characteristics of EPR_{4500} and EPR_{4900} for SGTR-relevant parameters,

A further increase of the initial pressuriser level would have no significant impact as far as the release to the atmosphere is concerned. The partial cooldown starts very soon after the initiating event and the variation of the initial pressuriser level has very little impact on the depressurisation and cooling phenomena.

6.6. LONG TERM STUDY

6.6.1. Case 3 – With LOOP

6.6.1.1. Choice of Single Failure and Preventive Maintenance

The single failure chosen reduces the capacity of the F1 systems to cool the RCP [RCS] and SG to the LHSI in RHR mode connection conditions. This failure can be either on:

- one RBS [EBS] pump to slow the boration,
- one VDA [MSRT] to slow the cooldown and SG depressurisation,
- one ASG [EFWS] pump to slow the cooldown,

There is no preventive maintenance case that would further reduce the cooldown, boration, or increase the activity release for the transient.

6.6.1.2. Conditions at the beginning of the long term phase

The long term phase starts at the end of the short term phase presented in Case 2. The conditions at the beginning of long term phase correspond to a maximum level in the affected SG and maximum activity transferred through the leak.

6.6.1.3. Specific Assumptions

The assumptions for the systems described in Case 2 remain valid. The additional assumptions and characteristics for the long term phase are the following:

MHSI: This is maintained until the end of RCP [RCS] cooldown phase, when RCP [RCS] temperatures in unaffected loops are less than 180°C. At this point it is shut down by the operator to allow the depressurisation of the RCP [RCS] and the affected SG or inventory reduction of SGa via the blowdown system.

[6] Comparison performed on the basis of the plant characteristics given in Appendix 14B.0.2 – Table 13 – EPR™ 4900MW and plant characteristics given in Sub chapter 14.1 – Table 13.

Accumulators: These are manually isolated when the MHSI is shut down. No accumulator injection occurs during the transient.

LHSI: Injects to the RCP [RCS] when the RCP [RCS] pressure reaches 20 bar.

RBS [EBS]: At the start of the plant cooldown, at 2500 seconds, one of two extra boration system pumps is started by the operator to provide boration. The pump is stopped if the cooldown is finished or if a connection from the affected SG to the partner SG via the transfer line is opened to reduce the inventory in SGa. The minimum capacity of the RBS [EBS] tank is sufficient to borate the RCP [RCS] inventory from a 10 ppm initial concentration to the 680 ppm required for LHSI in RHR mode conditions under EOC UO_2 assumptions. The concentrations quoted above are for natural boron.

VDA [MSRT]: For the RCP [RCS] cooldown phase, the VDA [MSRT] setpoint on the unaffected SGs is reduced to provide a 25^{o}C/hr cooldown gradient. This is required if only one RBS [EBS] pump is available. With the VDA [MSRT] capacity of 50% of nominal flow rate at 100 bar, this gradient can be maintained to a secondary pressure of about 3.5 bar. Subsequently, the VDA [MSRT] are fully open and the cooldown gradient decreases.

The VDA [MSRT] on the affected SG is opened by 20% to depressurise the RCP [RCS] and the affected SG at the end of the cooldown phase. This is performed once the inventory of the affected SG has been reduced using the transfer line.

ASG [EFWS]: In the long term phase, the ASG [EFWS] on the unaffected SGs is assumed to be controlled to the nominal wide range level measurement. To prepare the affected SG level reduction, the liquid level of SG4 is lowered to about 22% NR by the operator.

Transfer line: A transfer line connection from the affected SG to the partner SG is opened manually by the operator to lower the liquid level in the affected SG, before the final depressurisation is performed. This transfer line is the F1 part of the SG blowdown system.

6.6.2. Results

The sequence of events for Case 3 is given in Section 14.4.6 – Table 5.

Case 3 is not recalculated in the PCSR. The capability to reach the Safe Shutdown State and to meet the relevant criteria is derived from the results of the Case 3 analysis performed in BDR-99 and presented in Appendix 14B. The criteria considered are the steam relief to the atmosphere to keep radioactive releases significantly below the regulatory limits, no MSSV demand, and no SG overfilling. This assessment relies on the similar characteristics of the EPR_{4500} and EPR_{4900} relevant to the SGTR mitigation. There are no differences which could significantly modify the conclusions of the BDR-99 analysis.

The following assessment refers to the long term phase with the short term phase having been addressed in sub-section 6.5.3.2 of this sub-chapter.

6.6.2.1. Case 3: EPR_{4900} Accident Analysis of BDR-99 (1 SGTR, 2%FP, with LOOP, long term phase up to LHSI in RHR mode connection)

The results of the Case 3 analysis for BDR-99, performed with the S-RELAP5 code are presented in section 2.17.6.2 of Appendix 14B. They are summarised below:

- RCS Boration: The boron concentration in the primary coolant required for subcriticality at safe shutdown state conditions is reached at approximately 80 minutes, about 40 minutes after the start of RBS [EBS].

- Radioactive Releases: Before depressurisation of the RCP [RCS] and the affected SG at the end of the plant cooldown and boration phase, the liquid level in the affected SG is lowered by opening the transfer line to the partner SG. Thus no liquid is discharged to the atmosphere during the depressurisation of the affected SG via the VDA [MSRT]. When the depressurisation is stopped at an SG pressure of 20 bar, sufficient for LHSI in RHR mode operation, only approximately 42 te of contaminated steam have been discharged during this phase through the VDA [MSRT] of the affected SG. If the affected SG is depressurised to approximately 6 bar, 57 te of contaminated steam are discharged.

- Prevention of SG Overfilling: About 20 minutes after SGTR initiation, the narrow range level measurement of the affected SG reaches the MAX2 limit and partial cooldown is actuated. During the partial cooldown, the liquid inventory of the SG decreases. Once the partial cooldown is completed and the operator initiates isolation of the affected SG after a 30 minute grace period the level is 17.5 m. With steam-side isolation of the SG and VDA [MSRT] setpoint increase to 98 bar, the SG pressure increases slowly until pressure equilibrium with the primary side is reached. This equalisation is caused by the continued SGTR leakage. In this phase, the affected SG is filled to 20.8 m, with the top of the SG at 21.2 m. Subsequently, small condensation effects in the SG lead to a further liquid level increase, reducing the steam volume to 5 m^3 at 5.6 hours when the cooldown and boration phase is finished.

- Prevention of SGTR Backflow into RCP [RCS]: Prior to 5.6 hours, the risk of backflow is prevented by MHSI operation. At 5.6 hours, the operator opens a transfer line connection to an unaffected SG to reduce the liquid content of the affected SG. Within 2.8 hours, the liquid mass of the affected SG is lowered from 180 to about 140 te via the transfer line. This reduces the collapsed liquid level to about 17.5 m. At this point, the final depressurisation of the affected SG and the RCP [RCS] is performed using the VDA [MSRT] of the affected loop, without SG overfilling and without SGTR backflow.

- MSSV Setpoint: the MSSV are not demanded during the transient.

6.6.2.2. Case 3: Extrapolation of BDR-99 Results to EPR_{4500}

The results of Case 3 in BDR-99 for EPR_{4900} demonstrate that the F1A systems and functions are sufficient to reach the Safe Shutdown State without SG overfilling, without SGTR backflow and without an MSSV demand.

- RBS [EBS] performs the required F1 boration,

- ASG [EFWS] and VDA [MSRT] perform the required F1 heat removal and RCP [RCS] cooldown to LHSI in RHR mode connecting conditions

- VDA [MSRT] and SG blowdown perform the required F1 depressurisation of the RCP [RCS] and the affected SG down to LHSI in RHR mode connecting conditions.

Section 14.4.6 – Table 6 shows that, compared to EPR_{4900} characteristics, EPR_{4500} characteristics are similar for RCP [RCS] boration, RCP [RCS] cooldown, RCP [RCS]/SG depressurisation, and beneficial for heat removal.

The only potentially detrimental effects in EPR_{4500} compared to EPR_{4900} for the long term phase of transfer to LHSI in RHR mode are:

- The slightly longer duration of the transfer phase, due to a higher SG pressure at the initial hot shutdown conditions.

6.7. IMPACT OF THE MODIFICATION TO THE PARTIAL COOLDOWN RATE AND SAFETY CLASSIFICATION UPGRADE OF THE NORMAL SPRAY OPERATIONS

Consequences of the modification of the partial cooldown rate (Case 2 and Case 3)

In some of the current PCSR studies (Case 2 and Case 3), the cooldown rate of the partial cooldown is 100°C/h. In order to increase margins to safety criteria in LOCA studies, this rate is increased to 250°C/h while keeping the same other characteristics. It is performed using the four SG, down to 60 bar on the secondary side. Due to this rate increase, the related SG pressure drop signal setpoint is also increased from 2 bar/min to 5 bar/min.

In the frame of SGTR studies, the impact of the cooldown rate is not significant. The amount of energy removed from the RCP [RCS] remains the same whatever the RCP [RCS] cooldown rate. The only impact on the analysis is on the timescale for specific events.

The modification of the SG pressure drop signal will impact only the case with the single failure on the control valve of the SGa VDA [MSRT]. This modification will lead to a later isolation of SGa without modifying the amount of radioactive products released to the atmosphere. Whatever the SG pressure drop signal, the isolation pressure of the VDA [MSRT] remains at 40 bar and SGa has to depressurise to this isolation pressure. Only the timescale will be significantly different.

Impact of the safety classification change of the normal spray operations

In the current PCSR studies, the normal spray operations are not F1B classified. Consequently, it was not considered in the recovery route to reach the safe shutdown state.

The classification of the normal spray operations as F1B allows the operator to use a classified means to depressurise the RCP [RCS] in parallel with SGa, independent of the PSV. In the frame of the SGTR studies, the depressurisation of the RCP [RCS] relies on the connection between the primary side and the SGa. None of the depressurisation means of the pressuriser are considered.

6.8. CONCLUSION

The present analysis of the single tube 2A-SGTR accident shows that despite the worst single failure and preventive maintenance:

- the controlled state can be reached using only the following systems:
  - Manual RT 50 minutes after activity detection,
  - ASG [EFWS] and VDA [MSRT], including partial cooldown, for RCP [RCS] heat removal,

    - o RCV [CVCS] charging for RCP [RCS] coolant inventory control.

- the safe shutdown state is reached using only F1A and F1B systems:

    - o VIV [MSIV] closure, ARE/AAD/ASG [MFWS/SSS/EFWS] isolation, VDA [MSRT] setpoint increase for affected-SG isolation,
    - o MHSI and LHSI for RCP [RCS] coolant inventory control,
    - o ASG [EFWS] and VDA [MSRT] for RCP [RCS] cooling,
    - o RBS [EBS] for boration,
    - o SG transfer line connection for RCP [RCS] and affected SG depressurisation,
    - o LHSI in RHR mode for long term heat removal.

The total amount of contaminated fluid released to the atmosphere from the affected SG, assuming the worst single failure and preventive maintenance, is:

- At full power, with reactor coolant pumps on, and no LOOP:

    - o 218 tons of steam released directly to the atmosphere from the affected VDA [MSRT], and
    - o no overfilling or dry out takes place during the transient. No liquid is released to the atmosphere (except in the form of entrained vapour moisture). Thus, for the mass releases, only the entrained moisture in the vapour need be taken into account.

NB: The radioactive steam releases through the affected VDA [MSRT] are limited especially as due to the VIV [MSIV] being closed following the partial cooldown, the steam released comes from all four SGs.

The affected SG pressure has been kept below or equal to the RCP [RCS] pressure during the transient. Thus, any SGTR reverse flow is negligible and there is no criticality issue. The core remains covered throughout the transient, so the core cooling is never impaired and no clad heat-up is experienced.

Therefore, all the decoupling criteria and targets are met.

6.9. SYSTEM SIZING

The maximum MHSI pump delivery pressure is sized for the SGTR transient to allow leak termination with no overfilling of the affected SG.

SECTION 14.4.6 – TABLE 1

Initial Conditions for SGTR Accident Case 1

Parameter	Value
Initial reactor power	4590 MW
Initial average RCP [RCS] temperature	312.4 - 2.5 = 309.9°C
Initial reactor coolant pressure	155 + 2.5 = 157.5 bar
Reactor coolant flow rate	26986 m^3/hr/loop
Bypass flow rate • Total • Dome • Guide Tubes • Reflector	 5.53% 0.51% 3.01% 2.01%
Pressuriser level	56% + 5% = 61%
Initial unaffected SG level	49% + 10.5% = 59.5% NR
Main feedwater flow rate	649 Kg/s
SG Pressure (Exit)	71.9 bar
SG Saturation Pressure	72.1 bar
SG Pressure (MFW)	72.2 bar
Initial ARE [MFW] Enthalpy	991.18 KJ/kg
Initial ARE [MFW] Temperature	230°C
Steam line pressure (VDA [MSRT] connection)	71.3 bar
Steam header pressure	69.0 bar
RCV [CVCS] Injection temp. (letdown on)	270°C
RCV [CVCS] Injection temp. (letdown off)	50°C

SECTION 14.4.6 - TABLE 2

Plant Initial Conditions – EPR_{4900} Case 2

Parameters	Limiting values used
Reactor coolant system	
Initial reactor power (% of nominal power)	2
Reactor thermal power (MW)	85
Reactor coolant pump thermal power (MW)	app. 30
Initial average RCP [RCS] temperature (°C)	303.3 – 2.5 = 300.8
Initial reactor coolant pressure (bar)	155 + 2.5 = 157.5
Reactor coolant flow (kg/s)	app. 22135 (thermal hydraulic)
Pressuriser water volume (m3/%MR)	24.9 / 34 (nominal + 5% MR)
Steam generators	
Initial steam pressure (bar)	code calculation
Initial SG level (m)	16.05 (nominal + 5% NR)
Feedwater	
Main feedwater flow (% of nominal flow)	2
Initial ARE [MFWS] temperature (°C)	123

SECTION 14.4.6 - TABLE 3

Setpoints for Case 1

Function	Setpoint	Included bias
Intact VDA [MSRT]s	97 bar	+1.5
Affected VDA [MSRT]	94 bar	-1.5
PSV 1	175 bar	
PSV 2	178 bar	
PSV 2, 3	181 bar	
MSSVs	103.5 bar	-1.5
Low pressuriser pressure (MIN2)	133.5 bar	-1.5
Low-low pressuriser pressure (MIN3)	116.5 bar	+1.5
Low pressuriser level (MIN3)	12 %	
High SG level (MAX1)	71% NR	+2
High SG level (MAX2)	87% NR	+2
High SG level (MAX1)	91% WR	+2
High SG pressure dP/dt	-5 bar/min	-7 bar
Low SG level (MIN2)	38% WR	-2
Low SG pressure (MIN1)	51.5 bar	
Low SG pressure (MIN3)	38.5 bar	
Low ΔP_{SAT}	10 bar	

SECTION 14.4.6 - TABLE 4

Sequence of Events for Case 1

Event	Time (s)
2A SGTR occurs at bottom of tube bundle – cold side	0
Pressuriser level < MIN3	1053
Pressuriser heaters off	1055
Isolate letdown	1095
Reactor trip signal (manual) Turbine trip signal High load ARE [MFWS] stopped to all SGs	3000
Operator isolates the affected SG	3000
Operator actions: Low load ARE [MFWS] stopped to all SGs ASG [EFWS] 3 disabled in affected SG3 ASG [EFWS] 1, 2 & 4 flow starts	3000
Rod insertion begins	3000.4
Turbine isolated	3002.5
EFW 1, 2 & 4 flow starts	3015.9
VDA [MSRT] opening on high SG pressure (in affected SG3) at 94 bar, MSRCV failed open	3038
Low pressuriser pressure (133.5 bar)	3091
Low-low pressuriser pressure (116.5 bar) SI Signal Start partial cooldown	3135
MHSI/LHSI pumps started	3151
SG3 pressure < MIN1	3595
VIV [MSIV] 1, 2, 3 & 4 shut	3601
Partial cooldown complete in unaffected SG	3667
SG3 Pressure < MIN3	3833
SG3 MSRIV shut	3838
MSRT opening on high SG pressure (in affected SG3) at 99 bar, MSRCV failed open	8384
Partial cooldown complete and SG3 level > MAX2	8418
Stop RCV [CVCS] charging flow	8460
SG3 pressure < MIN3	8818
SG3 MSRIV shut	8824
Leak cancelled	14245
Transient terminated	14745

SECTION 14.4.6 - TABLE 5

Sequence of Events – EPR_{4900} Cases 2 and 3

0,0 s	SGTR occurrence, break area (2A) initial leak flow rate 25,1 kg/s feedwater flow rate steps to zero main steam bypass flow rate steps to zero
237 s	Main steam pressure > 93,0-1,5 bar; reactor trip turbine trip signal actuation of emergency power mode, coast down of RCPs, actuation of MS relief valves, actuation of emergency feedwater pumps (105 Mg/h per SG, 50°C)
265 s	Measured PZR water level more than 80 cm lower than setvalue; start of second CVCS pump (20 kg/s injection, 10 kg/s letdown)
598 s	Water level in affected SG (wide range) > 17,2 + 0,38 m; isolation of EFWS of affected loop
862 s	Measured PZR water level < 2,09 m (10 m^3); isolation of CVCS letdown cut off of PZR heaters
1236 s	Water level in affected SG (narrow range) > 18,1 + 0,14 m; actuation of partial cooldown (100K/h from 93,0 - 1,5 bar to 60 bar)
1810 s	SGTR leak flow compensated by CVCS makeup
	controlled state reached
1815 s	PZR pressure < 115,0 + 1,5 bar; ECC signal, actuation of MHSI / LHSI
2385 s	Partial cooldown finished
2500 s	Start of manual actions • cut-off of CVCS pumps (boration via CVCS not available) • steam side isolation of affected SG by closing the MSIV and increase of MSRV setvalue to 98 bar • start of EBS (1 pump with 2,8 kg/s available) • start of cooldown with 25 k/h
4830 s	Boron concentration in all parts of RCS > 620 ppm (required save shutdown concentration for EOC MOX)
6500 s	SGTR leak flow cancelled due to pressure equilibrium between RCS and SGa, i.e.
	End of short-term and beginning of long-term phase
16000 s	Manual isolation of EFWS supply to SG4 (decrease of water level in SG4) to prepare emptying of affected SG1 via blowdown line

- continued on next page -

20000 s	Steam side isolation of SG4 and opening of blowdown line between affected SG1 and SG4
20000 s	Manual isolation of MHSI and ACCUs
23600 s	Cut-off of EBS to finally stop SGTR leak rate
30000 s	Water level in affected SG 17,5 m (SG4 13 m); start of final depressurization by 20% opening of MS relief valve of affected SG
31710 s	Primary pressure < 20 bar; LHSI starts loop injection
35000 s	End of calculation; RCS pressure 20bar, RCS temperatures < 180°C RHR-Conditions reached
	safe shutdown state reached

Sequence of Events - Case 2/3
(Short & Long Term With LOOP)

SGa : affected SG
SG2 : related to the unaffected SG attached to PZR loop (used for transfer)
SG1 & SG4 : related to the unaffected SG

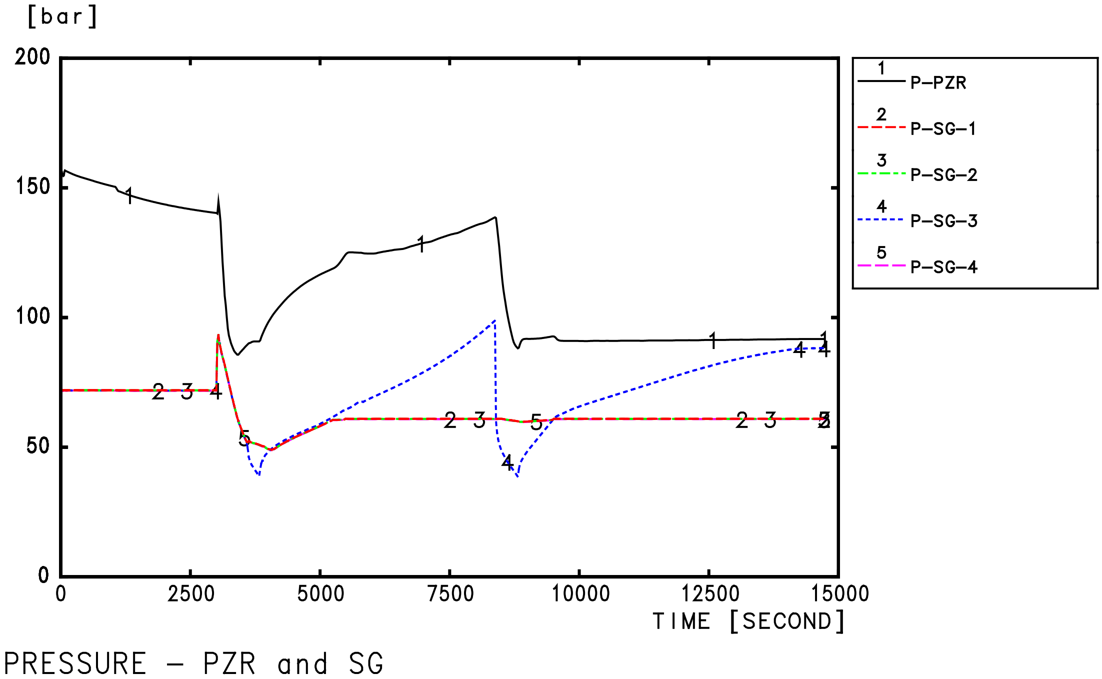
SECTION 14.4.6 - TABLE 6

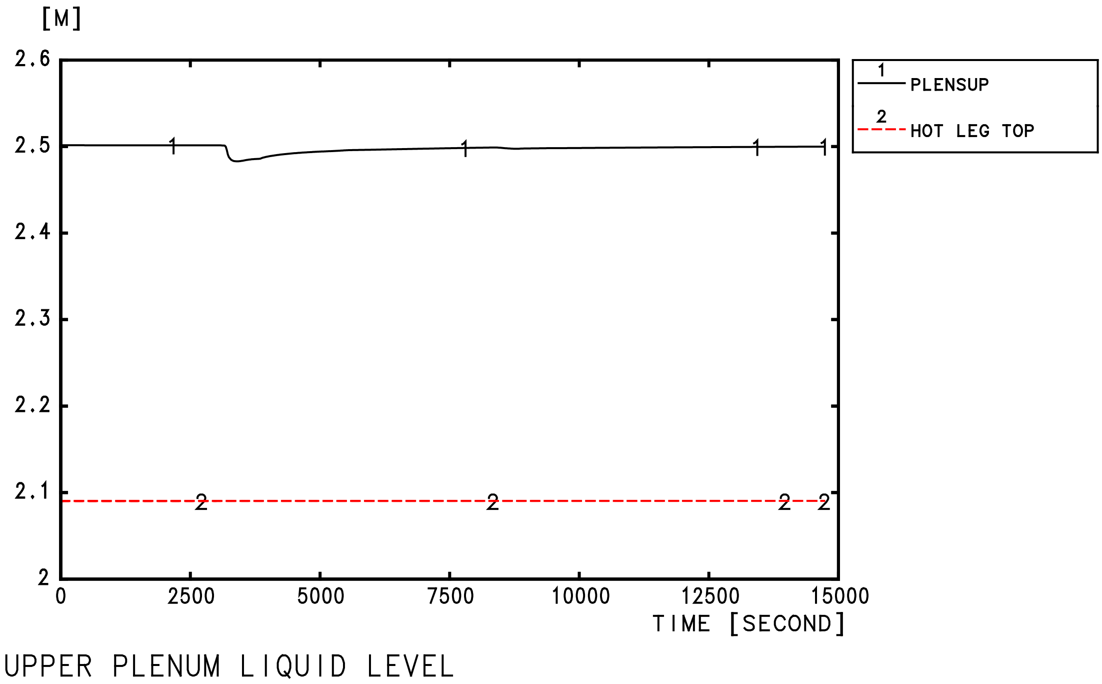
Comparison of EPR_{4900} to EPR_{4500} SGTR- Relevant Parameters

Parameter		EPR_{4900}	EPR_{4500}	Impact Assessment
Main Geometrical Data				
SG	free volume	247 m^3	238 m^3	Less than 5% difference
	heat transfer area	8171 m^2	7960 m^2	No significant impact on SG overfilling.
ASG [EFWS]	total effective liquid volume	1500 m^3	1680 m^3	Higher heat removal capacity.
RBS [EBS]	total tank volume, minimum	54 m^3	72 m^3	Increased capacity of boron injection. In the studies a value of 54 m^3 is credited conservatively.
Plant Initial Conditions				
Core Power	2% FP	98 MW	90 MW	About 8% less power in EPR_{4500} compared to EPR_{4900}.
Pressuriser level	0% FP	28%R, 21 m^3	31%R, 23 m^3	No significant impact on SG overfilling.
SG liquid mass	0% FP	116.3 t	87.6 t	More margin with respect to SG overfilling.
Protection System				
SG level	MAX2	18.1 m, 84%NR	18.0 m, 85%NR	Provides similar performance of SGTR mitigation with respect to countermeasures involving SG level.
	MAX1	17.2 m, 71%NR, 89%WR	17.0 m, 69%NR, 89%WR	
	NOM	16.2 m, 56%NR	15.7 m, 49%NR	
MSSV		102.5 bar	105.0 bar	Absolute values increased by about 2.5 bar, which is beneficial for limitation of SG overfilling.
VDA [MSRT]		93.0 bar	95.5 bar	
nominal at 100% FP		74.6 bar	78.0 bar	
Pressuriser Pressure	MIN2	135 bar	135 bar	
	MIN3	115 bar	115 bar	

SECTION 14.4.6 - FIGURE 1

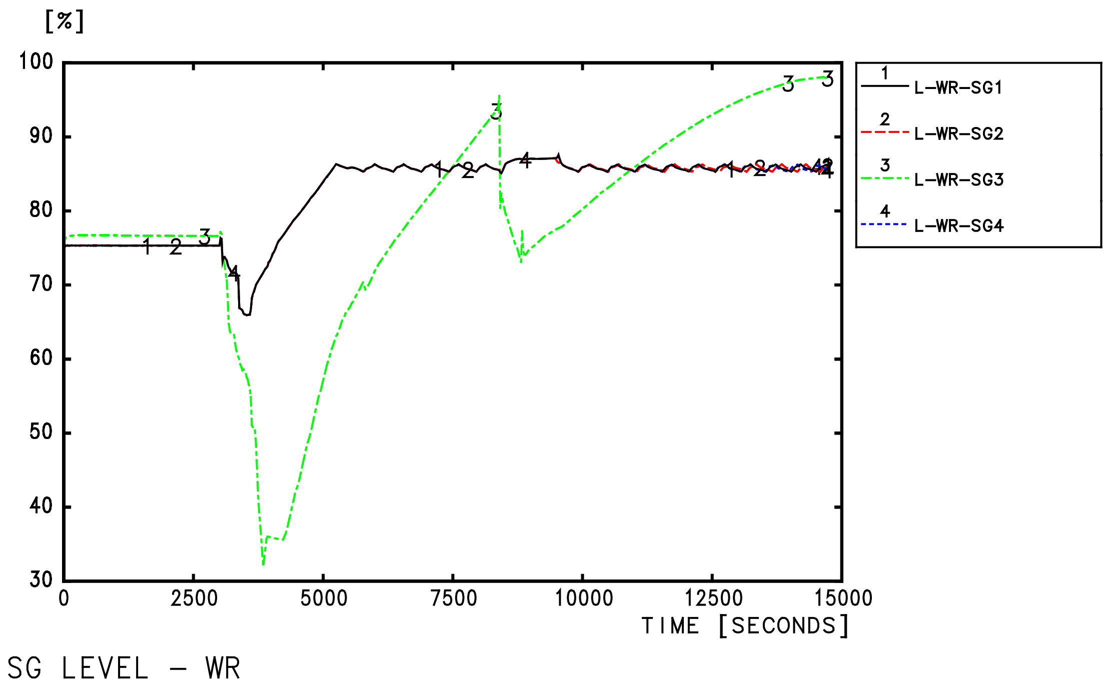
Case 1 - Primary and Secondary Pressures – Upper Plenum Level

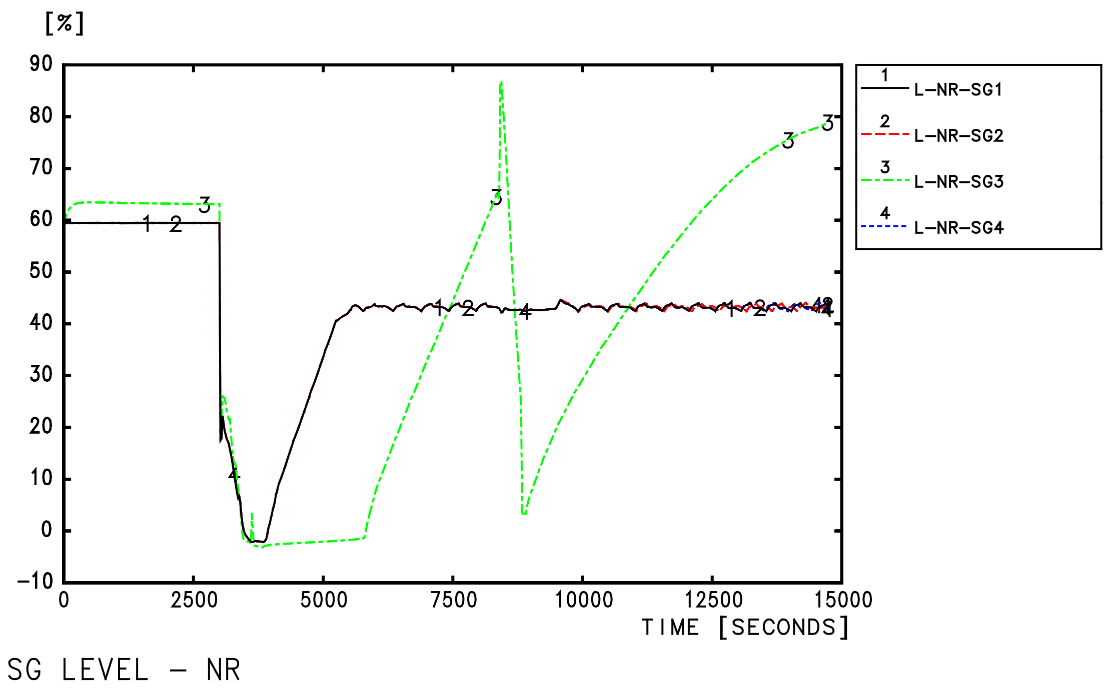




SECTION 14.4.6 - FIGURE 2

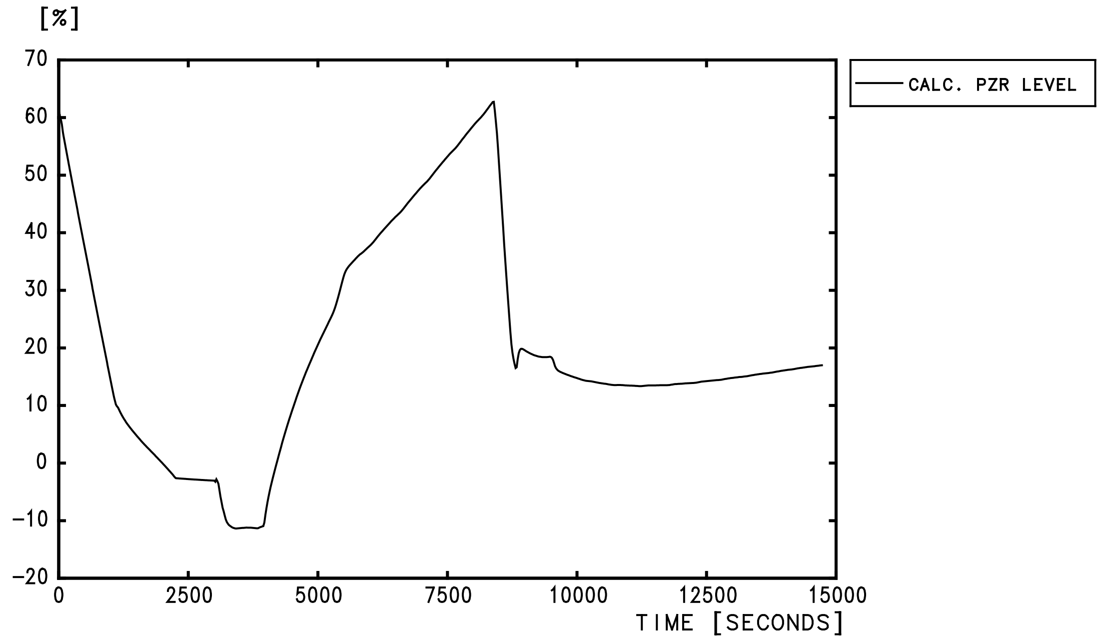
Case 1 - Steam Generator Levels



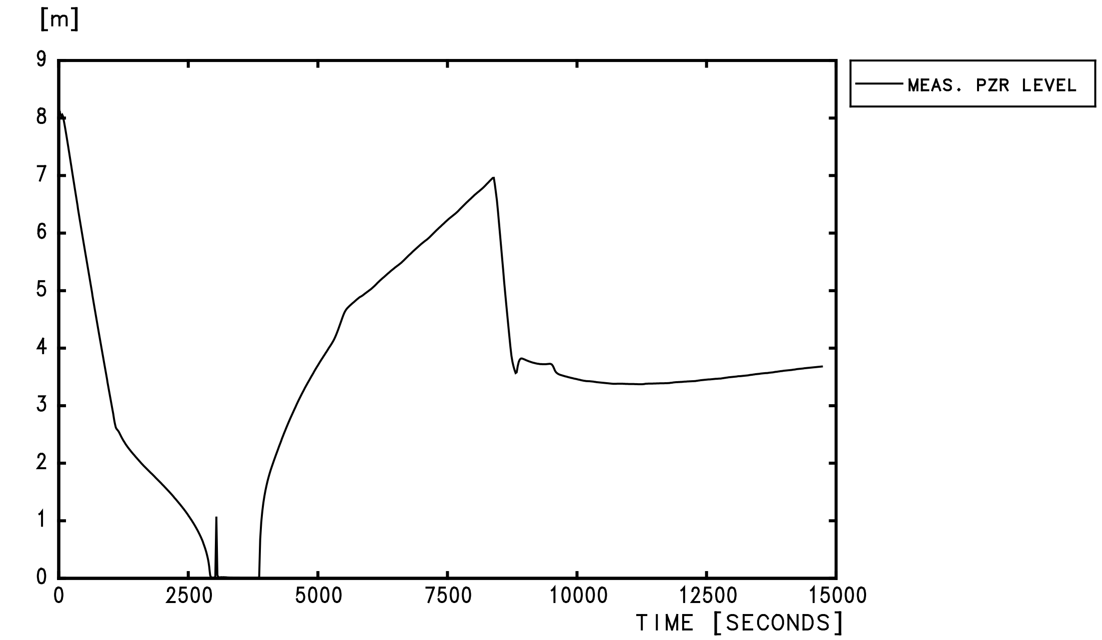


SECTION 14.4.6 - FIGURE 3

Case 1 - Pressuriser Level



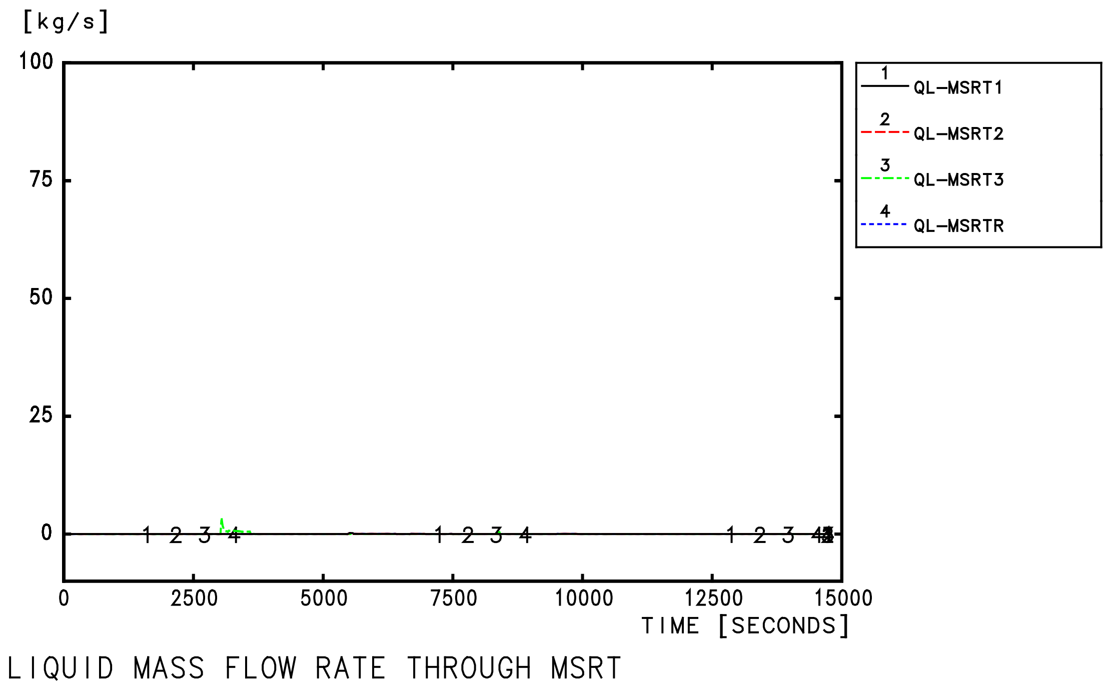
CALCULATED PRESSURIZER LEVEL

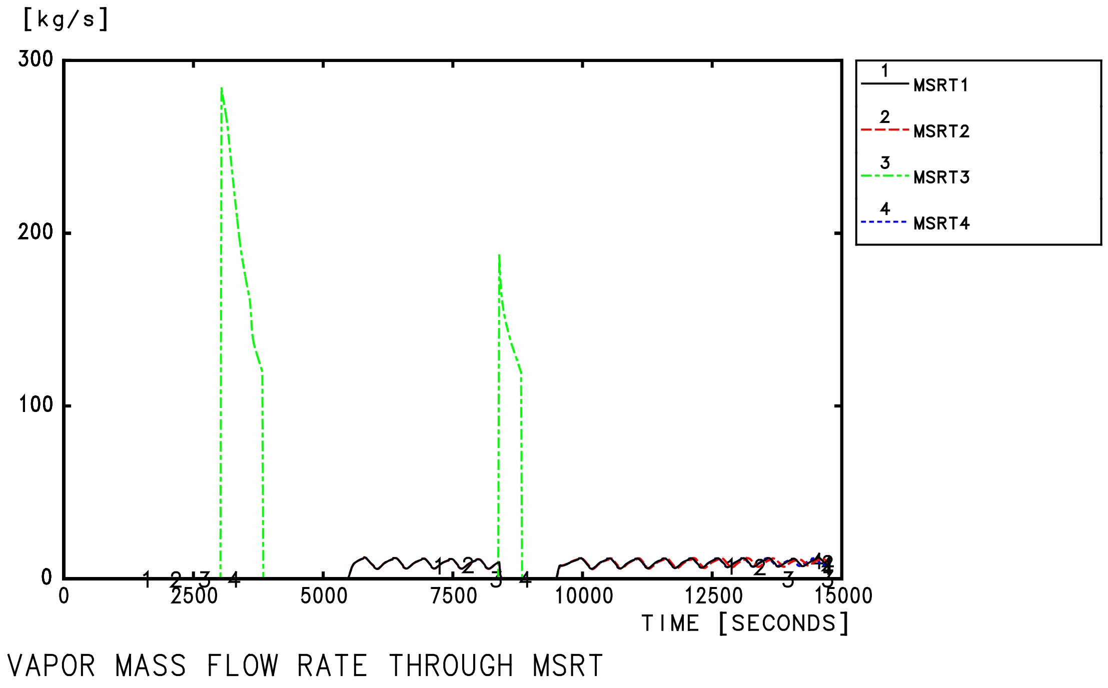


CATHARE MEASURED PRESSURIZER LEVEL

SECTION 14.4.6 - FIGURE 4

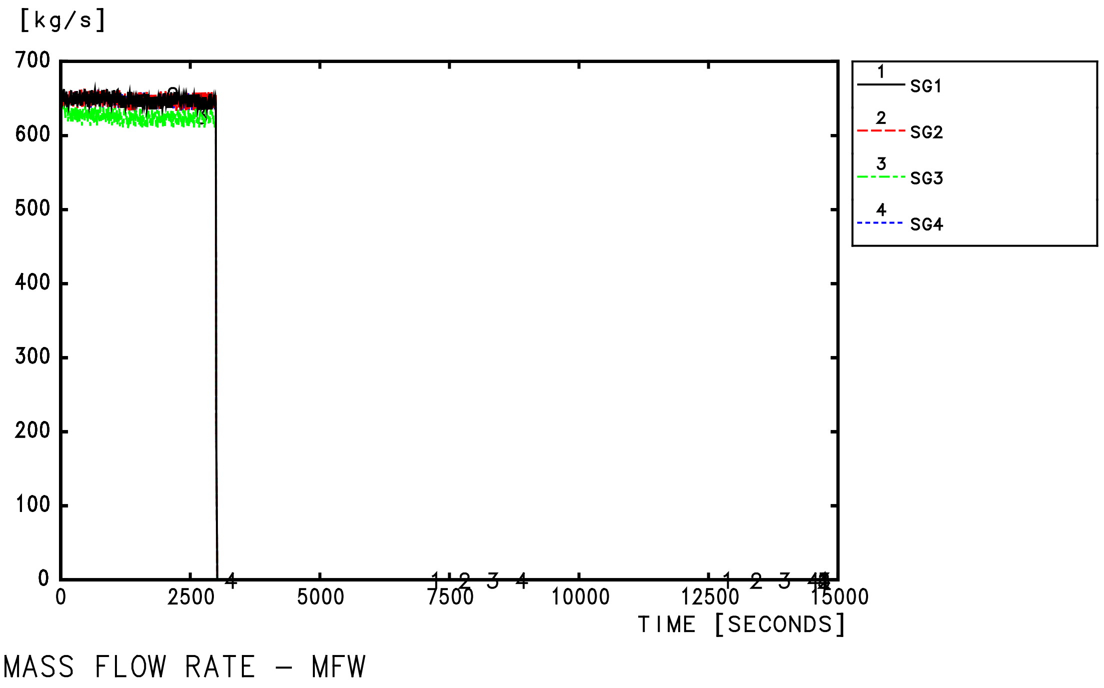
Case 1 - VDA [MSRT] Liquid and Steam Flow

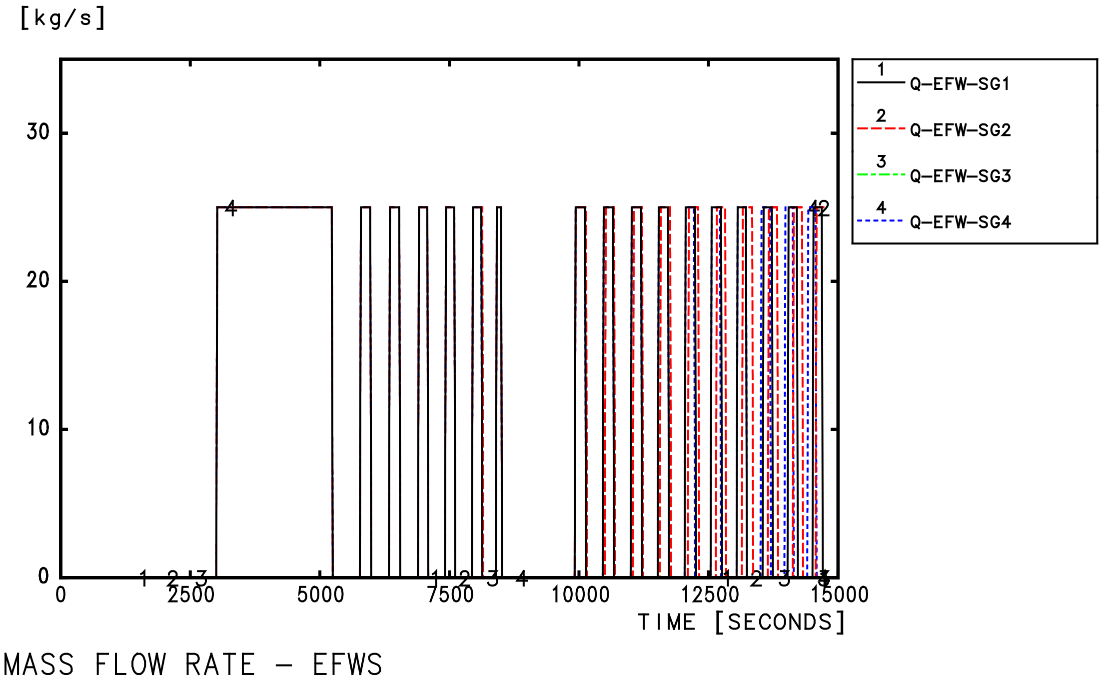




SECTION 14.4.6 - FIGURE 5

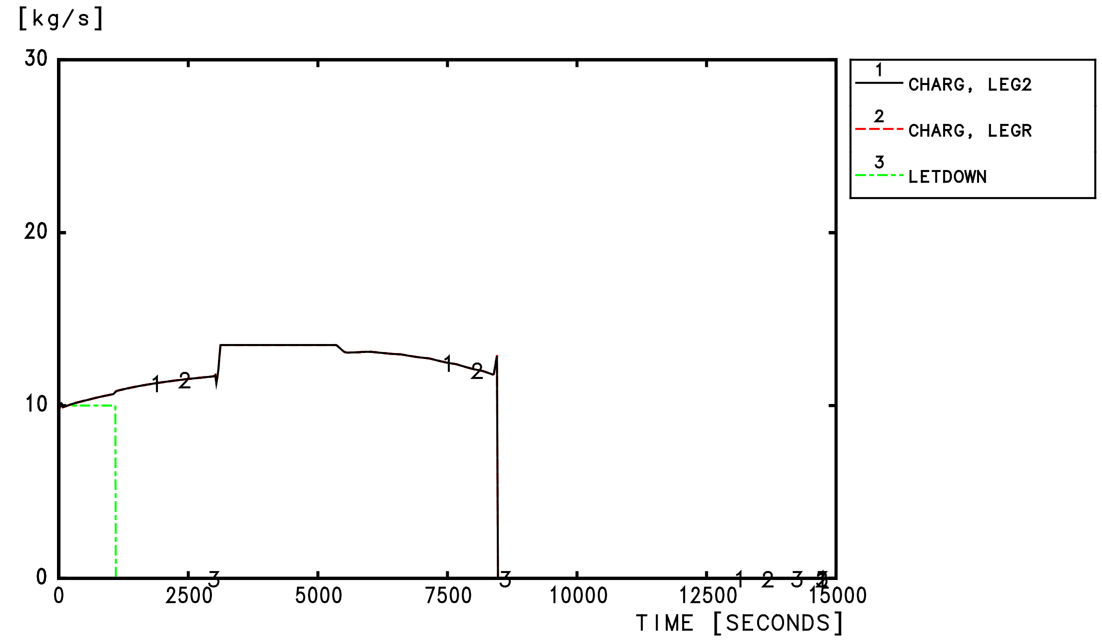
Case 1 - Main and Emergency Feedwater Flows



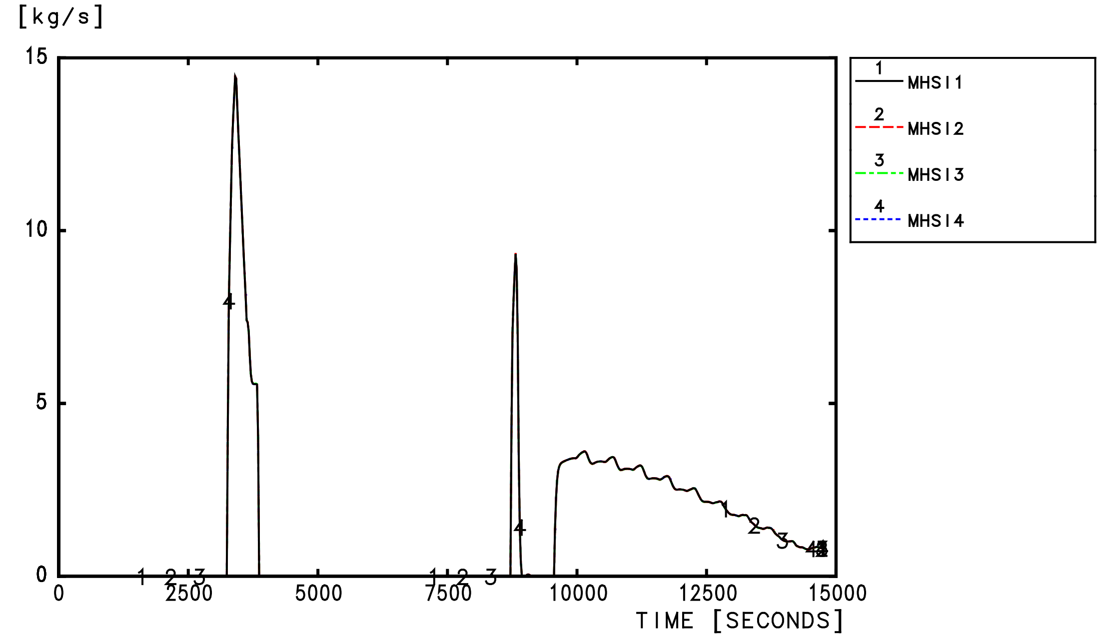


SECTION 14.4.6 - FIGURE 6

Case 1 - RCV [CVCS] and MHSI Flows



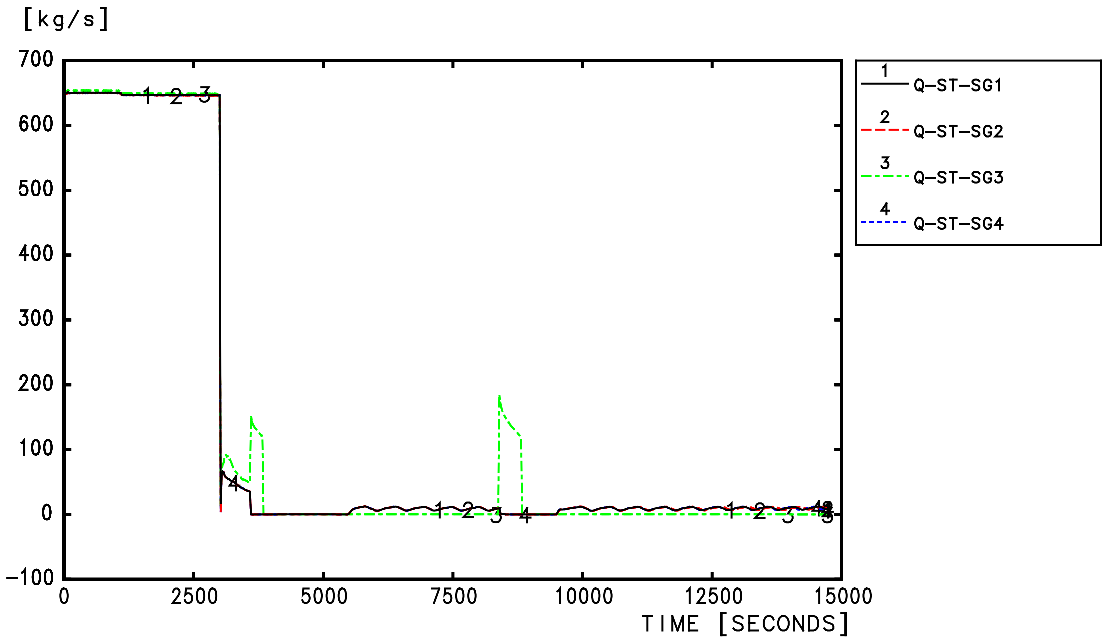
MASS FLOW RATE – CVCS



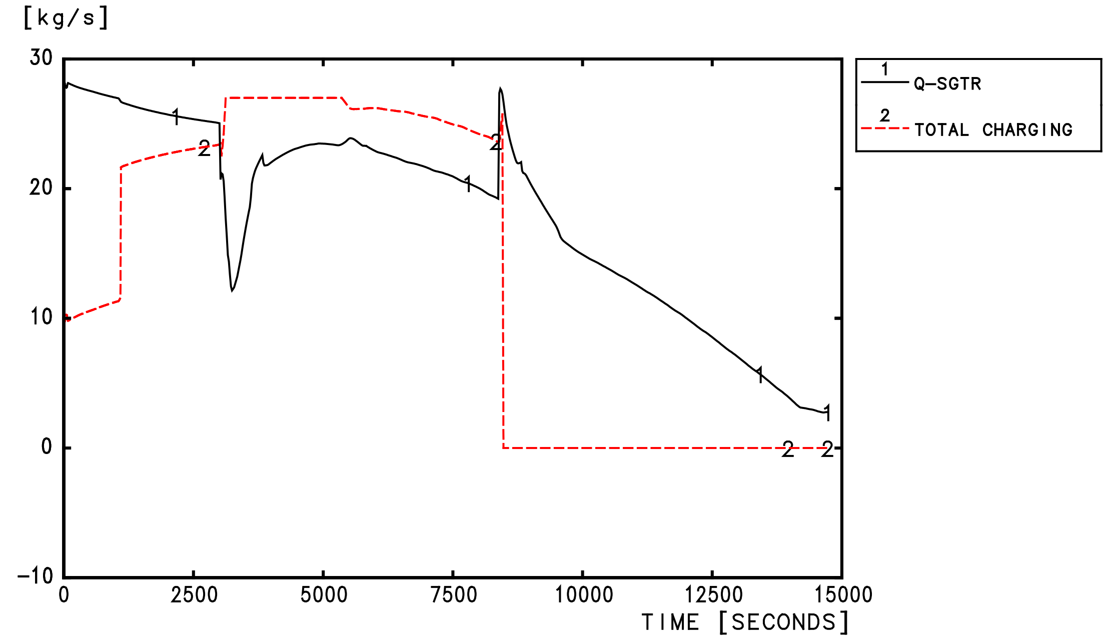
MASS FLOW RATE – MHSI

SECTION 14.4.6 - FIGURE 7

Case 1 - Steam Flows – RCV [CVCS] and Leakage Flows



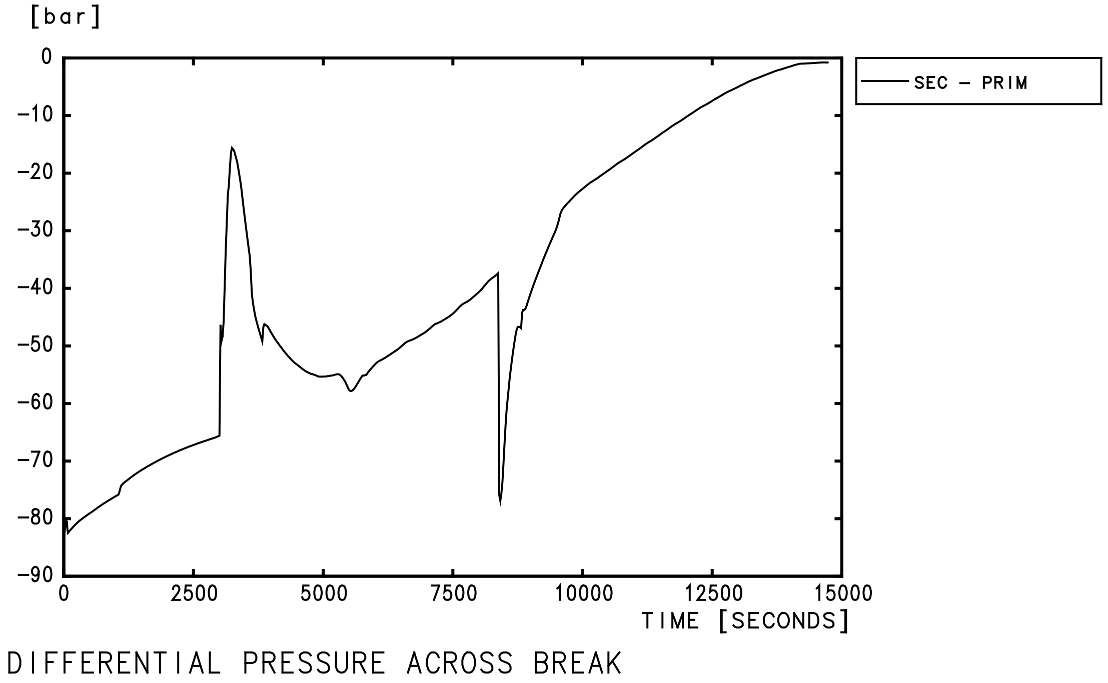
STEAM MASS FLOW RATE − SG

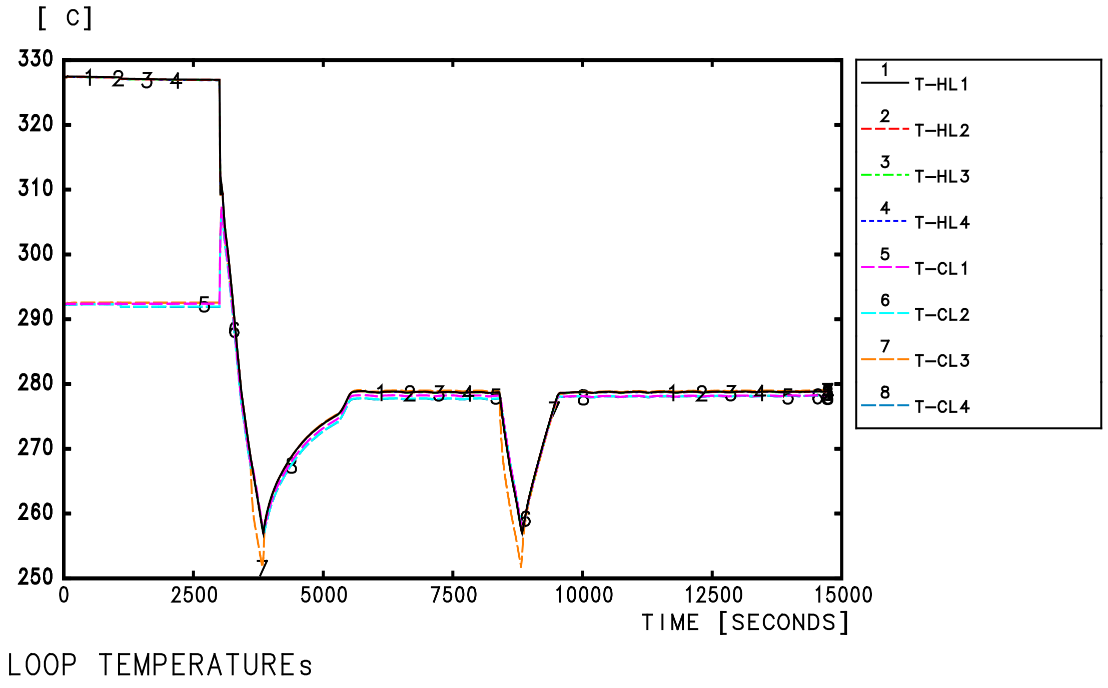


MASS FLOW RATE − SGTR AND TOTAL CVCS

SECTION 14.4.6 - FIGURE 8

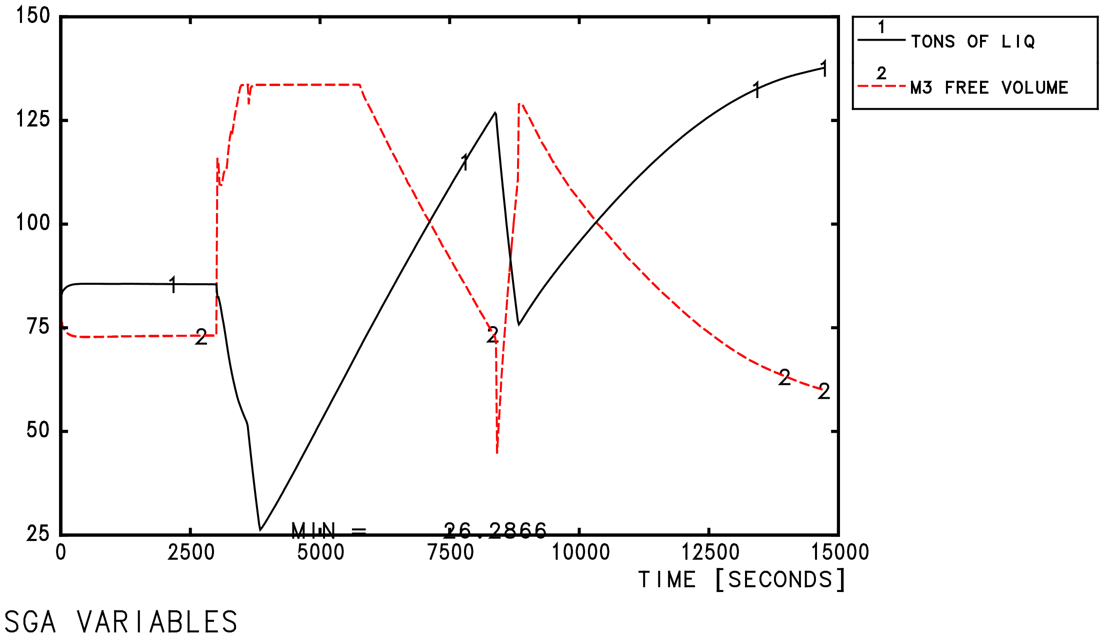
Case 1 - Break Differential Pressure – Loop Temperatures





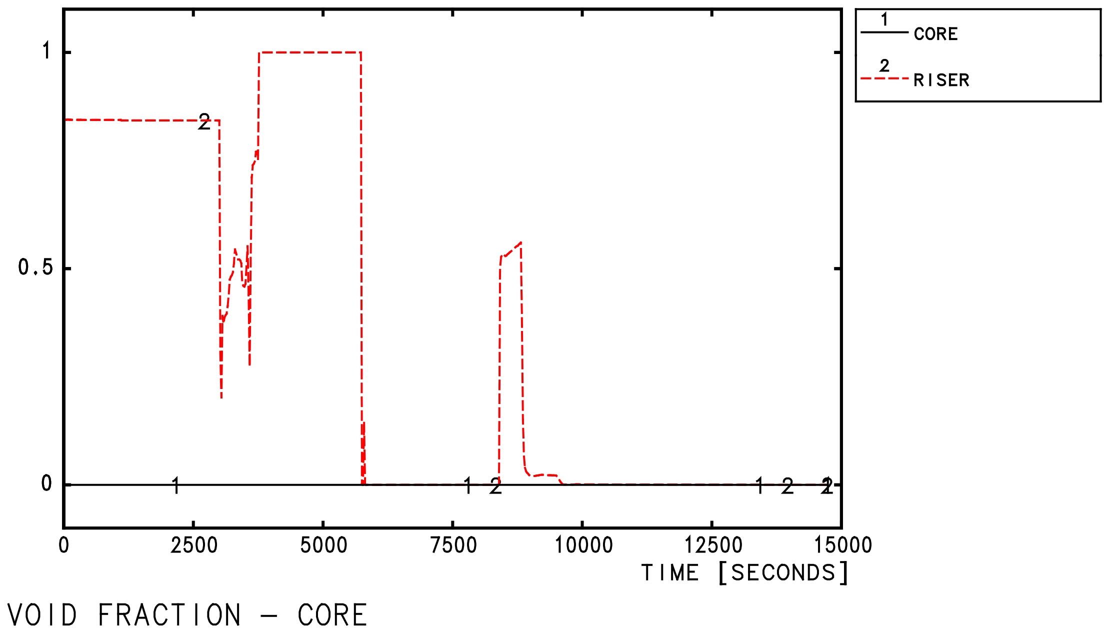
SECTION 14.4.6 - FIGURE 9

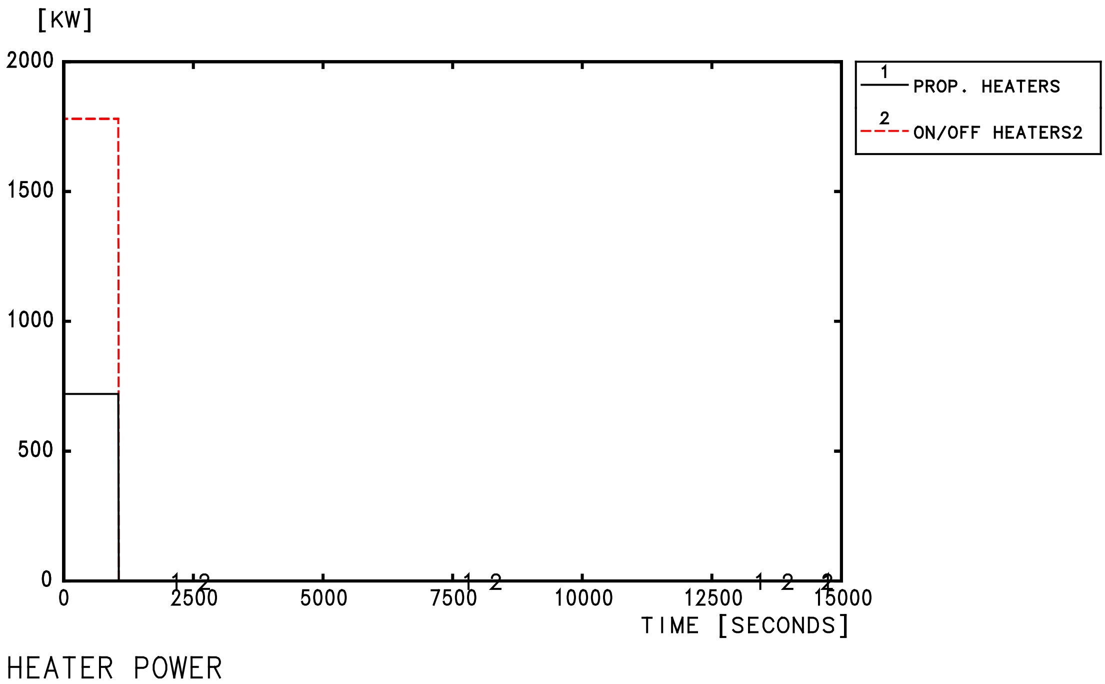
Case 1 - Liquid Mass and Free Volume in Affected SG



SECTION 14.4.6 – FIGURE 10

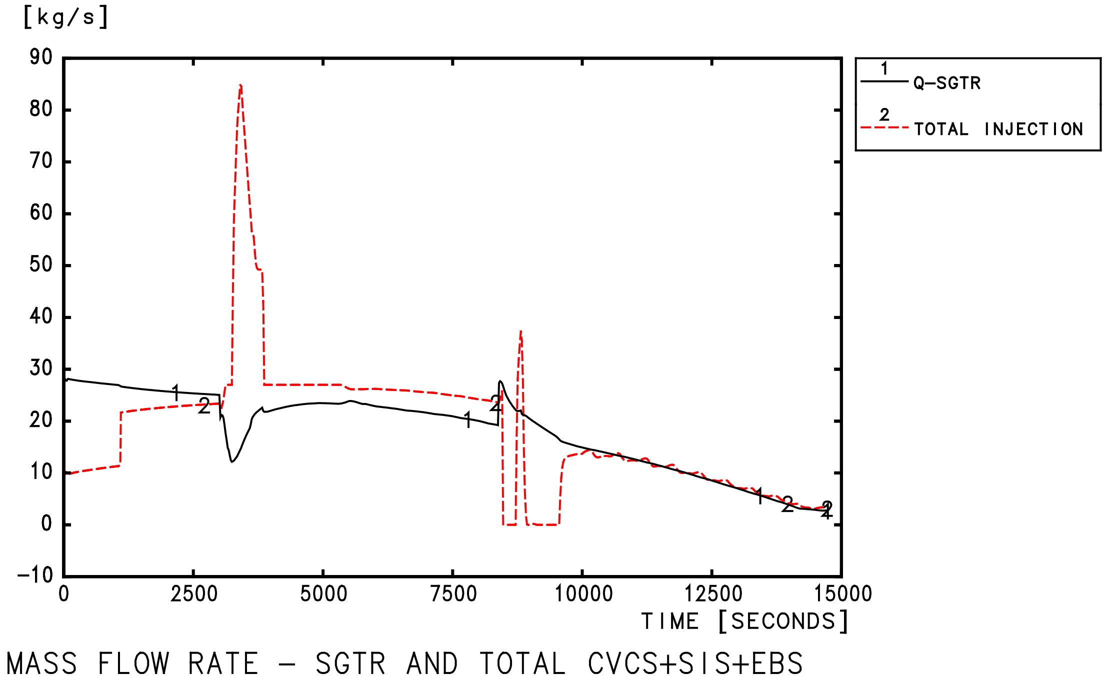
Case 1 - Core and SG Riser Void Fraction – Pressuriser Heater Power

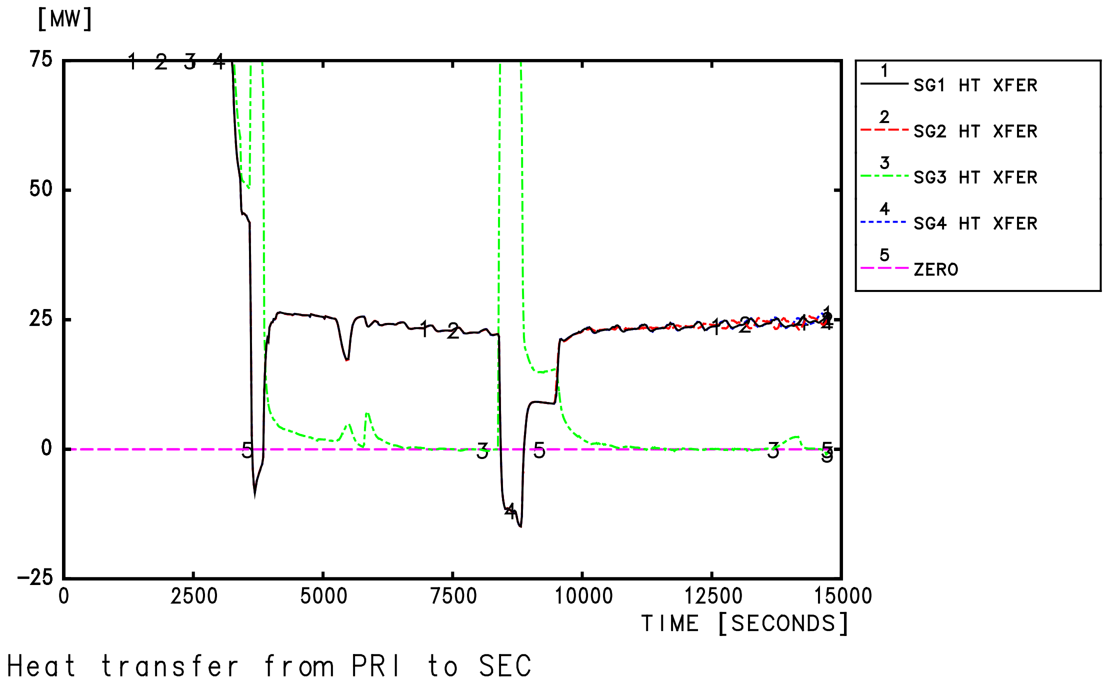




SECTION 14.4.6 – FIGURE 11

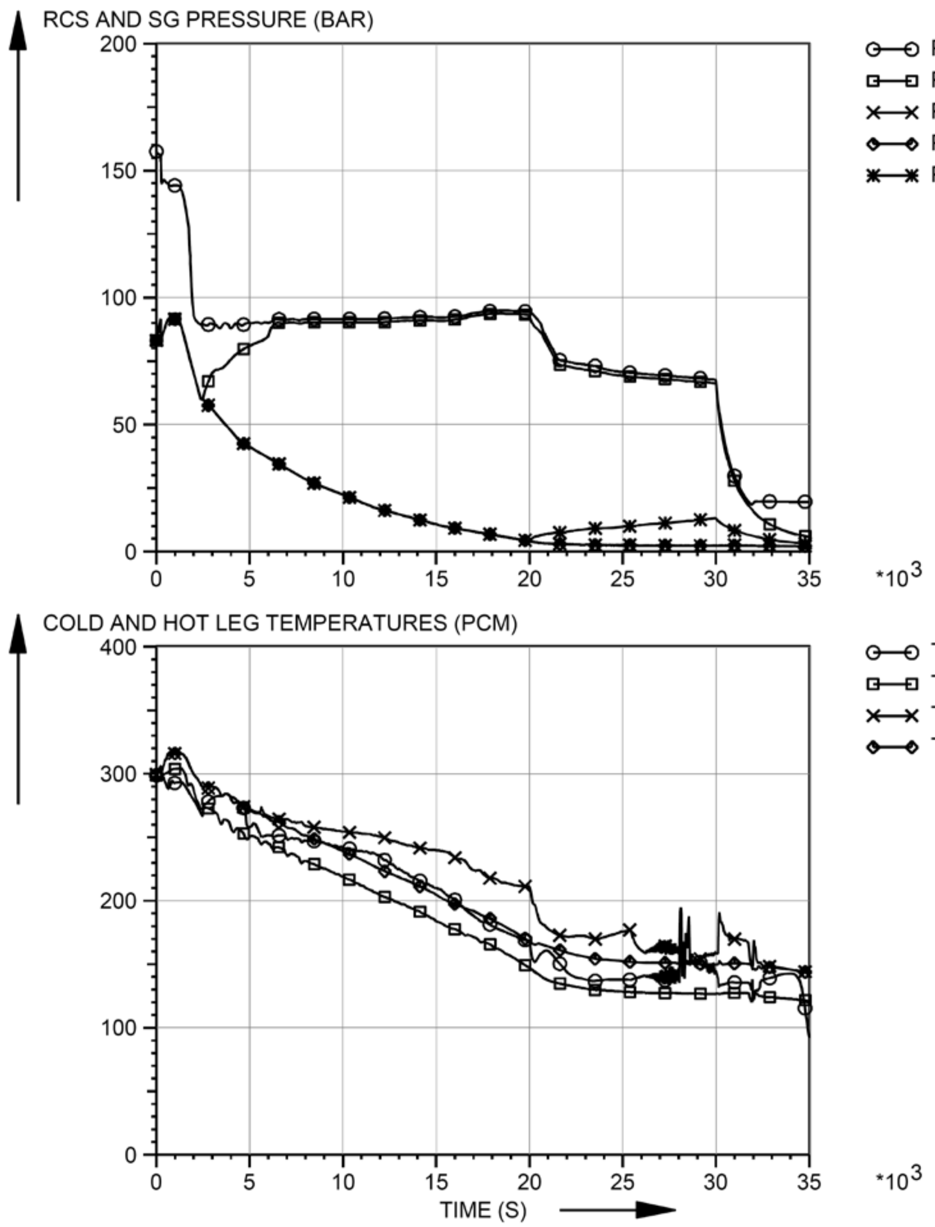
Case 1 - Total Injection Flow and Leak Flow – SG Heat Transfer





SECTION 14.4.6 - FIGURE 12

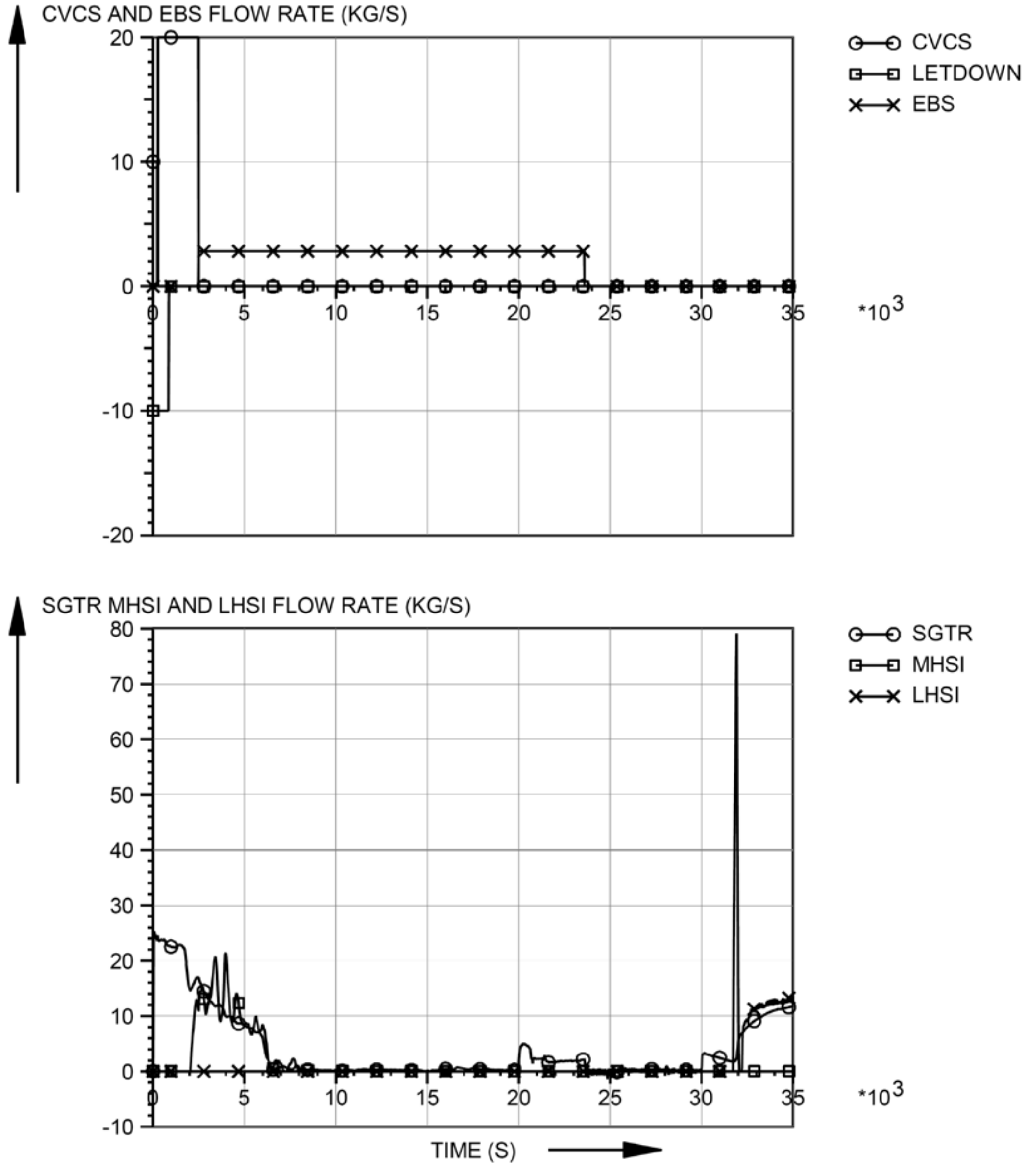
EPR$_{4900}$ Cases 2 and 3 – RCP [RCS] Pressure/ Hot and Cold Leg Temperatures



EPR-BDOP, 2A SGTR AT HOT STANDBY (2% POWER DECREASING)
EOC CONDITION, THERMAL DESIGN, BDR CALCULATION (15.2.17)
CODE RELAP-NLOOP, SIEMENS/KWU DEP NA-T/RN/RUN 98/7/41

SECTION 14.4.6 - FIGURE 13

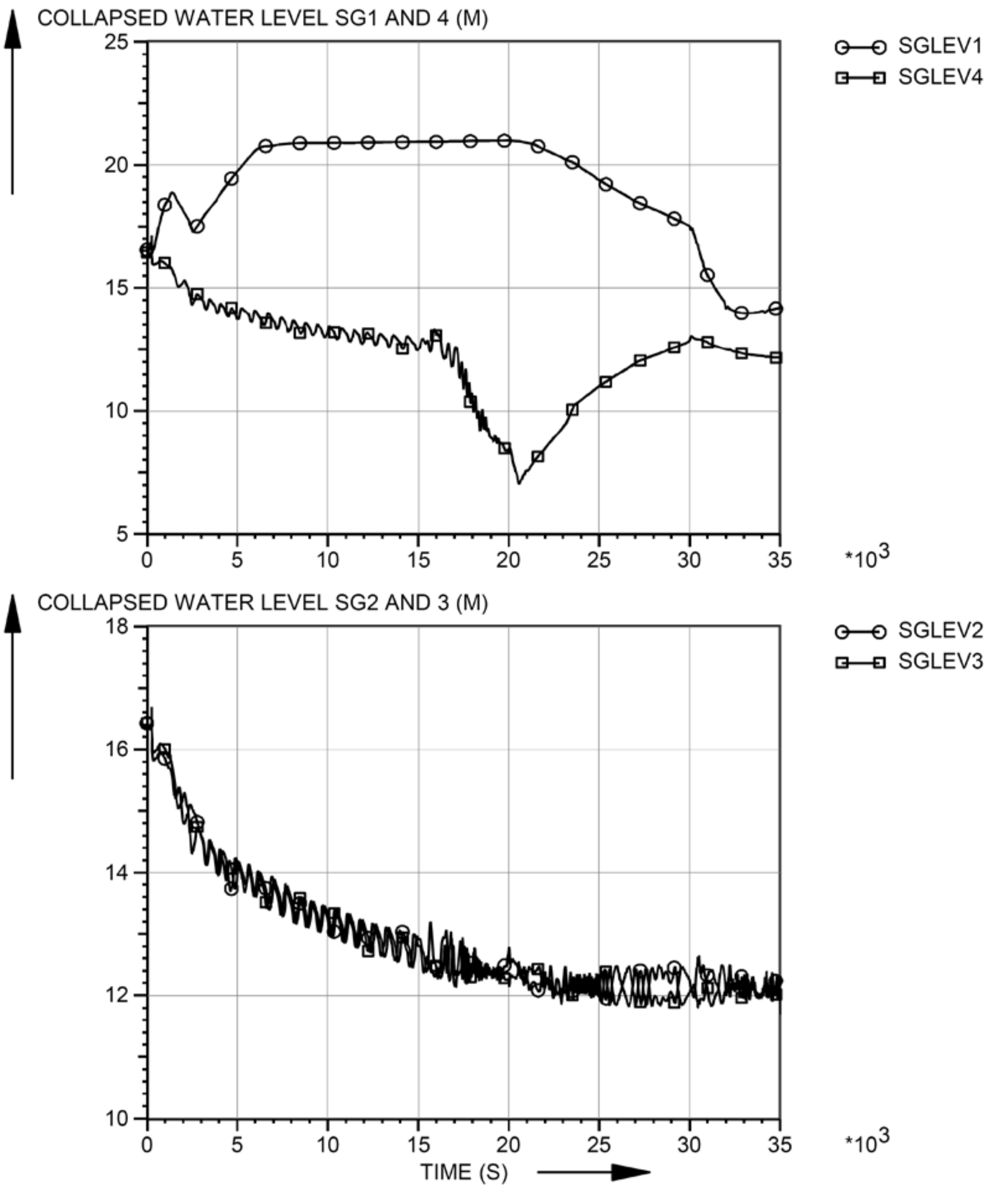
EPR$_{4900}$ Cases 2 and 3 - Mass Flow



EPR-BDOP, 2A SGTR AT HOT STANDBY (2% POWER DECREASING)
EOC CONDITION, THERMAL DESIGN, BDR CALCULATION (15.2.17)
CODE RELAP-NLOOP, SIEMENS/KWU DEP NA-T/RN/RUN 98/7/41

SECTION 14.4.6 - FIGURE 14

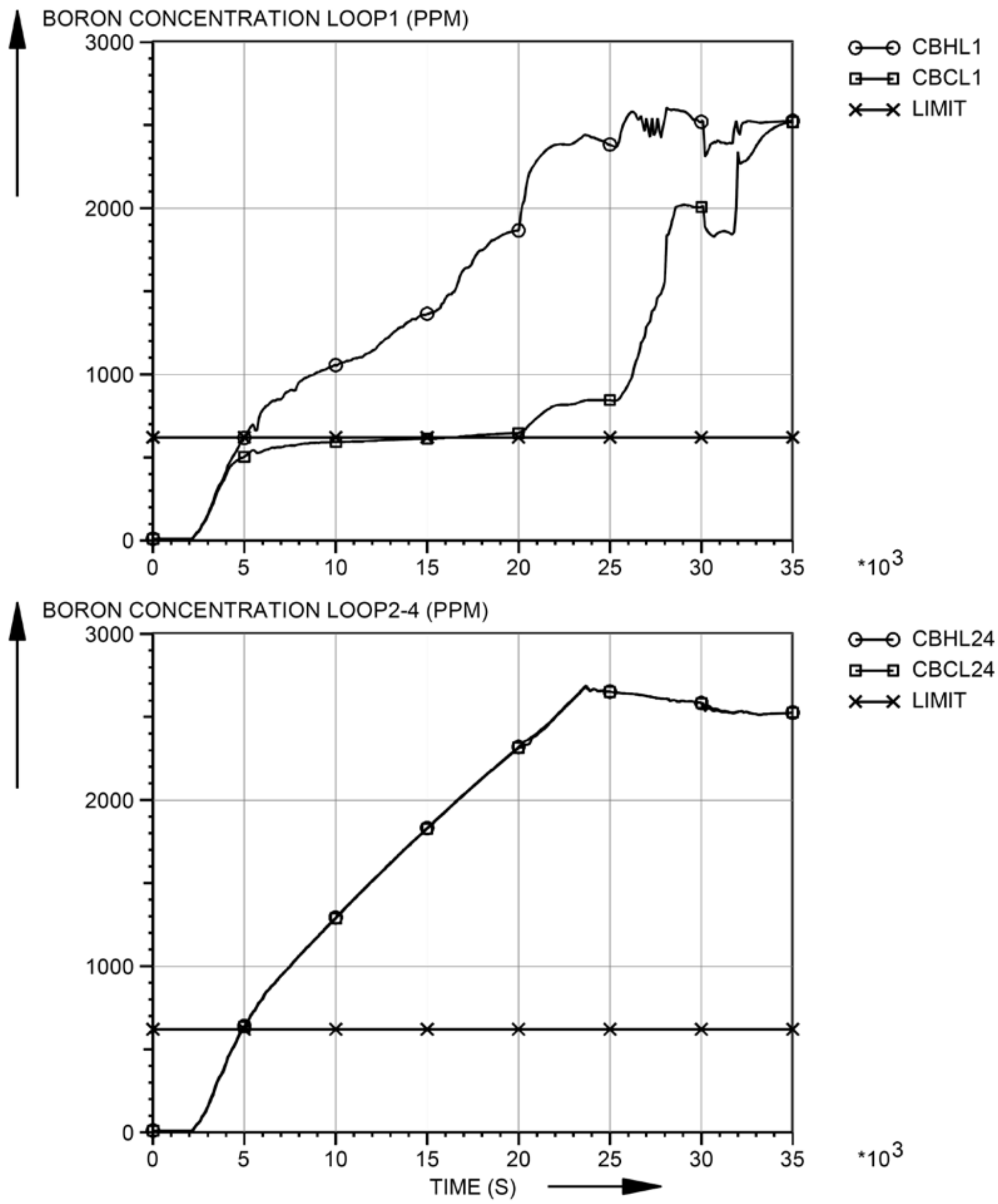
EPR_{4900} Cases 2 and 3 – SG Levels



EPR-BDOP, 2A SGTR AT HOT STANDBY (2% POWER DECREASING)
EOC CONDITION, THERMAL DESIGN, BDR CALCULATION
CODE RELAP-NLOOP, SIEMENS/KWU DEP NA-T/RN/RUN 98/7/41

SECTION 14.4.6 - FIGURE 15

EPR_{4900} Cases 2 and 3 – Boron Concentration



EPR-BDOP, 2A SGTR AT HOT STANDBY (2% POWER DECREASING)
EOC CONDITION, THERMAL DESIGN, BDR CALCULATION (15.2.17)
CODE RELAP-NLOOP, SIEMENS/KWU DEP NA-T/RN/RUN 98/7/41

7. INADVERTENT CLOSURE OF ONE OR ALL MAIN STEAM ISOLATION VALVES (PCC-3)

7.1. INTRODUCTION

The inadvertent closure of all VIV [MSIV] results in the isolation of all four main steam lines. This event is evaluated for overpressure, and is likely to be the limiting overpressure event. A reactor trip would occur as a result of the high primary or secondary pressures, which are limited by the pressuriser and steam generator relief systems respectively.

The inadvertent closure of all VIV [MSIV] is classified as a PCC-3 event.

7.2. INADVERTENT CLOSURE OF ALL VIV [MSIV] (STATE A, PCC-3)

7.2.1. Identification of causes and accident description

7.2.1.1. General Concern

The inadvertent closure of all of the VIV [MSIV] is an overheating event that leads to a risk of fuel failure due to a departure from nucleate boiling (DNB).

- Identification of causes

  The inadvertent closure of all VIV [MSIV] event is typically caused by a spurious I&C command to close the four VIV [MSIV].

- Precautions limiting the likelihood of the accident

  The 4 steam pilots of each VIV [MSIV] are assigned to separate electrical divisions. Therefore the probability of spurious closure of the 4 VIV [MSIV] due to electrical supply faults causing the spurious opening of 2 steam pilots in series in each VIV [MSIV] is very low.

7.2.1.2. Typical sequence of events

From the initiating event to the controlled state

The closure of all VIV [MSIV] results in the termination of all main steam flows. The reduced heat removal results in a secondary pressure and temperature increase, causing the primary pressure and temperature to increase.

Reactor Trip and Turbine Trip are automatically initiated, and the primary and secondary pressures are automatically limited by the pressuriser and SG relief devices.

Subsequently, the controlled state is reached. In this case it is defined as a hot shutdown state with residual heat being removed by the VDA [MSRT] and ASG [EFWS]. The ARE/AAD [MFWS/SSS] is not claimed as it is not a F1 classified system.

From the controlled state to the safe shutdown state

The safe shutdown state is reached when the LHSI/RHR can be connected.

The following sequence of actions should be performed by the operator to take the plant to the LHSI/RHR operating conditions:

- RCP [RCS] boration

  During the RCP [RCS] cooldown, the RCP [RCS] boration is performed by the RBS [EBS]. The RCV [CVCS] is not claimed as it is not a F1 classified system.

  Following completion of the boration, the operator stops the RBS [EBS].

- RCP [RCS] cooldown:

  The RCP [RCS] cooldown to LHSI/RHR connection conditions is carried out using the secondary side. It is undertaken by reducing the VDA [MSRT] setpoints at the required rate. The GCT [MSB] is unavailable as the VIV [MSIV] are closed by the initiating event.

  The rate of RCP [RCS] cooling is defined to be consistent with the capacity of the ASG [EFWS] tanks. Therefore, the LHSI/RHR operating conditions are reached before the ASG [EFWS] tanks empty.

  The EPR cooling rate is 50°C/h if two RBS [EBS] trains are available, or 25°C/h if only one RBS [EBS] train is available. The RBS [EBS] is designed so that the RBS [EBS] boration compensates for the reactivity insertion resulting from the RCP [RCS] cooling. The ASG [EFWS] tanks capacity is sufficient to keep the reactor coolant pumps in operation during the RCP [RCS] cooling phase if the shutdown of two of the four reactor coolant pumps has to be performed by the operator.

- RCP [RCS] depressurisation:

  After RCP [RCS] cooldown to 180°C in the hot legs, the operator will briefly open the PSV to depressurise the RCP [RCS] if the RCP [RCS] pressure is greater than the LHSI/RHR connection pressure of 30 bar. Normal pressuriser spray, or auxiliary pressuriser spray from the RCV [CVCS], are not claimed as they are not F1 classified systems. During the depressurisation phase, the LHSI maintains a minimum RCP [RCS] pressure of about 20 bar so that the RCP [RCS] keeps a certain sub-cooling margin. The LHSI/RHR connection conditions are met. One LHSI/RHR train is sufficient to remove the decay heat.

7.2.2. Safety Criteria

The safety criteria are the radiological limits for PCC-3/PCC-4 events.

The consequences of the inadvertent closure of all VIV [MSIV] are analysed to assess the following acceptance criterion: The number of fuel rods experiencing DNB must remain below 10%.

7.2.3. Definition of Studied Cases

- Fuel clad integrity

  The analysis is performed to demonstrate that the fuel clad integrity is maintained. The analysis therefore shows that the minimum DNBR remains above the DNBR criterion of 1.00 identified in Sub-chapter 14.1.

- Radiological consequences

  The safety criteria to be met are the dose equivalent limits in case of release to the atmosphere, as addressed in Sub-chapter 3.1: General safety design bases.

  The bounding transient for radiological releases is the PCC-2 ‘loss of condenser vacuum’ analysed in section 5 of Sub-chapter 14.3. The quantity of steam release to the atmosphere is identical in both cases.

7.2.4. Methods and Assumptions

7.2.4.1. Method of Analysis

The analysis of this accident is performed using the THEMIS code described in Appendix 14A.

The DNBR calculation is performed with the FLICA code described in Appendix 14A.

For the system transient analysed with the THEMIS code, the method of analysis is based on the following approach:

- Identification of the dominant phenomena
- Verification of the adequacy of the code to simulate these phenomena
- Application of conservative PCC analysis rules.

The dominant phenomena of this transient are:

- SG overpressure and heating, resulting from the loss of steam flow
- RCP [RCS] overpressure and heating, resulting from the SG heat removal changes
- Core power changes, resulting from neutronic feedbacks following changes in RCP [RCS] thermal-hydraulic conditions.

All these phenomena are within the capability of the THEMIS code. The neutronic calculations in this code rely on a point-kinetics model which accounts for all relevant aspects. The model includes feedbacks due to changes in moderator density, nuclear power, boron concentration, and rods position. The thermal-hydraulic qualification of this code is based on:

- The use of recognised and tested correlations, for example:
  - o Fluid-to-wall heat-transfer correlations covering normal and abnormal operating conditions. These include Dittus-Boelter for forced convection at the primary side of the SG tubes, and Jens-Lottes for nucleate boiling at the secondary side of the SG tubes
  - o Drift flux correlations, Zuber-Wallis, EPRI, and Patricia GV2 in the axial SG model
  - o Two-phase pressure drop correlation, Martinelli-Nelson in the axial SG model.
- The validation of specific models by test results from small-scale simulations, for example:
  - o Axial SG model without an economiser from the MB2 mock-up tests
  - o Axial SG model with an economiser of the N4 SG-type from the MEGEVE mock-up tests.

    The axial SG model refers to the modelling of the SG secondary side. The axial SG model is particularly well adapted to the calculation of plant transients experiencing secondary side overpressure resulting from loss of steam flow such as VIV [MSIV] closure. This is because the model has the capability to compute the internal recirculation flow, and to model the different regions of the SG, sub-cooled, two-phase, saturated, and superheated conditions, and their development during the transients.
- The overall verification of the code by simulation of PWR plants transients, for example:
  - o The THEMIS pressuriser model was validated by comparison with RCP [RCS] overpressure transients performed at the CRUAS 3-loops plant with heaters and start-up at various power and pressuriser levels,
  - o The SG axial model was validated in stable operating conditions by comparison with steady-state operation tests at different power levels, carried out at the BUGEY 3-loops and PALUEL 4-loops plants
  - o Load rejection and reactor trip at the PALUEL 4-loops plant were accurately simulated with THEMIS using the axial SG model. The calculated loop temperatures compared well with the measured values, thus validating the overall SG/RCP [RCS] heat transfer and the entire RCP [RCS] hydraulic calculations. The SG secondary side overpressure transient calculated by THEMIS closely agreed with the measured values. The results demonstrated the applicability of the SG axial model for these transient operating conditions.

The transient analysis relies on the application of the conservative PCC analysis rules described in Sub-chapter 14.0. Part of these rules is the deterministic application of conservatisms to all relevant boundary conditions, for the acceptance criteria being considered. These conservatisms address as a minimum:

- The characterisation of the initiating event to maximise the resultant impact

- Control dead band limits and maximum measurement uncertainties at the initial plant conditions
- Maximum uncertainty on each I&C measurement and signal delay, and on each system response time and capacity for protection and mitigation actions
- Conservative moderator and fuel Doppler coefficients for neutronic feedbacks

This methodology provides a conservative result which can be directly used to assess the acceptance criteria.

7.2.4.2. Protection and Mitigation Actions

The following F1A I&C functions provide protection following an inadvertent closure of all VIV [MSIV], for assessment against the DNBR criterion. The analyses for the other safety criteria, are discussed above.

- Reactor trip and turbine trip following a “SG pressure > MAX1” signal
- Reactor trip and turbine trip following a “pressuriser pressure > MAX2” signal
- Reactor trip and turbine trip following a low DNBR signal. However, this parameter is not claimed in the accident analysis presented.

The F1A I&C functions required to reach the controlled state are those related to decay heat removal via the SG, the ASG [EFWS] and VDA [MSRT], as described in section 5 of Sub-chapter 14.3: Loss of Condenser Vacuum.

The F1B devices, required to transfer the plant from the controlled state to the safe shutdown state, are described in section 5 of Sub-chapter 14.3: Loss of Condenser Vacuum.

7.2.5. Description of Studied Cases (from the Event Initiation to the Controlled State)

7.2.5.1. Choice of Single Failure and Preventive Maintenance

The minimum DNBR is reached at the start of rod drop. Therefore:

- The single failure assumed is a stuck rod;
- There is no preventive maintenance considered for this event, due to the short duration of the calculation phase. The calculation ends following the successful RT, a few seconds after the initiating event.

7.2.5.2. Initial State

The conservative initial conditions for the assessment of the minimum DNBR are 100% full power with uncertainties included. The relevant parameters are presented in Section 14.4.7 - Table 1.

7.2.5.3. Specific Assumptions

Neutronic data and decay heat

The fission power term A is calculated with a point-kinetic model.

A maximum moderator coefficient in absolute value (EOC) is assumed, which results in a power increase at the beginning of the transient, due to a moderator density increase.

The minimum Doppler coefficient in absolute value is chosen to maximise this power increase.

The maximum decay heat curve ‘term B+C with 1.645σ‘ is used, as described in Sub-chapter 14.1.

Assumptions related to non F1 systems

The control I&C functions are not modelled, as they either have no impact or they have beneficial effects on the minimum DNBR value.

In addition, the pressuriser spray is not taken into account.

Both of these assumptions lead to a higher minimum DNBR value.

Assumptions related to F1 systems

The reactor trip is the only F1 system, considered during this phase of the accident, relevant to the minimum DNBR value.

Other assumptions

The initial DNBR is equal to the minimum value allowed by the low DNBR LCO function. A detailed discussion of the limiting value of DNBR is provided in section 8 of Sub-chapter 14.1.

This minimum value results from the analysis of the short-term LOOP event described in section 6 of Sub-chapter 14.3.

7.2.5.4. Results and Conclusion

Minimum DNBR criterion

This PCC transient is not calculated in the PCSR. The assessment of the DNBR criterion is derived from the PCC transient analysis already performed for the EPR_{4900} in the BDR-99 presented in Appendix 14B, from the assessment is performed by comparing the relevant characteristics of the EPR_{4500} covered by the PSAR and the EPR_{4900} covered by the BDR-99 and their impact on the analysis results.

The BDR-99 accident analysis presented in section 2.4 of Appendix 14B shows that:

- At the beginning of the transient, the first two seconds, following the all VIV [MSIV] closure, the DNBR decreases. This is a consequence of the RCP [RCS] pressure increase which causes a moderator density increase. During this period the moderator temperature remains essentially unchanged. This results in a power increase, which then causes a decrease in DNBR.

- Subsequently, the DNBR then stabilises over approximately 1 second, then increases continuously for the remainder of the transient. This arises from a combination of the sharp RCP [RCS] pressure increase and the core temperature increase.

- The minimum DNBR is reached at about 2 seconds, before the RT signal "SG pressure > MAX1" which is generated at about 4 seconds. The minimum DNBR value is 1.13, for an initial DNBR value of 1.26. This demonstrates a comfortable margin to the DNBR criterion of 1.00. This result shows that the PCC event ‘inadvertent closure of four VIV [MSIV]’ is less onerous than the limiting PCC-2 short term Loss Of Off-site Power (LOOP) event. The initial DNBR value of 1.26 is derived from the LOOP transient. It is sufficient to maintain the DNBR above 1.00 during the RCP [RCS] flow rate decrease following the loss of the power to all reactor coolant pumps.

A comparison of the relevant parameters for the EPR_{4500} covered by the PCSR and the EPR_{4900} covered by the BDR-99 analysis shows that:

- The EPR_{4500} has an initial DNBR margin similar to that of the EPR_{4900} which has an initial minimum DNBR value of 1.26. The derivation of this value is discussed in section 6 of Sub-chapter 14.3.

- The EPR_{4500} has similar primary and secondary thermal-hydraulic characteristics to the EPR_{4900}. However, the EPR_{4500} has the potential for lower RCP [RCS] overpressure rates than the EPR_{4900}, due to a lower ‘power/pressuriser steam volume’ ratio. The power is decreased by 8% in the EPR_{4500} but the pressuriser steam volume is unchanged. This is a benefit for the DNBR. At the start of the transient the DNBR decreases because of the RCP [RCS] pressure increase before the moderator temperature has increased. The lower RCP [RCS] pressure increase causes a lower DNBR decrease in the period before RT.

Based on the above comparison:

- The PCC-3 event ‘inadvertent closure of four VIV [MSIV]’ is not limiting for the assessment of the minimum DNBR when compared to the PCC-2 LOOP event as the BDR-99 accident analysis provides a large margin for the DNBR criteria.

- The EPR_{4500} fulfils the DNBR criterion, DNBR $\geq$ 1, in the event of ‘inadvertent closure of four VIV [MSIV]’ as the results show no adverse effect for the EPR_{4500} compared with the EPR_{4900}.

Reaching the controlled state

The means to reach the controlled state at hot shutdown are the same as those identified for the PCC-2 event described and demonstrated in section 5 of Sub-chapter 14.3: ‘Loss of Condenser Vacuum’.

- Reactivity control: the initial power level and the shutdown rods worth are identical

- Residual heat removal: the same capabilities are available for the SG (VDA [MSRT] and ASG [EFWS])

- Coolant inventory: both events have no impact on the primary coolant inventory.

Influence of the pressuriser safety valves new design

The inadvertent closure of one or all main steam isolation valves has been performed with the SEBIM-type pressuriser safety valves modelled. The impact of the main PSV parameters change to SEMPELL-type pressuriser safety valves is as follows:

The BDR-99 accident analysis presented in Appendix 14B.2.4 - Table 2, shows that the minimum DNBR in the case of an inadvertent closure of all VIV [MSIV] occurs at 2.1 seconds.

Appendix 14B.2.4 - Figure 1 shows that the first PSV opens later than the time of minimum DNBR, at 5.8 seconds. The SEMPELL valve opening pressure is higher than the SEBIM opening pressure. Therefore, there is no risk of an earlier opening of a PSV that could have a detrimental impact on the DNBR.

Consequently, the change from SEBIM to SEMPELL pressuriser safety valves has no impact on the DNBR following this fault.

Thus, for the case of an inadvertent closure of all VIV [MSIV], the PSV are not challenged.

7.2.6. Description of Studied Cases (from the Controlled State to the Safe Shutdown State)

The safe shutdown state is reached once the LHSI/RHR is operational.

The transition from the controlled state to the safe shutdown state is covered by the PCC-2 turbine trip event discussed in section 5 of Sub-chapter 14.3 if reactor coolant pumps remain in operation. If all reactor coolant pumps are unavailable after turbine trip due to a loss of grid, it is covered by the PCC-3 long term LOOP event discussed in section 2 of this sub-chapter.

Impact of the safety classification of the normal spray operations

The F1B safety classification of the normal pressuriser spray allows credit to be claimed for this system in the transfer to LHSI/RHR connecting conditions. The analysis of this modification is given for the feedwater line break, section 3 of Sub-chapter 14.5 for ASG [EFWS] tank sizing, and on the loss of condenser vacuum, section 5 of Sub-chapter 14.3, concerning steam mass discharged to the atmosphere.

7.3. INADVERTENT CLOSURE OF ONE VIV [MSIV] (IN STATE A)

The inadvertent closure of one VIV [MSIV] could be caused by a spurious closure signal on one VIV [MSIV]. The closure of one VIV [MSIV] isolates the steam flow in one steam line. The heat removal reduction in the affected steam generator leads to a secondary pressure and temperature increase and consequently to a temperature increase in the corresponding primary loop. Reactor trip would occur following a high steam generator pressure or low DNBR signal. The secondary side pressure would be limited by the operation of the VDA [MSRT].

7.3.1. Identification of Causes and Accident Description

7.3.1.1. General Concern

The inadvertent closure of one VIV [MSIV] is an overheating event leading to a risk of fuel failure due to DNB, and a risk of Reactor Coolant Pressure Boundary (RCPB) and Secondary System Pressure Boundary (SSPB) failures due to primary and secondary system overpressure. This section deals with the risk of DNB.

The risk of excessive overpressure is addressed in section 1 of Sub-chapter 3.4, 'Topics specific to Mechanical Components'.

Identification of causes

The inadvertent closure of one VIV [MSIV] event could be initiated:

- By a spurious I&C command to close one VIV [MSIV], or
- By the spurious opening of two pilot valves in series on the VIV [MSIV].

Precautions limiting of the accident occurrence

The assignment to the 4 electrical divisions of the 4 steam pilots of one VIV [MSIV] is such that the probability of spurious closure of 1 VIV [MSIV] due to the spurious opening of 2 steam pilots in series is low.

The inadvertent closure of one VIV [MSIV] is classified as a PCC-3 event.

7.3.1.2. Typical Sequence of Events

From the initiating event to the controlled state

The closure of one VIV [MSIV] results in the termination of one main steam flow. The reduced heat removal leads to secondary pressure and temperature increases, causing the primary side pressure and temperature to increase.

Reactor Trip and Turbine Trip are automatically initiated, and the secondary side pressure is automatically limited by the SG relief valves.

Subsequently, the controlled state is reached, defined here as hot shutdown condition, with residual heat being removed by the VDA [MSRT] and the ASG [EFWS]. The GCT [MSB] is not claimed on the three unaffected SG as it is not an F1 classified system. The ARE/AAD [MFWS/SSS] is not claimed as they are not F1 classified systems.

From the controlled state to the safe shutdown state

The safe shutdown state is reached when the LHSI/RHR can be connected.

The sequence of actions to be performed by the operator to reach the LHSI/RHR operating conditions is:

- RCP [RCS] boration

  During the RCP [RCS] cooldown, the RCP [RCS] boration is performed by the RBS [EBS]. The RCV [CVCS] is not claimed as it is not an F1 classified system.

  After completion of the boration, the operator stops the RBS [EBS].

- RCP [RCS] cooldown

  The RCP [RCS] cooldown to LHSI/RHR connection conditions is carried out using the secondary side. It is performed by reducing the VDA [MSRT] setpoints. The GCT [MSB] is not claimed as it is not an F1 classified system.

  The rate of RCP [RCS] cooling is defined to be consistent with the capacity of the ASG [EFWS] tanks. Therefore, the LHSI/RHR operating conditions are reached before the ASG [EFWS] tanks empty.

  The EPR cooling rate is 50°C/h if two RBS [EBS] trains are available, or 25°C/h if only one RBS [EBS] train is available. This cooling rate may not be achievable in the long term due to the limited capacity of the VDA [MSRT] at low pressure. The RBS [EBS] is designed so that the RBS [EBS] boration compensates for the reactivity insertion resulting from the RCP [RCS] cooling. The ASG [EFWS] tank capacity is sufficient to keep two reactor coolant pumps in operation during the RCP [RCS] cooling phase assuming the shutdown of two of the four reactor coolant pumps has to be performed by the operator.

- RCP [RCS] depressurisation

  If the RCP [RCS] pressure is greater than the RIS/RRA [SIS/RHRS] connection pressure of 30 bar when the RCP [RCS] cooldown to 180°C in hot leg is completed, the operator can briefly open the PSV to depressurise the RCP [RCS]. The normal pressuriser spray[1], or auxiliary pressuriser spray from the RCV [CVCS] are not claimed as they are not F1 classified systems. During the depressurisation phase, the LHSI trains ensure a certain sub-cooling margin and limit the RCP [RCS] depressurisation at about 20 bar.

7.3.2. Safety Criteria

The safety criteria are the radiological limits for PCC-3/PCC-4 events.

The consequences of the inadvertent closure of one VIV [MSIV] are analysed for the following acceptance criterion: The number of fuel rods experiencing DNB must remain below 10%.

7.3.3. Definition of Studied Cases

Fuel cladding integrity

The analysis is performed to show that the minimum DNBR remains above 1.00, the decoupling criterion described in Sub-chapter 14.1, thus demonstrating that fuel cladding integrity is maintained.

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[1] The normal spray is F1B classified in the current design reference but not at the time of the study was performed.

The phase from the initiating event to the minimum DNBR is analysed in detail in the following sections.

The transfer to the controlled state and the safe shutdown state is assessed using a qualitative evaluation, by reference to other analyses in Chapter 14.

Radiological consequences

The safety criteria to be met are the dose equivalent limits following a release to the atmosphere, discussed in Sub-chapter 3.1.

The bounding transient for radiological releases is 'the loss of condenser vacuum' analysed in section 5 of Sub-chapter 14.3, 'Loss of Condenser Vacuum'. The quantity of steam release to the atmosphere is identical in both cases.

7.3.4. Methods and Assumptions

7.3.4.1. Method of Analysis

The analysis of this accident is carried out using the THEMIS code described in Appendix 14A.

The DNBR calculation is performed using the FLICA code described in Appendix 14A.

The methodology of analysis is the same as that used for the 'inadvertent closure of all VIVs [MSIVs]' event, presented in sub-section 7.2.4 of this sub-chapter. When compared to this accident, the 'inadvertent closure of one VIV [MSIV]' introduces an asymmetrical RCP [RCS] behaviour. The adequacy of the code and the methodology of analysis are discussed in 'Main Steam Line Break Accident' section 2 of Sub-chapter 14.5.

7.3.4.2. Protection and Mitigation Actions

F1A I&C protection for the DNBR criterion after an inadvertent closure of one VIV [MSIV].is provided by a reactor trip and turbine trip following a "SG pressure > MAX1" signal. The analyses for the other safety criteria are described in sub-section 7.2 of this sub-chapter.

> Note: the reactor trip on the low DNBR channel could be qualified for such an asymmetrical event. The reactor trip signal is not taken into account in the present accident analysis.

The F1A I&C functions required to reach the controlled state are those related to decay heat removal via the SG, the ASG [EFWS] and the VDA [MSRT] as described for the 'Loss of Condenser Vacuum' event in section 5 of Sub-chapter 14.3.

The F1B functions required to transfer the plant from the controlled state to the safe shutdown, are described in section 1 of the 'Loss of Condenser Vacuum' event described in section 5 of Sub-chapter 14.3.

7.3.5. Description of Studied Cases (from the Initiating Event to the Controlled State)

7.3.5.1. Choice of Single Failure and Preventive Maintenance

The minimum DNBR is reached at the start of rods drop. Thus:

- The single failure assumed is a stuck rod;
- The preventive maintenance is not considered for this event. Due to the short duration of the calculation phase an assumption of preventive maintenance would not impact the calculated results.

7.3.5.2. Initial State

The conservative initial conditions for the DNBR assessment are 100% full power, with uncertainties taken into account. The key parameters are presented in Section 14.4.7 – Table 2.

7.3.5.3. Specific Assumptions

Neutronic data and decay heat

The fission power term A is calculated with a point-kinetic model.

BOC neutronic data are considered, as they lead to the minimum decrease of power at the start of rods drop. In this case, EOC neutronic data do not lead to the limiting power increase, as they do for the inadvertent closure of all VIV [MSIV] discussed above.

The maximum decay heat curve 'term B+C with 1.645σ' is used as described in Sub-chapter 14.1.

Assumptions related to non F1 systems

The control I&C functions are not taken into account, as they have either no impact or are beneficial to the minimum DNBR value.

Assumptions related to F1 systems

The only F1 system demanded during this phase of the accident and relevant to the minimum DNBR value is the reactor trip.

Other assumptions

The initial DNBR is equal to the minimum value allowed by the low DNBR LCO function as discussed in section 8 of Sub-chapter 14.1.

7.3.5.4. Results and Conclusion

Regarding DNBR criterion

The PCC transient is not calculated in the Pre Construction Safety Report. The demonstration that the DNBR criterion is met is based on the result of PCC transient analysis already performed in the BDR-99 analysis for EPR_{4900} presented in Appendix 14B. The acceptability for EPR_{4500} can be judged from the comparison of relevant parameters for the EPR_{4500} covered by the PSAR and the EPR_{4900} covered by the BDR-99.

The BDR-99 accident analysis described in section 2.4 of Appendix 14B shows that:

- At the beginning of the transient, typically the first 4 seconds after one VIV [MSIV] closure, RCP [RCS] pressure and temperature have small variations which do not lead to a significant change to the DNBR value.

- DNBR starts to decrease. This is due to the core coolant temperature increase, combined with a relatively small pressure increase.

- Finally, the DNBR decrease stops at the start of the rod drop following a "SG pressure > MAX1" RT signal, which typically occurs around 5.5 seconds. Subsequently, the DNBR continues to increase.

- Assuming an initial DNBR value of 1.26, the minimum DNBR value is 1.16. This maintains a comfortable margin to the DNBR criterion of 1.00. This result shows that the PCC event ‘inadvertent closure of one VIV [MSIV]’ is less onerous than the limiting PCC-2 LOOP event. The initial DNBR value of 1.26 was defined to keep the DNBR above 1.00 during the RCP [RCS] flow rate decrease due to the loss of power to all four of the reactor coolant pumps.

Comparison of the relevant characteristics between the EPR_{4500} covered by the PCSR and the EPR_{4900} covered by the BDR-99 discussed in Sub-chapter 14.1 shows that:

- The EPR_{4500} has an initial DNBR margin similar to that of the EPR_{4900} with a minimum initial DNBR value of 1.26, as discussed in section 6 of Sub-chapter 14.3.

- The EPR_{4500} has similar primary and secondary side thermal-hydraulic characteristics compared to the EPR_{4900}. However, the EPR_{4500} has the potential for a lower RCP [RCS] heat-up rate than the EPR_{4900}, due to a lower ‘power/RCP [RCS] coolant mass’ ratio. In the EPR_{4500} power is decreased by 8% and RCP [RCS] coolant mass is unchanged from that for the EPR_{4900}. In addition the EPR_{4500} has a lower ‘power/SG water content’ ratio. The power is decreased by 15% and the SG water mass is decreased by 8% and the SG volume decreased by 4% compared with the EPR_{4900}. This is beneficial for the DNBR, as it reduces the rate of decrease.

Based on the above comparison:

- The results show a large margin to the DNBR criterion. The PCC-3 event of ‘inadvertent closure of one VIV [MSIV]’ is not limiting for DNB when compared to the PCC-2 LOOP event. This conclusion can be drawn even when no claims are made on the specific DNBR protection provided by the reactor trip on “low DNBR”.

- The comparison between the relevant characteristics for the EPR_{4500} and the EPR_{4900}, shows no significant detrimental effect for the EPR_{4500} compared with the EPR_{4900}.

It can be concluded that EPR_{4500} meets the DNBR criterion of DNBR $\geq$ 1 following an 'inadvertent closure of one VIV [MSIV]'.

The controlled state

The means to reach the controlled state, hot shutdown, are the same as for the PCC-2 event described in the section 'Loss of Condenser Vacuum', section 5 of Sub-chapter 14.3:

- Reactivity control: the initial power level and the shutdown rods worth are identical
- Residual heat removal: the same capabilities are available for the SG (VDA [MSRT] and ASG [EFWS])
- Coolant inventory: both events have no impact on the primary coolant inventory.

7.3.6. Description of Studied Cases (from the Controlled State to the Safe Shutdown State)

The safe shutdown state is reached once the LHSI/RHR is operational.

The transition from the controlled state to the safe shutdown state is covered by the PCC-2 event 'Loss of Condenser Vacuum', presented in section 5 of Sub-chapter 14.3, if reactor coolant pumps remain in operation. If the reactor coolant pumps are unavailable after turbine trip due to the loss of grid the transfer to the safe shutdown state is covered by the PCC-3 long term LOOP event described in section 2 of this sub-chapter.

Both the initial and final states are the same, and the capabilities of the F1B systems needed in both cases, the VDA [MSRT], ASG [EFWS] and RBS [EBS] are identical.

7.4. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

SECTION 14.4.7 - TABLE 1

Inadvertent Closure of All VIVs [MSIVs]
Main Assumptions
(4500 MWth)

Parameter	Limiting Value Assumed
Reactor Trip "SG pressure > MAX1"	
Setpoint	95.5+1.5 = 97.0 bar
Delay	1.2 s
Reactor Trip "pressuriser pressure > MAX2"	
Setpoint	166.5 + 1.5 = 168.0 bar
Delay	1.2 s
Initial conditions	
RCP [RCS] flow	T/h design flow
Core power	100% + 2% = 102% FP
Pressuriser pressure	155.0 - 2.5 = 152.5 bar
RCP [RCS] average temperature	312.8 + 2.5 = 315.3°C
Pressuriser level	56 + 5% = 61% of measuring range
SG level	49% of narrow range
SG heat transfer area	Nominal – 10%

SECTION 14.4.7 - TABLE 2

Inadvertent Closure of One VIV [MSIV]
Main Assumptions
(4500 MWth)

Parameter	Limiting Value Assumed
Reactor Trip "SG pressure > MAX1"	
Setpoint	95.5 + 1.5 = 97.0 bar
Delay	1.2 s
Initial conditions	
RCP [RCS] flow	T/h design flow
Core power	100% + 2% = 102% FP
Pressuriser pressure	155.0 - 2.5 = 152.5 bar
RCP [RCS] average temperature	312.8 + 2.5 = 315.3°C
Pressuriser level	56 - 5% = 51% of measuring range
SG level	49 - 5% = 44% of narrow range
SG heat transfer area	Nominal -10%

8. INADVERTENT LOADING OF A FUEL ASSEMBLY IN AN INCORRECT POSITION

8.1. DESCRIPTION

The reactor core loading pattern reflects a fuel assembly arrangement that allows the safety limits to be met throughout the fuel cycle. The “inadvertent loading of a fuel assembly in an incorrect position” refers to a core configuration that does not comply with the intended loading pattern.

The incorrect positioning of the assembly may occur during reactor refuelling operations (state E, see Sub-chapter 14.0). The consequences are:

Direct consequence

Non-compliance with the loading pattern may result in an unexpected increase in the core reactivity, namely a drop in the shutdown margin needed for refuelling operations.

Possible consequence

If the incorrect fuel loading pattern is not detected prior to power operations, it could lead to a change in the power distribution predicted for the core design and thereby exceed the safety limits.

8.2. IDENTIFICATION OF CAUSES, PREVENTION METHODS

Inadvertent loading of a fuel assembly in an incorrect position can be the result of:

- An error when the loading pattern was developed, or
- An error when the loading pattern was applied to the core, or
- An error during refuelling operations.

Various measures are implemented to prevent these errors.

8.2.1. Creation of the Loading Pattern and Application to the Core

The loading pattern development process where meeting the safety limits is justified is subject to two independent checks, one by the plant operator and the other by the fuel supplier.

The loading pattern is sent to the plant electronically. Using this information, the plant produces the assembly load sequence, a list of sequential assembly movements, for the monitoring station of the fuel handling system. The monitoring station carries out an automatic check of the pattern received via a checksum.

8.2.2. Reactor Reloading Operations

Measures are taken during the fuel handling system design stage, (monitoring function), to prevent the incorrect positioning of an assembly during reloading operations:

- Each sequence is automatically presented to the loading manager in the reactor building or to his assistant in the fuel building to confirm the acceptability of the fuel movement.
- After validation by the loading manager or his assistant, the assembly pick-up and put-down coordinates are sent to the equipment involved, either the fuel manipulator crane or the overhead crane.
- The fuel manipulator and overhead crane operators execute, using joystick control, the movements indicated by these automatic devices. The movement is interrupted immediately if the operators release the joystick.
- A camera is placed at the outlet point of the transfer device on the reactor building side to allow the operator to read the identity of each assembly transferred. The monitoring system only allows the operation to continue via subsequent movement to the defined sequence if the manually entered identity number read by the operator corresponds to the number expected in the sequencing plan.

8.3. DETECTION METHODS

8.3.1. During Reloading

The fuel handling system monitoring stations allow the progression of the core loading plan to be followed throughout the reloading operations. A core loading table can therefore be consulted in real time. At the end of loading, a complete core map is produced so that compliance with the required loading pattern can be confirmed.

This map guarantees that after reloading, the plant restarts with the assemblies in the correct positions in the core.

8.3.2. Restart Physics Tests

By comparing design predictions with measurement results, the restart physics tests enable a global verification that the core design meets the appropriate limits. They are based on the following measurements:

- Critical boron concentration levels
- Moderator temperature coefficient
- Control rod worth
- Flux maps
    - At low power
    - During power rise.

8.4. ANALYSIS OF THE CONSEQUENCES

Because of the detection methods outlined above, only the immediate consequences of the inadvertent loading of an assembly are analysed.

It is demonstrated that for refuelling conditions (state E), and for all of possible cases of replacement of an assembly with the fuel management's most reactive assembly, the shutdown margin remains above 1000 pcm.

Thus, it is shown that the 'Inadvertent loading of a fuel assembly in an incorrect position' does not lead to a risk of core criticality.

8.5. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

9. FORCED DECREASE OF REACTOR COOLANT FLOW (FOUR PUMPS)

9.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A complete loss of forced reactor coolant flow can be caused by a simultaneous fault in the power supplies to all the RCP [RCS] pumps caused by a drop in frequency on the external grid.

The bounding case studied corresponds to a supply frequency drop at 4 Hz per second to 0 Hz where it remains for an undetermined period.

The typical sequence of events during a forced decrease of reactor coolant flow is as follows.

From the initiating event to the controlled state

A fast supply frequency drop leads to a reversal of the motor torque, which reduces the reactor coolant pumps speed and coolant flow more rapidly than a voltage drop transients which is limited by the inertia of the flywheel.

If the reactor is at power at the time of the incident, the core power is essentially unchanged. As the primary coolant flow is decreasing, the margin to nucleate boiling is reduced. This could result in DNB with subsequent fuel damage, if the reactor is not tripped promptly.

The reactor trip is initiated by the “low RCP [RCS] pump speed” protection function, which is F1A classified.

A frequency drop of 4Hz/s leads to a complete Loss Of Off-site Power (LOOP).

The controlled state is defined below. This state is similar to the LOOP controlled state described in detail in section 6 of Sub-chapter 14.3.

- Nuclear power = 0% full power
- Inlet temperature = 303.3°C
- Pressure = 155 bar
- Boron concentration of the initial power state
- Xenon level higher than or equal to the initial xenon level
- All RCCA fully inserted
- Natural circulation.

The shutdown margin defined in section 1 of Sub-chapter 4.3 supports the required core subcriticality.

From the controlled state to the safe shutdown state

The safe shutdown state corresponds to the following conditions:

- Nuclear power = 0% full power
- Hot leg temperature = 180°C
- Pressure = 30 bar
- Decay heat removed by the steam generators or by LHSI/RHR mode.
- A boron concentration that ensures core subcriticality, even following the xenon depletion
- All RCCA fully inserted
- Natural circulation.

The following actions must be performed to reach the safe shutdown state:

- RCP [RCS] cooldown and depressurisation (F1 classified)
- Boration via the RBS [EBS] (F1 classified).

The forced decrease of the primary coolant flow event is bounded by the following events:

Criterion	Event
Reactivity Release	Loss of Condenser Vacuum described in section 5 of Sub-chapter 14.3
Subcriticality	RCV [CVCS] Malfunction described in section 13 of Sub-chapter 14.3
Heat Removal (DNB)	ASG [MFWS] System Pipe Break described in section 3 of Sub-chapter 14.5

The analysis covering the transient from the initiating event to the automatic reactor trip is presented below.

9.1.1. Safety and Decoupling Criteria

The complete loss of forced reactor coolant flow is a PCC-3 event.

The safety criteria are the dose equivalent limits following a release to the atmosphere, as addressed in Sub-chapter 14.0.

The decoupling criteria assessing the behaviour of the barriers are:

- The number of fuel rods entering DNB must not exceed 10% of the total core
- The average clad hot spot temperature must not exceed 1482°C

- The amount of fuel melting at the hot spot must not exceed 10%.

9.1.2. Reactor Protection System Actions

9.1.2.1. Measures Enabling the Incident Probability to be Limited

These measures are linked to grid reliability and to the reliability of the protection channels.

9.1.2.2. Reactor Protection Actions

A reactor trip to protect against the forced decrease of reactor coolant flow can be initiated by the following signals:

- The “low RCP [RCS] pumps speed” signal

The RCP [RCS] pumps are fitted with speed measurements. These measurements are used to generate a reactor trip signal when the speed in 2 out of 4 RCP [RCS] pumps falls below the setpoint.

- The “low-low reactor coolant flow rate (one loop)” signal

The loss of one RCP [RCS] pump is detected using loop flow measurements. These measurements are used to generate a reactor trip signal using 2 out of 4 logic when the loop flow rate falls below the setpoint. If the nuclear power level is below a low threshold, the low-low reactor coolant flow rate channel is inhibited to allow the plant to be operated with one of the 4 RCP [RCS] pumps stopped.

When the loss of flow rate affects all four RCP [RCS] pumps as a result of voltage or frequency disturbances, protection is provided by the two signals. The reactor trip is actuated by the “low RCP [RCS] pump speed” signal with a higher setpoint and shorter response time. When the loss of flow rate affects only one primary loop, protection is provided by the “low-low reactor coolant flow rate (one loop)” signal.

9.2. METHODS AND ASSUMPTIONS

9.2.1. Single Failure and Preventive Maintenance

The single failure and preventive maintenance assumptions are applied to the F1 systems in the most conservative way for the acceptance criterion being assessed as required by the general safety rules discussed in Sub-chapter 14.0.

The most significant single failure is the assumption that the RCCA with the highest absorption worth fails to enter the core during a reactor trip.

No preventive maintenance assumption is made for this transient.

9.2.2. Method of Analysis

The 3D kinetic neutronic computer code SMART, is coupled with THEMIS, modelling the thermal-hydraulic behaviour of the primary system as described in Appendix 14A, for analysis of the EPR_{4900} in BDR-99 discussed in Appendix 14B. The code suite calculates the following parameters during the transient:

- the core flow rate
- the nuclear power
- the heat flux
- the reactor coolant system pressure
- the reactor coolant system temperature.

The use of a 3D kinetic neutronic model allows the worst axial power shape to be modelled and its impact on the DNBR and reactor trip effectiveness assessed.

The parameters listed above are used by the thermal-hydraulic code FLICA, described in Appendix 14A, to evaluate the number of fuel rods that enter DNB.

9.2.3. Initial Conditions

The assumed initial operating conditions are the most conservative for DNBR. The nominal values are discussed in Sub-chapter 14.1:

- The maximum steady state power level is 102% of the nominal power level, including a 2% uncertainty on the thermal balance.
- The maximum steady-state average coolant temperature, taking into account measurement uncertainties, the average coolant temperature control dead band, and the effectiveness of the control system is the nominal temperature + 2.5° C.
- The minimum steady-state coolant pressure, taking into account measurement uncertainty and the effectiveness of the control system is the nominal pressure - 2.5 bar.
- The vessel flow is equal to 100% of the nominal thermal-hydraulic flow,
- For the 3D kinetic neutronic simulation, the axial power distribution and the nuclear $F\Delta H$ value are such that the initial DNBR value is equal to the low DNBR limiting condition of operation (LCO) value. The initial core Axial Offset (A0) value is 18%.
- For the DNBR calculations to confirm the design DNBR criterion is met, the axial power distribution and the nuclear $F\Delta H$ value are such that the initial DNBR value is equal to the DNBR limiting value. The initial core A0 value is 18%.

9.2.4. Core Related Assumptions

Moderator temperature coefficient

The lowest initial value of the moderator density coefficient is assumed. This value results in the maximum hot-spot heat flux during the initial part of the transient when the minimum DNBR is reached.

Doppler coefficient

This coefficient is set at its maximum absolute value. This reduces the negative reactivity addition during the reactor trip, thus increasing the heat flux when the minimum DNBR is reached.

Fuel-coolant heat transfer coefficient

This coefficient is minimised to maximise the thermal flux during the rod drop.

Shutdown margin

The RCCA having the greatest worth is assumed to be stuck above the core. The negative reactivity insertion following trip is thus minimised, leading to a minimum final subcriticality. The RCCA rod worth versus time is calculated using the conservative top peaked core initial condition with 18% axial offset.

9.2.5. Protection Actions

The reactor trip is initiated following a “low RCP [RCS] pump speed” signal.

The setpoint is assumed to be 91% of the nominal speed.

The total delay between the reactor trip setpoint being reached and the beginning of the RCCA drop is conservatively assumed to be 0.6 seconds.

9.2.6. Control Actions

The transient generates a heat-up within the reactor coolant system whose consequences are mitigated by the average coolant temperature and the pressure controls.

As the effect of the average coolant temperature control is beneficial, it is not modelled in the analysis.

The pressuriser spray flow rate is assumed to be its maximum value to limit the pressure increase.

9.3. RESULTS AND CONCLUSION

The justification that the safety and decoupling criteria are met for this event is based on the results of the PCC transient analysis for EPR_{4900} presented in section 2.6 of Appendix 14B and from the comparison of the relevant parameters for the EPR_{4500} covered by the PSAR and the EPR_{4900} covered by the BDR-99. The PCC transient is not recalculated in the PSAR.

9.3.1. BDR-99 results

All the decoupling acceptance criteria are met. The number of fuel rods experiencing DNB is less than 1%, and therefore significantly below the decoupling criterion of 10%.

9.3.2. Relevant Differences between the EPR_{4500} and the EPR_{4900}

A comparison of the EPR_{4500} with the EPR_{4900} is summarised below:

- Lower power level
- Same thermal-hydraulic core coolant flow coast-down as forced by the grid frequency decrease
- Higher RCP [RCS] average temperature (+1.5°C)
- Slightly lower core outlet temperature
- Same LCO for axial power distribution with an axial offset limit of 18%
- Same LCO for initial DNBR as described in section 6 of Sub-chapter 14.3
- Better RCCA effectiveness for reactor trip
- Similar bounding reactivity feedbacks for the moderator and fuel in absolute value.

9.3.3. Conclusions for the EPR_{4500}

The main parameters affecting the minimum DNBR reached during the transient are the primary flow transient, the core power distribution, limited by the LCO function on axial offset, the initial DNBR value, limited by its dedicated LCO function, and the slight power decrease resulting from reactivity feedback before reactor trip.

Since these parameters are not significantly different for the EPR_{4500}, only slight changes in the DNBR margin are expected when compared with the EPR_{4900}. This would only slightly change the number of rods which experience DNB and the number will remain below the decoupling criterion.

It can be concluded without the need for a dedicated calculation that the decoupling acceptance criteria are met for the EPR_{4500}.

9.4. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

10. LEAK IN THE GASEOUS WASTE PROCESSING SYSTEMS

This transient is only considered for the radiological consequence-related aspects. The relevant calculations are presented in the sub-chapter 'Radiological Consequences of Design Basis Accidents' (see Sub-chapter 14.6).

11. LOSS OF PRIMARY COOLANT OUTSIDE THE CONTAINMENT

11.1. INTRODUCTION

The scenario considered is a loss of coolant outside the containment. During normal plant operation several events could cause such a coolant loss. The representative accident sequence is described below. A loss of coolant accident outside the containment during the operation of the residual heat removal system is described in Sub-chapter 14.6.

This event is classified as a PCC-3 event.

The loss of primary coolant may result from a failure of:

- The Chemical and Volume Control System RCV [CVCS], including the RCV [CVCS] connecting lines, or
- The Nuclear Sampling System REN [NSS].

The most limiting pipe breaks for the Chemical and Volume Control System are:

- A 2A rupture of the RCV [CVCS] line, located between the Volume Control Tank (VCT) and the VCT suction valves
- A 2A rupture of a RCV [CVCS] connecting line, located in the Nuclear Auxiliary Building.

The most limiting pipe break for the Nuclear Sampling System REN [NSS] is:

- A 2A rupture of the pressuriser liquid sampling line, located between the containment penetration and the heat exchanger.

The RCP [RCS] is not significantly affected by such breaks as the break flow is matched by makeup or, in the event of a RCV [CVCS] pump trip, the connecting lines are isolated.

In each case the break isolation time is conservatively assumed to be 60 minutes after the pipe break occurs, based on the following:

- 30 minutes to actuate the alarm from either a Fuel Building sump level high or a Fuel Building activity high signal
- 30 minutes to perform the operator action of manual break isolation from the Main Control Room

Both these accident cases are analysed and the results given in Sub-chapter 14.6.

11.2. SYSTEM SIZING

This event sizes the closure time of the isolation valve in the RCV [CVCS] line.

12. UNCONTROLLED RCCA BANK WITHDRAWAL (STATES B, C, AND D)

This transient is classified as a PCC-3 event if it occurs in reactor states B, C, or D, but it is classified as a PCC-2 event if it occurs in reactor state A. The PCC-2 event is assessed in sections 9 and 10 of Sub-chapter 14.3.

A rod cluster control assembly (RCCA) bank withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks that leads to a power excursion. Such a transient could be caused by a malfunction of the reactor control system. It is assumed that such malfunction could not result in a withdrawal of shutdown RCCA (shutdown RCCA, unlike control RCCA, are not controlled by an automatic system). Therefore, the maximum reactivity insertion that can occur is from the simultaneous withdrawal of all (initially inserted) control RCCA at the maximum withdrawal speed.

The study of this transient relies on a dedicated protection function that eliminates this accident.

This dedicated protection function automatically cuts off the RCCA power supply when the plant leaves plant state A. It is based on primary temperature and primary pressure measurements. The RCCA power supply is automatically cut off when the temperature is lower than its setpoint or when the pressure is lower than its setpoint. Conversely, this permissive signal permits the manual connection of the RCCA power supply by the operator when the temperature is higher than the setpoint and when the pressure is higher than the setpoint.

The primary temperature and primary pressure measurements used in the permissive derivation are acquired from the Protection System and are F1A classified.

The F1 classification of this function allows the uncontrolled RCCA bank withdrawal scenario in states B, C and D to be removed from the Pre-Construction Safety Report.

In addition, there may be a requirement for the movement of the RCCA in shutdown states. To do so without challenging the deletion of the uncontrolled RCCA bank withdrawal scenario, the manual actuation of the RCCA power supply in states B to D is only possible if the coolant boron concentration is higher than the refuelling coolant boron concentration. This is defined to be sufficient to maintain core subcriticality with all RCCAs withdrawn.

13. UNCONTROLLED SINGLE CONTROL ROD WITHDRAWAL

13.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

13.1.1. Definition, Causes, and Description of the Transient

The accidental withdrawal of a single Rod Cluster Control Assembly (RCCA) at power can only occur in the following two, very unlikely cases:

- Case 1: The operator could deliberately withdraw a single control rod, the cluster control being transferred to manual mode, thinking that the clusters are misaligned or that one cluster has dropped.

- Case 2: If the reactor is operating in automatic control mode, several simultaneous electrical or mechanical failures can cause withdrawal of a single RCCA.

For each case, the RCCA withdrawal speed is limited to 75 cm/min. In the assessment described below the initial power level is 102% NP.

The typical sequence of events is described in the following sections.

13.1.1.1. From the Initiating Event to the Controlled State

The withdrawal of a single RCCA will result in an insertion of reactivity which leads to both an increase in average core power and temperature, and in the local power peak in the area of the core from which the RCCA has been withdrawn. The combination of conservative thermal-hydraulic conditions and distorted power distribution may result in a Departure from Nucleate Boiling (DNB) if the core is not adequately protected.

The protection against DNB is provided by the low DNBR channel, which calculates the DNBR on-line. To carry out its protection function for this fault, the calculated DNBR has to be representative of the real DNBR value during the single RCCA withdrawal transient. The calculated DNBR can fail to adequately represent the actual DNBR due to the distorted power distribution in the single RCCA withdrawal case, and taking account of the single failure criterion. This loss of accuracy has to be assessed and taken into account when basing protection claims on the low DNBR channel.

After the reactor trip initiated by the low DNBR channel which is F1A classified, the controlled state is as follows:

- Nuclear power = 0% full power
- Temperature = 303.3°C
- Pressure = 155 bar
- Boron concentration as for the initial power state
- Xenon level higher than or equal to the initial xenon level

- All RCCA fully inserted.

The shutdown margin calculation, discussed in section 5 of Sub-chapter 4.3, provides the core subcriticality in this state. This transient state is similar to that following a loss of condenser vacuum described in section 5 of Sub-chapter 14.3, where more details are provided.

13.1.1.2. From the Controlled State to the Safe Shutdown State

The safe shutdown state corresponds to a state with the following conditions:

- Nuclear power = 0% full power
- Hot leg temperature = 180°C
- Pressure = 30 bar
- Boron concentration sufficient for core subcriticality following the xenon depletion
- Residual heat is removed by the steam generators or the RRA [RHRS]
- All RCCA fully inserted.

The actions to be performed to reach the safe shutdown state are mainly:

- Cooldown via the VDA [MSRT] at a rate of 25°C/h if only one RBS [EBS] pump is available (or at 50°C/h if both RBS [EBS] pumps are available) down to a hot leg temperature of less than 180°C and then a reduction of primary pressure to less than 30 bar (RIS/RRA [SIS/RHRS] operating conditions) by means of the pressuriser safety valves.
- Boration with one or both RBS [EBS] pumps during the cooldown

The single RCCA withdrawal transient is bounded for the activity release by the loss of condenser vacuum discussed in section 5 of Sub-chapter 14.3.

This event is bounded for the heat removal capability by the feedwater line break transient, described in section 3 of Sub-chapter 14.5, as the four steam generators remain available.

This event is bounded for core subcriticality by the RCV [CVCS] malfunction which results in a decrease in boron concentration in the reactor coolant as discussed in section 13 of Sub-chapter 14.3.

The sequence of actions to be performed to reach the safe shutdown state is detailed in section 3 of Sub-chapter 14.5.

The analysis below addresses the transient from the initiating event to the reactor trip.

13.1.2. Safety and Decoupling Criteria

The single RCCA withdrawal at power is classified as a PCC-3 event.

The safety criteria are the dose equivalent limits following a release to the atmosphere discussed in Sub-chapter 14.6. The decoupling criterion is defined in terms of percentage of fuel rods that suffer a Departure from Nucleate Boiling (DNB). The criterion is that the percentage of fuel rods in DNB must be lower than 10%.

A more limiting criterion is imposed by verifying that there is no DNB.

13.1.3. Reactor Protection System Actions

The low DNBR protection function actuates a reactor trip, which protects the fuel against DNB during single RCCA withdrawal at power transients.

The low DNBR channel uses a on-line calculation of DNBR and Self Powered Neutron Detector (SPND) imbalance.

The DNBR calculations are based on the following:

- Power density distribution of the hot channel, which is directly derived from the neutronic in-core instrumentation (SPND). The signals from the SPNDs provide the integrated power along the hot channel using a polynomial function.
- Inlet temperature, derived from the cold leg temperature sensors
- Pressure, derived from the primary pressure sensors
- Relative core flow, derived from the reactor coolant pump speed sensors.

Twelve fuel assemblies are instrumented. Therefore the core is divided into 12 radial zones, each of which is surveyed by one SPND finger as shown in Section 14.4.13 - Figure 1. Thus, 12 on-line DNBR values are calculated. The axial location of the six SPND in their guide tube is shown in Section 14.4.13 - Figure 2.

The low DNBR protection channel follows the global architecture of the protection system with four divisions and a 2 out of 4 downstream vote. Each division uses all 12 on-line DNBR values. In order to avoid spurious actuation of the reactor trip in case of a single SPND failure during normal operation, the basic reactor trip actuation is based on the second lowest value of on-line DNBR.

The SPND imbalance calculation follows the global architecture of the protection system with four divisions and a 2 out of 4 downstream vote. Each division is fed with all 72 SPND signals. The calculation is based on the power density provided by the SPNDs. The differences in power density between two symmetrical SPND are calculated. These are SPNDs at the same axial level and symmetrical with respect to the centre of the core. The sum of the absolute value of these differences corresponds to the SPND imbalance.

The logic of the reactor trip actuation is shown in Section 14.4.13 - Figure 3 and described below:

- In the event of high SPND imbalance, the reactor trip actuation is based on the comparison of the lowest value of on-line DNBR with the setpoint covering imbalance and rod drop conditions. The setpoint covering imbalance and rod drop condition is designed to cover the additional loss of accuracy for the on-line DNBR calculation due to abnormal situations with high SPND imbalance. It is therefore higher and more conservative than the setpoint defined for normal operation.

- In all other situations, the reactor trip actuation is based on the comparison of the second lowest on-line DNBR with the setpoint.

Since the protection against DNB is provided through the low DNBR channel, the objective is to demonstrate that the on-line calculated DNBR is not higher than the actual value during a single RCCA withdrawal transient.

The further objective is to assess the loss of accuracy of the on-line DNBR calculation due to the accuracy of the SPNDs in single RCCA withdrawal situations. This loss of accuracy is considered as an uncertainty to be included in the event of a high SPND imbalance for determining the DNBR imbalance and rod drop setpoint value. A reactor trip is actuated for a calculated DNBR value below the DNBR imbalance and rod drop setpoint value only in cases with high SPND imbalance.

13.2. METHODS AND ASSUMPTIONS

13.2.1. Single Failure Selection

The single failure is applied to one SPND.

The high SPND imbalance setpoint will be reached whichever SPND is assumed to have failed, as several symmetrical SPND pairs would have a high imbalance in the event of a conservative single RCCA withdrawal case.

The worst case is to assume that the failed SPND belongs to the finger providing the lowest on-line calculated DNBR. Therefore, it is assumed that the on-line DNBR from this finger is unavailable. Consequently, the safety analysis must assess the loss of accuracy of the second lowest value of on-line DNBR.

13.2.2. Method of Analysis

13.2.2.1. Calculation of on-line DNBR in Single RCCA Withdrawal Situations

Static calculations are performed at nominal conditions of nominal power, inlet temperature, pressure and core flow.

For each zone, the on-line DNBR is calculated by an algorithm using the power density distribution provided by the SPND finger in the zone as an input.

The power density distribution consists of a polynomial fit built from the responses of the six SPNDs of the finger surveying the zone. The response of a SPND is equal to the product:

SPND calibration coefficient x Absorption rate density in the SPND

The two-energy-group, three-dimensional nodal diffusion code SMART is used to determine the absorption rate density in each of the 72 SPND.

The calibration coefficients are derived from three-dimensional SMART calculations at nominal power with all rods out. Each SPND is calibrated for the fuel rod with the maximum nuclear enthalpy rise in the zone that it surveys as shown in Section 14.4.13 - Figure 1:

$$\text{Calibration coefficient} = \frac{\text{integrated power to the SPND height in the fuel rod of the zone with maximum nuclear enthalpy rise}}{\text{absorption rate density of the SPND}}$$

13.2.2.2. Calculation of the Design DNBR for the Single RCCA Withdrawal Event

The design DNBR value for the single RCCA withdrawal event is determined by the FLICA code.

The three-dimensional SMART code is used to calculate the axial and radial power distributions needed as an input for FLICA.

13.2.2.3. Loss of Accuracy

One on-line DNBR value per zone, and therefore 12 on-line DNBR values for the core, are provided for the zones shown in Section 14.4.13 - Figure 1.

The loss of accuracy is obtained by comparing the second lowest value of on-line DNBR, 12 values for the core, to the actual DNBR.

$$\text{Loss of accuracy} = \frac{2^{nd}\ \text{lowest on line DNBR - actual DNBR}}{\text{actual DNBR}}$$

13.3. RESULTS AND CONCLUSIONS

13.3.1. Results for the EPR_{4250} [Ref-1]

The analysis for the EPR_{4250} is performed at the Beginning of Cycle (BOC) and the End of Cycle (EOC) for UO_2 and MOX fuel management schemes.

The analysis has also been performed at BOC and EOC for the initial core.

Deep RCCA bank insertion is a conservative assumption for a RCCA withdrawal event. The initial position of the control banks is defined according to the rod insertion strategy and rod insertion limits defined in Section 14.4.13 - Table 1.

The assignment of control RCCA to control banks may change during a cycle. Therefore, the analysis has been performed using each of the four sequences of rods positions shown in Section 14.4.13 - Figure 4.

Each individual RCCA was withdrawn from that set of initial conditions.

Cases leading to very high DNBR (>3) in these calculation are neglected. They generally correspond to EOC situations when the $F\Delta H$ is low at around 1.3.

The key results for the EPR_{4250} are presented below in Section 14.4.13 - Table 2.

For all the cases studied for RCCA withdrawal, the loss of accuracy is less than 9%. Consequently, to remain effective during RCCA withdrawal, the low DNBR setpoint value has to take account of this loss of accuracy. As an imbalance method is also used for a rod drop event, the setpoint value will be assessed using the highest loss of accuracy determined for those accidents.

The high setpoint value for SPND imbalance is assumed to be 300 W/cm with all relevant cases leading to a higher value.

13.3.2. Results for the EPR_{4500}

Dedicated calculations are not performed for the EPR_{4500} for the following reasons:

- The results should only be slightly different because the analysis has covered a large number of cases and the limits of RCCA insertion are unchanged for the EPR_{4500}.
- The objective of the analysis is the determination of the loss of accuracy associated with the accident. If, however, these values were higher for the EPR_{4500}, the low DNBR setpoint would be increased to compensate.

Adequate core protection during the uncontrolled withdrawal of single rod from power is therefore provided.

13.4. SYSTEM SIZING

This event is not limiting for the design of the claimed safety systems.

SECTION 14.4.13 – TABLE 1

EPR$_{4250}$ - Maximum Initial Insertion of the Control RCCA at Nominal Power

	RCCA Insertion (cm) at BLX				
	P1	P2	P3	P4	P5
set 1	200	0	0	0	0
set 2	200	100	0	0	0
set 3	200	100	100	0	0
set 4	200	100	100	100	0
set 5	200	100	100	100	100
set 6	100	0	0	0	0
set 7	100	100	0	0	0
set 8	100	100	100	0	0
set 9	100	100	100	100	0
set 10	100	100	100	100	100

	RCCA Insertion (cm) at EOL				
	P1	P2	P3	P4	P5
set 1	100	0	0	0	0
set 2	100	50	0	0	0
set 3	100	50	50	0	0
set 4	100	50	50	50	0
set 5	100	50	50	50	50
set 6	50	0	0	0	0
set 7	50	50	0	0	0
set 8	50	50	50	0	0
set 9	50	50	50	50	0
set 10	50	50	50	50	50

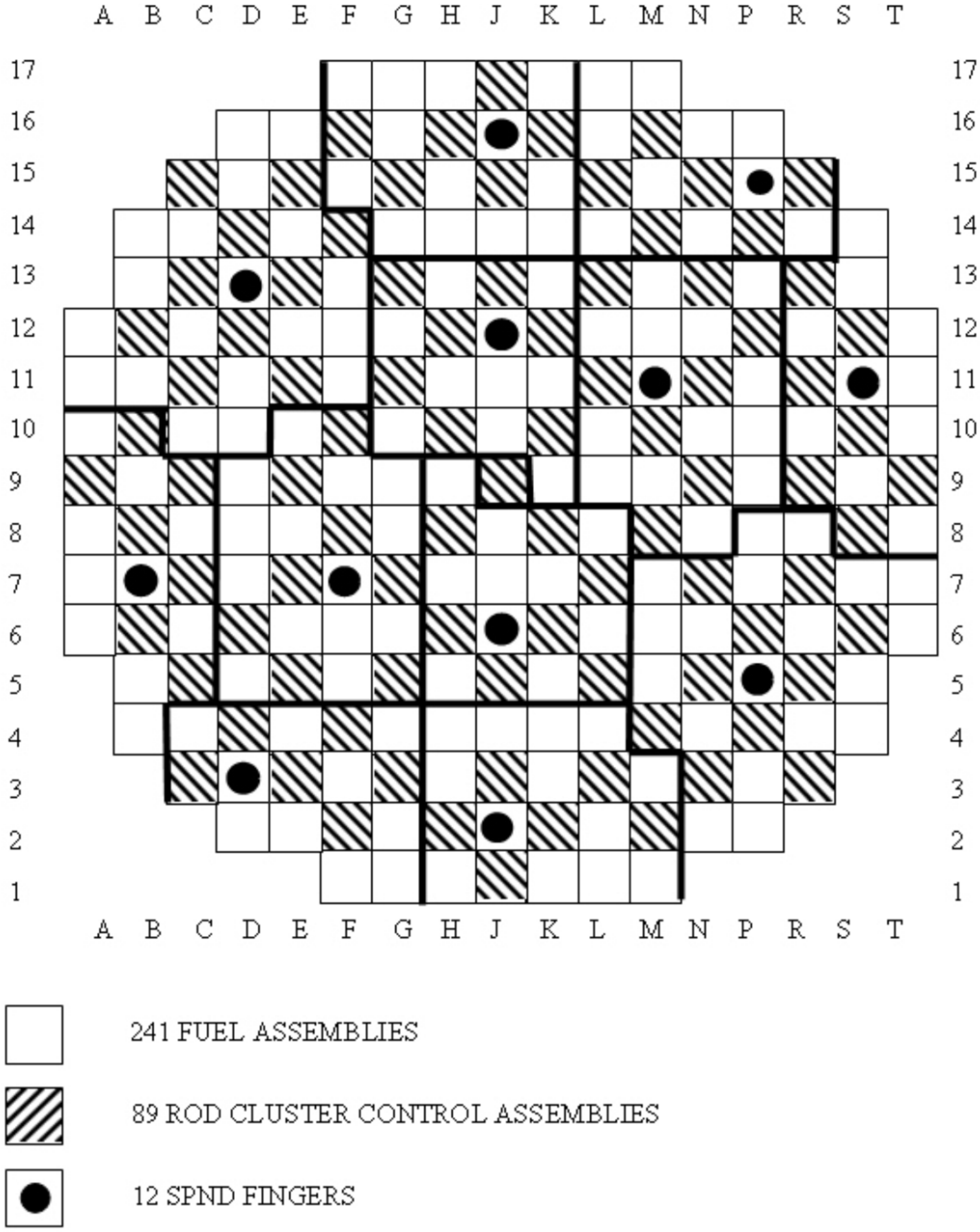
SECTION 14.4.13 – TABLE 2

EPR_{4250} – MAXIMUM LOSS OF ACCURACY (LOA) FOR THE RCCA WITHDRAWAL EVENT

Sequence of Rods Position	Location of Withdrawn Rod	Insertion of the Control RCCA	Fuel Management	Cycle	LOA 1st on-line DNBR (%)	LOA 2nd on-line DNBR (%)	SPND Imbalance (w/cm)
1	N3	set 1	First core	1, BLX	0	5.9	373
2	R7	set 2	UO_2 – INOUT-18 mo	equilibrium, BLX	4.8	6.5	398
3	C9	set 1	First core	1, BLX	-0.3	8.5	447
4	L3	set 1	First core	1, BLX	-0.1	8.5	399

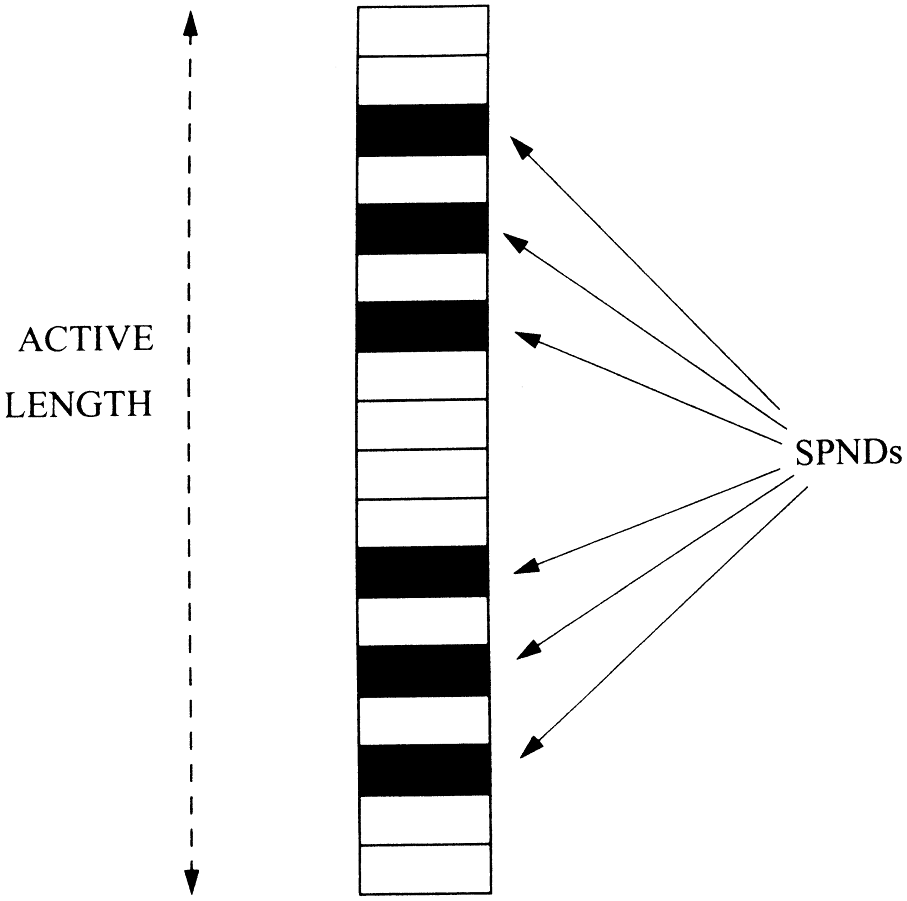
SECTION 14.4.13 - FIGURE 1

Radial Location of SPND Fingers and Radial Zones



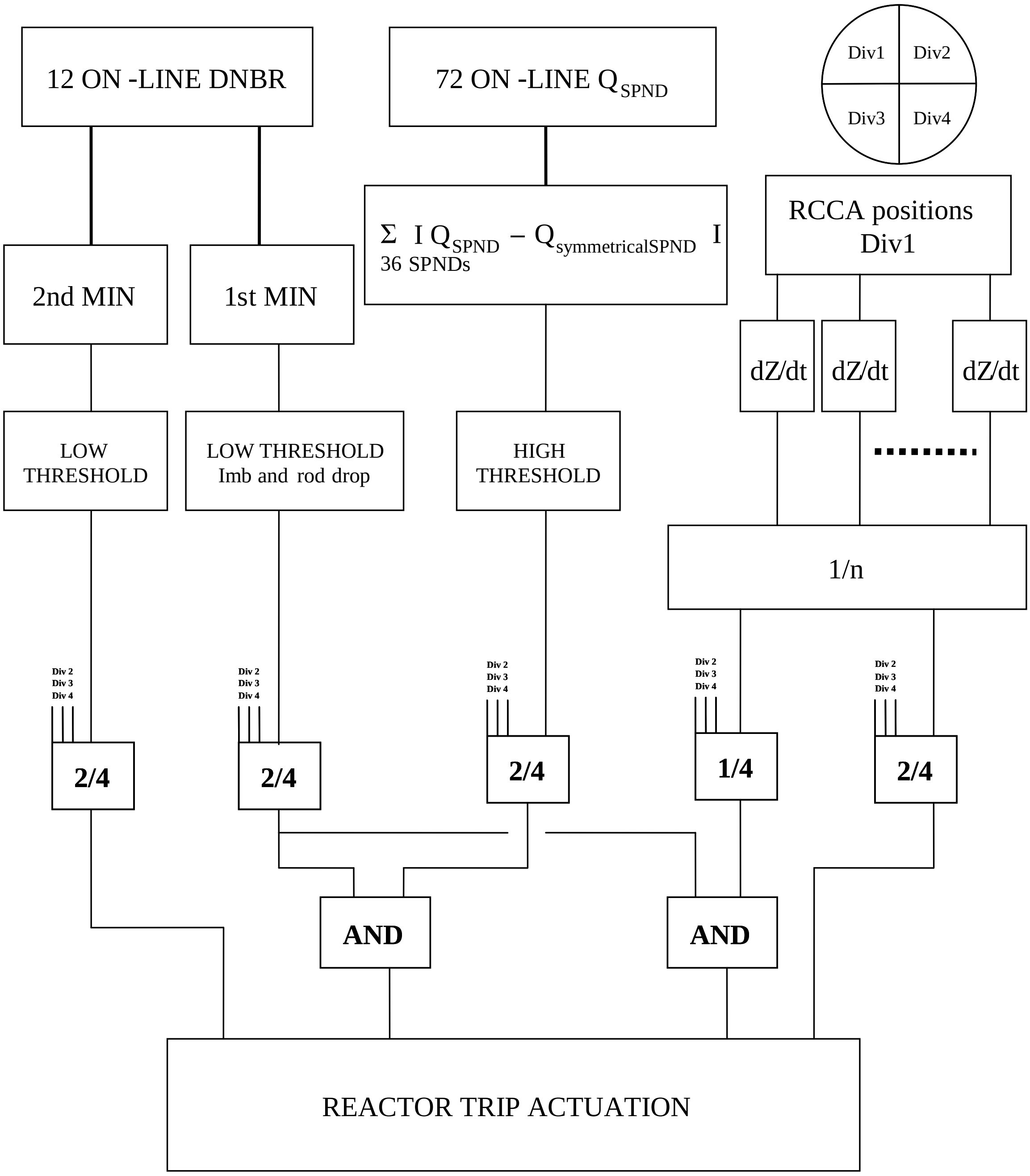
SECTION 14.4.13 - FIGURE 2

Axial Location of the SPNDs in a Fuel Assembly



SECTION 14.4.13 - FIGURE 3

Logic of Low DNBR Reactor Trip Actuation for Single Rod Withdrawal at Power



SECTION 14.4.13 - FIGURE 4 (1/4)

Definition of Control Banks Sequence No. 1
(Control Rods: P, Shutdown Rods: N)

17									N								
16						N		N		N		N					
15			N		P1		P3		P4		P3		P5		N		
14				N		N						N		N			
13			P5		P5		P4		P5		P2		P5		P1		
12		N		N				N		N				N		N	
11			P3		P2		N				N		P4		P3		
10		N				N		N		N		N				N	
09	N		P4		P5				P5				P5		P4		N
08		N				N		N		N		N				N	
07			P3		P4		N				N		P2		P3		
06		N		N				N		N				N		N	
05			P1		P5		P2		P5		P4		P5		P5		
04				N		N						N		N			
03			N		P5		P3		P4		P3		P1		N		
02						N		N		N		N					
01									N								
	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T

SECTION 14.4.13 - FIGURE 4 (2/4)

Definition of Control Banks Sequence No. 2
(Control Rods: P, Shutdown Rods: N)

17									N								
16						N		N		N		N					
15			N		P4		P3		P4		P1		P5		N		
14				N		N						N		N			
13			P5		P2		P5		P5		P3		P2		P4		
12		N		N				N		N				N		N	
11			P1		P3		N				N		P5		P3		
10		N				N		N		N		N				N	
09	N		P4		P5				P5				P5		P4		N
08		N				N		N		N		N				N	
07			P3		P5		N				N		P3		P1		
06		N		N				N		N				N		N	
05			P4		P2		P3		P5		P5		P2		P5		
04				N		N						N		N			
03			N		P5		P1		P4		P3		P4		N		
02						N		N		N		N					
01									N								
	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T

SECTION 14.4.13 - FIGURE 4 (3/4)

Definition of Control Banks Sequence No. 3
(Control Rods: P, Shutdown Rods: N)

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T
17									N								
16						N		N		N		N					
15			N		P3		P4		P1		P4		P5		N		
14				N		N						N		N			
13			P5		P5		P3		P5		P2		P5		P3		
12		N		N				N		N				N		N	
11			P4		P2		N				N		P3		P4		
10		N				N		N		N		N				N	
09	N		P1		P5				P5				P5		P1		N
08		N				N		N		N		N				N	
07			P4		P3		N				N		P2		P4		
06		N		N				N		N				N		N	
05			P3		P5		P2		P5		P3		P5		P5		
04				N		N						N		N			
03			N		P5		P4		P1		P4		P3		N		
02						N		N		N		N					
01									N								

SECTION 14.4.13 - FIGURE 4 (4/4)

Definition of Control Banks Sequence No. 4
(Control Rods: P, Shutdown Rods: N)

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T
17									N								
16						N		N		N		N					
15			N		P4		P1		P5		P3		P5		N		
14				N		N						N		N			
13			P5		P2		P4		P5		P3		P2		P4		
12		N		N				N		N				N		N	
11			P3		P3		N				N		P4		P1		
10		N				N		N		N		N				N	
09	N		P5		P5				P5				P5		P5		N
08		N				N		N		N		N				N	
07			P1		P4		N				N		P3		P3		
06		N		N				N		N				N		N	
05			P4		P2		P3		P5		P4		P2		P5		
04				N		N						N		N			
03			N		P5		P3		P5		P1		P4		N		
02						N		N		N		N					
01									N								

14. LONG TERM LOSS OF OFFSITE POWER (LOOP): FUEL POOL COOLING ASPECTS (STATE A)

14.1. INTRODUCTION

Loss of offsite power (LOOP) results in the loss of the electrical supply to all plant auxiliaries. For the fuel pool cooling function, this means that all pumps in the main PTR [FPCS] trains and the support systems are stopped. In this situation, the four emergency diesel generators (EDGs) start and restore electrical supplies to the components connected to the busbars supplied by the EDG.

A simplified flow diagram of the system, the electrical supply to the PTR [FPCS] pumps and the methodology used for the analysis presented in this section are described in Sub-chapter 14.3.

In the present section, the PCC-3 event 'Long term loss of offsite power in state A' is analysed with regards to its effects on the spent fuel pool cooling. This case also bounds the PCC-4 event 'Long term loss of offsite power in state C' and the PCC-2 event 'Short term loss of offsite power (< 2 hours) in states A, C, and D'.

The grace period before reaching a water temperature of 80°C, which is applied as a decoupling criterion for this fault, is in all cases, significantly longer than 2 hours. This is discussed in section 14.3 below. Therefore no countermeasures are necessary for the PCC-2 event where offsite power is recovered within 2 hours of the initiating event.

14.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

To address state A, which corresponds to reactor power operation, it is necessary to perform analyses for both end of cycle (EOC) and beginning of cycle (BOC) PTR [FPCS] conditions. In both cases, the fuel pool is cooled in normal operation by a main train with one in-service PTR [FPCS] pump. The PTR [FPCS] heat exchanger is cooled by an RRI [CCWS] train.

<u>Case 1</u>: At EOC, i.e. before the beginning of shutdown for refuelling, maintenance of a main PTR [FPCS] can be scheduled as the power in the pool is at a minimum for state A. In these conditions the maximum heat load to be removed from the fuel pool is 2.99 MW [Ref-1] [Ref-2].

Note: More information on preventive maintenance assumptions is provided in Sub-chapter 14.0.

<u>Case 2</u>: At BOC, i.e. at the start of a new reactor operating cycle when the heat load in the fuel pool is the maximum for state A, the maximum heat load to be removed from the fuel pool is 5.85 MW. Maintenance can be performed on a support system, for example, the RRI [CCWS] in these circumstances [Ref-1] [Ref-2].

Note: BOC is the most onerous time for this type of maintenance and therefore is bounding for assessing the impact of maintenance during the complete cycle.

An initial fuel pool water temperature of 50°C is assumed which bounds all operating situations.

14.3. GRACE PERIOD

An initial water volume in the fuel pool of 1463 m^3 during normal operation is assumed, corresponding to a water level of 18.90 m at the suction line (no passive failure i.e. leakage, is considered for this PCC [Ref-1][1]).

For this volume, the grace period without any cooling means is calculated as:

<u>For EOC:</u> The water temperature in the fuel pool will reach 80°C at 16.5 hours after a total loss of the cooling function. Boiling of the fuel pool water will start after a period of 27.4 hours. These values are calculated for the limiting case with MOX fuel

<u>For BOC</u>: The water temperature in the fuel pool will reach 80°C at 8.4 hours after a total loss of the cooling function. Boiling of the fuel pool water will start after 14.0 hours. These values are calculated for the limiting case with MOX fuel

14.4. BOUNDARY CONDITIONS

The transient is analysed using conservative assumptions, consistent with the approach adopted for all other PCC events. Hence, the fuel pool heat loads for EOC and BOC are calculated with the bounding decay heat value for BOC, i.e. 5.85 MW, which includes a safety margin.

Single failure and preventive maintenance are combined with the event.

The precautionary start-up of the third PTR [FPCS] train during maintenance work could be taken into account, but would not have any positive effect on this transient as the initiating event (LOOP), makes all trains unavailable, including the third train. A subsequent restart of the third train (which would require a state change) cannot be considered in the PCC study as this train is classified F2.

14.5. DECOUPLING CRITERIA

The PCC-3 ‘without fuel pool drainage’ decoupling criterion for the PTR [FPCS] design states that the water temperature in the fuel pool must not exceed 80°C.

14.6. TRANSIENTS

The most onerous configurations are analysed below. These two cases bound all other initiating events, as well as all possible combinations of single failure and preventive maintenance.

[1] This analysis is a Flamanville 3 (FA3) study that takes into account some specific features of the FA3 design but the results are bounding for the UK design. In particular, the analysis considers that the third PTR [FPCS] train is lower than for the UK design. Therefore, the calculations are conservative.

Case 1: EOC: in this situation, maintenance may be performed on either a complete main train or sections of this train, e.g. the pump, heat exchanger, pipework common to the two RRI [CCWS] trains, main switchboard or switchboard supplying the two pumps in one main train. In the most onerous case, maintenance is considered on the heat exchanger of a main train (for example main train 2) so the entire main train is declared unavailable. (Case 1 is calculated using the bounding decay heat value for BOC, i.e. 5.85 MW (see section 14.4)

After the loss of offsite power (LOOP), the pumps in all trains are stopped including the extra third train. A signal to start the EDGs is generated on loss of voltage on the plant emergency buses: the EDGs restore supplies to the four main PTR [FPCS] pumps and the pumps of the support systems (RRI [CCWS]). If the most onerous single failure is assumed - on the diesel of the electrical division supplying main train 1 (EDG 2), main PTR [FPCS] train 1 cannot restart.

In this situation it is necessary to install cross-connection no. 28 to electrical division 1 to recover the electrical supply to PTR [FPCS] pumps 1 and 2 and to start these two pumps [Ref-1]. This approach is possible, since the grace period of 16.5 hours before the fuel pool water temperature reaches 80°C is sufficient for operators to perform the installation of this cross-connection and start PTR [FPCS] train 1. In this situation, with cooling provided by one main PTR [FPCS] train, the stabilised fuel pool temperature will remain lower than that assessed in case 2 below.

If preventive maintenance is being performed on one PTR [FPCS] pump or electrical switchboard at the beginning of the event, rather than on a heat exchanger, the situation is less severe. Following the LOOP, in this case the available equipment is sufficient to start one main PTR [FPCS] train with one RRI [CCWS] available for cooling, even allowing for the single failure. It is not necessary to install the electrical supply cross-connections.

Case 2: BOC: in this situation maintenance may be performed on one support system, for example on the RRI 3 [CCWS 3] line. Initially, the first main PTR [FPCS] train is in operation, cooled by RRI 1 [CCWS 1].

After the loss of offsite power (LOOP), the pumps in all trains are stopped including that in the third train. A signal to start the EDGs is generated on loss of voltage on the plant emergency buses and the EDG supplies are restored to the four main PTR [FPCS] pumps and the pumps of the support systems (RRI [CCWS]).

In this situation, the single failure can be applied anywhere in the system or support systems, diesel generators, switchboards, PTR [FPCS] pumps, RRI [CCWS] pumps, etc. Whichever failure is assumed, at least one PTR [FPCS] pump cooled by at least one RRI [CCWS] train including the relevant cooling chain, remains available. As a result, assuming the most onerous configuration and decay heat in the fuel pool, the long-term water temperature will not exceed 52°C (assuming a RRI [CCWS] design temperature of 38°C).

14.7. CONCLUSION

In all cases (BOC and EOC), the water temperature in the fuel pool remains below 80°C throughout the transient. Therefore the PCC-3 ‘without fuel pool drainage’ decoupling criterion is fulfilled following a ‘Long term LOOP, fuel pool cooling aspects (state A)’ event.

The main assumptions and results of this study are detailed in Section 14.4.14 - Table 1 below.

14.8. EFFECT OF A REACTOR EVENT ON THE PTR [FPCS]

In addition to the studies concerning the PTR [FPCS], PCC-3 or PCC-4 events affecting the reactor have been assessed with regard to their effect on the PTR [FPCS] and pool temperature.

The events have been assessed by calculating the pool temperature in steady state conditions, assuming:

- Normal operation of the PTR [FPCS] cooling trains,
- Different PTR [FPCS] configurations: beginning and end of cycle, and the end of refuelling, with either one or two main trains in service,
- Maximum decay heat for a MOX fuel management scheme with safety margin,
- a PTR [FPCS] pool water volume for normal operation of 1463 m^3,
- a decoupling RRI [CCWS] temperature of 45°C, which is representative of the maximum temperature which may occur in the RRI [CCWS] during a PCC-3 or PCC-4 transient affecting the reactor core.

The maximum fuel pool temperature calculated is 59°C, for a decay heat of 5.85 MW including safety margin, and with only one main PTR [FPCS] train in operation.

This result demonstrates that PCC-3 or PCC-4 core events do not have a significant effect on the PTR [FPCS] system.

SECTION 14.4.14 - TABLE 1

Main assumptions and results (4500 MWth)

		End of cycle	Beginning of cycle
Decay heat (MW)	**With margins**	**2.99**	**5.85**
$T_{SEC\ [ESWS]}$ / $T_{RRI\ [CCWS]}$ / $T_{SRU\ [UCWS]}$ (°C)		**30 / 38 / 30**	**30 / 38 / 30**
$T_{fuel\ pool}$ (initial) (°C)		**50**	**50**
Fuel pool water volume (m^3)		**1463**	**1463**
Grace period without any cooling (hours)			
	To reach 80°C	**16.5**	**8.4**
	To reach 100°C	**27.4**	**14.0**

Decay heat (MW)	**With margins**	**2.99**	**5.85**
Decay heat used for calculation(MW)	**With margins**	**5.85**	**5.85**
$T_{fuel\ pool}$ (final) (°C)		**<52**	**<52**

15. LOSS OF ONE TRAIN OF THE FUEL POOL COOLING SYSTEM (PTR [FPCS]) OR OF A SUPPORTING SYSTEM (STATE F)

15.1. INTRODUCTION

The loss of a cooling train of the PTR [FCPS] system is studied in state F, i.e. during refuelling. In this situation, the core is being unloaded from the reactor vessel and placed in the spent fuel pool. Two PTR [FPCS] main trains provide cooling during this phase of operation.

The principal flow diagram, the electrical supply of the pumps of the PTR [FPCS] and the methods used for the analyses given in this section, are described in Sub-chapter 14.3.

15.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

In normal operating conditions for this plant state, two PTR [FPCS] main trains with one pump operating in each train are used to cool the fuel pool.

The maximum heat load in the spent fuel pool occurs just as the last fuel element has been unloaded from the reactor vessel and placed inside the fuel pool. This occurs approximately 111 hours after shutdown and corresponds to a decay heat of 20.81 MW [Ref-1].

An initial fuel pool water temperature of 50°C is assumed, which bounds all operating states.

15.3. GRACE PERIOD

An initial water volume in the fuel pool of 1463 m^3 is assumed during normal operation. This corresponds to a water level of 18.9 m at the suction line (see note below) [Ref-1][2] [Ref-2]. In these circumstances, the grace period from the loss of cooling before the fuel pool water temperature reaches 80°C is 2.4 hours: the grace period to the onset of boiling is 3.9 hours.

Note: no passive failure is considered in this PCC-3 study. Fuel pool drainage is studied in section 16.

15.4. BOUNDARY CONDITIONS

The transient is analysed using conservative assumptions, consistent with the approach adopted for all other PCC events. Hence, a fuel pool heat load for state F of 20.81 MW is calculated, with a safety margin included.

Single failure and preventive maintenance are combined with the event. In this case, the maintenance of two LJ switchboards and the two corresponding RRI [CCWS] trains is assumed. Planned maintenance of the PTR [FPCS] main trains is not expected in this plant state.

[2] This analysis is a Flamanville 3 (FA3) study that takes into account some specific features of the FA3 design but the results are bounding for the UK design. In particular, the analysis considers that the third PTR [FPCS] train is lower than for the UK design. Therefore, the calculations are conservative.

A loss of offsite power (LOOP) occurring following an earthquake without additional failure is considered in the analyses, consistent with the general study rules discussed in Sub-chapter 14.0.

The third PTR [FPCS] train is usually started during maintenance works as a preventive measure. This gives an additional safety margin, but this preventive start-up is not necessary in this case as shown by the analyses below.

15.5. DECOUPLING CRITERIA

The PCC-3 ‘without fuel pool drainage’ decoupling criterion for the PTR [FPCS] design states that the water temperature in the fuel pool must not exceed 80°C.

15.6. TRANSIENTS

The most onerous configurations are analysed below. These cases bound all other initiating events, as well as all possible combinations of single failure and preventive maintenance. In particular, the failure of the main or the dedicated switchboards as well as the failure of the RRI [CCWS] pumps is covered by case 2 below.

In both cases analysed below, preventive maintenance is assumed on RRI [CCWS] trains 2 and 4 and on the two corresponding LJ switchboards.

PTR [FPCS] main trains 1 and 2 are in operation assuming:

- PTR [FPCS] pump 1 in operation with the heat exchanger cooled by RRI 1 [CCWS 1]
- PTR [FPCS] pump 3 in operation with the heat exchanger cooled by RRI 3 [CCWS 3]

Case 1: If the PTR [FPCS] pump 1 is affected by the initiating event and the single failure applied on the PTR [FPCS] pump 2, then the PTR [FPCS] main train 1 is lost.

The PTR [FPCS] components of main train 2 are not impacted by the event and hence remain available and in operation.

Case 2: If the heat exchanger of the PTR [FPCS] train 1 is affected by the initiating event, then PTR [FPCS] main train 1 is lost. The single failure should not be applied to the components of main train 2, as these components are in operation before the event and consequently remain available and in operation as no change in state is required. Therefore PTR [FPCS] main train 2 remains available and in operation.

With only one main train in operation, the fuel pool water temperature stabilises at 65°C (assuming an RRI [CCWS] design temperature of 38°C).

In case of LOOP (without applying the single failure criterion), all PTR [FPCS] trains lose power. The emergency diesel generators then start up and power is restored to the main PTR [FPCS] trains and their support systems, enabling pool cooling to be restored (once the safe state has been achieved).

15.7. CONCLUSION

The water temperature in the fuel pool remains lower than 80°C throughout the transient. Therefore the PCC-3 ‘without fuel pool drainage’ decoupling criterion is fulfilled for the ‘Loss of one train of the fuel pool cooling system (PTR) [FPCS] or of a supporting system (state F)’ transient.

The main assumptions and results of this study are detailed in Section 14.4.15 - Table 1 below.

SECTION 14.4.15 - TABLE 1

Main assumptions and results (4500 MWth)

		End of refuelling
Decay heat (MW)	**With margins**	**20.81**
$T_{SEC\ [ESWS]}$ / $T_{RRI\ [CCWS]}$ / $T_{SRU\ [UCWS]}$ (°C)		**30 / 38 / 30**
$T_{fuel\ pool}$ (initial) (°C)		**50**
Fuel pool water volume (m^3)		**1463**
Grace period without any cooling (hours)		
	To reach 80°C	**2.4**
	To reach 100°C	**3.9**
$T_{fuel\ pool}$ (final) (°C)		**65**

16. ISOLATABLE PIPING FAILURE ON A SYSTEM CONNECTED TO THE SPENT FUEL POOL (STATE A TO F)

16.1. INTRODUCTION

A piping failure resulting in leakage from the fuel pool may occur in one of the following systems, all connected to the fuel pool, depending on its initial configuration:

- In the fuel building, a purification line or a skimming line connected to the spent fuel pool, the transfer compartment or the underwater loading pit when one of these is connected to the spent fuel pool in states A to F,
- In the reactor building, a purification line or a skimming line connected to the reactor cavity, the core internal storage or the reactor building transfer compartment in state E,
- A main PTR [FPCS] cooling train, when operating in states A to F,
- The extra (third) PTR [FPCS] cooling train, when it is operating, i.e. during maintenance work on a main PTR [FPCS] cooling train or a support system in state A

Diagrams describing these systems are provided in Sub-chapter 9.1.3 - Figure 2.

The methodology used in the analyses presented in this section is described in Sub-chapter 14.3.

The upper end of a fuel assembly that is being handled is at a level of +16.20 m and that of an assembly stored in the spent fuel pool rack is at a level of +10.30 m [Ref-1][3] [Ref-2].

16.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

During normal operating conditions in states E and F, two PTR [FPCS] main trains with one pump operating per train are used to cool the fuel pool.

In state A only, the extra (third) train is started during maintenance work on the PTR [FPCS] system or on a support system as a preventive measure.

Purification lines may be opened in the fuel building in states A to F and in the reactor building in state E, except during maintenance work on the electrical switchboards supplying one or both redundant isolation valves.

[3] This analysis is a Flamanville 3 (FA3) study that takes into account some specific features of the FA3 design but the results are bounding for the UK design. In particular, the analysis considers that the third PTR [FPCS] train is lower than for the UK design. Therefore, the calculations are conservative.

For failures occurring on a purification skimming or water make-up line, or a main PTR [FPCS] cooling train, the most onerous case assumes that the maximum possible heat load has to be removed from the spent fuel pool. This occurs just as the last fuel element has been unloaded from the reactor vessel and placed inside the fuel pool. This occurs approximately 111 hours after shutdown and corresponds to a decay heat of 20.81 MW [Ref-1].

A maximum decay heat of 5.85 MW, corresponding to the beginning of cycle (BOC), will be assumed in the study of failures related to the third PTR [FPCS] train as a bounding case for state A.

An initial fuel pool water temperature of 50°C is assumed, which bounds all operating states.

16.3. GRACE PERIOD

For PCC events involving draining of the fuel pool, the grace period is calculated with a reduced initial fuel pool water volume of:

- 1227 m^3 if the failure occurs on a main PTR [FPCS] train. This corresponds to a water level of 16.85 m after which the leakage is stopped by the anti-siphon device.

  In this case, the fuel pool water temperature will reach 97°C at 3.1 hours after the loss of the cooling function, assuming a decay heat of 20.81 MW.

- 1153 m^3 if the failure occurs on the third PTR [FPCS] train. This corresponds to a water level of 16.20 m after which the leakage is stopped by uncovery of the suction line.

  In this case, the fuel pool water temperature will reach 97°C at 10.3 hours after the loss of the cooling function, assuming a decay heat of 5.85 MW.

The water volume considered in the grace period calculations is conservatively assumed to consist only of the water initially resident in the spent fuel pool and does not include any make-up water. In addition, the analysis conservatively ignores any benefit due to the reduced make-up water temperature.

16.4. BOUNDARY CONDITIONS

The transient is analysed using conservative assumptions, consistent with the approach adopted for all PCC events. Therefore, the fuel pool heat load is considered with a safety margin.

Single failure and preventive maintenance are combined with the event.

A loss of offsite power (LOOP) occurring following an earthquake is considered in the analyses, without additional failure, consistent with the general study rules discussed in Sub-chapter 14.0.

The start-up of the third PTR [FPCS] train during maintenance work as a preventive measure is considered in the study.

16.5. DECOUPLING CRITERIA

For PCC events involving draining of the fuel pool, the decoupling criterion for the PTR [FPCS] design is avoidance of pool boiling throughout the transient. Therefore, the studies aim to show that the fuel pool water temperature remains below a maximum temperature of 97°C.

In the long term, following restoration of a main PTR [FPCS] train, the water temperature in the fuel pool must not exceed 80°C.

16.6. TRANSIENTS

16.6.1. Piping failure on a purification line in the fuel building

In the event of a break on a purification pipe in the fuel building, redundant motorised valves located on the drain lines connected to the back of the transfer compartment and the underwater loading pit automatically close. This closure is initiated following the detection of a low water level in the spent fuel pool at +18.40 m. The closure provides a double isolation.

The draining through the suction pipe descending into the spent fuel pool and the discharge pipes located in each compartment is stopped at a fuel pool water level of +18.05 m by the anti-siphon devices installed on each line [Ref-1].

A controlled state with the draining stopped is therefore reached automatically before the PTR [FPCS] pumps are automatically shutdown by a low-level signal at +18.00 m, and hence without any loss of the PTR [FPCS]. In addition, no water make-up is required to reach the controlled state. Consequently, the safe shutdown state is equivalent to the controlled state.

16.6.2. Piping failure on a skimming line in the fuel building

The spent fuel pool surface skimming system is installed above the +18.00 m level. [Ref-1] Therefore, siphoning due to a break on this system would not lead to automatic shutdown of the PTR [FPCS] pumps at +18.00 m.

The controlled state is therefore automatically reached without any loss of the PTR [FPCS] and without any requirement for water make-up. Consequently, the safe shutdown state is equivalent to the controlled state.

16.6.3. Piping failure on a fuel pool water make-up line in the fuel building

The water make-up lines do not descend below the +18.00 m level in the spent fuel pool. Therefore, siphoning due to a break on a make-up line would not lead to automatic shutdown of the PTR [FPCS] pumps at +18.00 m [Ref-1].

The controlled state is therefore automatically reached, without any loss of the PTR [FPCS] and without any requirement for water make-up. Consequently, the safe shutdown state is equivalent to the controlled state.

16.6.4. Piping failure on a purification line in the reactor building

Following detection of the draining, one of the actions is the closure of the transfer tube valve, if not previously closed, to isolate the fuel building pool from the reactor building pool. The subsequent transient progression would therefore be different depending on whether the transfer tube is closed or open.

Case 1: Transfer tube closed

In the event of a break on a purification pipe in the reactor building, redundant motorised valves located on the drain lines connected to the back of the reactor cavity, the core internal storage or the reactor building transfer compartment automatically close. This closure signal is initiated on detection of a low water level in the reactor building transfer compartment at +17.90 m. The closure provides a double isolation.

The drainage through the discharge pipes descending into each compartment is stopped at a fuel pool water level of +18.05 m by the anti-siphon devices installed on each line [Ref-1].

The controlled state is therefore automatically reached and, as the transfer tube is closed, the spent fuel pool water level is stabilised above the setpoint for the PTR [FPCS] pumps automatic shutdown signal of +18.00 m. Consequently, the safe shutdown state is reached.

Case 2: Transfer tube open

The motorised isolation valves, located on the drain lines connected at the back of the reactor cavity, the core internal storage or the reactor building transfer compartment, are automatically closed on detection of a low water level of +18.40 m in the spent fuel pool. The closure provides a double isolation.

Also, the drainage through the discharge pipes descending into each compartment is stopped at a fuel pool water level of +18.05 m by the anti-siphon devices [Ref-1].

The controlled state is therefore automatically reached without any loss of the PTR [FPCS]. It is reached before the PTR [FPCS] pumps are automatically shutdown by a low-level signal at +18.00 m, and without any water make-up being required. Consequently, the safe shutdown state is equivalent to the controlled state.

16.6.5. Piping failure on a skimming line in the reactor building

A floating device provides the surface skimming of the reactor building pools. A break on this system leads to draining at a low flow rate of less than 50 m^3/h. This leaves sufficient time (more than 1 hour) for the operator to remove the floating device and stop the draining.

Following detection of the draining, one of the actions is the closure of the transfer tube isolation valve, if not previously closed, to isolate the fuel building pool from the reactor building pool. The subsequent transient progression would therefore be different depending on whether the transfer tube is closed or open.

Case 1: Transfer tube closed

Following a break on a reactor building skimming system pipe, the time between the first significant water level alarm at +18.90 m and the actuation of the PTR [FPCS] pump automatic shutdown signal at +18.00 m, is 2.3 hours. This period is sufficient for the operator to remove the floating device.

The controlled state is therefore reached and, as the transfer tube is closed, the spent fuel pool water level is stabilised above the setpoint of the PTR [FPCS] pumps automatic shutdown of +18.00 m. Consequently, the safe shutdown state is reached.

Case 2: Transfer tube open

Following a break on a reactor building skimming system pipe, the time between the first significant water level alarm at +18.90 m and the actuation of the PTR [FPCS] pumps automatic shutdown signal at +18.00 m, is 4.8 hours. This period is sufficient for the operator to remove the floating device.

The controlled state is therefore reached, without any loss of the PTR [FPCS] and without a requirement for water make-up. Consequently, the safe shutdown state is equivalent to the controlled state.

16.6.6. Piping failure on a main PTR [FPCS] cooling train

16.6.6.1. Introduction

This sub-section describes the analysis of a pipe break on a main train, leading to a complete but temporary loss of fuel pool cooling when the pool level drops to +18.00 m due to automatic shutdown of the main PTR [FPCS] pumps.

Maintenance is not authorised on the main trains in states E or F, but can be carried out on the two electrical divisions in state F. Interconnections are installed prior to maintenance in this case.

This sub-section does not specifically consider the scenario of a pipe break in a main train during maintenance on the other train in state A, as it would not lead to a loss of pool cooling (provided by the third PTR [FPCS] train started as a preventive measure whose switch-off only occurs when the pool water level reaches +16.40 m) or to uncovery of a fuel assembly during maintenance, as fuel handling is prohibited once the third PTR [FPCS] train is operational.

In the case of LOOP, the third PTR [FPCS] train also shuts down and fuel pool cooling is then completely lost. The emergency diesel generators (EDGs) are then started up and electrical power is restored to the third train and its support systems, enabling fuel pool cooling to be restored.

This sub-section describes the analysis of piping failures leading to a complete and temporary loss of the fuel pool cooling when the pool level drops to +18.00 m due to automatic shutdown of the main PTR [FPCS] pumps [Ref-1][4].

16.6.6.2. Transient description

A piping failure on this system may occur in any reactor state from A to F.

The resulting fuel pool draining causes an automatic shutdown of the PTR [FPCS] pumps when the fuel pool water level drops to +18.00 m, resulting in the loss of the fuel pool cooling function.

[4] This analysis is a Flamanville 3 (FA3) study that takes into account some specific features of the FA3 design but the results are bounding for the UK design. In particular, the analysis considers that the third PTR [FPCS] train is lower than for the UK design. Therefore, the calculations are conservative.

A LOOP is assumed to occur at the moment of the drainage initiating event. Even though power is automatically restored to the PTR [FPCS] main trains following start-up of the EDGs, it is assumed, as a bounding hypothesis, that the PTR [FPCS] trains are started manually and that the spent fuel pool is without cooling for one hour. This one hour loss of cooling is added to the duration of loss of cooling due to the low water level (below +18.00 m) once the water level is above +18.00 m. An initial level of 19.00 m is therefore assumed, to maximise the drainage duration and the duration of loss of cooling due to LOOP. In addition, the analysis conservatively ignores the beneficial cooling effect once the PTR [FPCS] trains have restarted.

The draining through the discharge and suction pipes is finally stopped by the following mechanisms:

- for rapid drainage, at a fuel pool water level of +16.85 m via the anti-siphon devices fitted to each suction pipe in the pool. The controlled state is therefore achieved passively. The PTR [FPCS] train containing the break is located and manually isolated using two redundant isolation valves on the suction pipe. Once this isolation has been performed, the safe state is reached with long term fuel pool cooling restored.

- for slow drainage, at a fuel pool water level higher than +16.85 m by manual isolation of the suction for each PTR [FPCS] train that was initially operational and also via the anti-siphon device fitted to the discharge pipe at +18.05 m [Ref-1]. The controlled state is therefore reached.

Water make-up is then performed, to a minimum water level of +18.05 m in the fuel pool. This level is sufficient to restart the intact PTR [FPCS] main train and therefore to reach the safe state.

In state E, following the detection of the draining, one possible action is the closure of the transfer tube isolation valve, if not previously closed. This isolates the fuel building pool from the reactor building pool. The bounding situation is failure to close the transfer tube, as it maximises the duration of loss of cooling.

<u>Case 1</u>: transfer tube closed in states A to D and F.

Different drainage flow values are analysed from a set of values ranging between 50 and 700 m^3/h. The most limiting value of the drainage flow in terms of the duration of loss of cooling is 217 m^3/h and this bounding case is described below.

Between water levels of +19.00 m and +18.00 m, loss of cooling is considered, taking account of a LOOP: the duration of loss of cooling corresponds here to the duration of this water level drop, as this is less than one hour.

The level at which drainage stops is estimated based on the time taken for drainage to reduce the water level from +18.40 m to the minimum level of +16.85 m. The signal that the +18.40 m level has been reached is the first significant alarm to be used as the basis for which local isolation actions can be performed after 1 hour (the level alarm at +18.90 m is not considered, as a safety margin). In this case, drainage is stopped passively at +16.85 m.

Once the break is isolated, water make-up is provided by the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS] - F1B) at a flow rate of 150 m^3/h. The make-up is initiated by local actions at a minimum of 1 hour after the pool water level drops to +18.40 m. The cooling of the spent fuel pool is restored once the water level rises to +18.05 m, enabling the start-up of the PTR [FPCS] train.

The most limiting drainage situation leads to a loss of cooling for a total of 2.6 hours, which corresponds to:

- the drainage time from +18.90 m to +18.00 m = 0.65 hours,
- the drainage time to reach +16.85 m = 0.75 hours,
- the JAC/JPI [NIFPS] make-up delay time = 1.13 hours.

The average temperature reached in the spent fuel pool during the transient does not exceed 61°C in states A to D and 89°C in state F (with decay heat values of 5.85 MW and 20.81 MW respectively, and assuming for safety margin purposes a minimum volume of water in the spent fuel pool of 1227 m^3). The safe state is reached and the average fuel pool water temperature is stabilised at 52°C in states A to D and at 65°C in state F in the long term, with one main PTR [FPCS] cooling train in operation (assuming a RRI [CCWS] design temperature of 38°C).

Case 2: transfer tube open in state E

Different drainage flow values are analysed from a set of values ranging between 70 and 800 m^3/h. Drainage rates of less than 70 m^3/h are not studied because there is a delay of more than 4 hours before the fuel pool water level drops to +18.00 m, during which time the isolation actions of the PTR [FPCS] train or the start-up of make-up water from JAC/JPI [NIFPS] would be performed. The most limiting value of the drainage flow in terms of the duration of loss of cooling is 560 m^3/h and this bounding case is described below.

Between water levels of +19.00 m and +18.00 m, loss of cooling is considered, assuming a LOOP: the duration of loss of cooling corresponds here to the duration of this water level drop, as this is less than one hour.

The level at which drainage stops is estimated based on the time taken for drainage to reduce the water level from +18.40 m to the minimum level of +16.85 m. The signal that the +18.90 m level has been reached is the first significant alarm to be used as the basis for which local isolation actions can be performed after a delay of 1 hour. With a drainage rate of 560 m^3/h, the drainage time from +18.90 m to +16.85 m is less than 1 hour, precluding any operator actions, and therefore the water level is stopped passively at +16.85 m.

Once drainage is stopped, water make-up is provided by the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS] - F1B) at a flow rate of 150 m^3/h. The make-up is initiated by local actions a minimum of 1 hour after the pool water level drops to +18.90 m.

Make-up from the MHSI system pumping the In-Containment Refuelling Water Storage (IRWST) to the primary system, at injection flow rates between 146 and 210 m^3/h, is also claimed.

Prior to the MHSI start up, the instrumentation lance compartment is drained into the IRWST to provide a water level sufficient for full function of the MHSI pumps. This drainage action is initiated from the main control room when the pool water level drops to +18.40 m, no sooner than 30 minutes after the first significant warning that +18.90 m has been reached.

An MHSI pump is therefore started from the main control room 75 minutes after the opening of the instrumentation lance compartment (the time period used for complete drainage of lance compartment). A maximum make-up volume of 45 m^3 is possible via MHSI before exceeding the limiting water level in the IRWST of +0.5 m.

Cooling of the fuel pool is restored as soon as the water level reaches +18.05 m through start-up of the PTR [FPCS] train.

The maximum overall make-up time by JAC/JPI [NIFPS] (150 m^3/h) and MHSI (146 m^3/h) is 2.05 hours for a make-up volume of 328 m^3/h.

The most limiting drainage situation leads to a loss of cooling of 3.1 hours, which corresponds to:

- start-up time for JPI [NIFPS] make-up water = 1.00 hour,
- and the overall JPI [NIFPS] and MHSI make-up time = 2.05 hours.

The fuel pool cooling function is therefore restored 3.1 hours after it was lost. The maximum average fuel pool water temperature reached during the transient is 97°C, assuming a minimum MHSI injection flow rate of 146 m^3/h and a heat load of 20.81 MW including safety margins. The calculation conservatively assumes that only the volume of the spent fuel pool at a level of +16.85 m contributes to the heat sink. A safe state is reached and the long-term fuel pool water temperature is stabilised at 65°C, with one main PTR [FPCS] cooling train in operation (assuming a RRI [CCWS] design temperature of 38°C).

16.6.7. Piping failure on the extra (third) PTR [FPCS] train

16.6.7.1. Introduction

This sub-section describes the analysis of a piping failure on the third PTR [FPCS] train. This train is started in state A while maintenance work is being performed on a PTR [FPCS] main train or a support system as a preventive measure.

16.6.7.2. Transient description

Different drainage flow values are analysed from a set of values ranging between 50 and 500 m^3/h. The most limiting value of the drainage flow in terms of the duration of loss of cooling is 50 m^3/h and this bounding case is described below.

The resulting fuel pool draining leads to an automatic shutdown of the PTR [FPCS] pumps in the main trains at a level of +18.00 m, and to shutdown of the third PTR [FPCS] train at a level of +16.40 m, resulting in a loss of the fuel pool cooling function.

Between water levels of +19.00 m and +18.00 m, loss of cooling is considered, taking account of a LOOP: the duration of loss of cooling is 1 hour in this case, corresponding to the bounding time for manually restarting a PTR [FPCS] train.

The draining through the discharge pipe is stopped at a pool water level of +18.05 m by the anti-siphon device installed on the descending discharge line. The draining through the suction pipe is stopped at +16.20 m by the uncovering of the suction line. The controlled state is therefore passively reached at a level of + 16.20 m.

The PTR [FPCS] train containing the break is located and manually isolated by two redundant isolation valves on the suction pipe, to allow the plant to be taken to the safe state with long term heat removal from the fuel pool restored.

Water make-up is performed with the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) at a flow rate of approximately 150 m^3/h. This is initiated a minimum of 1 hour after the pool water level drops to +18.40 m, which is the first significant alarm of fuel pool drainage (the level alarm at +18.90 m is not considered, as a safety margin). The volume of water to be provided is 215 m^3, representing a make-up time of 1.8 hours. This restores the water level to +18.05 m in the fuel pool which is sufficient to start the remaining PTR [FPCS] main train and therefore to reach the safe state.

The most limiting drainage situation (drainage rate of 50 m^3/h) leads to a maximum duration for loss of cooling of 7.9 hours, which corresponds to:

- the loss of cooling due to LOOP between levels +18.90 m and +18.00 m = 1 hour
- the drainage time from +18.00 m to +16.20 m = 5.08 hours
- and the JPI [NIFPS] make-up time = 1.74 hours.

The fuel pool cooling function is thus restored 7.9 hours after it was lost, assuming conservatively that a total loss of the cooling function occurred at +18.00 m. The fuel pool water temperature does not exceed 86°C during the transient, assuming a heat load of 5.85 MW including safety margins. The calculation conservatively assumes that only the volume of the spent fuel pool at a level of +16.20 m contributes to the heat sink. The safe state is reached and the long-term fuel pool water temperature is stabilised at 52°C, with one main PTR [FPCS] cooling train in operation (assuming a RRI [CCWS] design temperature of 38°C).

16.7. CONCLUSION

In all cases, the average water temperature in the fuel pool does not exceed 97°C during the transient, and is below 80°C in the long term after a PTR [FPCS] train has been restored. Therefore, the decoupling criterion for PCC events involving fuel pool draining is fulfilled for the transient ‘Isolatable piping failure on a system connected to the spent fuel pool (states A to F)’.

The main assumptions and results of this study are summarised in Section 14.4.16 - Table 1 below.

SECTION 14.4.16 - TABLE 1

Main assumptions and results (4500 MWth)

Note: Transients that do not lead to a complete and temporary loss of the fuel pool cooling are not detailed here

	Main PTR [FPCS] piping failure			Third PTR [FPCS] piping failure
	State A	State F (core unloaded)	State E (end of core unloading)	State A
Transfer tube	Closed	Closed	Open	Closed
Decay heat (MW) (with margins)	5.85	20.81	20.81	5.85
$T_{RRI\ [CCWS]}$ / $T_{SRU\ [UCWS]}$ (°C)	38 / 30	38 / 30	38 / 30	38 / 30
$T_{fuel\ pool}$ (initial) (°C)	50	50	50	50
Fuel pool water volume (m^3)	1227	1227	1227	1153
Grace period without any cooling (hours)				
To reach 97°C	11.0	3.1	3.1	10.3
Transient duration (without any cooling) (hours)	2.6	2.6	3.1	7.9
Maximum $T_{PTR\ [FPCS]}$ without any cooling (°C)	61	89	97	86
$T_{fuel\ pool}$ (°C) in the long term (after restoration of the cooling function)	52	65	65	52

SUB-CHAPTER 14.4 – REFERENCES

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

4. INADVERTENT OPENING OF AN SG RELIEF TRAIN OR A SAFETY VALVE (STATE A)

4.1. FAILURE OF ONE MAIN STEAM RELIEF CONTROL VALVE

[Ref-1] Y Benoit. EPR RF02 MSRIV description (Main steam relief isolation valve). EMLR DC 0022 Revision B. AREVA. October 2003. (E).

4.3. SPURIOUS OPENING OF ONE MAIN STEAM SAFETY VALVE (MSSV)

[Ref-1] J L Riviere. MSSS / VDA System specification. ITSR DC 0178 Revision E. AREVA. January 2004. (E).

5. SMALL BREAK LOCA (DN < 50) INCLUDING A BREAK IN THE RBS [EBS] INJECTION LINE (STATES A AND B)

5.1. SMALL BREAK LOCA IN STATE A

5.1.1. Identification of Causes and Accident Description

5.1.1.1. General Comments

[Ref-1] N Jarrin. UK EPR - Safety Case for the Inherent Boron Dilution following LOCA. PEPR-F DC 24 Revision A. AREVA. July 2011. (E)

5.1.5. Description of Cases Analysed, from Initiating Event to the Controlled State

5.1.5.3. Specific Assumptions

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E).

5.1.5.4. Results

[Ref-1] L Foucart. Safety Report Chapter 15.2.3e Small Break LOCA < DN 50 (State A), notably break occurring on the Extra Boration System Injection Line (4500 MWth update).] NFPSR DC 1044 Revision B. AREVA. July 2005. (E)

5.2. SMALL BREAK LOCA ≤ DN 50 IN STATE B

5.2.1. Accident definition

[Ref-1] N Jarrin. UK EPR - Safety Case for the Inherent Boron Dilution following LOCA. PEPR-F DC 24 Revision A. AREVA. July 2011. (E)

5.2.5. Description of studied cases from the initiating event to the controlled state

5.2.5.5. Results

5.2.5.5.2. State B2: Accumulators Unavailable (Isolated)

[Ref-1] L Foucart. Safety Report Chapter 15.2.3e Small Break LOCA < DN 50 (State A), notably break occurring on the Extra Boration System Injection Line (4500 MWth update). NFPSR DC 1044 Revision B. AREVA. July 2005. (E)

6. STEAM GENERATOR TUBE RUPTURE (1 TUBE)

6.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

6.1.2. Typical Sequence of Events

6.1.2.1. From rupture initiation to leak termination (short term)

a) From rupture initiation to the controlled state

[Ref-1] T Bruyères. Steam Generator Tube Rupture Mitigation Strategy. PEPR-F DC 38 Revision D. AREVA. October 2012. (E).

6.3. METHODS AND ASSUMPTIONS

6.3.1. Methods of analysis

Case 1 – without LOOP

[Ref-1] A Barbier. CATHARE2 – Code Synthetic qualification assessment. EPD DC 490 Revision C. AREVA. September 2007. (E).

Cases 2 and 3 – with LOOP

[Ref-2] S-RELAP5 Models and correlation. Code Manual EMF-2100 Revision 6. (E).

[Ref-3] Validation of the S-RELAP5 program. KWU NDS1/97/1002. (E).

13. UNCONTROLLED SINGLE CONTROL ROD WITHDRAWAL

13.3. RESULTS AND CONCLUSIONS

13.3.1. Results for the EPR_{4250}

[Ref-1] EPR Preliminary Safety Analysis Report (PSAR 4250) – Section 15.2.3P. “Uncontrolled Single Control Rod Withdrawal (State A)”. Edition 2003. AREVA. (E)

14. LONG TERM LOSS OF OFFSITE POWER (LOOP): FUEL POOL COOLING ASPECTS (STATE A)

14.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

[Ref-1] EPR Preliminary Safety Analysis Report, Section 15.2.3r “PCC-3: Loss of electrical supplies (> 2 hours), fuel pool cooling aspects (State A)”. EDF. Edition 2006. (E)

[Ref-2] Residual Decay Heat Curves for Major Components Design Purposes Heat Load inside the Fuel Pool. NEPC-F DC 164 Revision B. AREVA NP. November 2008. (E)

14.3. GRACE PERIOD

[Ref-1] Functional study on the treatment of PCCs and RRC-As involving spent fuel pool cooling loss and draining. ECEF080499 Revision B1. EDF. November 2012. (E)

14.6. TRANSIENTS

[Ref-1] System Design Manual Fuel Pool Purification and Cooling Systems (PTR [FPPS/FPCS]) Part 5 – Instrumentation and Control. SFL EF MF 2006.751 Revision F1. Sofinel. September 2009. (E)

SFL EF MF 2006.751 Revision F1 is the English translation of SFL EF MF 2006.751 Revision F.

15. LOSS OF ONE TRAIN OF THE FUEL POOL COOLING SYSTEM (PTR [FPCS]) OR OF A SUPPORTING SYSTEM (STATE F)

15.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

[Ref-1] Residual Decay Heat Curves for Major Components Design Purposes Heat Load inside the Fuel Pool. NEPC-F DC 164 Revision B. AREVA NP. November 2008. (E)

15.3. GRACE PERIOD

[Ref-1] Functional study on the treatment of PCCs and RRC-As involving spent fuel pool cooling loss and draining. ECEF080499 Revision B1. EDF. November 2012. (E)

[Ref-2] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2 System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

SFL–EF MF 2006.712 Revision G1 is the English Translation of SFL–EF MF 2006.712 Revision G.

16. ISOLATABLE PIPING FAILURE ON A SYSTEM CONNECTED TO THE SPENT FUEL POOL (STATE A TO F)

16.1. INTRODUCTION

[Ref-1] Functional study on the treatment of PCCs and RRC-As involving spent fuel pool cooling loss and draining. ECEF080499 Revision B1. EDF. November 2012. (E)

[Ref-2] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2 System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

SFL–EF MF 2006.712 Revision G1 is the English Translation of SFL–EF MF 2006.712 Revision G.

16.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

[Ref-1] Residual Decay Heat Curves for Major Components Design Purposes Heat Load inside the Fuel Pool. NEPC-F DC 164 Revision B. AREVA NP. November 2008. (E)

16.6. TRANSIENTS

16.6.1. Piping failure on a purification line in the fuel building

[Ref-1] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2 System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

SFL–EF MF 2006.712 Revision G1 is the English Translation of SFL–EF MF 2006.712 Revision G.

16.6.2. Piping failure on a skimming line in the fuel building

[Ref-1] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2 System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

SFL–EF MF 2006.712 Revision G1 is the English Translation of SFL–EF MF 2006.712 Revision G.

16.6.3. Piping failure on a fuel pool water make-up line in the fuel building

[Ref-1] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2 System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

SFL–EF MF 2006.712 Revision G1 is the English Translation of SFL–EF MF 2006.712 Revision G.

16.6.4. Piping failure on a purification line in the reactor building

[Ref-1] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2 System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

SFL–EF MF 2006.712 Revision G1 is the English Translation of SFL–EF MF 2006.712 Revision G.

16.6.6. Piping failure on a main PTR [FPCS] cooling train

16.6.6.1. Introduction

[Ref-1] Functional study on the treatment of PCCs and RRC-As involving spent fuel pool cooling loss and draining. ECEF080499 Revision B1. EDF. November 2012. (E)

16.6.6.2. Transient description

[Ref-1] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2 System Operation, SFL–EF MF 2006-712 Revision G1. Sofinel. August 2009. (E)

SFL–EF MF 2006-712 Revision G1 is the English Translation of SFL–EF MF 2006-712 Revision G.