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01	Second issue for INSA review of sub-sections 14.3.3, 14.3.13 and 14.3.18	04-01-2008
02	Integration of co-applicant and INSA review comments.	27-04-2008
03	Marking of RESTRICTED	16.01.09
04	PCSR June 2009 update including: - Text clarifications - Inclusion of references - Technical update to account for December 2008 design freeze, notably partial cooldown rate, pressuriser safety valves and normal spray classification	28-06-2009
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06	Consolidated Step 4 PCSR update: - Minor editorial changes - Clarifications regarding uncontrolled level drop (§16) - Update of the spent fuel pool fault analysis (§18)	27-03-2011

Continued on next page

REVISION HISTORY (Cont'd)

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07	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 - Correction of cross-reference to Sub-chapter 4.4 (§9.2.2)	26-11-2012

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TABLE OF CONTENTS

SUB-CHAPTER 14.3 - ANALYSIS OF PCC-2 EVENTS

1. MAIN FEEDWATER SYSTEM (ARE [MFWS]) MALFUNCTION CAUSING A REDUCTION IN FEEDWATER TEMPERATURE

1.1. ACCIDENT DESCRIPTION

During power operation, a reduction in ARE [MFWS] temperature leads to either a new steady state condition at increased power level or to a reactor trip. Potential trip parameters include "high reactor power" or "low DNBR" depending on the burn up of the core and the detailed reactor trip criteria. This PCC-2 event is therefore bounded by other PCC-2 events that involve an increase in reactor power, such as the “uncontrolled RCCA bank withdrawal" discussed in section 9 of this sub-chapter. In that event the power increase is accompanied by increased coolant temperature, presenting a greater challenge to the core safety limits.

At shutdown conditions, a reduction in feedwater temperature leads to a reduction of the core shutdown margin. The extent of this reduction depends on the uncontrolled insertion of reactivity from the resultant cooldown. The high shutdown margin of the EPR design provides an inherent design safeguard against such an event. It is expected that the resulting overcooling transient will lead to a reactivity transient less severe than the PCC-4 event “steam line break” discussed in section 2 of Sub-chapter 14.5. That analysis shows the “no DNBR” acceptance criterion is met with a large margin, even without taking credit for core boration.

The reduction in feedwater temperature event is preliminarily assessed as being bounded by other events.

1.2. SYSTEMS SIZING

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

2. MAIN FEEDWATER SYSTEM (ARE [MFWS]) MALFUNCTION CAUSING AN INCREASE IN FEEDWATER FLOW

2.1. ACCIDENT DESCRIPTION

ARE [MFWS] malfunctions that can increase feedwater flow to one or more steam generators are the failure or incorrect operation of the feedwater control system. This results in a cooldown transient. Excessive feedwater addition will cause an increase in core power by decreasing the reactor coolant temperature. The impact of such transients is reduced by the thermal capacities of the secondary side and the RCP [RCS]. The “low DNBR” trip prevents any significant power increase.

The ARE [MFWS] malfunction is classified as a PCC-2 event.

The plant is designed to terminate the event automatically by closing the Main Feedwater System (ARE [MFWS]) isolation and control valves following a “High SG level” reactor trip signal in the affected SG. The high-load (HL) and low-load (LL) main feedwater isolation and control valves are all closed by this trip signal.

A reactor trip could be activated during power operation by a signal other than “High SG Level” e.g. by the “High Core Power Level”, or “Low DNBR” core protection signals. The severity of the initiating event would be reduced automatically by the isolation of all ARE [MFWS]-HL lines, which is initiated by all reactor trip signals. The failure of an isolation valve to close would allow the filling of the faulted SG until the SG level reaches the “High SG Water Level” signal. ARE/AAD [MFWS/SSS] isolation signal is then generated, thereby closing the downstream ARE [MFWS] isolation valve.

2.2. SYSTEM SIZING

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

3. EXCESSIVE INCREASE IN SECONDARY STEAM FLOW

3.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

3.1.1. General concern

An Excessive Increase in Secondary Steam Flow is classified as a PCC-2 event.

An excessive increase in secondary steam flow may result from:

- An inadvertent actuation of partial cooldown
- An inadvertent opening of a main steam bypass valve
- A failure to close a main steam relief valve after use.

During power operation, core protection is provided by a reactor trip signal, initiated by the Protection System, which includes the DNBR protection signal (F1A classified). The automatic actuation of the reactor trip prevents core damage before the reactor is shutdown.

After reactor shutdown, the core overcooling transient continues for as long as the secondary side system depressurisation continues, with a potential return to core criticality. The severity of the event depends on this potential return to core criticality after the reactor trip has occurred.

The Protection System set points and responses for F1A classified signals, used in this sub-chapter, are summarised in section 5 of Sub-chapter 14.1 and Sub-chapter 14.1 – Table 9. I&C signal delays and safeguard action delays are summarised in Sub-chapter 14.1 – Table 11 and Sub-chapter 14.1 – Table 12.

3.1.1.1. Spurious actuation of partial cooldown (PC)

The partial cooldown consists of the automatic lowering of the Main Steam Relief Train (VDA [MSRT]) setpoint from 95.5 bar to 60 bar, and the simultaneous reduction of the Main Steam Bypass (GCT [MSB]) setpoint from 90 bar to 55 bar at a rate corresponding to a cooldown rate of 100°C/h, as described in section 5 of Sub-chapter 14.1. Consequently, an inadvertent partial cooldown, resulting from a spurious demand from the I&C system, leads to an overcooling event that stops when the steam generator pressure reaches 55 bar.

The core control and shutdown rods are designed to ensure core sub-criticality following an automatic reactor trip, down to a RCP [RCS] temperature of 260°C [Ref-1], which includes the end of the partial cooldown (Tsat at 55 bar = 270°C). This design criterion is met under conservative assumptions, without core boration, and assuming the most negative reactive control rod stuck in its upper position (see Sub-chapter 4.3).

When the spurious partial cooldown has finished, the plant is stabilised with the core sub-critical.

3.1.1.2. Spurious opening of a Main Steam Bypass (GCT [MSB]) valve

The spurious opening of a GCT [MSB] valve causes an uncontrolled depressurisation of the main secondary system. The Main Steam Isolation Valves (VIV [MSIV]) are automatically closed by the Protection System (F1A classified signal) when the steam pressure reaches 50 bar.

When all the VIV [MSIV] are closed, the cooldown is terminated. This inadvertent core cooling is bounded by the one that occurs when a VDA [MSRT] fails to close after use, where the steam isolation occurs later (at 40 bar), as described in sub-section 3.1.1.3 of this sub-chapter.

3.1.1.3. Failure of a VDA [MSRT] to close after use

In PCC analyses, the VDA [MSRT] is actuated after reactor/turbine trip; the GCT [MSB] is not claimed because it is not F1 classified. Each VDA [MSRT] operates as follows:

- Complete opening of the VDA [MSRT] isolation valve (MSRIV), which is initially closed
- Partial closing of the VDA [MSRT] control valve (MSRCV), which is initially fully open, down to the position needed for pressure control.

Should the single failure occur, causing a MSRCV to fail to close, an uncontrolled SG depressurisation would occur, leading to an uncontrolled RCP [RCS] overcooling event.

When the main steam pressure reaches MIN3 (40 bar), the VDA [MSRT] is automatically isolated by closing both the MSRCV and the MSRIV. This isolation is effective despite the failed MSRCV, due to the redundancy provided by the MSRIV.

Because the single failure has already been applied to the MSRCV (see single failure rules in Sub-chapter 14.0), all control rods are assumed to enter the core.

This event bounds any PCC-2 event that does not affect the design shutdown margin, when considering reactivity. This assumes the application of the single failure to one MSRCV that fails to close after the VDA [MSRT] has operated.

3.1.2. Typical sequence of events

3.1.2.1. From the initiating event to the controlled state

Following any PCC-2 event, the VDA [MSRT] are used in all the steam generators to perform heat removal following reactor shutdown. The GCT [MSB] is not claimed because it is not F1 classified. The single failure may be assumed to be in the MSRCV of any one steam generator, which may stay fully open at its initial position after the corresponding MSRIV has been opened.

This MSRCV failure causes an uncontrolled increase in steam flow, leading to an uncontrolled SG depressurisation. All VIV [MSIV] are automatically closed following a "SG pressure drop" actuation signal. After the main steam lines have been isolated, only the affected steam generator continues to depressurise. The energy removal from the RCP [RCS] causes a reduction of primary coolant temperature and pressure. Due to the negative moderator temperature coefficient, the cooldown results in a reduction in the core shutdown margin.

When the affected SG pressure reaches the “MIN3” setpoint (40 bar), the VDA [MSRT] is automatically isolated by closing its isolation and control valves. The VDA [MSRT] is closed by the MSRIV, which is separate from the failed-open MSRCV.

Following the closure of the VDA [MSRT] on the affected SG, the event reaches the controlled state. This occurs with stable heat removal via the three unaffected steam generators, with feed injected by the ASG [EFWS] (if the ARE [MFWS] are unavailable), and steam removal through the VDA [MSRT].

The automatic actuation of a partial cooldown is highly probable from the RIS [SIS] signal on "pressuriser pressure < MIN3" (115 bar). In that case, the controlled state is reached at the end of the partial cooldown, with a steam pressure of 60 bar in the three unaffected SG. If no partial cooldown signal is generated, the three unaffected SG are held at the hot shutdown setpoint of 95.5 bar.

3.1.2.2. From the controlled state to the safe shutdown state

The safe shutdown state is defined as a state where the LHSI/RHR operating conditions are reached.

The sequence of actions, initiated by operator action, is:

RCP [RCS] boration

During the cooldown, RCP [RCS] boration is performed via the RBS [EBS]. The RCV [CVCS] is not claimed as it is not F1 classified. After completing the required boration, the operator stops the RBS [EBS].

RCP [RCS] cooldown

The RCP [RCS] cooldown to RIS/RRA [SIS/RHRS] conditions is performed by the three unaffected SG by decreasing the VDA [MSRT] setpoints. In this case (the GCT [MSB] is unavailable as the VIV [MSIV] are closed.

The RCP [RCS] cooldown rate is consistent with the ASG [EFWS] tank capacity so that the LHSI/RHR operating conditions are reached before the ASG [EFWS] tanks are emptied.

The EPR design cooling rate is 50°C/hr if two RBS [EBS] trains are available, or 25°C/hr if only one RBS [EBS] train is available [Ref-1]

RCP [RCS] depressurisation

If the RCP [RCS] pressure remains above 30 bar after the cooldown, the operator will momentarily open the PSV to depressurise the RCP [RCS].

During this depressurisation phase, the LHSI maintains a minimum RCP [RCS] pressure of about 20 bar so that the RCP [RCS] sub-cooling margin is maintained.

3.2. SAFETY CRITERIA

The safety criteria are the radiological limits for normal operation.

The consequences of a main steam system depressurisation are analysed for the following decoupling criteria:

- Fuel cladding integrity;

- Reactor coolant pressure boundary;
- Quantity of radioactive products released.

3.3. DEFINITION OF CASES STUDIED

Fuel cladding integrity

This analysis is performed to demonstrate that the following core damage prevention criterion is met:

> Departure from nucleate boiling (DNB) will not occur following an automatic reactor trip for a steam discharge equivalent to the complete opening of a VDA [MSRT], with isolation on a “low SG pressure” signal (40 bar).

Because the postulated single failure is assumed to be the VDA [MSRT] control valve, as explained above, all of the control rods are assumed to be inserted.

The period between the initiating event and reaching the controlled state is studied in detail below.

The demonstration used to show that the safe shutdown state can be reached is based on a qualitative assessment using other studies described in Chapter 14.

Reactor coolant pressure boundary

The uncontrolled RCP [RCS] cooldown can cause thermal shock to the reactor vessel. The effects of cooldown on the RCP [RCS] are analysed in Chapter 3.

Radiological consequences

The safety criteria to be met are the dose equivalent limits for release to the atmosphere, as described in Sub-chapter 3.1.

The bounding transient, with regard to radiological releases, is the loss of condenser vacuum analysed in section 5 of this sub-chapter. The amount of steam released to the atmosphere is similar in the two cases.

3.4. METHODS AND ASSUMPTIONS

3.4.1. Methods of analysis

The transient is calculated from the VDA [MSRT] fully opening at hot shutdown conditions, to the VDA [MSRT] isolation at 40 bar.

The methodology of analysis is the one described in section 2 of Sub-chapter 14.5 for steam system piping breaks, which belong to the same event family. However, as there is no return to criticality in the case studied, and hence no power excursion which might have an impact on the transient, the thermal-hydraulic calculation can be decoupled from the neutronic calculation.

The transient calculation is performed in two steps:

- The thermal-hydraulic transient is first calculated by the THEMIS code (Appendix 14A), with reactor power set to zero after reactor trip (no neutronic computation). This conservative approach provides the minimum temperature level at the core inlet during the transient.
- The neutronic behaviour is then calculated by the 3D-code SMART (Appendix 14A): based on the thermal-hydraulic core parameters calculated by THEMIS (core pressure, core inlet temperatures, and flow rates). SMART computes the maximum reactivity variation and validates the initial hypothesis of non return to criticality.

3.4.2. Protection and mitigation actions

The following F1A I&C systems provide protection following an accidental depressurisation of the main steam system, when assessing the DNBR criterion (for analyses with regard to other criteria, see sub-section 3.3 of this sub-chapter):

- Reactor trip on:
  - core power level > MAX3
  - DNBR < MIN3
  - Pressuriser pressure < MIN2
  - SG pressure drop > MAX1
  - SG pressure < MIN1.
- Safety injection actuated when:
  - Pressuriser pressure < MIN3.
- Closure of all main steam isolation valves on:
  - SG pressure drop > MAX1
  - SG pressure < MIN1.
- Closure of main feedwater high-load line in all SG on:
  - SG pressure drop > MAX1
  - SG pressure < MIN1
  - SG level > MAX1.
- Closure of main feedwater low-load line of the SG on:
  - SG pressure drop > MAX2
  - SG pressure < MIN2

    - o SG level > MAX1 (after RT).

- Isolation of the VDA [MSRT] of the affected SG on:

    - o SG pressure < MIN2.

The F1B systems, required to transfer the plant from the controlled state to the safe shutdown state, are described in "Loss of condenser vacuum" (section 5 of this sub-chapter).

3.5. DESCRIPTION OF SENSITIVITY CASES (FROM THE INITIATING EVENT TO THE CONTROLLED STATE)

3.5.1. Choice of single failure and preventive maintenance

The single failure is postulated as one MSRCV that stays stuck open after using the corresponding VDA [MSRT].

By definition, the single failure initiates the accident event[1]. Consequently, no additional failure has to be postulated during the transient (e.g. the MSRIV is available to close on demand at 40 bar and all control rods are inserted when the reactor is tripped).

No preventive maintenance is assumed since it has no significant negative impact on the transient.

3.5.2. Initial state

The severity of this event (particularly the spurious opening of the GCT [MSB]) is a function of power level.

Accidental main steam release is more serious in terms of reactivity insertion when the plant is at hot shutdown conditions:

- When the reactor is at full power, the RCP [RCS] contains more energy than when it is at hot shutdown conditions, since there is additional energy stored in the fuel. This additional energy in the form of heat will reduce the cooling effect of the accident after the reactor trip.

- In addition, since the initial SG fluid mass and SG pressure are greater at hot shutdown, the extent and duration of the RCP [RCS] cooldown are greater.

Therefore, this analysis simulates the initial plant state at hot shutdown conditions. It is representative of the plant state when the VDA [MSRT] can be required following a reactor trip.

End of Life (EOL) conditions are assumed to maximise the reactivity insertion during the RCP [RCS] cooldown. The RCP [RCS] boron concentration is assumed to be zero.

The initial conditions are presented in Section 14.3.3 - Table 1.

[1] In fact, the single failure is superimposed on a PCC-2 event leading to the opening of the VDA [MSRT]. The accident begins at the moment the MSRCV fails to close.

3.5.3. Specific assumptions

3.5.3.1. Neutronic data and decay heat

The thermal-hydraulic calculation is performed at hot zero power conditions (no neutronic computation: term A nil), and with no credit for decay heat (terms B+C set to zero). This conservative approach provides the minimum temperature level at the core inlet during the transient.

This calculation assumes that all of the control rods are inserted, the boron concentration is zero and that the xenon concentration is at the equilibrium level. Even with the most onerous fuel management strategy (MOX), the calculation shows that the core remains subcritical (see sub-section 3.5.4 of this sub-chapter).

The neutronic calculation is performed using the core thermal-hydraulic conditions previously defined, and using the most onerous neutronic conditions with regard to core reactivity:

- The most onerous fuel management is retained, between UO_2 and MOX,
- All neutronic data correspond to EOL operating conditions,
- The EOC shutdown margin refers to the hot full power equilibrium xenon level, with all the control/shutdown rods inserted. (Core physics studies show that this margin is ensured even under the most unfavourable conditions; in particular at the end of equilibrium cycle when the temperature coefficient reaches its highest value [Ref-1]).

Under those conditions, the initial shutdown margin is conservatively assumed to be 4000 pcm with all rods inserted [Ref-2].

3.5.3.2. Assumptions related to non-F1 systems

- ARE [MFWS]:

  It is assumed that a maximum ARE [MFWS] flow of 30% of nominal flow is delivered to each SG until the ARE [MFWS] low-load line is closed.

  This maximum flow corresponds to the maximum capacity of the ARE [MFWS] low-load line, which is the only ARE [MFWS] line open after the reactor trip. The ARE [MFWS] flow control is not claimed.

- No other control systems are taken into account as they have either a beneficial impact or no significant impact on the event.

3.5.3.3. Assumptions related to F1 systems

- VIV [MSIV] (F1A):

  All VIV [MSIV] are closed following a "SG pressure drop > MAX1" signal with a setpoint of 2 bar/min. The setpoint of this signal is adjusted to 8.5 bar (7 bar + 1.5 bar uncertainty) below the SG pressure, with a maximum value of 75 bar. The delay for steam lines isolation consists of a 0.9 second channel delay, plus a 5 seconds valve closing time.

- ARE [MFWS] isolation (F1A):

  The ARE [MFWS] low-load line of the affected steam generator is closed following a "SG pressure drop > MAX2" signal with a setpoint of 2 bar/min. The setpoint of this signal is adjusted to 18.5 bar (17 bar + 1.5 bar uncertainty) below the SG pressure, with a maximum value of 65 bar. The delay for ARE [MFWS] isolation consists of a 0.9 seconds channel delay, plus a 15 second valve closure time.

  The ARE [MFWS] low-load lines of the unaffected steam generators are closed following a "SG level > MAX1" signal with a setpoint of 69% of the narrow range + 2% uncertainty. The delay for ARE [MFWS] isolation consists of a 1.5 second channel delay, plus a 15 second valve closure time.

- MHSI (F1A):

  - Minimum safety injection capability is assumed as discussed in Sub-chapter 14.1 - Table 13).
  - To simplify the calculation, it is assumed that the MHSI injects no boron. This is a conservative bounding assumption.
  - Safety injection and partial cooldown are actuated following a "pressuriser pressure < MIN3" signal at a setpoint of 115 bar - 1.5 bar uncertainty.
  - The delay for MHSI injection consists of a 0.9 second channel delay, plus 10 seconds to start the pumps.

- VDA [MSRT] (F1A):

  - The maximum capacity of the stuck open VDA [MSRT] is conservatively assumed to be 1270 t/hr, or 55% of nominal steam flow at 100 bar.
  - The VDA [MSRT] of the affected steam generator is isolated by closing its MSRIV following a "SG pressure < MIN3" signal at a setpoint of 40 bar - 1.5 bar uncertainty.
  - The delay for VDA [MSRT] isolation consists of a 0.9 second channel delay, plus a 5 second valve closure time.

3.5.3.4. Other assumptions

- A maximum SG heat transfer coefficient is used assuming no fouling and no plugging.
- The steam flow through the open VDA [MSRT] is calculated using the Moody pressure drop – flow correlation at each calculation time step. The backpressure is always assumed to be atmospheric pressure.
- A perfect moisture separation in the steam generator is assumed.
- No credit is taken for heat stored in the metal structures other than the fuel rods and the steam generator tubes.
- A minimum loop-flow mixing is assumed inside the RPV. Specifically, a maximum value of 86% flow entering the RPV through inlet nozzle ‘i’ remains in the associated core quadrant ‘i’ at core inlet.

- It is more onerous to assume forced circulation when assessing the criterion of no return to criticality. Therefore the reactor coolant pumps are kept running throughout the transient.

The main assumptions are summarised in Section 14.3.3 - Table 1.

3.5.4. Results

The result of the thermal-hydraulic calculation performed in the BDR-99 for the EPR_{4900} is used to define a conservative set of thermal-hydraulic conditions to be used for the EPR_{4250} at the time of VDA [MSRT] isolation. The main parameter is the pressure setpoint for the VDA [MSRT] isolation, which is not changed between the EPR_{4900} and the EPR_{4250}. This pressure determines the minimum core inlet temperature.

On this basis, the neutronic calculation is performed using the EPR_{4250} neutronic characteristics. A dedicated EPR_{4250} calculation is performed, because some important neutronic characteristics are beneficial (e.g. shutdown margin) and others are adverse (e.g. moderator coefficient) for the EPR_{4250} compared with the EPR_{4900}.

Thermal-hydraulic calculation: EPR_{4900} accident analysis in BDR-99

The BDR-99 analysis results for the EPR_{4900} are presented in Appendix 14B. They are summarised as follows:

- The stuck-open VDA [MSRT] occurs at time 0.
- All steam lines are isolated following the first SG pressure drop signal at about 140 seconds. Feedwater flow in the affected SG is terminated following the second SG pressure drop signal at about 175 seconds.
- The SI signal occurs at about 150 seconds and the MHSI starts injecting into RCP [RCS] at approximately 200 seconds.
- At about 400 seconds, the SG pressure reaches the setpoint of 40 bar in the affected SG, and the VDA [MSRT] is automatically isolated. This terminates the RCP [RCS] cooling transient.

The main thermal-hydraulic parameters at the time of VDA [MSRT] isolation are:

- Core pressure = 65 bar
- Core boron concentration = 0 ppm
- Core inlet temperature in the affected loop = 254°C
- Core inlet temperature in the unaffected loop = 262°C
- Core flow = 22230 kg/s.

The reactivity calculated under these conditions is about -600 pcm. The results demonstrate that there is no return to critical conditions.

Following the stabilisation of the thermal-hydraulic parameters, the controlled state is reached using only F1A systems. This corresponds to the end of partial cooldown operating conditions with the primary temperature at 275°C. This corresponds to the saturation temperature for a SG pressure of 60 bar. At this time the pressuriser pressure is 87 bar.

The reactivity excursion is under control, decay heat is removed by the intact steam generators and the core coolant inventory is in a stable condition.

Thermal-hydraulic calculation: extrapolation of EPR_{4900} results to EPR_{4250}

The important parameter for reactivity assessment is the core inlet temperature. The BDR-99 thermal-hydraulic results presented in Appendix 14B show that the core inlet temperature in the affected loop is close to the saturation temperature at the VDA [MSRT] isolation pressure setpoint. The saturation temperature for 40 bar is 250°C. The core inlet temperature is slightly higher because of the mixing effect in the RPV downcomer and lower plenum, having a flow contribution from the hotter unaffected loops.

VDA [MSRT] isolation is actuated at the same pressure setpoint of 40 bar for the EPR_{4250} as for the EPR_{4900}. As a result, the following set of thermal-hydraulic design data is defined for EPR_{4250}:

- Core pressure = 65 bar
- Core boron concentration = 0 ppm
- Core inlet temperature in the affected loop = 245°C
- Core inlet temperature in the unaffected loop = 245°C
- Core flow = 22245 kg/s the EPR_{4250} thermal hydraulic design flow.

The above design data provides a conservative set of thermal-hydraulic data for the EPR_{4250} neutronic calculation:

- The core inlet temperature of 245°C includes a comfortable margin to the 250°C saturation temperature at 40 bar. This temperature bounds the pressure sensor uncertainty of 1.5 bar. This is shown by the following three sets of saturation temperatures:

  250°C = T_{sat} 40 bar , 248°C = T_{sat} 38.5 bar , 245°C = T_{sat} 36.5 bar.

- The same core inlet temperature of 245°C is applied to all loops to cover the lower flow mixing data for the EPR_{4250} compared to the one previously used for the EPR_{4900}. 86% of loop flow from loop 'i' enters core quadrant 'i' without interacting with other loop flows in the EPR_{4250}, compared of 65% for the EPR_{4900}. It is the only significant penalty for the EPR_{4250} compared with the EPR_{4900} for the core thermal-hydraulic parameters when VDA [MSRT] isolation occurs.

Neutronics calculation: EPR_{4250} SMART calculation

A SMART calculation is performed using the EPR_{4250} neutronic characteristics and the previously determined set of thermal-hydraulic core data [Ref-1]. The SMART code is described in Appendix 14A.

The calculation assumes all rods are inserted, no boron, and xenon at the equilibrium level. For the most onerous fuel management scheme, MOX, the calculation shows that the core remains subcritical with a margin of more than 600 pcm.

Global extrapolation of the results from EPR_{4250} to EPR_{4500}

The pressure range in the secondary circuit and the protection settings are identical for the EPR_{4250} and the EPR_{4500} [Ref-2] (see Sub-chapter 14.1). With all the rods inserted, the EPR_{4500} has a larger negative reactivity than the EPR_{4250} and very similar neutronic characteristics [Ref-3] [Ref-4]. It is therefore reasonable to conclude that the results obtained for the EPR_{4250} are applicable to the EPR_{4500}. Therefore core subcriticality will be maintained during the transient.

Consequences of the modification of the partial cooldown rate and associated setpoints

The consequences of the partial cooldown rate increase will be the following:

- The closure of all the main steam isolation valves on a ''steam generator pressure drop > MAX1'' signal will occur a few seconds later. This results from the increase in the SG pressure drop signal from 2 bar/min to 5 bar/min.

- The closure of the main feedwater low load isolation and control valves of the affected steam generator on a ''steam generator pressure drop > MAX2'' signal will also occur a few seconds later due to the increase in the SG pressure drop signal from 2 bar/min to 5 bar/min.

The cooling of the primary side by the secondary side will be slightly higher and the minimum core pressure can be a little lower than 65 bar.

The minimum secondary pressure corresponding to the VDA [MSRT] isolation on SG pressure < 40 bar will remain the same.

Therefore the minimum cold leg temperature will remain equal to about 250°C, the saturation temperature corresponding to 40 bar.

Calculations of core reactivity have been made with conservative data considering a temperature at the core inlet equal to 245°C for all loops. Due to the pressure decrease the coolant density will be lower which will lead to a lower reactivity increase from the moderator density feedback.

Therefore it can be concluded that the decoupling criterion of no core DNB will be fulfilled.

3.5.5. Conclusion

The core remains subcritical in the case of a single failure of one VDA [MSRT] to close after use. The automatic closure of the MSRIV at 40 bar, actuated by the F1A signal "SG pressure < MIN2", stops the uncontrolled overcooling caused by the stuck-open MSRCV without core criticality being reached.

Following the stabilisation of the thermal-hydraulic conditions, the controlled state is achieved using only F1A systems. This corresponds to the operating conditions at the end of partial cooldown, at a primary temperature of 275°C (Tsat 60 bar) and a pressuriser pressure of approximately 90 bar. The reactivity is under control, the core power is removed via the unaffected steam generators, and the core coolant inventory is stable.

This study generally demonstrates that the single failure applied to the VDA [MSRT] has no adverse effect on DNBR and no consequences on reactivity control. There is no return to criticality following reactor trip. This applies in all PCC-2 to PCC-4 events where the initiating event does not affect the shutdown margin or does not cause a SG cooling greater than that associated with a partial cooldown. The PCC events that are not covered by this general demonstration are addressed in the “Steam Line Break (States A, B)” presented in section 2 of Sub-chapter 14.5.

3.6. DESCRIPTION OF CASES STUDIED (FROM THE CONTROLLED STATE TO THE SAFE SHUTDOWN STATE)

The safe shutdown state is defined as a state where the RIS/RRA [SIS/RHRS] operating conditions are reached.

Transition from the controlled state to the safe shutdown state is bounded by the loss of condenser vacuum fault assessed in section 5 of this sub-chapter. The controlled state following secondary system depressurisation is less onerous than after a loss of condenser vacuum because of the lower primary pressure and temperature, with the same shutdown margin. In addition, the demands on F1B systems are identical in both instances, the VDA [MSRT], ASG [EFWS], and RBS [EBS].

3.7. SYSTEM SIZING

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

SECTION 14.3.3 - TABLE 1

Stuck-Open Main Steam Relief Train Main Assumptions (from Sub-chapter 14.1 – Tables 2 and 3)

PARAMETER	UNITS	VALUE USED
INITIAL CONDITIONS		
Reactor power (min)	% FP	0
Shutdown margin (min)	pcm	4000
RCP [RCS] boron concentration (min)	Pcm	0
RCP [RCS] loop flow (design)	m^3/h	27180
Average RCP [RCS] temperature (nominal)	°C	303.3
Pressuriser pressure (nominal)	bar a	155
Pressuriser level (max)	% MR	31 + 5
SG level (nominal)	% NR	49
F1A ACTIONS		
VDA [MSRT]		
VDA [MSRT] stuck-open capacity (max)	te/h	1270 (55% nom) at 100 bar a
VDA [MSRT] isolation on SG pressure < MIN3	bar a	40 - 1.5
VDA [MSRT] closing delay (max)	S	0.9 (I&C) + 5 (valve)
VIV [MSIV]		
VIV [MSIV] isolation in all steam lines on SG pressure drop > MAX1	bar/min	2, with setpoint adjusted 7+1.5 bar below SG pres. (max 75 bar)
VIV [MSIV] closing delay (max)	s	0.9 (I&C) + 5 (valve)
ARE [MFWS]		
ARE [MFWS] flow per SG (max low-load line)	te/h	650 (30% nom)
ARE [MFWS] isolation in affected SG on SG pressure drop > MAX2	bar/min	2, with setpoint adjusted 17+1.5 bar below SG pres. (max 65bar)
ARE [MFWS] isolation in unaffected SG on SG level > MAX1	% NR	69 + 2
ARE [MFWS] low-load line closing delay (max)	s	1.5 (I&C) + 15 (valves)
MHSI		
MHSI actuation on pressuriser pressure < MIN3	bar a	115 - 1.5

FP : Full Power
MR : Measuring Range
NR : Narrow Range, calibrated at 100% power

4. SPURIOUS TURBINE TRIP

4.1. ACCIDENT DESCRIPTION

A "turbine trip" is defined as a spurious loss of turbine flow to zero, or any turbine trip signal triggered while all neutronic and thermal hydraulic plant parameters are at their nominal operating values (including normal fluctuations and uncertainties).

A turbine trip is classified as a PCC-2 event.

The “turbine trip” transient is covered by the one resulting from the PCC-2 event "Loss of condenser vacuum" analysed in section 5 of this sub-chapter, since the loss of the condenser causes a turbine trip.

4.2. SYSTEM SIZING

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

5. LOSS OF CONDENSER VACUUM

5.1. INTRODUCTION

A Loss of Condenser Vacuum results in a turbine trip, and behaves similarly to a Loss of Load or Turbine Trip event. This event, classified as a PCC-2 event, is analysed in reactor state A only, since this state causes the highest possible steam release to the atmosphere and thus is limiting in terms of activity release in a PCC-2 event.

The analysis is performed to demonstrate that the DNBR remains above the acceptance criterion and that the radiological limits for PCC-2 are not exceeded during the transition to the safe shutdown state. For this purpose, both the transients to the controlled state and then to the safe shutdown state are analysed in detail. Generally only F1-classified systems are claimed, so that the steam release to the atmosphere is maximised. Bounding PCC-2 radiological calculations are performed on the basis of this event (see Sub-chapter 14.6).

The Protection System (RPR [PS]) set points and responses for F1A classified signals used in this sub-chapter are summarised in section 5 of Sub-chapter 14.1 and Sub-chapter 14.1 – Table 9. I&C signal delays and safeguard action delays are summarised in Sub-chapter 14.1 – Table 11 and Sub-chapter 14.1 – Table 12.

5.2. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

Following loss of condenser vacuum, the turbine trip is actuated. Since the loss of condenser vacuum also causes the main steam bypass GCT [MSB] to be unavailable, the main steam relief trains VDA [MSRT] are actuated for heat removal. The reactor trip is triggered on high primary or secondary pressure.

The purpose of this transient analysis is to determine the thermal hydraulic parameters, such as:

- Minimum SG water mass inventory

- Maximum SG integrated steam mass release to the atmosphere.

In order to calculate the activity release and doses, it is assumed that one main steam relief control valve (MSRCV) fails to close after the activation of the VDA [MSRT]. As a result of this continuous excess heat removal, the RCP [RCS] and secondary side cool down so that the emergency core cooling and secondary side isolation criteria are reached.

The controlled state is reached after the automatic closing of the main steam relief isolation valve (MSRIV) when the main steam pressure falls below 40 bar. For further plant cooldown, including boration to reach the safe shutdown state, the following manually initiated actions are necessary:

- 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) [Ref-1] down to a hot leg temperature of less than 180°C and then a reduction of primary pressure to less than 30 bar (LHSI/RHR operating conditions) by means of the pressuriser safety valves.

- Boration with one or both RBS [EBS] pumps during the cooldown.

With the operation of the RCV [CVCS] (normally available), adequate boration and depressurisation can be achieved without activation of RBS [EBS] and PSV.

Controlled and Safe Shutdown State

It must be demonstrated that the controlled and the safe shutdown state can be reached with F1A and F1B classified functions based on the following safety and acceptance criteria:

- Safety Criteria: Radiological limits for normal operation must not be exceeded.
- Acceptance Criterion: No departure from nucleate boiling (DNBR limit: 1).

The following F1A classified functions are available to achieve the controlled state:

- The reactor trip is initiated from one of the following signals:
  - Pressuriser pressure >166.5 bar
  - SG pressure > 95.5 bar
- Four VDA [MSRT] for secondary side heat removal and pressure limitation actuated on attainment of "SG pressure > 95.5 bar"
- Four ASG [EFWS] trains for secondary side water supply actuated on attainment of "SG level < 7.85 m", (only required in case of total loss of main feedwater)
- Three pressuriser safety valves (PSV) for RCP [RCS] pressure limitation with their setpoints at 174 bar (1st PSV) and 178 bar (2nd and 3rd PSV)
- Isolation of non-affected SGs by the signal "SG pressure drop > MAX1 (2 bar/min)" initiating closure of all VIV [MSIV],
- Closing of the affected SG MSRIV from SG pressure < MIN3 (40 bar),
- Actuation of MHSI from safety injection signal "PZR pressure < MIN3 (115 bar)".

For the transfer from the controlled state to the safe shutdown state, the following (at least) F1B classified functions are available:

- Four main steam relief trains VDA [MSRT] for cooldown to RIS/RRA [SIS/RHRS] operating conditions (manual action)
- One RBS [EBS] for boration during cooldown
- Four ASG [EFWS] trains for feedwater supply to the SG (automatic or manual action)
- One pressuriser safety valve to lower the primary pressure to 30 bar
- Shut-off of the MHSI pumps.

5.2.1. Methods and Assumptions

Calculations are performed using the NLOOP computer code described in Appendix 14A.

5.2.1.1. Important phenomena of the models used in NLOOP

Two families of transients; loss of secondary heat sink leading to RCP [RCS] heat-up and secondary overcooling resulting in RCP [RCS] overcooling; are combined in one event sequence.

Primary Side

Overcooling occurs during the inadvertent opening of the main steam relief train VDA [MSRT] with a consequent emptying of the pressuriser. After the VDA [MSRT] closes, the primary temperature and pressure increase again.

Secondary side

Initially, main steam pressure increases due to the closure of the turbine valves. The main steam relief train opens quickly and re-closes to maintain a constant main steam pressure at zero power in the three unaffected loops. In the affected loop, the main steam relief train remains open and the excessive loss of steam leads to overcooling until the main steam relief train is closed when the main steam pressure of the affected SG falls to 40 bar. Thus, the affected SG level decreases. Consequently, heat transfer in that SG is reduced substantially.

5.2.2. Initial and boundary conditions

Conservative initial and boundary conditions for the analysis are selected to maximise the total integrated mass release to the atmosphere and minimise the water mass content in the SGs until both controlled and safe shutdown states are reached.

The main initial conditions are listed in Section 14.3.5 - Table 1. These initial conditions, such as reactor power, core inlet temperature, and primary pressure take into account the total uncertainties and maximum variations caused by the control systems. The coolant flow corresponds to the thermal design flow.

The analysis takes into account the effect of heat storage capacity of the metal structures of the RCP [RCS] and SGs.

The event is analysed under EOL core conditions with the largest moderator and Doppler reactivity feedback.

5.2.3. Choice of Single Failure and Preventive Maintenance

In order to maximise the steam release to the atmosphere, a single failure is assumed: the non-closing of the MSRCV after opening of the MSRIV. In addition, during the transfer from the controlled to the safe shutdown state, the failure of one RBS [EBS] pump is selected. This delays the cooldown phase and thus results in an increase in the steam release to the atmosphere. There is no preventive maintenance required for the scenario**.**

In order to provide bounding results for the secondary side heat removal, unavailability of the main feedwater is assumed at the start of the event. This will show the heat removal capacity of the ASG [EFWS] under adverse conditions.

5.3. RESULTS AND CONCLUSIONS

5.3.1. From the initiating event to the controlled state

The PCC transient is not calculated in this sub-chapter. The ability to fulfil the safety and the acceptance criteria is demonstrated using the results of the PCC transient analysis contained in Section 15.2.5 of the BDR-99 presented in section 2.5.2 of Appendix 14B and from the comparison of relevant parameters between the EPR_{4500} and the EPR_{4900} covered by the BDR.

5.3.1.1. BDR-99 status

The results of the loss of condenser vacuum transient for the EPR_{4900} are presented in section 2.5.2 of Appendix 14B.

The initial heat-up phase following the loss of condenser/turbine trip leads to reactor trip following a "pressuriser pressure > MAX2” signal at a setpoint of 166.5+1.5 bar. A brief opening of the first PSV occurs following a "pressuriser pressure > 175.5 (174+1.5) bar" signal. The cooldown due to the failing to close of one MSRCV is terminated by the automatic closing of all the VIV [MSIV] following a "SG pressure < MIN1 signal at a setpoint of 50-1.5 bar". Shortly after, the affected MSRIV is closed following a "MS pressure < MIN3” signal with a setpoint of 40-1.5 bar which provides further isolation.

There is no risk of DNB because the core cooling conditions are far better than in the limiting case of loss of offsite power described in section 6 of this sub-chapter, as there is no loss of forced RCP [RCS] flow.

The MHSI is also activated following a "PZR pressure < MIN3” signal with a setpoint of 113 (115-2) bar. However its impact is negligible because it starts simultaneously with the termination of the cooldown following MSRIV closure.

The controlled state is safely reached by the following actions:

- Reactor trip and closing of both all VIV [MSIV] and the affected MSRIV ensure subcriticality throughout the transient evolution. The minimum reactor inlet temperature with a failed main steam relief control valve is 250°C.
- The availability of three VDA [MSRT] and four ASG [EFWS] trains ensure heat removal.

5.3.1.2. Comparison of the EPR_{4500} and the EPR_{4900}

For the DNBR assessment, the following features of EPR_{4500} compared to EPR_{4900} have to be considered for minimum SG water level and maximum steam release to the atmosphere:

- Lower reactor power level and lower decay heat by about 9%
- Practically identical RCP [RCS] flow conditions
- Higher margin (2.5 bar) between VDA [MSRT] opening setpoint (95.5 bar) and limits for VIV [MSIV] and VDA [MSRT] closing, at 50 bar and 40 bar respectively, which are kept unchanged
- Practically identical power-related VDA [MSRT] capacities

- EPR_{4500} SG heat transfer surface is about 6% larger relative to the power level
- Practically identical power-related SG water inventory
- Power-related EPR_{4500} ASG [EFWS] pump capacities slightly higher
- Slightly higher EPR_{4500} ASG [EFWS] tank sizes and inventories
- Practically identical RCCA worth
- Practically identical bounding reactivity feedbacks for moderator and fuel
- Higher EPR_{4500} MHSI shutoff head
- Lower ratio of reactor power to RCP [RCS] volume in the EPR_{4500} as the RCP [RCS] volume is practically unchanged.

5.3.1.3. Conclusions for the EPR_{4500}

The DNBR criterion for the EPR_{4500} is considered to be the same as for the EPR_{4900}, since the reactor power is lower and the core flow rate is practically the same.

The slightly higher margin between VDA [MSRT] opening setpoint and the closing pressures of the VIV [MSIV] and MSRIV for the ERP_{4500}, compared to the EPR_{4900}, tend to be lead to a higher steam discharge to the atmosphere through the VDA [MSRT]. However, the resulting slight theoretical extension of the blowdown period is compensated for by the lower reactor power level and consequent lower steam production.

A sufficient margin to the emptying of the affected SG is maintained during the transient. The minimum SG water level will be higher than in the EPR_{4900} because of the slightly higher ASG [EFWS] pump capacity and unchanged power-related initial SG water inventory, and lower reactor core power.

Overall, the steam discharge to the atmosphere will be comparable between the EPR_{4500} and the EPR_{4900}. As a result, based on the high margins to the radiological limits obtained for the EPR_{4900}, the EPR_{4500} radiological limits are also met with high margins.

Subcriticality is also maintained for the EPR_{4500}, as the minimum RCP [RCS] temperatures are the same as for the EPR_{4900} as the lower limit of MSRIV closure setpoint was not changed.

5.3.2. From the controlled state to the safe shutdown state

The PCC transient is not calculated in this sub-chapter. The capability to fulfil the safety and acceptance criteria is derived from the results of the PCC transient analysis contained in Section 15.2.5 of the BDR-99 presented in section 2.5.2 of Appendix 14B, and from comparison of the relevant parameters of the EPR_{4500} and the EPR_{4900} covered by the BDR.

5.3.2.1. BDR-99 status

The results of the loss of condenser vacuum transient for the EPR_{4900} are presented in section 2.5.2 of Appendix 14B.

The transition from the controlled state to the safe shutdown state is assessed against the relevant acceptance criteria:

- Sub-criticality is maintained by boration via one RBS [EBS] pump.
- Heat removal is provided by three VDA [MSRT] and four ASG [EFWS] trains.
- Activity release is under control since none of the barriers are challenged.
- Reduction of primary pressure to 30 bar is provided by shut down of the MHSI and RBS [EBS] pumps, and the opening one of the pressuriser safety valves for a brief period of time. The pressuriser safety valve is not required to lower the primary pressure if the RBS [EBS] pump is shut off when the necessary boron concentration for the safe shutdown state is reached.

The minimum SG water inventory and maximum integrated mass release from the main steam flow via the main steam relief valves are listed below:

Steam Generator Inventory (Mass content)

SG	Time (s)	Minimum inventory (te)
SG 1(affected)	1040	15
SGs 2-4	960	28

Integrated Mass Release through MS-flow

SG	Time (s)	Integrated MS-flow via MS-relief per loop (te)
SG 1(affected)	16000	249
SGs 2-4	16000	295
Sum of all SGs	16000	1134

5.3.2.2. Relevant differences between EPR_{4500} and EPR_{4900}

In addition, in comparing the EPR_{4500} and the EPR_{4900} (Appendix 14B) it is noted that:

- The ASG [EFWS] tank is, relative to power, larger for the EPR_{4500}. The RBS [EBS] capacity is also relatively higher with respect to the slightly lower RCP [RCS] volume
- The pressuriser safety valve capacity is higher relative to power, since it is unchanged compared to the EPR_{4900}.

5.3.2.3. Conclusions for EPR_{4500}

The transition to the safe shutdown state for the EPR_{4900} is fully applicable and even bounds the conditions for the EPR_{4500} because of the following boundary conditions:

- Sub-criticality is maintained by the RBS [EBS]
- Heat removal via the VDA [MSRT] and the ASG [EFWS] is provided with higher margins because of slightly higher ASG [EFWS] capacities and lower core power. Consequently, the total steam discharge to the atmosphere, and the resulting activity release, are lower than for the EPR_{4900}. Furthermore, it is possible to depressurise the affected SG using the dedicated main steam bypass and thus achieve LHSI/RHR conditions quicker.
- Depressurisation via the PSV is also possible if required.

The overall conclusion is that the analysis for the EPR_{4900} fully bounds the EPR_{4500} plant design.

5.3.3. Impact of the new design of pressuriser safety valves

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

Since the limiting transient for DNBR is loss of offsite power described in section 6 of Sub-chapter 14.3, the analysis of the impact of the pressuriser safety valve change from SEBIM to SEMPELL for the DNBR criterion is done for that case.

5.3.4. Impact of the safety classification change of the normal spray operations

The F1B qualification of normal spray allows credit for this system to be claimed to reach the LHSI/RHR connecting conditions. As a consequence, the pressuriser safety valves are not used between the controlled and the safe shutdown states. This modification requires the energy, formerly released by PSV, to be released by the steam generators. Thus, this modification can have consequences on the amount of steam discharged from steam generators.

In the BDR-99 analysis at 4900 MW, section 2.5.2 of Appendix 14B, the pressuriser safety valves are used for an operator action to decrease RCP [RCS] pressure to reach the LHSI/RHR connecting pressure. It is conservatively assumed that the valves remain open for 30 seconds. The SEBIM pressuriser valve steam capacity is 300 te/hour per valve. Thus the amount of steam released is 2.5 te per valve. Assuming the steam enthalpy at 30 bar at saturation of 2803 MJ/te, the energy released through the PSV is 2.1×10^4 MJ. The amount of steam produced for that energy value in the steam generator, considering a phase change enthalpy of 2030 MJ/te (under 9 bar saturated), equals 10.4 te.

In the BDR-99 analysis at 4900 MW, section 2.5.2 of Appendix 14B, the total amount of steam released from the four steam generators is 1134 te, when normal spray is not F1B classified. As a result, if normal spray is classified, the steam released to the atmosphere will be increased by 12 te, or 1%. This value is bounded by the PCSR power of 4500 MW, being 8% lower than the BDR-99 assumption of 4900 MW.

As a consequence, the steam mass release for the loss of condenser vacuum at 4500 MW with normal spray F1B classified is bounded by the study at 4900 MW without normal spray F1B classified.

5.4. SYSTEM SIZING

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

SECTION 14.3.5 - TABLE 1

Initial Conditions, Loss of Condenser Vacuum

Parameters	Values
Reactor coolant system	
Initial reactor power (%)	102 (100+2)
Initial average RCP [RCS] temperature (°C)	315.3 (312.8+2.5)
Initial reactor coolant pressure (bar)	157.5 (155+2.5)
RPV coolant flow (kg/s)	22225 Thermal-hydraulic design flow rate)
Pressurizer level (%)	61 (56+5)
Steam generators	
Initial steam pressure (bar)	Based on RCP [RCS] temperature
SG heat transfer area	Nominal
SG level (%)	47 (49-2)

6. SHORT-TERM LOSS OF OFFSITE POWER (<2 HOURS)

This event is analysed in reactor state A at power only. This state is the most challenging for the necessary countermeasures of decay heat removal in the very short term and those required to reach the controlled and safe shutdown state within the relevant decoupling and safety criteria.

6.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A complete loss of non-emergency AC power results in the loss of all power to the plant auxiliaries. Power is lost to the reactor coolant pumps, the condensate pumps, ARE [MFWS] pumps, and others. Consequently, the event causes an overheating of the primary and secondary sides with a risk of DNB and over pressurisation.

The loss of offsite power may be caused by a complete loss of the offsite grid or by a loss of the onsite AC distribution system.

The analysis of the short term total loss of offsite power also covers the slow decay of network frequency typically up to 2.5 Hz/s. Higher frequency decay rates are considered to be PCC-3 events and are discussed in section 9 of Sub-chapter 14.4.

The decrease in heat removal by the secondary system is accompanied by a flow coast-down, which further reduces the capacity of the primary coolant to remove heat from the core. The heat removal rate stabilises once actuation of the VDA [MSRT] occurs which maintains a maximum steam pressure of approximately 95.5 bar.

The main characteristics of a Loss Of Offsite Power event are described below:

- Instruments and safety-related valve-motors are supplied from batteries without interruption. The other safety-related motors are supplied according to the load sequence from the emergency DC power sources.

- As the secondary pressure rises following the turbine trip, the main steam relief isolation valves (MSRIV) upstream of the Main Steam Relief Control Valves (MSRCV) are automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump because it is not an F1 qualified system. If the steam flow rate through the relief trains is not available, the steam generator safety valves may lift to dissipate the heat of the fuel and coolant plus the residual heat produced in the reactor.

- As the no-load, hot standby, temperature is approached, the steam generator main steam relief valves, or safety valves if the relief valves are not available, are used to remove the residual heat and to maintain the plant at the hot shutdown condition.

- The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply essential plant loads following the prescribed loading sequence.

- The Emergency FeedWater System (ASG [EFWS]) begins operation when the SG level MIN2 setpoint[1] is reached. This provides the long-term primary system heat removal. The ASG [EFWS] injection is initiated automatically or by operator action.

Protection

The reactor trip signal ''low-reactor coolant pump speed in 2 of 4 loops'' provides protection for the complete loss of forced coolant flow due to LOOP:

The event sequence can be divided into two distinct phases that culminate when the controlled state at hot standby is reached.

- A short term phase covering the first seconds of the event which is characterised by reduced DNB margins
- A long term phase, during which decay heat removal must be maintained and the activity release to the atmosphere via the VDA [MSRT] must be evaluated until the controlled state is reached.

Controlled and Safe Shutdown State

It must be shown that the controlled state can be reached using only F1A functions and that the safe shutdown state can be reached using only F1A and F1B functions. The following safety and decoupling criteria must be met:

Safety Criterion: Radiological limits for normal operation must not be exceeded.

Decoupling Criterion: No DNB with a DNBR limit of 1.0

The following F1A functions are available to achieve the controlled state:

- The reactor trip is initiated following a reactor coolant pump speed < 91% signal
- Four VDA [MSRT] for secondary side heat removal and pressure limitation actuated at a high SG pressure of 95.5 bar
- Four ASG [EFWS] trains for secondary side water supply actuated following a low SG level signal at a setpoint of 7.85 m
- Three pressuriser safety valves for RCP [RCS] high pressure limitation with their setpoints at 174 bar (1st PSV) and 178 bar (2nd and 3rd PSV).

For the transition from the controlled state to the safe shutdown state, at least the following F1B functions are available

- Four VDA [MSRT] for cooldown to RIS/RRA [SIS/RHRS] level activated by the operator
- Two RBS [EBS] trains for boration during cool down initiated by the operator
- Four ASG [EFWS] trains for feedwater supply to the SG either initiated automatically or by manual action.

[1] See Sub-chapter 14.1

6.2. METHODS AND ASSUMPTIONS

The two phases of this transient, discussed above, are studied separately. The DNBR behaviour is investigated in a short-term study. In addition, a long-term study is focused on confirming that there is adequate heat removal and that the maximum system pressure remains acceptable. The codes for the calculations differ, using three dimensional kinetic calculations for the short term and point kinetic calculations for the long term.

The NLOOP and PANBOX/COBRA codes are used to analyse the LOOP transient. These codes are described in Appendix 14A.

For these studies, it is assumed that at time = 10 seconds (in the calculation with the NLOOP code, Version LIBNLPEPR4) and at time = 0 seconds (in the calculation with PANBOX/COBRA, Version 2.1R7), LOOP occurs that results in a turbine trip.

Choice of Single Failure and Preventive Maintenance

Time Period	Single Failure	Maintenance/Additional Unavailability
Short term phase	No impact	No impact
Long term phase	One MSRIV sticks in closed position	No impact

6.2.1. Important Phenomena and Qualification of the NLOOP and PANBOX/COBRA Models

In the LOOP transient all reactor coolant pumps are lost, together with the loss of the secondary side feedwater supply and the turbine condenser.

Phenomena

Primary Side

The transient involves loss of coolant flow and a reduction in the primary/secondary heat transfer. An early reactor trip at a relatively high coolant flow is followed by a reduction in the reactor power to decay heat levels and subsequent heat transfer by natural circulation. The decay heat is removed by the main steam relief valves at a constant steam pressure. Following reactor trip the primary coolant temperature and pressure increase and the PZR safety valves may be actuated to limit the maximum pressure. This leads to in-surges and out-surges from the pressuriser, with corresponding pressure changes in the PZR and primary circuit.

Secondary Side

During the transient, steam pressure increases due to the closing of the turbine/MSB valves. Fast opening and re-closing of the VDA [MSRT] occurs, to maintain a constant steam pressure at zero load. The heat transfer area of the SG tubes is not reduced and the separators are not overfilled.

Core Power and DNBR

Heat transfer in the average and hot channel until control rods insertion due to reactor trip

Neutron flux reactor power (integral and local) depends on the thermal-hydraulic parameters. Coolant flow behaviour depends on the pressure drop and cross-flow.

Qualification

Qualification of NLOOP Models for Primary and Secondary Side Phenomena

An integral test was carried out on a PWR power plant in which most of the primary and secondary phenomena are addressed [Ref-1]. This event was analysed using the NLOOP code and the results were benchmarked against the plant test results [Ref-2]. The benchmark showed good agreement between the test and analysis.

Additionally, some further integral tests of real plants and test facilities were performed and then compared with NLOOP calculations. These benchmarks also showed a good agreement with the plant or test facility behaviour for the single phenomena on the secondary and primary side [Ref-3]. A list of these tests is provided below:

Event	Plant/test facility	Phenomena	Reference
Emergency power mode	KNU1	See above	[Ref-1]
Turbine trip without main steam-bypass	Biblis B	Secondary and primary side pressure increase. PZR water level behaviour, primary coolant heat up under forced RCP [RCS] circulation.	[Ref-2]
LOOP followed by a total loss of FW supply	PKL3	Reactor coolant pumps coast-down, primary heat-up until attaining natural circulation at decay heat in the core.	[Ref-3]

Qualification PANBOX/COBRA Models for DNBR Related Phenomena

A reactor coolant pump shaft break event at full power in a PWR power plant in which DNB related phenomena were addressed is described in [Ref-4]. This event was analysed with PANBOX and the results showed a good agreement with the real plant behaviour.

Additionally, a further integral test on a PWR plant, a reactor trip from full load, was also analysed with PANBOX [Ref-5]. The analysis also showed a good agreement with the plant behaviour for the single phenomena of reactor trip.

To validate the COBRA hot channel calculations, the code has been benchmarked against a wide range of single heated bundle tests results, as follows: (a) Critical Heat Flux (CHF) tables, (b) pressure losses at different void fractions, (c) enthalpy and mass flux distribution, (d) slip flow

The list below shows the tests / phenomena used for the validation of PANBOX/COBRA:

Event	Plant/test facility	Phenomena	Reference
Reactor coolant pump shaft break	KKG	Fast flow reduction in one loop	[Ref-4]
Reactor trip	KKU	Locally and time-dependent reactor power due to rod insertion	[Ref-5]

6.2.2. Short-Term Study (Evaluation of Minimum DNBR)

Method of Analysis

Calculations described in this sub-section have been performed using the PANBOX/COBRA-3-CP computer codes.

The analysis of the DNBR transient is based on the following design approach:

The loss of reactor coolant flow due to a loss of offsite power is the limiting PCC-2 event where DNBR must not drop below the design limit of 1.0. To ensure this design criteria is met, the initial DNBR prior to the event must at least be equal to a minimum initial DNBR, the so-called DNBR-LCO described in section 8 of Sub-chapter 14.1. Consequently, the present analysis determines the DNBR-LCO that ensures the limit of DNBR=1.0 is met, i.e. no DNB occurs during the event.

The approach used to determine the DNBR-LCO is as follows:

- The initial state of the plant is defined, taking into account the best-estimate F∆H covering the fuel management schemes and the applicable LCOs including the dead-bands and uncertainties in important parameters. These parameters include RCP [RCS] pressure and temperature, power level, and axial offset. The resulting DNBR curve obtained using coupled neutronic/thermal-hydraulic calculation is called DNBR1.

- Starting from this initial state, an arbitrary increase in the local F∆H from arbitrarily increasing the power of the assembly containing the hot channel and surrounding channels is analysed. This analysis uses coupled neutronics/ thermal-hydraulic methods, until the minimum DNBR is equal to the physical criterion of 1.0. In these analyses the increase of F∆H includes a factor accounting, in an uncoupled manner, for the overall uncertainty of the low DNBR surveillance channel of 32%.

- The final calculation of this parametric study gives the variation of DNBR during the transient and the two lower initial limiting values of DNBR. These values correspond to the low DNBR-LCO, without uncertainties for the low DNBR surveillance channel called DNBR2, and the absolute minimum physical DNBR, including uncertainties due to the low DNBR surveillance channel, called DNBR3.

This approach assumes an increase of the hot channel F∆H and an increase in the hot channel exit void fraction. The density reactivity feedback will naturally decrease the hot channel power axial offset.

Initial and Boundary Conditions

The initial conditions were selected to be the most conservative for the assessment of the DNB limit during steady-state operation. The initial conditions are listed in Section 14.3.6 - Table 1.

Reactor Trip

In all cases for which the reactor is operating at power, the reactor trip is assumed to be generated by a low reactor coolant pump speed signal, the slowest trip signal during the event. The low reactor coolant pump speed setpoint is 91%, including measurement uncertainties. A conservative delay of 0.6 seconds is used between the setpoint actuation and the beginning of rod drop as discussed in Sub-chapter 14.1 – Table 11.

The following assumptions are used for the calculation of the RCCA worth versus time:

- Bottom peaked conservative power shape for the NLOOP overall plant analysis. With a bottom peaked power shape, the RCCA reactivity at the beginning of rod insertion is lower than for a top peaked power shape.

- RCCA worth as a function of RCCA position calculated by PANBOX on the basis of the actual three-dimensional power distribution for the DNBR analysis.

Flow Coast-Down and Heat Transfer to the Fuel-Clad

The core flow coast-down curve assumes a conservatively reduced value of the reactor coolant pump inertia, reduced by 15% from the calculated value.

The heat transfer coefficient between fuel and clad (α-gap) is assumed to be 9700 W/m^{2}K [Ref-1].

6.2.3. Long-Term Study Regarding Overall Plant Behaviour until the Controlled State

Method of Analysis

The analysis has been performed using the NLOOP code.

Initial and Boundary Conditions

Initial and boundary conditions were selected to be the most conservative for heat removal and for primary and secondary pressure. The initial conditions are listed in section 14.3.6 - Table 1. The following assumptions are also used within the analysis:

- The availability of the Main Steam Bypass (MSB) in the first 10 seconds after the accident is not considered.

- The analysis includes the effect of heat-retaining structures in the RCP [RCS] and SG.

- The setpoint for the reactor trip on low reactor coolant pump speed is the same as for the short-term analysis.

- The main steam relief valves and pressuriser safety valves setpoints include uncertainties.

- The worst single failure is assumed in one MSRIV stuck in the closed position

- Unavailability due to preventive maintenance of any safety-related component has no impact on the results and therefore is not considered.

6.2.4. Long-Term Study Regarding Maximum Activity Release

The long-term study of the maximum activity release is judged to be bounded by the “Loss of condenser vacuum” assessed in section 5 of this sub-chapter for the following reasons:

- The early automatic phase of the accident is nearly the same as for the “Loss of condenser vacuum” scenario. This is mainly the inventory loss of the SG with the single failure in one MSRIV. The only difference is the coast-down of the reactor coolant pump, which does not significantly influence the loss of the SG inventory.

- The maximum amount of steam released to the atmosphere in both the short- and long-term phases of the transient is lower because the reactor coolant pump power does not have to be removed.

6.3. RESULTS AND CONCLUSIONS

6.3.1. From the Initiating Event to the Controlled State

The PCC transient is not calculated in the PCSR. The ability to fulfil the safety and decoupling criteria is assessed using the results of the PCC transient analysis contained in section 15.2.5.1 of BDR-99 presented in section 2.5.1 of Appendix 14B, and from the comparison of relevant parameters between the EPR_{4500} and the EPR_{4900}, which is covered by BDR-99.

6.3.1.1. BDR-99 Status

The BDR-99 study concludes that, in order to meet the DNBR design limit, the initial DNBR physical value must not be lower than 1.26 (DNBR3). Assuming a low DNBR surveillance channel uncertainty of 32%, the corresponding onsite setting for the DNBR-LCO threshold (DNBR2) is fixed at 1.66.

As for the overall plant behaviour, the heat removal is provided by the VDA [MSRT] and the maximum primary pressure is 173 bar which is below the setpoint of the first pressuriser safety valve.

6.3.1.2. Relevant Differences between EPR_{4500} and EPR_{4900}

The main differences or features of the EPR_{4500} compared to the EPR_{4900} are listed below:

- Lower power level

- Same core thermal-hydraulic behaviour

- Higher RCP [RCS] average temperature (+1.5°C)

- Slightly lower core outlet temperature

- Same LCO for axial power distribution (axial-offset limit: 18%)
- Better RCCA efficiency
- Similar bounding reactivity feedbacks, in absolute value, for moderator and fuel
- Equivalent capacities for VDA [MSRT]
- Similar margin between MS operating pressure and VDA [MSRT] setpoints (17.5 bar)
- Same margin between RCP [RCS] pressure and PSV setpoints (19 bar)
- SG heat transfer surface corresponds to 106% power level.

6.3.1.3. Conclusions for EPR_{4500}

The main parameters impacting the DNBR decrease and the DNBR-LCO threshold during the LOOP transient are the primary flow transient, the core power distribution, limited by the LCO on the axial-offset, and the slight power decrease due to reactivity feedback prior to the reactor trip.

In light of the above comparison between the EPR_{4500} and the EPR_{4900}, it can be concluded that the DNBR-LCO threshold is not significantly impacted. A value of approximately 1.66 for DNBR-LCO threshold will remain adequate to protect the core against DNB.

Similarly, the BDR-99 demonstration of sufficient heat removal and the maximum pressures also apply to the EPR_{4500} conditions.

6.3.2. From the Controlled State to the Safe Shutdown State

The safe shutdown state is defined as a state where the RIS/RRA [SIS/RHRS] connection conditions are reached.

This transition is not analysed explicitly since it is bounded by the analyses of several other events, the reference cases. The table below presents the reference cases and their compliance with the safe shutdown criteria: (1) subcriticality, (2) activity release, and (3) decay heat removal.

Criteria	Reference case	Remark/Reason
Subcriticality	(see section 13 of this sub-chapter) (uncontrolled boron dilution)	Reactor trip/shutdown is demonstrated in the reference case, which is more severe due to one stuck control rod.
Maximum activity release	(see section 5 of this sub-chapter) (loss of condenser vacuum)	The reference case is more severe, as one SG is completely emptied
Decay heat removal	(see section 3 of Sub-chapter 14.5) (Feedwater system piping break)	The reference case is more severe, as only one train is available for cooldown

6.3.3. Impact of the new design of pressuriser safety valves

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.

Appendix 14B.2.5.1 - Figure 2 (3/5) of Appendix 14B shows that the RCP [RCS] pressure goes up to the maximum value of 173 bar. This is below the first SEBIM PSV opening setpoint of 174 bar, and thus there is no PSV opening during this sequence.

The SEMPELL pressuriser safety valve has a first opening threshold of 175 bar, higher than the SEBIM value opening setpoint of 174 bar. The maximum RCP [RCS] pressure stays below the SEMPELL opening pressure. Thus, there is no impact on the analysis of the change from the SEBIM type valve to the SEMPELL type valve.

6.3.4. Impact of the safety classification change of the normal spray operations

The F1B qualification of normal spray allows credit to be taken for this system in the transfer to reach the LHSI/RHR connecting conditions. The impact of this modification is shown on 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 this sub-chapter, concerning steam mass discharged to the atmosphere.

6.3.5. Systems Sizing

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

SECTION 14.3.6 - TABLE 1

Initial Conditions, Short-Term Loss of Offsite Power
(from Sub-chapter 14.1 – Tables 2 and 3)

Parameters	Values Used
Reactor Coolant System	
Initial reactor power (%)	102 (100+2)
Initial average RCP [RCS] temperature (°C)	315.3 (312.8+2.5)
Initial reactor coolant pressure (bar)	152.5 (155-2.5)
RPV coolant flow (kg/s)	22225 (Thermal-hydraulic design flow rate)
PZR level (%)	61 (56+5)
Steam Generators	
Initial steam pressure (bar)	According to RCP [RCS] temperature level
SG heat transfer area	Nominal

7. LOSS OF NORMAL FEEDWATER FLOW (LOSS OF ALL ARE [MFWS] PUMPS AND AAD [SSS] PUMPS)

7.1. ACCIDENT DESCRIPTION

The study of the loss of normal feedwater transient addresses core protection aspects and heat removal from the primary system.

The core is not affected because the latest automatic reactor trip that will occur is on the protection signal SG level < MIN1, at which point the heat transfer capability of the steam generators (SG) has not been significantly reduced. Each SG is subsequently supplied by the emergency feedwater system (ASG [EFWS]) once its level falls below MIN2 setpoint.

The rise in primary pressure during the transient, caused by the reduction in heat removal, remains within the limits calculated for the worst transient described in section 1 of Sub-chapter 3.4 dealing with protection of the primary system against over-pressure.

7.2. SYSTEMS SIZING

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

8. PARTIAL LOSS OF CORE COOLANT FLOW (LOSS OF ONE REACTOR COOLANT PUMP)

This event is only analysed in reactor state A.

8.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A partial loss of coolant flow can result from a mechanical or electrical failure in a reactor coolant pump or from a fault in the power supply or I&C to the pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the event, the immediate effect of the loss of coolant flow is an increase in the coolant temperature. This increase could result in DNB with subsequent fuel rod damage if the reactor is not tripped.

The limiting event for a partial loss of reactor coolant flow is the loss of one reactor coolant pump (RCP [RCS] pump) and it is classified as a PCC-2 event (see Sub-chapter 14.0)

The necessary protection for a partial loss of coolant flow event is provided by the following signals:

- Partial trip following a “low loop flow rate (one loop)” signal (“loss of one reactor coolant pump” limitation) which reduces the reactor power to approximately 50% is not taken into account in this analysis because it is not an F1A qualified system.

- Reactor trip following a low-low reactor coolant flow signal with a setpoint of < 25% in conjunction with reactor power > 75%. This is a preliminary value for this setpoint.

With loss of more than one RCP [RCS] pump, reactor trip is immediately actuated following a reactor coolant pump speed signal with a setpoint of below < 91% or a “low loop flow rate (two loops)” signal.

The Protection System setpoints and responses for F1A classified signals, used in this sub-Chapter, are summarised in section 5 of Sub-chapter 14.1 and Sub-chapter 14.1 – Table 9. I&C signal delays and safeguard action delays are summarised in Sub-chapter 14.1 – Table 11 and Sub-chapter 14.1 – Table 12.

Controlled and Safe Shutdown State

It must be shown that the controlled state can be reached using only F1A functions and that the safe shutdown state can be reached using only F1A and F1B functions. The following safety and decoupling criteria must be met

<u>Safety Criterion:</u> Radiological limits for normal operation.

<u>Decoupling Criterion:</u> No DNB

The following F1A safety functions are available to achieve the controlled state:

- The reactor trip is initiated following a “flow rate in one loop” < 25%. signal

- Four VDA [MSRT] for secondary side heat removal and pressure limitation are actuated following a "High SG pressure” > 95.5 bar" signal.

- Four ASG [EFWS] trains for secondary side water supply are actuated following a "Low SG level" < 7.85 m signal.

For the transition from the controlled state to the safe shutdown state the following F1B functions are available:

- Four VDA [MSRT] for cooldown to RIS/RRA [SIS/RHRS] level (manual action).
- Two RBS [EBS] trains for boration.
- Four ASG [EFWS] trains for feedwater supply to the SG (automatic or manual action).

8.2. METHODS AND ASSUMPTIONS

The NLOOP and PANBOX/COBRA-3-CP computers codes have been used to perform the analysis of the partial loss of core coolant flow transient. These codes are described in Appendix 14A.

The NLOOP code (Version LIBNLPEPR4) has been used to calculate the main plant parameters and the PANBOX/COBRA-3-CP code (Version 2.1R7) calculates the core behaviour and the DNBR values. The overall results are obtained through an iterative calculation between NLOOP and PANBOX/COBRA-3-CP.

8.2.1. Important Phenomena and Qualification of the Models Used in NLOOP and PANBOX/COBRA

The family of transients considered is the total or partial loss of primary forced reactor coolant flow. The phenomena addressed here are basically the same as those considered in section 6. Therefore, the scope of code qualification mentioned there also applies here.

A specific example of the present case, backflow in the affected loop, was verified by a recalculation of a corresponding event at the KKG/BAG NPP [Ref-1].

8.2.2. Initial and Boundary Conditions

The main initial conditions used in the analysis, such as reactor power, core average temperature, and primary pressure include the dead-band, total uncertainties, and maximum typical control deviations. The coolant flow corresponds to the thermal-hydraulic design flow.

Initial operating conditions assume the limiting DNBR during steady-state operation. These initial conditions are identical to those of the LOOP event described in section 6 of this sub-chapter.

<u>Reactor Trip</u>

The low-low loop coolant flow setpoint of 25%, including measurement uncertainties, is considered. A conservative delay between the reactor trip setpoint actuation and the beginning of rod drop is modelled.

The RCCA worth versus time is calculated on the basis of the:

- Bottom-peaked conservative power shape for the NLOOP overall plant analysis
- RCCA worth as a function of RCCA position calculated by PANBOX on the basis of the actual 3D power distribution used for the DNBR analysis.

Reactivity Feedback

- Moderator Temperature Coefficient

  The lowest initial value of the moderator temperature coefficient is assumed. This results in a low density feedback and in the maximum hot-spot heat flux during the initial part of the transient when the lowest DNBR is reached.

- Doppler Coefficient

  This coefficient is selected at its maximum absolute value to increase the positive reactivity addition when the power is reduced because of the density feedback.

- Density Feedback

  The density reactivity feedback is a code result calculated from the core initial condition with 18% axial offset, the lowest moderator temperature feedback, and the maximum absolute Doppler coefficient.

Flow Coast-Down and Heat transfer between Cladding and Coolant

The core flow coast-down curve is based on a conservatively reduced value of RCP [RCS] pump inertia with a -15% reduction assumed.

The heat transfer coefficient between fuel and clad (α-gap) is 9700 $W/m^2/K$ [Ref-1].

Choice of Single Failure and Preventive Maintenance

Single Failure:

- No single failure is assumed because there is no impact on the results of the analysis.

Preventive Maintenance:

- For the analysis of the transition to the controlled state, no preventive maintenance is assumed since it has no impact on the results of the analysis.
- For the analysis from the controlled to the safe shutdown state, the assumptions for the reference cases apply as discussed below.

8.3. RESULTS AND CONCLUSIONS

8.3.1. From Initiating Event to Controlled State

The PCC transient is not recalculated for this sub-chapter. The ability to meet the safety and decoupling criteria is based on the results of the PCC transient analysis contained in BDR-99 discussed in section 2.7 of Appendix 14B and from the comparison of the relevant parameters of the EPR_{4500} and the EPR_{4900} covered by the BDR-99.

8.3.1.1. BDR-99 Status

In the BDR-99 study the minimum DNBR, including uncertainties, is 1.1 and meets the decoupling criterion of 1.0. The DNBR value is obtained using conservative decoupling assumptions, such as the assumed value of F∆H which is increased to a value of 1.9, above the core design limit of 1.65. Consequently, the maximum linear heat rate (> 500 W/cm) is higher than the maximum allowed value of 470 W/cm obtained in the LOCA analysis (See Sub-chapter 4.3 – Table 1).

8.3.1.2. Relevant Differences between the EPR_{4500} and the EPR_{4900}

The relevant differences between the EPR_{4500} and the EPR_{4900} are the same as those outlined in section 6 of this sub-chapter for LOOP.

8.3.1.3. Conclusions for the EPR_{4500}

The differences between the EPR_{4500} and the EPR_{4900} are not likely to change the bounding character of the LOOP observed in BDR-99. Since the decoupling criterion "no DNB" is met for the LOOP event through an appropriate setting of the DNBR-LCO setpoints, as discussed in section 6 of this sub-chapter, it can be concluded that this criterion is also met by the loss of one RCP [RCS] pump event.

8.3.2. From the Controlled State to the Safe Shutdown State

The transition from the controlled state to the safe shutdown state is not analysed explicitly since it is covered by analyses of other events, the reference cases. The table below identifies the reference cases for demonstrating safe shutdown and compliance with the three criteria: ‘’subcriticality”, ‘’decay heat removal” met by achieving RIS/RRA [SIS/RHRS] connection conditions, and ‘’activity release/barrier integrity” to stay within the PCC limits.

Criteria	Reference Case	Remark/Reason
Subcriticality	section 13 of this sub-chapter Uncontrolled boron dilution	Reactor trip/shutdown is demonstrated in the reference case, which is more severe due to one stuck control rod.
Maximum activity release	section 5 of this sub-chapter Loss of condenser vacuum	The reference case is more severe, as one SG is completely emptied
Heat removal	section 3 of Sub-chapter 14.5 Feedwater system piping break	The reference case is more severe, as only one train is available for cooldown

8.3.3. Systems Sizing

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

9. UNCONTROLLED RCCA BANK WITHDRAWAL AT POWER

The Uncontrolled RCCA Bank Withdrawal transient is classified as PCC-2 event if it occurs in reactor state A. It is classified as a PCC-3 event if it occurs in reactor states B, C, or D and is discussed in section 12 of Sub-chapter 14.4.

The detailed analysis presented in the following sections is performed in state A when the reactor is at power. The cases of hot standby (core critical) and hot shutdown (core subcritical) are addressed in section 10 of this sub-chapter.

A conservatively high withdrawal speed of 75 cm/min for all RCCAs is assumed for this assessment (See Sub-chapter 4.3 – Table 1).

The Protection System setpoints and responses for F1A classified signals, used in this sub-Chapter, are summarised in section 5 of Sub-chapter 14.1 and Sub-chapter 14.1 – Table 9. I&C signal delays and safeguard action delays are summarised in Sub-chapter 14.1 – Table 11 and Sub-chapter 14.1 – Table 12.

9.1. IDENTICATION OF CAUSES AND ACCIDENT DESCRIPTION

The uncontrolled RCCA withdrawal at power results in an increase in the core heat flux. Since heat removal by the steam generators lags behind the core power generation, prior to the steam generator pressure reaching the relief or safety valves setpoint, there is a net increase in the reactor coolant temperature and pressure.

In the event of a slow reactivity insertion transient, the increase of the coolant temperature follows the nuclear power increase. If sufficient, this could eventually result in DNB.

In the event of a fast reactivity insertion transient the nuclear power increases very rapidly. In contrast, the coolant temperature increases slowly. These conditions could eventually lead to fuel damage due to DNB or a high linear power density.

The uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at power can occur due to:

- an operator error
- a control system error
- an equipment failure.

The transient can be divided into two distinct phases:

From the Initiating Event to the Controlled State

The reactivity insertion causes an increase of both the nuclear power and the heat flux, and possibly the coolant temperature.

During this phase, a reactor trip is actuated by either the low DNBR protection channel or the high neutron flux rate of change protection, which are both F1A classified.

The controlled state is the hot shutdown state defined as (See Sub-chapter 14.0):

- Nuclear power = 0% full power
- Coolant temperature = 303.3°C
- Primary pressure = 155 bar
- RCCAs fully inserted
- Boron concentration at the initial power value
- Xenon level equal to the initial xenon level
- Reactor coolant pumps are running.

The shutdown margin described in section 5 of Sub-chapter 4.3 keeps the core subcritical after reactor trip.

The controlled state is similar to that following the loss of condenser vacuum transient discussed in section 5 of this sub-chapter where more details are provided.

From the Controlled State to the Safe Shutdown State

The safe shutdown state corresponds to a state where the LHSI is operating in SIS-RHR mode, or where the RIS/RRA [SIS/RHRS] connection conditions are reached. It is defined as follows (see Sub-chapter 14.0):

- Nuclear power = 0% full power
- Hot leg coolant temperature = 180°C
- Primary pressure = 30 bar
- RCCAs fully inserted
- Boron concentration sufficient to maintain core subcriticality after the xenon depletion
- Decay heat is removed by the steam generators or by the LHSI in RIS/RRA [SIS/RHRS] mode.

The principal actions to be performed to reach the safe shutdown state are:

- RCP [RCS] cooldown and depressurisation (F1 classified)
- Boration performed by the RBS [EBS] (F1 classified).

The activity release during an uncontrolled RCCA bank withdrawal at power is bounded by the loss of condenser vacuum event discussed in section 5 of this sub-chapter.

The heat removal capability for this event is bounded by the feedwater system piping break event discussed in section 3 of Sub-chapter 14.5. The four steam generators remain available and consequently the ratio between the heat flux produced in the RCP [RCS] and the number of available SG is lower for the uncontrolled RCCA bank withdrawal at power.

This event is bounded by the uncontrolled boron dilution event discussed in section 13 of this sub-chapter for its impact on the subcriticality criterion.

The sequence of operator actions needed to reach the safe shutdown state is detailed in section 3 of Sub-chapter 14.5. The analysis presented in the following section deals with the phase of the transient from the initiating event to the reactor trip.

9.1.1. Safety and Decoupling Criteria

This event is classified as a PCC-2 event. The safety criteria are the radiological limits for normal operation.

The decoupling criteria, to maintain the integrity of the barriers to radiological release, are:

- Critical Heat Flux limit:
    - This limit is satisfied if the minimum DNBR during the transient remains above the DNBR design limit as discussed in Sub-chapter 14.1,
- Fuel temperature limit:
    - This limit is satisfied if the maximum linear power density at the hot spot remains below 590 W/cm (see Sub-chapter 4.4 - Table 1)

9.1.2. Reactor Protection System Actions

The following reactor trip setpoints of the reactor protection system provide the protection of the core during this type of transient:

- Low DNBR
- High neutron flux rate of change
- High Linear Power Density
- High core power level
- Pressuriser pressure > MAX2[1]
- Pressuriser level > MAX1[1].

The low DNBR and HLPD protection channels provide effective core protection for most reactivity insertion transients, with the exception of the fastest transients where the protection must be activated within a very short period. This is provided by the high neutron flux rate of change protection channel.

[1] See Sub-chapter 14.1

9.2. METHODS AND ASSUMPTIONS

9.2.1. Single Failure Selection

The worst single failure and unavailability due to maintenance on F1 classified systems must be assumed as required by the PCC accident analysis rules defined in Sub-chapter 14.0.

This is identified as the highest worth rod remaining stuck above the core during reactor trip.

There is no preventive maintenance applicable to this transient.

9.2.2. Method of Analysis

This transient is simulated using the THEMIS code, a multi-loop system code with a point kinetics neutronics model, described in Appendix 14A. This code calculates the evolution of the following parameters during the transient:

- Nuclear power
- Heat flux
- Pressuriser pressure
- Temperature in the loops and at the core inlet.

The variation of the DNBR during the transient is then calculated using the thermal-hydraulic design code FLICA described in Appendix 14A. Simultaneously, a global processing algorithm is used to determine the evolution of the real time DNBR value calculated by the protection system. These two calculations are performed with the same axial power distribution and the same nuclear $F\Delta H$. The axial power shape and the nuclear $F\Delta H$ are kept constant throughout the transient.

A general description of the low DNBR protection channel as simulated in the global processing algorithm is provided in the following paragraphs. A simplified diagram of the algorithm is shown in Section 14.3.9 - Figure 1. The time constants for the relevant modules, lead-lag modules and filter modules are given in the same figure. A full technical description of the algorithm is presented in section 2.2.1 of Sub-chapter 4.4.

The on-line DNBR is calculated using the following measurements:

- The nuclear power distribution derived from the nuclear in-core instrumentation by the Self Powered Neutron Detectors (SPND)
- The pressure derived from the primary pressure sensors
- The core flow derived from the reactor coolant pump speed sensors
- The inlet temperature derived from the cold leg temperature sensors.

The SPND measurements are processed using a filtering module. The time constant of this module is chosen to take into account the delay between the variations of the nuclear power, the parameter measured for calculation of the DNBR, and the variations of the heat flux, the relevant parameter for the value of the physical DNBR.

The cold leg temperature measurements are processed using a filtering module and a lead-lag module. This lead-lag module compensates for the delay due to the temperature sensor and the consequent difference between the cold leg temperature measurement, used for the DNBR calculations, and the core inlet temperature, the relevant parameter for the physical DNBR.

The calculated DNBR value is also processed using a lead-lag module, mainly to compensate for the total delay between low DNBR setpoint being reached and the start of rod drop following reactor trip signal.

The purpose of the analysis is to optimise the response time, time constants and protection setpoints to achieve the following:

- Provide effective protection at all reactivity insertion rates
- Limit the DNBR variation during the transient
- Provide margins for operational transients.

The on-line low DNBR protection channel provides the core protection for a wide range of reactivity insertion transients. However, its response time is not rapid enough to cope with very fast reactivity insertion transients. Protection against these reactivity insertion transients is provided by the high neutron flux rate of change protection channel.

9.2.2.1. Initial Conditions

The analysis of the transient is performed at full and intermediate power levels.

The initial conditions are chosen conservatively for the assessment of the DNBR. The initial values for power, average coolant temperature, and reactor coolant pressure are the extreme values allowed for operation in steady-state conditions.

For the 100% Nominal Power transients, the nuclear F∆H value is chosen so that the initial DNBR is equal to the DNBR limiting value.

For intermediate power levels, an initial decoupling nuclear F∆H value is considered, taking into account the insertion of control rods.

9.2.2.2. Core Related Assumptions

Reactivity Coefficients

Two cases are analysed for each initial power level and for each reactivity insertion rate:

- Minimum reactivity feedback:
    - The moderator density coefficient is zero
    - The Doppler coefficient takes its minimum absolute value
    - Kinetic coefficients are assumed to be at their minimum values.
- Maximum reactivity feedback:
    - The moderator density coefficient is at its maximum value

- o The most negative Doppler coefficient is assumed
- o Kinetic coefficients are assumed to be at their maximum values.

Fuel-to-Coolant Heat Transfer Coefficient

A maximum value of the fuel-to-coolant heat transfer coefficient is used to maximise the thermal power during the transient.

Shutdown Margin

The rod having the greatest worth is assumed stuck above the core. The negative reactivity insertion following the trip is thus minimised, resulting in a minimum final shutdown margin (see Sub-chapter 4.3). In addition, the most conservative negative reactivity insertion curve, as a function of time, is used.

9.2.2.3. Reactivity Insertion Rate

To verify that the different means of reactor trip provide protection in all possible situations, a wide range of reactivity insertion rates are considered. These cover all possible cases of RCCA withdrawal over the whole set of fuel management cycles, starting from different initial power levels.

9.2.2.4. Protection Actions

The setpoint values include instrumentation and setpoint uncertainties. The maximum time delays are assumed within the analysis.

9.2.2.5. Control Actions

The pressure control system is assumed to be operational and the pressuriser spray flow rate is assumed to be at its maximum value. This limits the reactor coolant pressure increase during the transient and is conservative for the assessment of DNB.

9.3. RESULTS AND CONCLUSIONS

The low DNBR or the ex-core high neutron flux rate of change channel, depending on the reactivity insertion rate, provides protection for this event.

9.3.1. Conclusions for the EPR_{4900}

The results for the EPR_{4900} are provided in section 2.10 of Appendix 14B. The conclusions from the EPR_{4900} study are outlined here:

- From an initial power equal to 100% Nominal Power, the low DNBR channel provides core protection up to 32 pcm/s for minimum reactivity feedback, and for the whole range of reactivity insertion rates for maximum reactivity feedback.

- From 32 pcm/s to 90 pcm/s with minimum reactivity feedback, core protection is provided by the high neutron flux ex-core channel.

- From an initial power equal to 10% Nominal Power, the low DNBR channel provides protection for the whole range of reactivity insertion rates and reactivity feedbacks.

- The low DNBR and the excore high neutron flux rate of change channels provide adequate protection over the entire range of possible reactivity insertion rates.

- The analysis of reactivity insertion transients concludes that the minimum DNBR value for initial conditions that meet the DNBR design limit is 1.26.

9.3.2. Conclusions for the EPR_{4500}

The uncontrolled RCCA bank withdrawal at power transient has not been recalculated within this sub-chapter, as the results should not be significantly different from the results for the EPR_{4900} presented in Appendix 14B for the following reasons:

- The initiating event is considered in a conventional and conservative manner. A reactivity insertion rate range of 0 to 90 pcm/s independent of the real core is assumed.

- The main result of this study is to establish the limiting reactivity insertion rate for which the low DNBR protection channel provides protection and to calculate the loss of DNBR margin with the reactor trip initiated by the high neutron flux rate of change protection channel. This depends mainly on the settings of the protection channels, which, at this stage of the project, remain unchanged from the EPR_{4900}.

- If subsequent analysis leads to the transient becoming worse than the loss of offsite power (short term) transient, the DNBR design limit will have to be adjusted slightly.

9.3.3. Impact of the new design of pressuriser safety valves

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.

Appendix 14B.2.10 - Figure 7 shows that the maximum value of RCP [RCS] pressure does not reach 174 bar. The first SEBIM PSV opening setpoint is 174 bar and therefore there is no PSV opening during the transient.

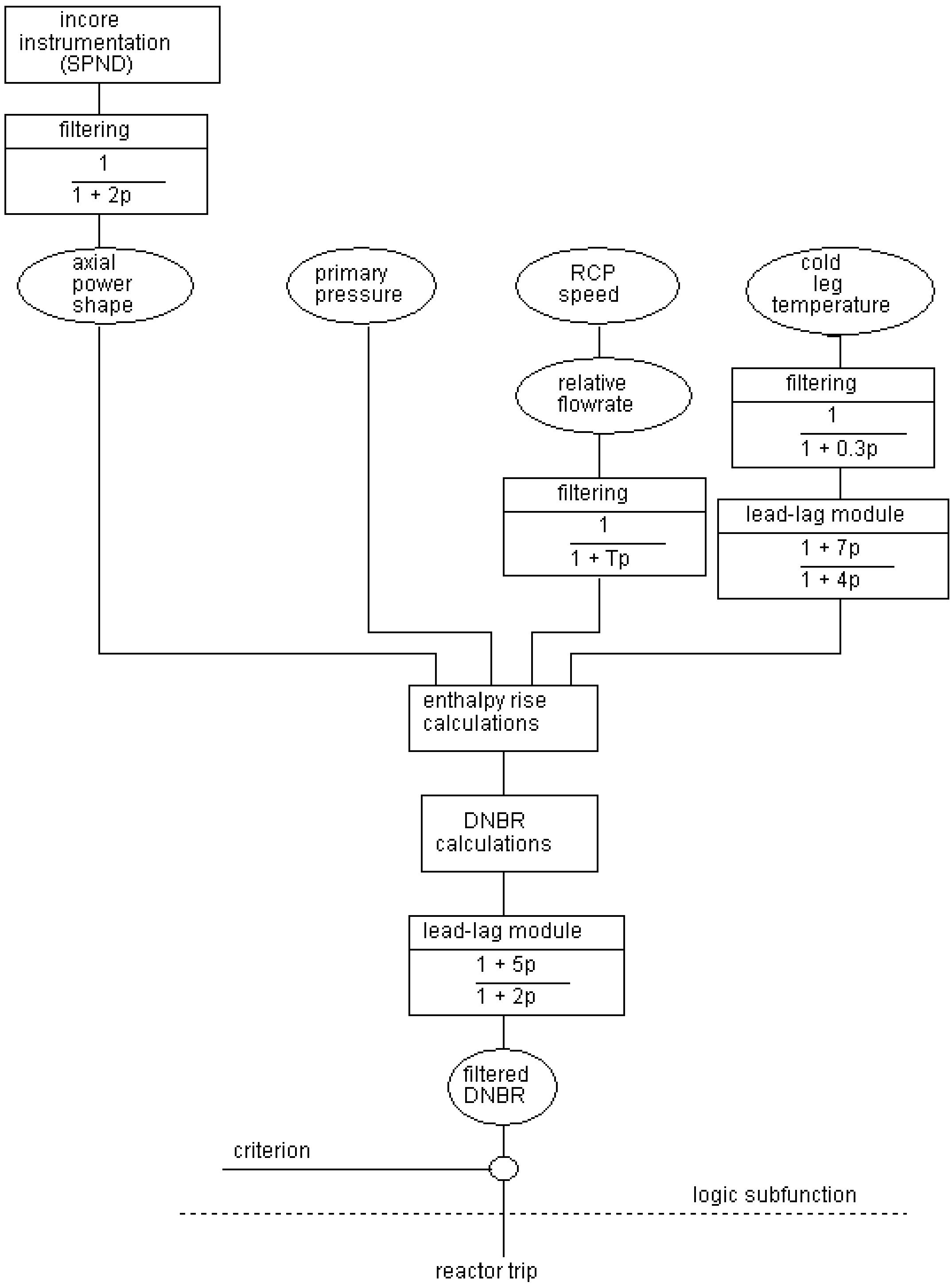
The SEMPELL pressuriser safety valves have the first opening pressure at 175 bar which is higher than the SEBIM value. The maximum RCP [RCS] pressure stays below the SEMPELL valve opening pressure and therefore there is no impact on the analysis results from the change of the valve type from SEBIM to SEMPELL.

9.3.4. Systems Sizing

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

SECTION 14.3.9 - FIGURE 1

Simplified Diagram of the Low DNBR Protection Channel



10. UNCONTROLLED RCCA BANK WITHDRAWAL FROM HOT ZERO POWER CONDITIONS

This is a PCC-2 category transient if it occurs in reactor state A. It is classified as a PCC-3 category transient if it occurs in reactor states B, C, or D and is discussed in section 12 of Sub-chapter 14.4.

In state A, the detailed analysis presented below is performed for two conditions. It covers: hot standby with the reactor critical and hot shutdown with the reactor subcritical. The case at power is addressed in section 9 of this sub-chapter.

The Protection System setpoints and responses for F1A classified signals, used in this sub-Chapter, are summarised in section 5 of Sub-chapter 14.1 and Sub-chapter 14.1 – Table 9. I&C signal delays and safeguard action delays are summarised in Sub-chapter 14.1 – Table 11 and Sub-chapter 14.1 – Table 12.

10.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

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 and resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control system.

It is assumed that a malfunction of the reactor control system could not result in a simultaneous removal of both the shutdown and the control banks. Therefore, the maximum reactivity insertion is that occurring with the simultaneous withdrawal of either all initially inserted control RCCA at the maximum control rod speed, or all shutdown RCCA at the maximum control speed. In the following assessment, the shutdown bank refers to all shutdown RCCA and the control bank refers to all control RCCA. The distribution of these two sets of RCCA within the core is shown in Section 14.3.10 - Figure 1.

The accident scenario may vary depending on the initial state:

- Hot shutdown: both the control and the shutdown banks are initially fully inserted. Either the shutdown bank is withdrawn with the control bank remaining inserted or the control bank is withdrawn with the shutdown bank remaining inserted,

- Hot standby: the shutdown bank is initially fully withdrawn, while the control bank is fully inserted[1] then withdrawn. The accidental withdrawal of the shutdown bank is studied at Beginning Of Cycle (BOC) only to cover possible mishandling during measurement tests. In such cases, the control bank is initially fully withdrawn, while the shutdown bank is fully[1] inserted then withdrawn.

[1] The initial positions may differ a little in the transient study so as to provide maximum reactivity insertion from of the RCCA being withdrawn when the core returns to power.

The uncontrolled addition of reactivity to the reactor core by an uncontrolled RCCA withdrawal results in a power excursion. The core response to the continuous reactivity insertion is characterised by a very fast rise in neutron flux, limited by the negative Doppler reactivity feedback. This self limitation of the power excursion is important because it limits the core power during the time interval before the protection system acts. The neutron flux is measured during the transient. If the detected flux exceeds the setpoint value, reactor trip will be initiated, dropping all the RCCA.

The transient is terminated by the reactor trip initiated by the high neutron flux rate of change reactor protection channel. The two following signals may also be actuated during an uncontrolled RCCA withdrawal:

- Low doubling time (intermediate range)
- High neutron flux. (intermediate range)

Due to the rapid power increase during the uncontrolled RCCA withdrawal, all the reactor trip setpoints are reached nearly simultaneously. Reactor trip is conservatively assumed to occur when the maximum power level is reached.

Controlled and Safe Shutdown State

It must be shown that the controlled state can be reached using only F1A functions and the safe shutdown state reached using only F1A or F1B functions. It must also be shown that the following safety and decoupling criteria are met:

Safety Criterion:

Radiological limits for normal operation.

Decoupling Criteria:

- No DNB for a DNBR design limit: 1.21, as discussed in Sub-chapter 14.1
- Maximum fuel clad temperature of 1482°C (see Sub-chapter 14.0)
- No fuel melting. The melting temperature of un-irradiated $U0_2$/MOX is 2810/2737°C and decreasing by 7.6/4°C per 10,000 MWd/teU burnup. (See Sub-chapter 4.4).

The following F1A functions are available to achieve the controlled state:

Reactor Trip:

The reactor trip is initiated by one of the neutron flux measurement signals described above. The reactor trip is conservatively assumed to occur when the maximum power level is reached.

Systems:

- Four VDA [MSRT]
- Four ASG [EFWS] trains

Signals:

- Actuation of the VDA [MSRT] at a SG pressure of 95.5 bar

- Actuation of the ASG [EFWS] trains at a low SG water level < 7.85 m.

For the transition from the controlled state to the safe shutdown state, at least the following F1B functions are available:

- Four VDA [MSRT] for cooldown to RIS/RRA [SIS/RHRS] injection (manual action)
- Two RBS [EBS] trains for boration during the cooldown (manual action)
- Four ASG [EFWS] trains for feedwater supply to the SG (automatic or manual action).

10.2. METHODS AND ASSUMPTIONS

10.2.1. Method of Analysis

The analysis of the uncontrolled RCCA withdrawal has been performed using the SMART/FLICA computer codes described in Appendix 14A. The analysis relies on transient calculations using three-dimensional neutron kinetics and including local thermal-hydraulic feedback in an open channel configuration. A hot spot analysis is performed simultaneously, enabling the computation of hot fuel rod temperature and DNBR in a hot channel. The safety-relevant parameters are derived from the hot spot analysis.

Important Phenomena and Qualification of the Models used in SMART/FLICA

The family of transients is fast reactivity increases due to inadvertent RCCA movements.

Phenomena

Primary Side

A sudden power increase from zero load with limited primary temperature and pressure increase due to early reactor trip.

Secondary Side

Due to early reactor trip the impact on the secondary side is negligible. The core inlet temperature and reactor pressure remain practically unchanged before reactor trip.

Core Behaviour - Core Power and DNBR

Heat transfer in the average and hot channel until RCCA insertion due to reactor trip.

Neutron flux and reactor power, both total and local, depend on the thermal-hydraulic parameters. Coolant flow behaviour depends on the pressure drop and cross-flow.

Since the core inlet temperature does not change, a specific analysis of the overall plant behaviour is not needed.

10.2.2. Boundary Conditions and Assumptions

Conservative boundary conditions are used in the analysis to cover realistic future core loadings and uncertainties on relevant operating parameters. The impact of most assumptions is to achieve an increase in total power level or an increase in local power density for the analysis. These assumptions result in higher fuel temperatures, higher coolant temperatures, and lower DNBR values. Uncertainties in the main plant parameters are treated in a deterministic way.

The uncontrolled RCCA withdrawal is investigated for core burnup states at Beginning of Cycle (BOC) and End of Cycle (EOC).Different sets of initial operating parameters for calculations of DNBR, cladding temperature, and fuel centreline temperature are defined for core burnup states at BOC and EOC. These initial conditions and other accident specific input data are listed in Section 14.3.10 - Table 1. Additional information on initial conditions is given in Sub-chapter 14.1.

The main boundary conditions for uncontrolled RCCA withdrawal and their effects are as follows:

- A low initial power level at HZP

  Effect: Faster power excursion and thus a higher power peak.

- RCCA withdrawal velocity: The analyses assume the maximum velocity of 75 cm/min (see Sub-chapter 4.3 – Table 1)

  Effect: A higher positive reactivity insertion rate and thus a higher power peak.

  The reactivity released is calculated on the basis of a maximum RCCA worth. The values used are presented in Section 14.3.10 - Table 1. These maximum values refer to an RCCA withdrawal from positions of 30-40 cm (BOC) and 20-30 cm (EOC) respectively. This allows the core to reach prompt criticality, $\rho \approx \beta_{eff}$, if the accident is initiated from hot standby conditions

  The total delay until the reactor trip consists of the following three contributions:

  - Time delay until the trip setpoint, assumed to occur at the power peak, is reached. This is computed as part of the transient calculation.
  - An additional delay between reactor trip signal and the start of RCCA drop of 0.6 seconds as defined in Sub-chapter 14.1 – Table 11
  - The drop time of the RCCA.

- The RCCA worth as a function of time is calculated on the basis of the following:

  - RCCA drop characteristic. The RCCA position as a function of time is described in Sub-chapter 14.1 for an RCCA dropped from the top of the core
  - A highly conservative drop time for the RCCA released from inside, or slightly outside, the dashpot. This is the case for RCCA being withdrawn at time of reactor trip
  - RCCA worth as a function of RCCA position, calculated by SMART on the basis of the three-dimensional power distribution computed during the transient analysis.

- Two fuel-cladding gap heat transfer regimes are considered:
  - A constant high value of 150000 W/(m^2.K) to simulate a closed gap
  - A constant low value of 2400 W/(m^2.K)

The results show that a constant high value for the gap conductivity is conservative for calculating low DNBR, high fuel cladding temperature, and high fuel temperature as this assumption leads to much higher power levels.

The analysis has been performed for the following set of fuel management options corresponding to a 4250 MW power level:

- UO_2 – 18 Month INOUT Cycle 1
- UO_2 – 18 Month INOUT Cycle 2
- UO_2 – 18 Month INOUT Cycle 3
- UO_2 – 18 Month INOUT Equilibrium Cycle
- UO_2 – 18 Month OUTIN Equilibrium Cycle
- MOX – 18 Month INOUT Equilibrium Cycle
- UO_2 – 12 Month INOUT Equilibrium Cycle

Details of these seven fuel management options are given in PSAR 4250. AREVA [Ref-1]. The core model is varied to cover all fuel management options and calculation uncertainties. The main assumptions varied within the cases studied are:

- A conservative fraction of delayed neutrons, the bounding values are discussed in Section 14.3.10 - Table 1

  Effect: Based on a sensitivity study the high bounding value for the fraction of delayed neutrons is used in this analysis. The high delayed neutron fraction value leads to higher fuel temperatures and a lower DNBR during an uncontrolled RCCA withdrawal.

- A minimum fuel temperature feedback coefficient using the bounding values defined in Section 14.3.10 - Table 1

  <u>Effect:</u> Higher peak power value, since the power peak is limited by the Doppler coefficient.

- A reduced moderator feedback using the bounding values defined in Section 14.3.10 - Table 1.

  <u>Effect</u>: Higher power level during the transient due to a minimum negative reactivity insertion.

- In addition, an axial offset for full power is used for this zero load case as defined in Section 14.3.10 - Table 1.

10.2.3. Method of Analysis

Single failure assumption:

The assumption of a stuck RCCA at reactor trip is the worst single failure for this accident.

Preventive Maintenance:

- No preventive maintenance is assumed for the period from the initiating event to reaching the controlled state since it has no impact.

- From the controlled to the safe shutdown state, the assumptions for the reference cases apply as discussed in section 10.3.2 below.

10.3. RESULTS AND CONCLUSIONS

10.3.1. From the Initiating Event to the Controlled State

10.3.1.1. Conclusion for the EPR_{4250}

The uncontrolled RCCA withdrawal is modelled by withdrawing either the control bank or the shutdown bank at the maximum RCCA withdrawal velocity (See Sub-chapter 4.3 - Table 1). When the inserted reactivity exceeds the value of the fraction of delayed neutrons, the reactor becomes prompt critical, which leads to a strong power peak. The power excursion causes an increase in fuel temperature and the prompt power peak is mainly limited by the Doppler reactivity feedback. As a result, core reactivity becomes lower than the fraction of delayed neutrons and the prompt power excursion stops.

The transient is ultimately terminated by the reactor protection system as the detected neutron flux exceeds the reactor trip setpoint. This is assumed coincident with the peak reactor power.

Maximum and minimum values for the relevant safety parameters are determined when the reactor trip occurs from the transient calculations. These values are summarised in Section 14.3.10 - Table 2. The time dependent behaviour of the reactor power and the relevant safety parameters are shown in Section 14.3.10 - Figure 2, sheets 1 to 6.

The plots are presented only for the withdrawal of the shutdown bank in hot shutdown at EOC as this is the most limiting case. The other cases show a similar transient behaviour.

It is shown that the safety and decoupling criteria are met in the event of an uncontrolled RCCA withdrawal from hot shutdown or hot standby, even with very conservative assumptions.

The increase of fuel and clad temperatures is small and the minimum DNBR remains above the limit value. Thus fuel or cladding damage will not occur.

The heat removal in the controlled state is provided by the VDA [MSRT] and the ASG [EFWS]. Since the integrity of the barriers is not impaired, the activity release is within the limits of PCC-2.

10.3.1.2. Conclusions for the EPR_{4500}

Dedicated calculations for the present Pre-Construction Safety Report are not carried out, since the results are not expected to be significantly different for the following reasons:

- The analysis of the EPR_{4250} shows large margins to the relevant criteria with only slight increases in fuel temperature, and a minimum DNBR significantly above the criterion.
- The operating parameters and the neutronics data for the EPR_{4500} are comparable with the EPR_{4250} as shown in Section 14.3.10 - Table 3.

10.3.2. From the Controlled State to the Safe Shutdown State

This transition is not analysed explicitly since it is covered by analyses of other events, the reference cases. The table below presents these reference cases and their compliance with the three criteria of "subcriticality", "decay heat removal provided by the RIS/RRA [SIS/RHRS]'', and "activity release/barrier integrity" to justify being within the PCC limits.

Criteria	Reference Case	Remark/Reason
Subcriticality	section 13 of this sub-chapter RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant	Reactor trip/shutdown is demonstrated in the reference case, which is more severe due to one stuck control rod.
Maximum Activity Releas	section 5 of this sub-chapter Loss of condenser vacuum	The reference case is more severe, as one SG is completely emptied
Decay Heat removal	section 3 of Sub-chapter 14.5 Feedwater system piping break	The reference case is more severe, as only one train is available for cooldown

10.3.3. Systems Sizing

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

SECTION 14.3.10 - TABLE 1 (1/2)

Initial Reactor State Before Uncontrolled RCCA Withdrawal
EPR 4250 MW

Withdrawal of Control Bank[2]:

	Unit	BOC		EOC	
Initial reactor state		Hot shutdown	Hot standby	Hot shutdown	Hot standby
Reactor power	(-)	10^{-13}	10^{-13}	10^{-13}	10^{-13}
Initial subcriticality	(pcm)	-2000	0	-2000	0
RCCA configuration		All RCCA fully inserted	Control bank inserted Shutdown bank fully extracted	All RCCA fully inserted	Control bank inserted Shutdown bank fully extracted
Axial offset at Hot Full Power (HFP)	(%)	-30[3]			
Initial axial offset at Hot Zero Power (HZP)	(%)	-85.7	-17.1	-80.9	52.7
Reactor trip reactivity		Code result			
Control rod drop time for RCCA falling from top of core	(s)	4			
Control rod drop time for RCCA falling from inside dashpot	(s)	2			
Effective fraction of delayed neutrons incl. 5% uncertainty	(pcm)	754		559	
Xenon at HFP		3			
Fuel R-feed back at HZP incl. 20% uncertainty	(pcm/°C)	-2.2		-2.6	
Moderator R-feed back at HZP including 3.6 pcm/°C uncertainty	(pcm/°C)	-5.4		-38.0	
Core flow rate	(m^3/h)	102740			
Core inlet temperature	(°C)	305.8			
Primary pressure	(bar)	152.9			
Single failure		Stuck RCCA			
Maximum RCCA withdrawal efficiency	pcm/cm	87.0	40.5	118.0	58.9

[2] Values include uncertainties
[3]The top peaked xenon distribution used yields a bottom peaked axial power distribution peaked up to a target value for axial offset of -30%, all rods out.

SECTION 14.3.10 - TABLE 1 (2/2)

Initial Reactor State before Uncontrolled RCCA Withdrawal
EPR 4250 MW

Withdrawal of Shutdown Bank[4]:

	Unit	BOC		EOC	
Initial reactor state		Hot shutdown	Hot standby	Hot shutdown	
Reactor power	(-)	10^{-13}	10^{-13}	10^{-13}	
Initial subcriticality	(pcm)	-2000	0	-2000	
RCCA configuration		All RCCA fully inserted	Shutdown bank inserted Control bank fully extracted	All RCCA fully inserted	
Axial offset at HFP	(%)	-30[5]			
Initial axial offset (at HZP)	(%)	-85.7	-45.3	-80.9	
Reactor trip reactivity		code result			
Control rod drop time for RCCA falling from top of core	(s)	4			
Control rod drop time for RCCA falling from inside dashpot	(s)	2			
Effective fraction of delayed neutrons incl. 5% uncertainty	(pcm)	754		559	
Xenon at HFP		[5]			
Fuel R-feed back at HZP incl. 20% uncertainty	(pcm/°C)	-2.2		-2.6	
Moderator R-feed back at HZP incl. 3.6 pcm/°C uncertainty	(pcm/°C)	-5.4		-38.0	
Core flow rate	(m^3/h)	102740			
Core inlet temperature	(°C)	305.8			
Primary pressure	(bar)	152.9			
Single failure		stuck RCCA			
Maximum RCCA withdrawal efficiency	pcm/cm	110.2	60.8	151.9	

[4] Values contain uncertainties
[5] The top peaked xenon distribution used yields a bottom peaked axial power distribution peaked up to a target value for axial offset of -30%, all rods out

SECTION 14.3.10 - TABLE 2 (1/2)

Results for EPR 4250 MWth [Ref-1]

Withdrawal of Control Bank:

	BOC				EOC			
Initial state	Hot shutdown		Hot standby		Hot shutdown		Hot standby	
Initial rod insertion	All RCCA fully inserted		Control bank inserted Shutdown bank fully extracted		All RCCA fully inserted		Control bank inserted Shutdown bank fully extracted	
Assumption on pellet-to-cladding heat transfer	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity
Time of nuclear flux peak (s)	32.0	32.0	30.9	30.9	24.3	24.3	23.8	23.8
Maximum Nuclear Power − Pnuc (fraction of Nominal Power − NP[6])	0.796	0.751	0.596	0.578	0.847	0.818	0.962	0.981
Start of rod drop (s)	32.6	32.6	31.5	31.5	24.9	24.9	24.4	24.4
Time of maximum thermal flux (s)	32.6	32.6	31.5	31.6	24.9	24.9	24.4	24.5
Maximum thermal heat flux (fraction of nominal thermal heat flux[6])	0.115	0.061	0.137	0.076	0.099	0.054	0.159	0.091
Time for max. fuel centreline temperature (s)	33.3	33.6	32.6	33.1	25.7	26.0	25.5	26.1
Max. fuel centreline temperature (°C)	697	697	612	633	661	668	587	609
Time for max. cladding	32.6	32.8	31.5	31.7	24.9	25.1	24.5	24.7

[6] The nominal value of one parameter is its value at full power normal operation.

	BOC				EOC			
Initial state	Hot shutdown		Hot standby		Hot shutdown		Hot standby	
Initial rod insertion	All RCCA fully inserted		Control bank inserted Shutdown bank fully extracted		All RCCA fully inserted		Control bank inserted Shutdown bank fully extracted	
Assumption on pellet-to-cladding heat transfer	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity
temperature (s)								
Max. cladding temperature (°C)	368	336	354	330	361	333	347	328
Time for min DNBR (s)	32.6	32.8	31.5	31.7	24.9	25.0	24.5	24.7
Min. DNBR	7.2	15.9	7.5	15.2	9.9	20.7	10.4	20.2

SECTION 14.3.10 - TABLE 2 (2/2)

Results for EPR 4250 MWth [Ref-1]

Withdrawal of Shutdown Bank:

	BOC				EOC	
Initial state	Hot shutdown		Hot standby		Hot shutdown	
Initial rod insertion	All RCCA fully inserted		Shutdown bank inserted Control bank fully extracted		All RCCA fully inserted	
Assumption on pellet-to-cladding heat transfer	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity	High gap heat conductivity	Low gap heat conductivity
Time of nuclear flux peak (s)	27.6	27.6	20.4	20.4	20.1	20.1
Max Pnuc (fraction of NP[6])	1.675	1.485	1.340	1.234	1.841	1.642
Start of rod drop (s)	28.2	28.2	21.0	21.0	20.7	20.7
Time of maximum thermal flux (s)	27.6	28.2	21.0	21.1	20.1	20.1
Maximum thermal heat flux (frac. nom. [6])	0.146	0.070	0.173	0.098	0.15	0.06
Time for max. fuel centre line temperature (s)	28.9	29.2	22.0	22.5	21.6	21.9
Max. fuel centre line temperature (°C)	836	816	664	686	841	825
Time for max. cladding temperature (s)	28.2	28.3	21.1	21.3	20.9	21.0
Max. cladding temperature (°C)	392	347	360	333	383	344
Time for min DNBR (s)	28.2	28.3	21.1	21.3	20.8	21.0
Min. DNBR	5.7	14.9	6.8	13.9	9.5	19.8

SECTION 14.3.10 - TABLE 3 (1/2)

Uncontrolled RCCA Bank Withdrawal – Hot Shutdown / Hot Standby – Deviation
4500 MWth / 4250 MWth

NEUTRONIC DATA:

	Penalty	Uncertainty	4250 MW	4500 MW	Deviation (negative if less penalty)	
Moderator temperature coefficient	Min (pcm /°C)	-3.6 pcm/°C	-5.4	-6.4	-1	BOC
			-38	-38.7	-0.7	EOC
Doppler coefficient	Min (pcm /°C)	20 %	-2.2	-2.2	0	BOC
			-2.6	-2.6	0	EOC
β_{eff}	Max (pcm)	5%	754	754	0	BOC
			559	560	+1	EOC
Initial shutdown margin	Min (pcm)	500 pcm	-2000	-3400	-1400	

SECTION 14.3.10 - TABLE 3 (2/2)

Uncontrolled RCCA Bank Withdrawal – Hot Shutdown/ Hot Standby – Deviation 4500MWth / 4250 MWth

Efficiency of the RCCA Banks (given in pcm/cm) with 10% Uncertainty:

		Hot Shutdown		Hot Standby	
		Withdrawal of control bank (Shutdown bank completely inserted)	Withdrawal of control bank (Control bank completely inserted)	Withdrawal of control bank (Shutdown bank completely inserted)	Withdrawal of control bank (Control bank completely inserted)
BOC	4250 MW	87	110.2	40.2	60.8
	4500 MW	85	101	44	56.8
	Deviation	**-2**	**-9.2**	**+3.9**	**-4**
EOC	4250 MW	118	151.9	58.9	N/A
	4500 MW	119.9	160.6	58	N/A
	Deviation	**+ 1.9**	**+ 8.7**	**- 0.9**	**-**

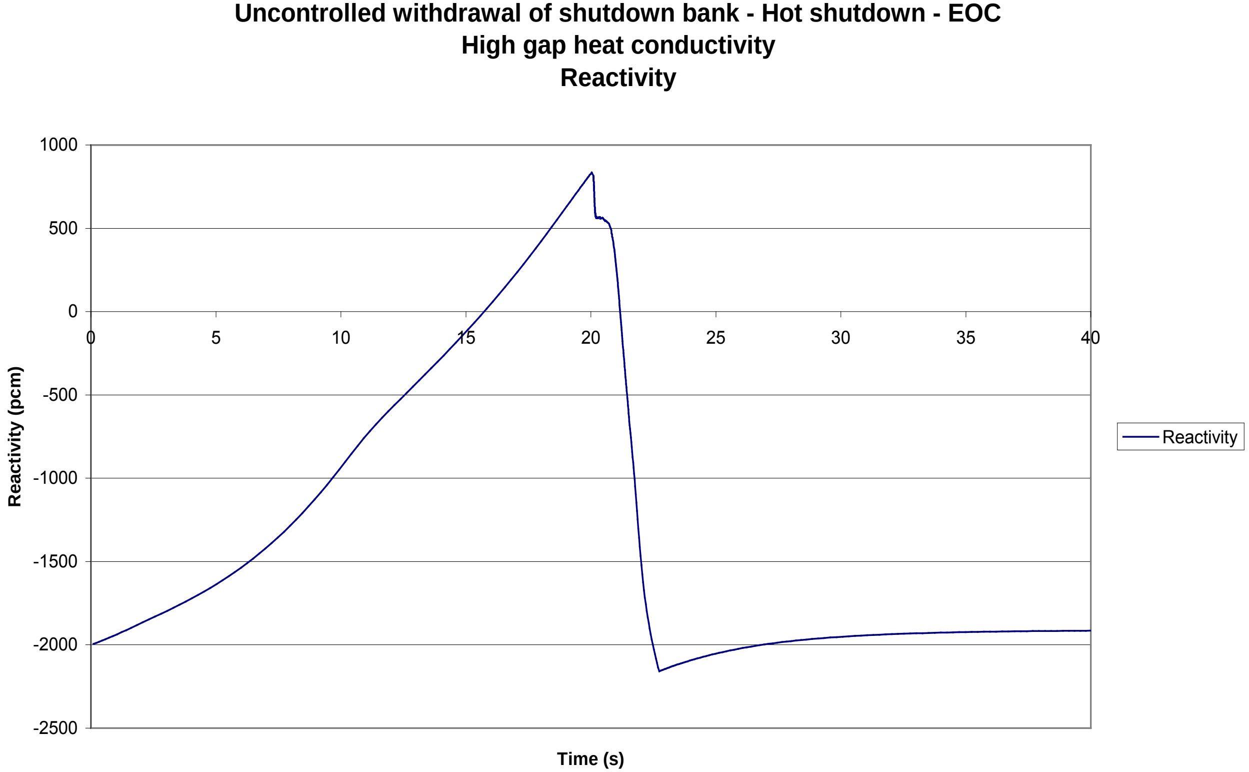
SECTION 14.3.10 – FIGURE 1

89 RCCA Pattern
(control rods: P, shutdown rods: N)

17									N								
16						N		N		N		N					
15			N		P		P		P		P		P		N		
14				N		N						N		N			
13			P		P		P		P		P		P		P		
12		N		N				N		N				N		N	
11			P		P		N				N		P		P		
10		N				N		N		N		N				N	
09	N		P		P				P				P		P		N
08		N				N		N		N		N				N	
07			P		P		N				N		P		P		
06		N		N				N		N				N		N	
05			P		P		P		P		P		P		P		
04				N		N						N		N			
03			N		P		P		P		P		P		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

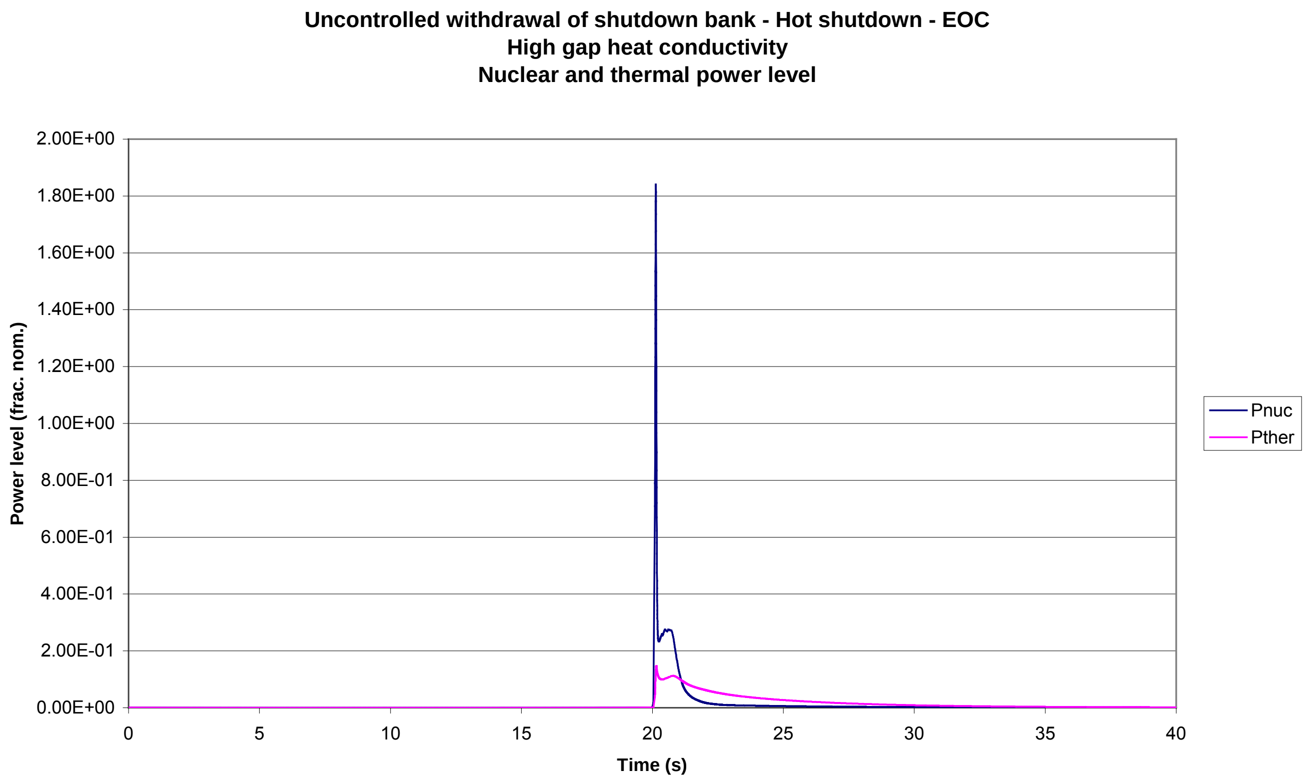
SECTION 14.3.10 - FIGURE 2 (1/6)

Results for EPR 4250 MWth [Ref-1]



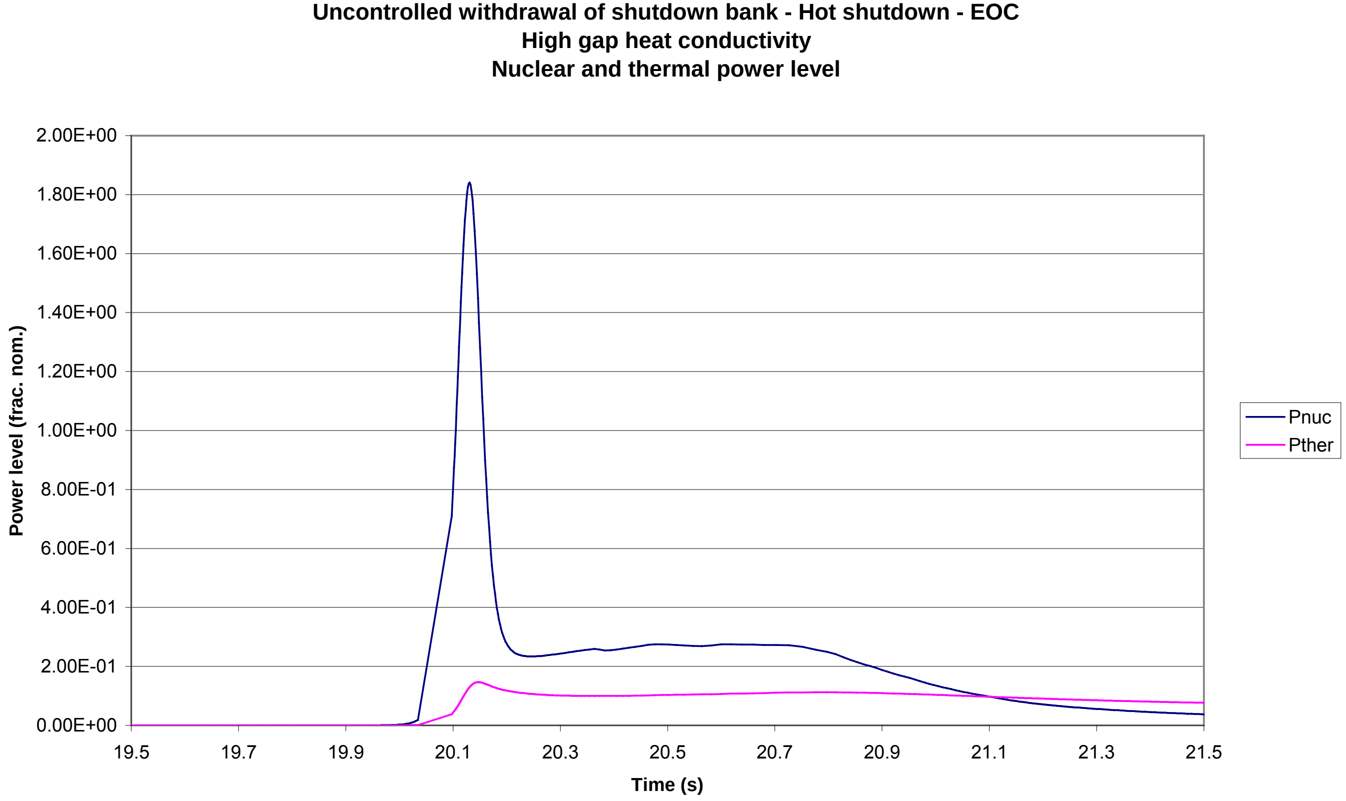
SECTION 14.3.10 - FIGURE 2 (2/6)

Results for EPR 4250 MWth [Ref-1]



SECTION 14.3.10 - FIGURE 2 (3/6)

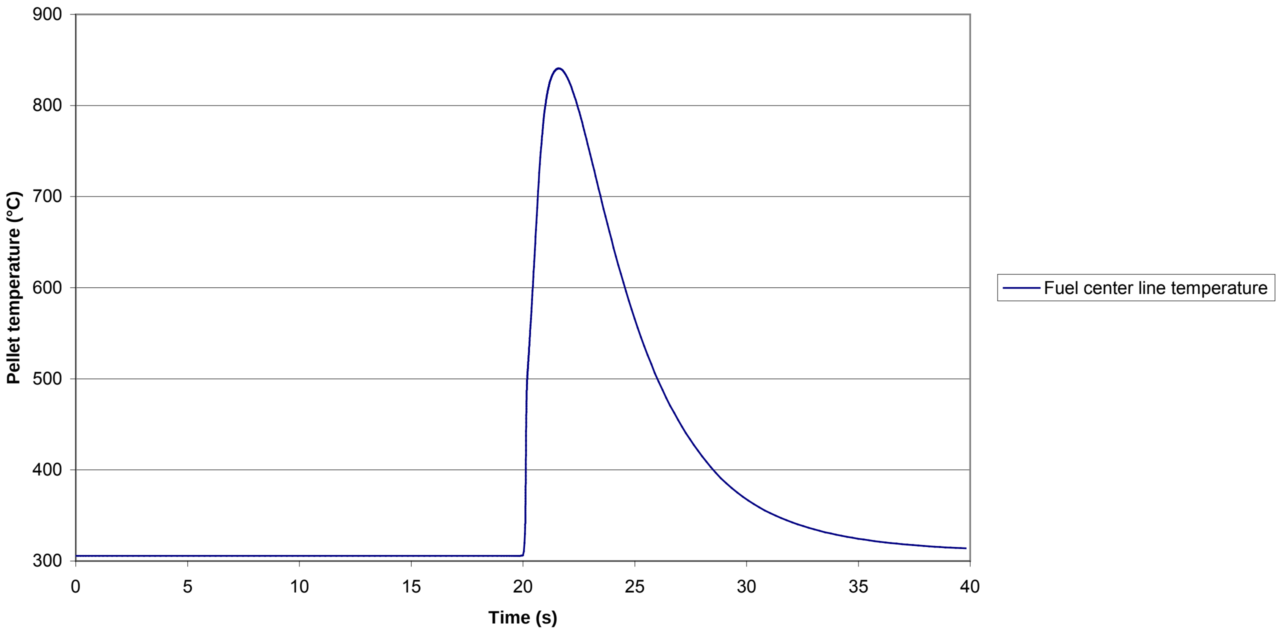
Results for EPR 4250 MWth [Ref-1]



SECTION 14.3.10 - FIGURE 2 (4/6)

Results for EPR 4250 MWth [Ref-1]

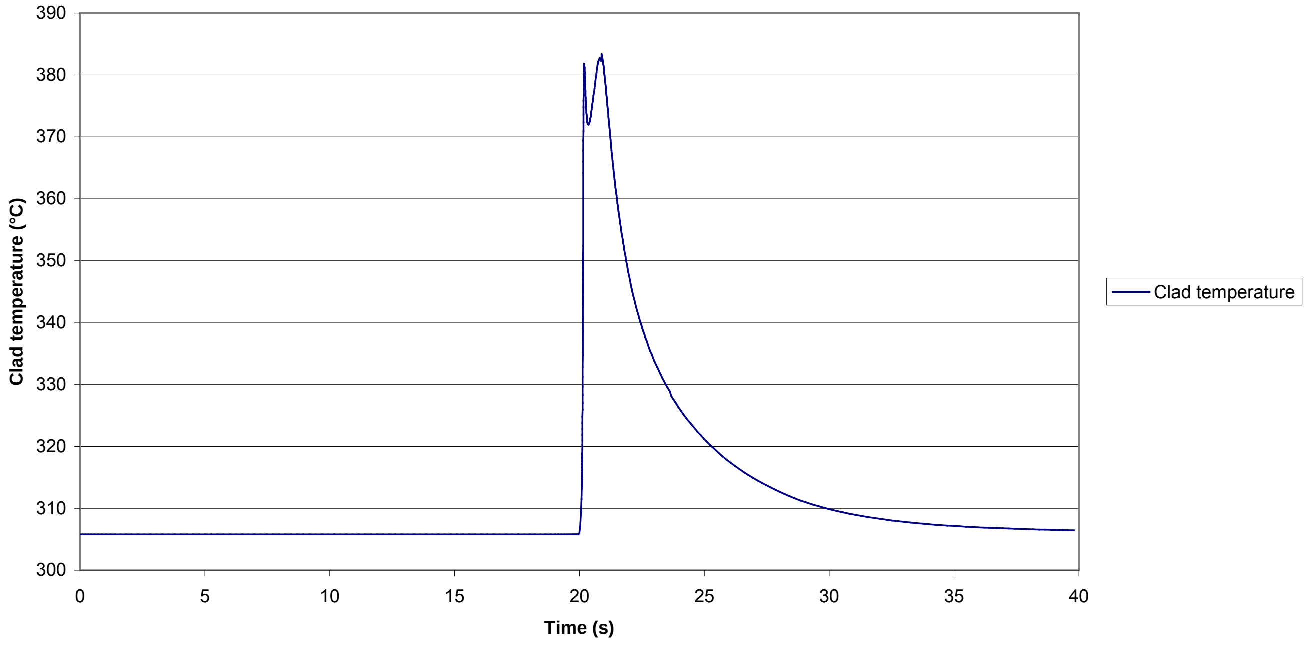
Uncontrolled withdrawal of shutdown bank - Hot shutdown - EOC
High gap heat conductivity
Fuel center line temperature



SECTION 14.3.10 - FIGURE 2 (5/6)

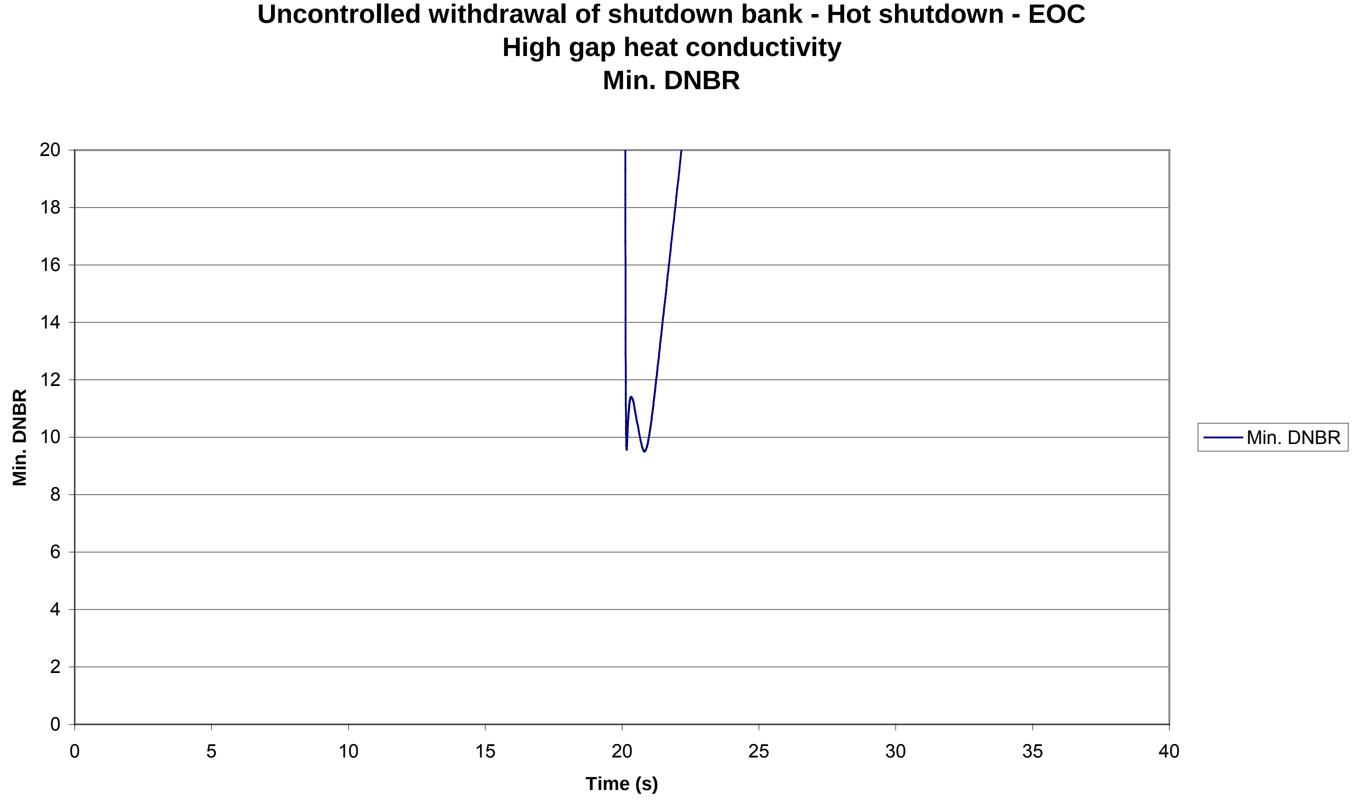
Results for EPR 4250 MWth [Ref-1]

Uncontrolled withdrawal of shutdown bank - Hot shutdown - EOC
High gap heat conductivity
Clad temperature



SECTION 14.3.10 - FIGURE 2 (6/6)

Results for EPR 4250 MWth [Ref-1]



11. RCCA MISALIGNMENT UP TO ROD DROP, WITHOUT CONTROL SYSTEM ACTION

11.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

11.1.1. Definition, Causes and Description of the Transient

Rod Cluster Control Assemblies (RCCA) are always moved in pre-selected banks. Each of these banks consists of several RCCA. The dropping of one RCCA may be caused either by a fault in the control rod drive mechanism or by a failure of the electrical power supply to the RCCA lift coils. The dropping of one sub-bank may be caused by a failure of the electrical power supply system for a sub-bank.

The layout of the 89 RCCA is shown in Section 14.3.11 - Figure 1.

The typical sequence of events following a rod drop is described below.

11.1.1.1. From the Initiating Event to the Controlled State

A drop of one or more RCCAs into the core is characterised by a negative reactivity insertion and a subsequent drop in core power. The core power decreases and the subsequent primary-secondary power mismatch leads to a thermal-hydraulic transient governed by the reactivity feedback effects and the temperature control logic. Normal control of the turbine acts to maintain a constant load. Thus, the core power increases to a new primary-secondary equilibrium. The combination of this power increase and a distorted power distribution, caused by the presence of dropped rods, may result in Departure from Nucleate Boiling (DNB) if the core is not adequately protected.

The protection against DNB is achieved by the low DNBR channel, which calculates the Departure from Nucleate Boiling Ratio (DNBR) on-line and through a direct reactor trip actuation if two or more rods are dropped. To carry out its protection function, the on-line DNBR must be representative of the actual DNBR value during the rod drop transient. Due to the distorted power distribution in this situation, and considering the Single Failure Criterion, the on-line DNBR can become unrepresentative and this must be considered when basing protection claims on the low DNBR channel.

After reactor trip, actuated by the low DNBR channel which is F1A classified, the controlled state is:

- Power = 0% full nuclear power
- 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.

The shutdown margin calculations, discussed in section 5 of Sub-chapter 4.3, maintain core subcriticality in this state. This transient is similar to that following a loss of condenser vacuum discussed in section 5 of this sub-chapter.

11.1.1.2. From the Controlled State to the Safe Shutdown State

The safe shutdown state has the following conditions:

- Power = 0% full nuclear power
- Temperature = 180°C
- Pressure = 30 bar
- Boron concentration is sufficient to keep the core subcritical after the xenon depletion
- Residual heat is removed by the steam generators or the SIS-RHR system
- All RCCA fully inserted.

The main actions to be performed to reach the safe shutdown state are:

- 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 rod drop transient is bounded for the activity release by the loss of condenser vacuum discussed in section 5 of this sub-chapter.

The rod drop transient is bounded for the cooling capability by the feedwater line break in state A discussed in section 3 of Sub-chapter 14.5 since the four steam generators remain available.

The rod drop transient is bounded for core subcriticality by the RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant (states A to E) discussed in section 13 of this sub-chapter.

The sequence of actions necessary to reach the safe shutdown state is given in detail in section 3 of Sub-chapter 14.5.

The analysis below considers the transient from the initiating event to the reactor trip.

11.1.2. Safety and Decoupling Criteria

Dropping one, two, or three RCCA from the same sub-bank, and dropping one sub-bank, are classified as PCC-2 events. The safety criteria are the radiological limits for normal operation.

The decoupling criterion is no Departure from Nucleate Boiling (DNB).

11.1.3. Reactor Protection System Actions

If needed, the low DNBR protection function actuates a reactor trip that protects the fuel against DNB during rod drop transients. The low DNBR channel uses on-line DNBR calculations, rod position measurements, and SPND imbalance. The on-line DNBR calculations are based on:

- Power density distribution of the hot channel derived directly from the neutronic incore instrumentation, the Self Powered Neutron Detectors (SPND). The signals from the incore detectors provide the integrated power along the hot channel using a polynomial function (See Sub-chapter 4.4).
- Inlet temperature: derived from the cold leg temperature sensors
- Pressure: derived from the primary pressure sensor
- Relative core flow: derived from the reactor coolant pump speed sensors.

As 12 fuel assemblies are instrumented, the core is divided into 12 radial zones, each one being surveyed by one SPND finger as shown in Section 14.3.11 - Figure 2. These provide the on-line DNBR values. The axial location of the six SPNDs in a guide tube is shown in Section 14.3.11 - Figure 3.

The on-line DNBR channel follows the global architecture of the protection system with four divisions and a two out of four vote downstream. Each division uses all 12 on-line DNBR values (see Section 14.3.11 - Figure 4). In order to avoid spurious actuation of the reactor trip if a single SPND fails during normal operation, the basic reactor trip actuation is based on the second lowest value of the on-line DNBR.

The RCCA position rate of change measurement protection channel is described in Section 14.3.11 - Figure 4 This system follows the global architecture of the protection system for reactor trip activation, with four divisions and a two out of four vote downstream. Each division is supplied with RCCA position measurements from a different quadrant of the core.

In addition, the RCCA position measurement rate of change supplies a downstream one out of four vote. This enables the protection actuation to be switched from the second lowest to the minimum value of the on-line DNBR, and to compare it to a specific setpoint to be used with a misaligned or dropped rod (see Section 14.3.11 - Figure 4). This is one of the possible signals that could be used to cover this event and is used in the current analysis. However, other options based on the SPND signals may be considered in the future.

The SPND imbalance calculation (see Section 14.3.11 - Figure 4) follows the global architecture of the protection system with four divisions and a downstream two out of four vote. Each division is fed with all 72 SPND signals. It is based on the power density provided by the SPNDs. The differences in the power density between two symmetrical SPNDs with the same axial level and symmetrical with respect to the centre of the core are computed. The sum of the absolute value of these differences corresponds to the SPND imbalance.

For the rod drop event, the logic of the reactor trip actuation is based on either:

- RCCA position rate of change signal in two out of four divisions
- The comparison of the minimum value of on-line DNBR with the specific imbalance/rod drop setpoint as shown in Section 14.3.11 - Figure 4.

The rod drop event is generally detected by both RCCA position rate of change signal and SPND imbalance, with the following exceptions:

- Single rod drop may be detected only by SPND imbalance assuming a single failure on the corresponding RCCA position measurement.
- Two or four symmetrical rod drops are only detected by RCCA position rate of change measurement, as they do not generate an SPND imbalance.

As the protection against DNB is provided by the low DNBR channel, the objective is to show that the on-line calculated DNBR remains in good agreement with the actual value in the event of a rod drop. Therefore, it is necessary to assess the loss of accuracy of the on-line DNBR due to the use of SPNDs in rod drop situations. This loss of accuracy is included as an uncertainty to be applied when setting the DNBR setpoint.

The actual value of the DNBR is assessed by design calculations.

11.2. METHODS AND ASSUMPTIONS

11.2.1. Single Failure Selection for the Analysis

The single failure can be applied to:

- One individual rod cluster position indicator
- One SPND, which results in the on-line DNBR from the corresponding finger being unavailable. It also results in reduced capability of the SPND imbalance calculation.

11.2.1.1. Single Rod Drop

Assuming that the single failure is applied to one individual rod cluster position indicator, it is reasonable to assume that the lowest on-line DNBR remains available.

The worst case is to assume that the failed SPND belongs to the finger providing the lowest on-line DNBR. Therefore, it is assumed that the on-line DNBR from this finger is unavailable. Consequently, it is necessary for the safety analysis to assess the loss of accuracy of the second lowest value from the on-line DNBR.

The detection of this event is thus performed by RCCA position measurement with the SPND imbalance providing redundant detection.

11.2.1.2. Two Rods Drop

If the failure is applied to one SPND, there is a rod drop detection by the RCCA position measurement with two out of four logic and a direct reactor trip actuation.

The worst case is to apply the failure to one RCCA position measurement. The event is no longer detected by the RCCA position measurement with two out of four logic. It is still detected by the RCCA position measurement with one/four logic. Except for symmetric rod drops, SPND imbalance will provide redundant detection. The lowest on-line DNBR remains available and the loss of accuracy for on-line DNBR has to be assessed for this lowest on-line DNBR.

As a consequence, it is necessary for the safety analysis to assess the loss of accuracy of the second lowest value of on-line DNBR for the drop of a single rod and of the minimum value of on-line DNBR for the drop of two rods.

11.2.1.3. Three or More Rods Drop

Whatever single failure is applied, the dropping of three of more control rods is detected by the RCCA position measurement with two out of four logic. Thus, three and four rods dropping are not considered for specific on-line DNBR analysis.

The logic of the approach for a direct reactor trip is to avoid too high a setpoint for the on-line DNBR shown in Section 14.3.11 - Figure 4. The requirement for this direct actuation could be removed by implementation of a second, higher, setpoint for the on-line DNBR if three or more rods are dropped.

11.2.2. Method of Analysis

11.2.2.1. Calculation of on-line DNBR in Rod Drop 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 (see Section 14.3.11 – Figure 2). 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, described in Appendix 14A, is used to determine the absorption rate density in each of the 72 SPNDs.

For the analysis, the calibration coefficients are derived from three-dimensional SMART calculations at nominal power with all rods out. Each SPND is calibrated on the fuel rod with the maximum nuclear enthalpy rise of the zone that it surveys as shown in Section 14.3.11 - Figure 2, as follows:

$$\text{Calibration coefficient} = \frac{\begin{bmatrix}\text{Integrated power at the SPND height in the fuel rod} \\ \text{of the zone with maximum nuclear enthalpy rise}\end{bmatrix}}{\left[\text{Absorption rate density of the SPND}\right]}$$

11.2.2.2. Calculation of the Design DNBR for rod drop events

The design DNBR value for rod drop events is determined by the FLICA code, described in Appendix 14A.

The three-dimensional SMART code, described in Appendix 14A, is used to calculate the axial and radial power distributions, which are used as an input for the FLICA code.

11.2.2.3. Loss of Accuracy

One on-line DNBR value per zone, and thus 12 on-line DNBR values for the core, is provided as shown in Section 14.3.11 - Figure 2.

For single rod drop events, the loss of accuracy in the on-line DNBR is obtained by comparing the second lowest value of on-line DNBR of the 12 values for the core to the actual DNBR.

$$\text{Loss of accuracy} = \frac{\text{2nd lowest real time DNBR} - \text{actual DNBR}}{\text{actual DNBR}}$$

For two rods drop events, the loss of accuracy is obtained by comparing the lowest value of on-line DNBR of the 12 values for the core to the actual DNBR.

$$\text{Loss of accuracy} = \frac{\text{Lowest real time DNBR} - \text{actual DNBR}}{\text{actual DNBR}}$$

11.3. RESULTS AND CONCLUSIONS

Calculations are not performed at this stage of the Pre-Construction Safety Report. It is considered sufficient to use the experience gained from previous EPR studies.

11.3.1. Results for the EPR_{4250}

These studies, described in detail in section 15.2.2P of PSAR 4250. AREVA [Ref-1] have been performed for equilibrium cycles of UO_2 and MOX fuel management, and for the first core cycle. The results are summarised in this section.

All cases of rod drop have been investigated. The main results shown in Section 14.3.11 – Table 1 are presented below:

- For a single rod drop, the distortion of the core is limited in most cases. Only a few rod drops lead to relatively high losses of accuracy, with a maximum value of 4.3%. However, such rod drops also cause high values of SPND imbalance and can therefore be easily detected.
- For two or four rods dropping, the distortion of the core is bigger and leads to a higher loss of accuracy.
- For three rods dropping, very high values of loss of accuracy appear. Taking these values into account would produce too high a value for the low DNBR setpoint with imbalance and rod drop. A direct reactor trip is thus required via another parameter.

For all the cases of one or two rod drops studied, the loss of accuracy is less than 9%. Consequently, to remain effective during a rod drop event, the low DNBR setpoint value has to take into account of this loss of accuracy. As the SPND imbalance treatment is also used for single RCCA withdrawal at power as discussed in section 13 of Sub-chapter 14.4, the final setpoint value will be assessed by the highest loss of accuracy found in those accidents.

The high setpoint value for SPND imbalance is set to 300 W/cm..

With these provisions, the protection of the core is provided for a rod drop event.

11.3.2. EPR$_{4500}$ Conclusions

Calculations are not performed at this stage of the Pre-Construction Safety Report. It is considered sufficient to use the experience gained from previous EPR studies.

Given the large set of cases studied, at this stage it can be concluded that similar provisions will also provide adequate protection for the EPR$_{4500}$.

A complete calculation of the loss of accuracy and subsequent setting of the low DNBR and SPND imbalance setpoints will be provided prior to operation of the plant.

11.3.3. Systems Sizing

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

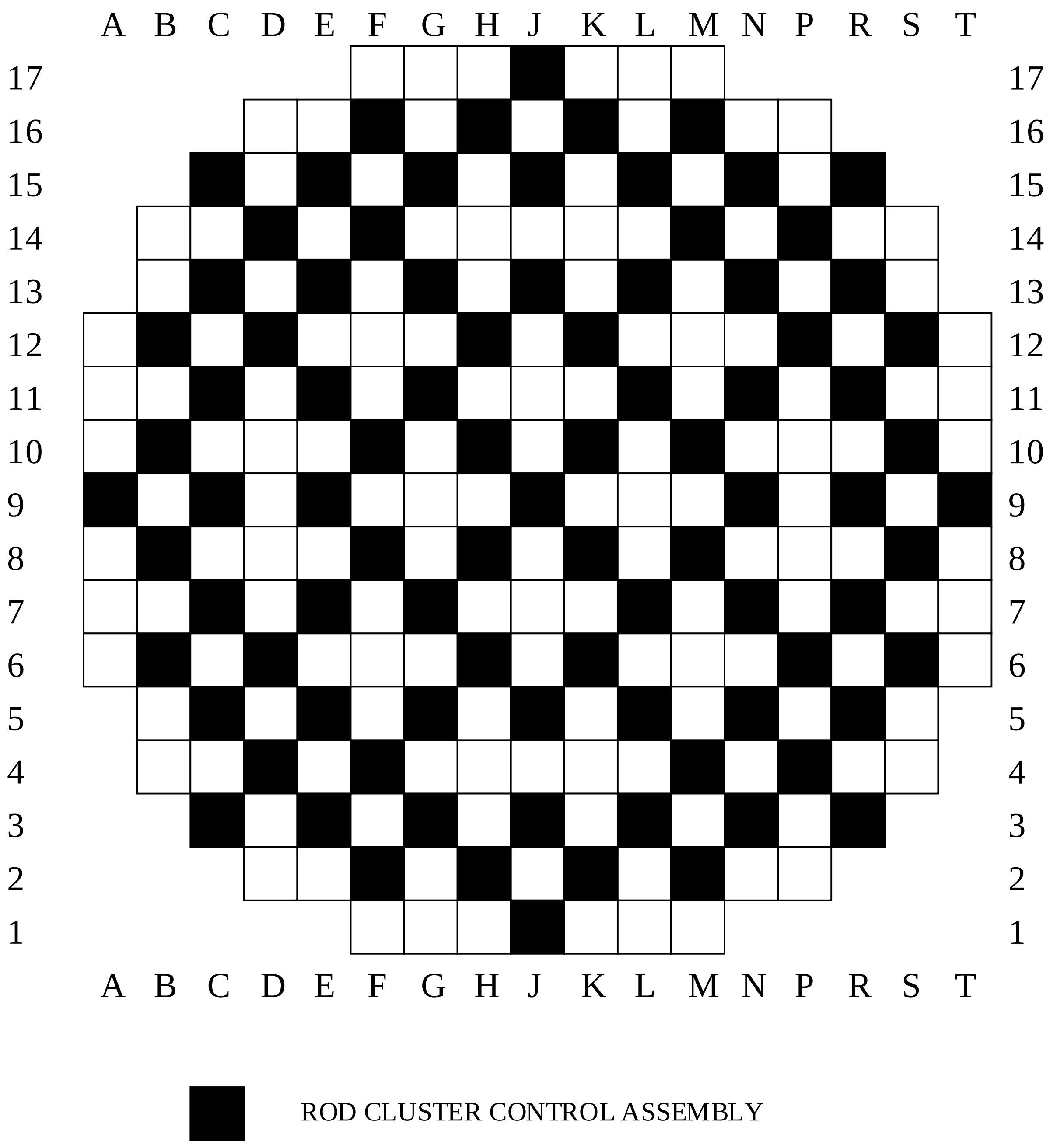
SECTION 14.3.11 - TABLE 1

Maximum Loss of Accuracy in Rod Drop Conditions
EPR 4250 MW [Ref-1]

Number of Rods Dropped	Dropped Rods Location	Fuel Management	Burnup	Loss of Accuracy 1st on-line DNBR (%)	Loss of Accuracy 2nd on-line DNBR (%)	SPND Imbalance (W/cm)
1	F8	UO_2 – INOUT – 18 month	Equilibrium, BOC	0.4	4.3	987
1	H2	First core	Cycle 1, EOC	2.0	2.2	644
2	G5, L13	UO_2 – INOUT – 18 month	Equilibrium, BOC	9.0	9.0	0
2	G5, L13	UO_2 – OUTIN – 18 month	Equilibrium, EOC	5.8	5.8	0
3	B8, K2, H16	First core	Cycle 1, BOC	4.4	15.7	1599
3	G5, N7, L13	UO_2 – OUTIN – 18 month	Equilibrium, EOC	4.8	5.5	334
4	H2, B8, K16, B10	UO_2 – OUTIN – 18 month	Equilibrium, BOC	5.7	5.7	0
4	H2, B8, K16, B10	First core	Cycle 1, EOC	4.7	4.7	0

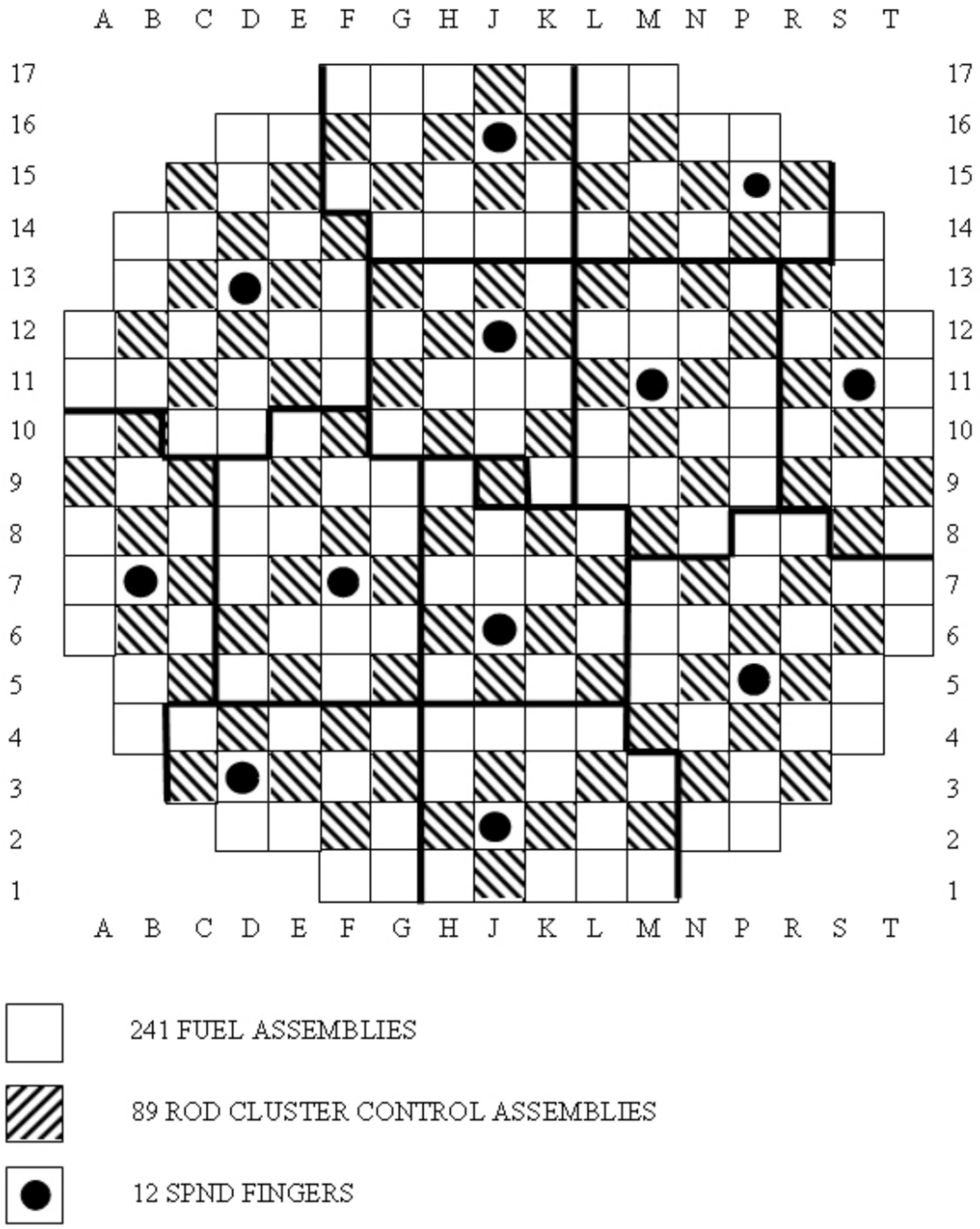
SECTION 14.3.11 - FIGURE 1

89 RCCA Pattern



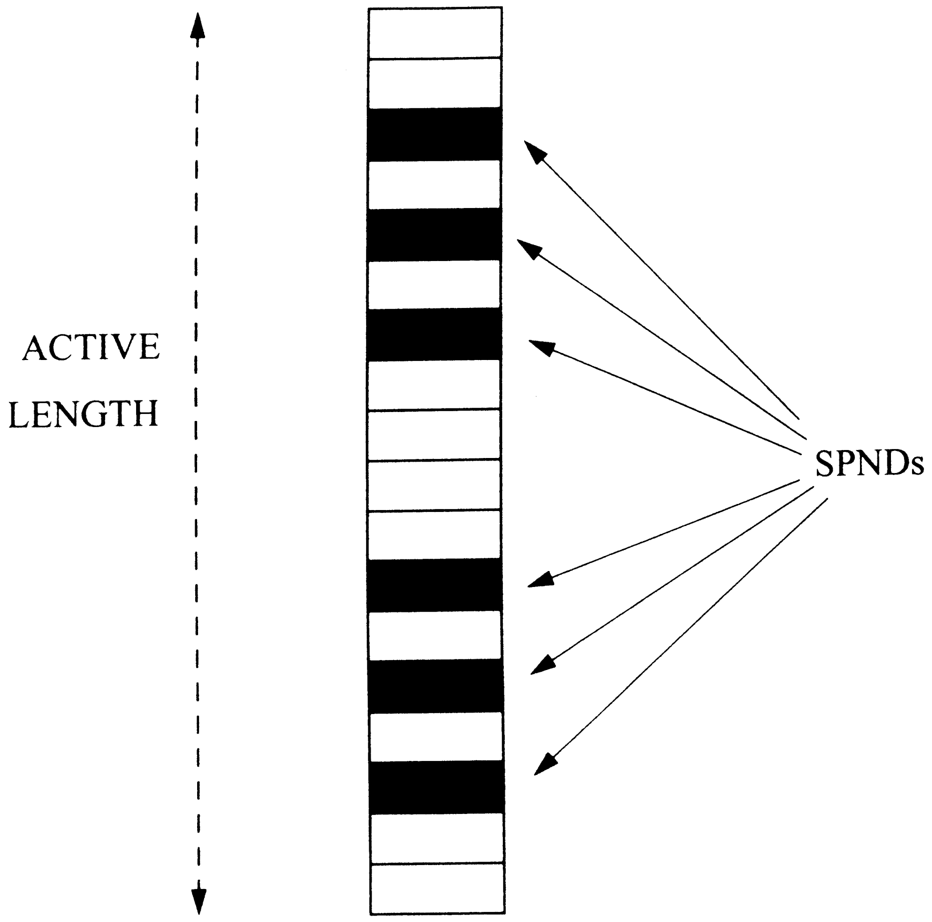
SECTION 14.3.11 - FIGURE 2

Radial Location of SPND Fingers and Radial Zones



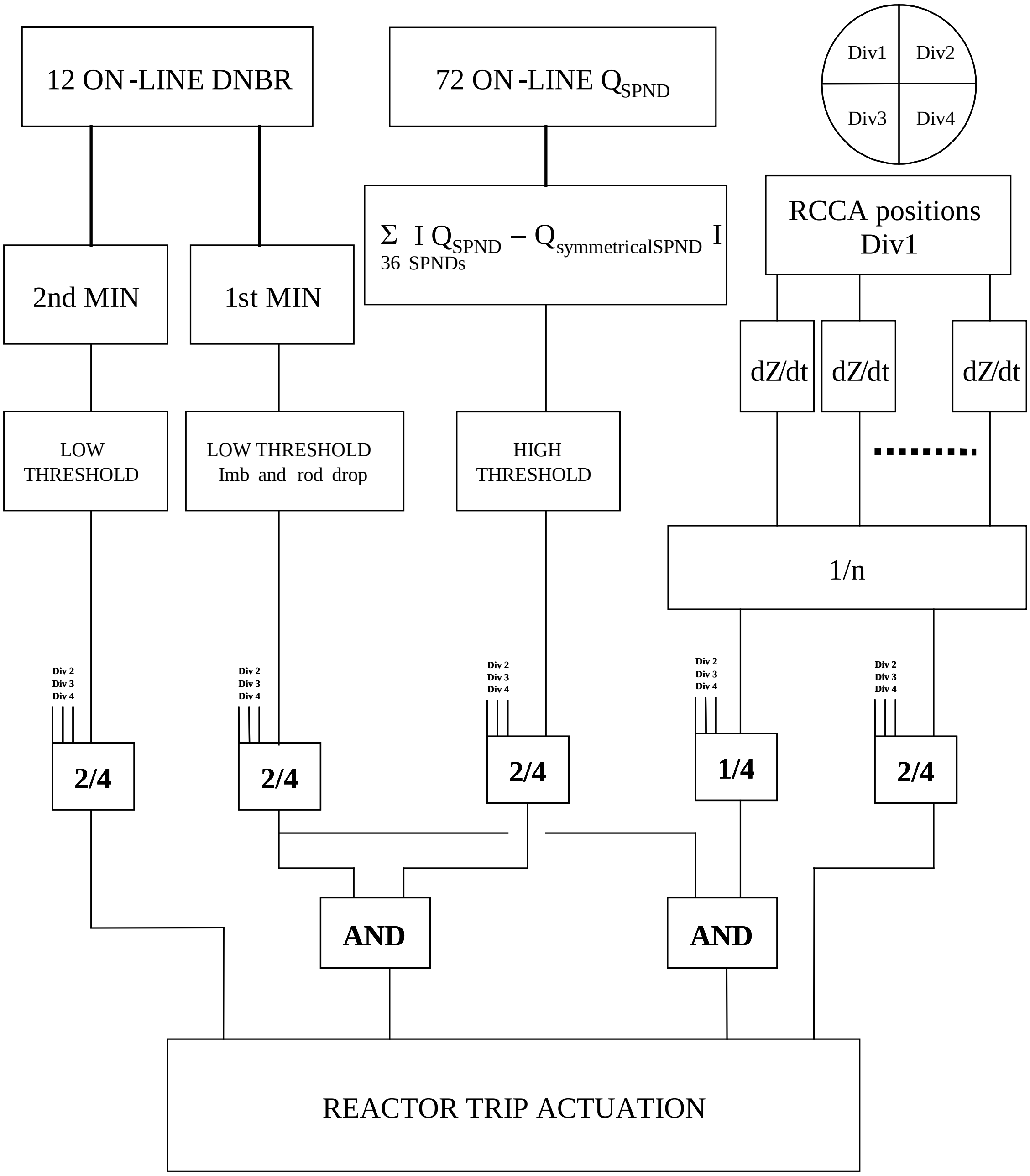
SECTION 14.3.11 - FIGURE 3

Axial Location of the SPND in a Fuel Assembly



SECTION 14.3.11 - FIGURE 4

Logic of Low DNBR Reactor Trip Actuation for Rod Drop



12. START-UP OF AN INACTIVE REACTOR COOLANT LOOP AT AN INCORRECT TEMPERATURE

12.1. ACCIDENT DESCRIPTION

The start-up of an inactive reactor coolant loop at an incorrect temperature is classified as a PCC-2 event.

The transient for the start-up of an inactive reactor coolant pump at an incorrect coolant temperature is not analysed, because protection against unacceptable reactor power transients is provided by the design of the automatic reactor trip function. In general, mitigation of this type of event aims at avoiding automatic reactor trips.

12.2. SYSTEM SIZING

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

13. RCV [CVCS] MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT

13.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

13.1.1. Definition, Causes, and Description of the Accident

The accident is assumed to occur as a result of a failure in the Reactor Boron and Water Makeup System (REA [RBWMS]) or in the Chemical and Volume Control System (RCV [CVCS]). This accident is classified as a PCC-2 event.

The initial plant states for this type of accident can be:

- Power operation (state A)
- Standard hot or cold shutdown (state A, B or C)
- Cold shutdown for refuelling or maintenance, including:
    - Mid-loop operation level – ¾ loop - with closed vessel (state C)
    - Mid-loop operation level – ¾ loop - with open vessel (state D)
    - RPV flange level operation with open vessel (state D)
    - Reactor cavity flooded for refuelling (state E).

Injecting water into the Reactor Coolant System (RCP [RCS]) can increase the core reactivity.

For each initial plant state, a typical sequence of events leading from an initiating event to the controlled state is described below. A short description of possible initiating events is also presented.

13.1.1.1. Power Operation

The uncontrolled boron dilution causes a reactivity insertion which is balanced by a control rod insertion under automatic control, and can potentially lead to a power and temperature rise under manual control.

The progress of the transient could lead to:

- A Departure from Nucleate Boiling (DNB)
- A potential loss of shutdown margin due to RCCA bank insertion during the dilution which prevents the controlled state and the safe shutdown state from being reached.

During this phase of the transient, automatic protection would be initiated by either the shutdown margin LCO function (F2), or the limitation function (F2), or the reactor power LCO and limitation functions (F2). However, the safety analysis is performed without claiming these F2 classified channels.

The F1A anti-dilution in power conditions protection channel will automatically isolate sources of the spurious dilution, terminating the dilution, and the core remains at power.

If the malfunction that caused the spurious dilution can be easily corrected, the operating procedures allow the operator to restore the normal conditions and to keep the reactor at power. If the malfunction cannot be easily corrected, the operator applies normal operating procedures or initiates a manual reactor trip (F1A) to reach the controlled state. The operator actions are not claimed until 30 minutes after the anti-dilution in power conditions protection channel actuation.

The controlled state, defined in Sub-chapter 14.0, is reached when the dilution source is isolated and the following conditions are met:

- Core is subcritical
- Xenon level is equal to the initial xenon level
- Boron concentration is below the initial boron concentration
- RCCAs are inserted
- Core coolant temperature is at Hot Zero Power (HZP) conditions.

The single failure is assuming with the highest worth rod stuck above the core.

The anti-dilution in power conditions protection channel, described in Section 14.3.13 – Figure 1 initiates RCV [CVCS] isolation as soon as the derived boron concentration in the reactor coolant is below the setpoint value. This setpoint value corresponds to the critical boron concentration for the core at HZP with all the rods in, except the highest worth rod which is stuck above the core (ARI-1), without xenon. The setpoint, which includes calculation uncertainties, ensures the available shutdown margin from the RCCA banks is sufficient to take the core subcritical at HZP in all cases, irrespective of the xenon level.

If a reactor trip occurs before the actuation of this protection channel, the protection will be provided by the anti-dilution in standard shutdown states conditions protection channel.

13.1.1.2. Standard Hot or Cold Shutdown

For all shutdown conditions the RCCA are fully inserted. The uncontrolled boron dilution causes a reactivity insertion that would lead to an unintentional criticality if no protection action occurs.

The progress of the transient could lead to transitory states, at power, where the thermal-hydraulic and reactivity conditions would not be under control.

During the phase from the initiating event to the protection channel actuation, automatic protection could be initiated by Non Classified (NC) or F2 functions, but the safety analysis is performed without claiming these functions.

The shutdown high neutron flux alarm (a F1A classified signal) is actuated before reaching criticality. The operators are informed but, based on the rules for operator actions described in section 2.5 of Sub-chapter 14.0, the safety assessment does not claim manual actions for 30 minutes after the first significant information is available to the operator.

The anti-dilution in standard shutdown states conditions protection channel initiates RCV [CVCS] isolation before reaching criticality as soon as the derived boron concentration in the reactor coolant falls below the setpoint. The setting of this setpoint for the critical boron concentration of the assumed shutdown condition, depending on the core temperature, including calculation uncertainties ensures the core remains subcritical. This covers all RCV [CVCS] dilution rates, including the most conservative flow of 26 kg/s (94 te/h) discussed in sub-section 13.1.2 of this sub-chapter.

The core remains subcritical throughout the transient.

The controlled state, defined in Sub-chapter 14.0, is reached when the dilution source is isolated and the following conditions are met:

- Initial thermal and hydraulic conditions are not modified
- Core is subcritical
- Boron concentration is below the initial boron concentration
- RCCAs are inserted.

13.1.1.3. Cold Shutdown for Refuelling or Maintenance

For these cases, the initial boron concentrations are higher than those in normal cold shutdown, the reactor coolant pumps are stopped and the RCP [RCS] level can be at mid-loop or at RPV flange level.

The worst condition is mid-loop operation where the coolant inventory is low. During this scenario the RCV [CVCS] and REA [RBWMS] are operating to provide mid-loop level control.

A short time after the dilution begins, before reaching criticality, the anti-dilution in shutdown conditions with reactor coolant pumps not in operation protection channel initiates RCV [CVCS] isolation. This occurs as soon as the boron concentration in the charging line is below the setpoint value. This setpoint for the boron concentration required under maintenance outage and refuelling outage conditions, minus built-in margins to avoid spurious actuations ensures the core remains subcritical. The setpoint is sufficient to protect the core irrespective of the RCV [CVCS] dilution rate including the maximum flow of 26 kg/s (94 te/h) discussed in sub-section 13.1.2 of this sub-chapter.

The core remains subcritical throughout transient.

The controlled state, defined in Sub-chapter 14.0, is reached when the dilution source is isolated and the following conditions are met:

- Initial thermal and hydraulic conditions are not modified
- Core is subcritical

- Boron concentration is below the initial boron concentration
- RCCAs are inserted.

Transient analyses in state D bound those in state E.

13.1.2. Causes, Initiating Events

The main dilution scenarios related to a ’RCV [CVCS] malfunction that result in a decrease in boron concentration in the reactor coolant’ event (PCC-2) are the following:

- Malfunction in the coolant degasification system
- Malfunction in the coolant purification system
- Malfunction in the coolant storage and treatment system
- Malfunction in the reactor boron and water makeup system.

These systems are described in Chapter 9 of the PCSR, and a diagram of the RCV [CVCS] and related systems is included.

A short description of the dilution scenarios is given below.

Malfunction in the Coolant Degasification System

During standby, the evaporator and the degasifier column contain coolant which originates from the last operating period and has a boron concentration that can be almost 0 ppm. If the degasifier is put in service without discharging this coolant to the Coolant Storage Tanks, an unwanted injection of unborated water into the RCP [RCS] can occur.

The unborated water can be delivered to the RCP [RCS] with a maximum flow rate of 20 kg/s as determined by the RCV [CVCS] capacity.

The reduction in the boron concentration of the reactor coolant is limited by the volume of the evaporator and degasifier column.

Malfunction in the Coolant Purification System

An injection of unborated coolant flow into the RCP [RCS] via the RCV [CVCS] can occur if an insufficiently borated demineraliser, which was on standby, is put into service. This is possible under all RCP [RCS] operating states.

A limited amount of boron can be removed from the reactor coolant.

The unborated water can be delivered to the RCP [RCS] with a maximum flow rate of 20 kg/s as determined by the RCV [CVCS] capacity.

Malfunction in the Coolant Storage and Treatment System

Malfunction of the boron concentration measurements can result in too low a boron concentration of the boric acid produced in the evaporator column that is to be reused for injection into the RCP [RCS]. Under the most pessimistic assumptions of a long operation without detection of the failure, a very low boron concentration in the product occurs. For an initially empty boric acid tank this can lead to a dilution of the reactor coolant when an injection of boric acid is required. This can occur for the following reasons:

- Leakage make up to RCP [RCS]
- Rod position control
- Boration of the RCP [RCS] to reach a shutdown state.

The injection rate of the almost boron-free coolant that should contain 7000 ppm boron is limited by the delivery rate of the two boric acid pumps of 6 kg/s each.

Malfunction in the Reactor Boron and Water Makeup System (REA [RBWMS])

Demineralised water may be injected in error via the RCV [CVCS] into the RCP [RCS] with one or both of the demineralised water pumps of the REA [RBWMS] when either no changes of the reactivity are required or the corresponding boric acid injection is missing. Injection of the demineralised water can be initiated by one of the following:

- Leakage makeup to RCP [RCS]
- Rod position control
- Xenon compensation
- Load variations
- Manual command.

This event is possible in all RCP [RCS] operating states.

The maximum injection rate is determined by the capacity of the demineralised water pumps of 13 kg/s each. The maximum dilution flow rate is thus 2 x 13 kg/s or 26 kg/s which is a bounding value with two demineralised water pumps running.

This scenario is a bounding one for the safety analyses of the ‘RCV [CVCS] malfunction that result in a decrease in boron concentration in the reactor coolant’ (PCC-2) event.

13.1.3. Decoupling Criteria

The boron dilution due to a RCV [CVCS] malfunction is classified as a PCC-2 event.

The safety criteria are the radiological limits for PCC-2 events defined in section 2.1 of Sub-chapter 14.0. The consequences of an uncontrolled boron dilution event are analysed against the following decoupling criteria:

- Fuel cladding integrity

- Reactor Coolant Pressure Boundary (RCPB) integrity.

With regard to the RCPB criteria, the uncontrolled boron dilution analysis is not limiting.

For power operation, the following decoupling criteria ensure the integrity of the fuel cladding:

- No departure from nucleate boiling with the DNBR value higher than the criterion
- No high linear power density with the linear power density lower than the criterion.

For all the shutdown states, these criteria are met by ensuring the core remains subcritical throughout the transient.

13.1.4. Reactor Protection System

To cover the entire range of standard reactor states, three protection channels are defined according to the reactor status (see Section 14.3.13 - Figures 1 to 3):

- Anti-dilution in shutdown conditions with reactor coolant pumps not in operation protection channel
- Anti-dilution in standard shutdown states conditions protection channel (all rods in (ARI))
- Anti-dilution in power conditions protection channel (not all rods in (not ARI)).

All these protection channels use the measured boron concentration in the charging line as an input signal. This analogue signal is provided by the boron meter system.

Except for the case with reactor coolant pumps not in operation, the detection of spurious dilution relies on an real time derivation of the boron concentration of the reactor coolant based on a boron mass balance, and assuming a conservative primary system volume, using the following inputs:

- The boron concentration in the charging line, provided by the boron meter system
- The total RCV [CVCS] charging flow
- The core (cold leg) temperature (not used in power states).

All these protection channels actuate the automatic closure of two redundant valves downstream of the RCV [CVCS] tank (Volume Control Tank), thereby isolating the main RCV [CVCS] sources of spurious dilution. In addition, the charging pumps then switchover to take suction from the In-containment Refuelling Water Storage Tank (IRWST), although this action is not classified F1A. Operation of these protection channels thus ensure the dilution is terminated once they are actuated.

In shutdown conditions, the protection channels ensure the core remains subcritical throughout the transient.

13.1.4.1. Power Operation

The fuel clad integrity is maintained by a reactor trip that may occur on any of the following protection system channels (F1A), described in section 5 of Sub-chapter 14.1:

- Low DNBR
- High linear power density
- High core power level
- High pressuriser level (> MAX1[1])
- High pressuriser pressure (> MAX2[1]).

The effectiveness of the shutdown in making the core subcritical is ensured by the anti-dilution in power conditions protection channel (F1A) (see Section 14.3.13 - Figure 1).

Within the anti-dilution in power conditions channel, the derived boron concentration is compared to a protection setpoint corresponding to the predetermined critical boron concentration of the core at HZP with all rods in. The derived boron concentration is compared to a protection setpoint corresponding to the predetermined critical boron concentration of the core at HZP with all rods in (except the highest worth rod stuck above the core (ARI-1)) and without xenon, including the calculation uncertainties. A periodic adjustment of the setpoint is required, to account for changes in fuel burnup. This protection action is enabled by a signal representing a reactor power condition, a "Not ARI" permissive signal.

A schematic representation of the anti-dilution in power conditions protection channel is given in Section 14.3.13 - Figure 1.

13.1.4.2. Standard Hot or Cold Shutdown

During standard hot or cold shutdown, protection is provided by the anti-dilution in standard shutdown states conditions protection channel (F1A) (see Section 14.3.13 - Figure 2).

Within the anti-dilution in standard shutdown states conditions protection channel, the derived boron concentration is compared to a temperature-dependent protection setpoint corresponding to the critical boron concentration appropriate to the shutdown state. This depends on the core temperature and is calculated with all rods in (ARI) and without xenon, including calculation uncertainties. A periodic adjustment of the setpoint is required, to account for changes in fuel burnup. This protection action is enabled by a signal representing a reactor shutdown condition, an "ARI" permissive signal, and disabled by a signal that indicates the reactor coolant pumps are shutdown.

A schematic representation of the anti-dilution in standard shutdown states conditions protection channel is given in Section 14.3.13 - Figure 2.

[1] See Sub-chapter 14.1

13.1.4.3. Cold Shutdown for Refuelling or Maintenance

During cold shutdown for refuelling or maintenance conditions, protection is provided by the anti-dilution in shutdown conditions with reactor coolant pumps not in operation protection channel (F1A) (see Section 14.3.13 - Figure 3).

The anti-dilution in shutdown conditions with reactor coolant pumps not in operation protection channel is designed to mitigate the risk of heterogeneous dilution when no reactor coolant pumps are in operation, and provides the protection against homogeneous dilution for all shutdown states where the reactor coolant pumps are not running.

The input signal is the boron concentration in the charging line provided by a boron meter system. The boron concentration measurement is compared with a setpoint corresponding to the boron concentration required under maintenance outage and refuelling outage conditions, minus built-in margins to avoid spurious actuations.

This protection action is enabled by a signal indicating the reactor coolant pumps are shutdown.

A schematic representation of the anti-dilution in shutdown conditions with reactor coolant pumps not in operation protection channel is given in Section 14.3.13 - Figure 3.

13.2. METHODS AND ASSUMPTIONS

13.2.1. Choice of Single Failure and Preventive Maintenance

13.2.1.1. Power Operation

A single failure is applied to the protection system by assuming the highest worth rod is stuck above the core.

To reach the controlled state, only RCV [CVCS] isolation (F1A) actuated by the protection system (F1A) and a manual reactor trip (F1A) are necessary.

Preventive maintenance of other F1A systems does not impact the safety assessment.

13.2.1.2. Shutdown States

No specific single failure has any impact on the safety assessment.

No preventive maintenance has any impact on the safety assessment.

13.2.2. Method of Analysis

The safety analysis primarily consists of a determination of the values for the protection setpoints.

13.2.2.1. General Assumptions

The assumptions for the safety analyses are considered in determining the setpoint value for the protection channel, and in the design of the algorithm in the protection channels that provides on-line boron concentration in the reactor coolant.

Assumptions for the definition of the algorithm

- A conservative volume of borated water in the reactor coolant system, excluding the pressuriser, the surge line, and the dead volume at the top of the reactor vessel, of 336 m^3 is assumed.

- As homogeneous dilution is the concern, it is assumed that the fluid let-down is at the same boron concentration as the reactor coolant.

Assumptions for the determination of the setpoint value

The following assumptions are used, together with those described above for the definition of the algorithm:

- The response delay of the channel (up to isolation actuation) is lower than 66 seconds.

- Isolation of the RCV [CVCS] is completed in 40 seconds

- The analysis uses a maximum assumed water makeup system dilution flow of 26 kg/s as discussed in sub-section 13.1.2 of this sub-chapter

- The RCV [CVCS] piping volume swept by pure water from the IRWST suction to the boron meter location is 1.0 m^3.

The boron concentrations have been calculated by the two-energy-group, three-dimensional nodal diffusion code SMART described in Appendix 14A. The uncertainty in the boron concentration level, U(BC), is 100 ppm for this calculation model

13.2.2.2. Power Operation

For fuel clad integrity, the uncontrolled boron dilution transient is less onerous than the uncontrolled withdrawal of control rod banks at power discussed in section 9 of this sub-chapter. As the dilution transient is a slow phenomenon resulting in a reactivity insertion of less than 2 pcm/s), the DNBR criterion is satisfied by the low DNBR protection channel.

The anti-dilution in power conditions protection channel initiates RCV [CVCS] isolation as soon as the derived boron concentration in the reactor coolant is below the setpoint value. The setting of this setpoint ensures the shutdown margin is sufficient to take the core subcritical at hot zero power. The predetermined critical boron concentration of the core is calculated at hot zero power with the highest worth rod stuck above the core and without xenon, and includes the overall calculation uncertainties.

The protection setpoint is defined by the following relationship:

Setpoint = critical boron concentration (HZP, ARI – 1, XE=0)
+ calculation uncertainty on critical BC: U(BC)
+ derivation uncertainty: U(rec.)
+ temperature uncertainty, impact on HZP condition: U(T)
+ drop in boron concentration during the system response time: ΔBC

13.2.2.3. Standard Hot or Cold Shutdown

The anti-dilution in standard shutdown states conditions protection channel initiates RCV [CVCS] isolation as soon as the derived boron concentration in the reactor coolant is below the value of the setpoint. This setpoint ensures the core remains subcritical irrespective of the RCV [CVCS] dilution, even for the most conservative assumption for RCV [CVCS] dilution flow of 26 kg/s. The critical boron concentration appropriate to the shutdown state is dependant on the core temperature, and includes the overall calculation uncertainties)

The protection setpoint is defined by the following relationship:

Setpoint = critical boron concentration (CZP to HZP, ARI, XE=0)
+ calculation uncertainty on critical BC: U(BC)
+ derivation uncertainty: U(rec.)
+ temperature uncertainty, impact on critical BC: U(T)
+ drop in boron concentration during the system response time: ΔBC

13.2.2.4. Cold Shutdown for Refuelling or Maintenance

The anti-dilution in shutdown conditions with reactor coolant pumps not in operation protection channel initiates RCV [CVCS] isolation as soon as the boron concentration of the charging line is below the setpoint.

This channel is partly devoted to the mitigation of the risk of heterogeneous dilution when no reactor coolant pumps are in operation. The setpoint value is then as close as possible to the required boron concentration in the reactor coolant pump shutdown state. The required boron concentration value for refuelling and maintenance is the IRWST boron concentration. This value is used throughout the cycle, from Beginning Of Life (BOL) to End Of Life (EOL). The minimum value of 'on-site' IRWST boron concentration includes calculation uncertainties, allowances, and provisions.

The protection setpoint is the boron concentration required under maintenance outage and refuelling outage conditions, minus built-in margins to avoid spurious actuations:

Setpoint = minimum value of 'on-site' IRWST boron concentration
- measurement uncertainty (monitored BC)

13.2.3. Initial Conditions

For all the reactor states, the nominal conditions have no impact on the safety assessment.

13.3. RESULTS AND CONCLUSIONS

13.3.1. Power Operation

The dilution transient is equivalent to an uncontrolled withdrawal of RCCA banks at power with a reactivity insertion rate of less than 2 pcm/s. Fuel clad integrity is ensured by the low DNBR protection channel.

During automatic control, the reactivity insertion arising from the dilution transient is balanced by insertion of RCCA banks. During manual control, it is balanced by an increased power level and temperature.

Nominal values of the boron concentrations expected at power conditions of full power, all rods out with equilibrium xenon are given in Section 14.3.13 - Table 1.

The decrease in boron concentration in the reactor coolant due to a RCV [CVCS] malfunction is monitored by the anti-dilution in power conditions protection channel. The actuation of this channel initiates the RCV [CVCS] isolation and maintains sufficient shutdown margin in the RCCA banks to take the core subcritical at hot zero power, even if the highest worth rod is stuck above the core, irrespective of the xenon level.

If a reactor trip on the Low DNBR protection channel occurs before the actuation of this protection channel, the protection will be provided by the anti-dilution in standard shutdown states conditions protection channel. A consequence of this is that the setpoint value of the anti-dilution in standard shutdown states conditions protection channel must take account of a stuck rod. Therefore it is equal to the setpoint of the anti-dilution in power conditions protection channel.

The critical boron concentrations used for the determination of the setpoint value for the protection channel are given in Section 14.3.13 - Table 2.

13.3.2. Standard Hot or Cold Shutdown

At the beginning of the transient, the boron concentration is the required boron concentration for the appropriate shutdown state.

The boron concentrations required in standard hot shutdown states are given in Section 14.3.13 - Table 1 part 1/2. The boron concentrations required in standard cold shutdown states are given in Section 14.3.13 - Table 1.

The decrease of boron concentration in the reactor coolant due to a RCV [CVCS] malfunction is monitored by the anti-dilution in standard shutdown states conditions protection channel. This channel initiates the RCV [CVCS] isolation before criticality is reached and ensures that the core remains subcritical for all RCV [CVCS] dilutions, up to the maximum RCV [CVCS] dilution flow of 26 kg/s at cold shutdown.

The critical boron concentrations used for the determination of the setpoint value of the protection channel are given in Section 14.3.13 - Table 2.

At hot shutdown, the setpoint for the anti-dilution in standard shutdown states conditions protection channel must be the same as the anti-dilution in power conditions protection channel setpoint as discussed above.

13.3.3. Cold Shutdown for Refuelling or Maintenance

At the beginning of the transient the boron concentration is the required boron concentration for the cold shutdown for refuelling and maintenance as shown in Section 14.3.13 - Table 1.

The boron concentration of the charging line is monitored by the anti-dilution in shutdown conditions with reactor coolant pumps not in operation protection channel. As soon as this monitored boron concentration is below the boron concentration required under maintenance outage and refuelling outage conditions, minus built-in margins to avoid spurious actuations, this channel initiates the RCV [CVCS] isolation and ensures the core remains subcritical irrespective of the RCV [CVCS] dilution rate.

13.4. TRANSITION TO THE SAFE SHUTDOWN STATE

The safe shutdown state is defined as a state where the core is subcritical even after xenon decay and where the Low-Head Safety Injection pumps (LHSI) in Residual Heat Removal (SIS-RHR) mode are operational, providing decay heat removal by a closed-loop cooling chain.

The uncontrolled boron dilution transients are bounded by the loss of condenser vacuum event discussed in section 5 of this sub-chapter for any potential activity release for the dilution events starting from power conditions. For the dilution events starting from shutdown states, the core remains subcritical throughout the transient.

The capability for heat removal and ASG [EFWS] tank inventory following, the uncontrolled boron dilution transients are bounded by the feedwater line break transient discussed in section 3 of Sub-chapter 14.5.

Boration is needed for any of the PCC-2 to PCC-4 events to compensate for increases in reactivity resulting from the RCP [RCS] cooldown. to allow the transfer of the plant from the controlled state to the safe shutdown state. In this state the LHSI can be initiated in SIS-RHR mode, The uncontrolled boron dilution during the hot shutdown state bounds all the other PCC-2 events in this respect, because it starts with the lowest core boron concentration and with the core being just subcritical.

The actions to be performed by the operator to reach the safe shutdown state are:

Boration

Boration is needed to compensate for increases in reactivity resulting from RCP [RCS] cooldown.

The boration is performed by manually actuating the Extra Boration System (RBS [EBS]), which is F1A classified.

The single failure is assumed to occur in one of the two RBS [EBS] trains to minimise the available boration capability.

The boration ends when the RCP [RCS] boron concentration required for the safe shutdown state is achieved.

The volume of the RBS [EBS] tanks is designed to support this requirement as discussed in Chapter 6.

The constant flow rate delivered by one RBS [EBS] train discussed in Chapter 6 is designed to compensate for the increase of reactivity caused by the RCP [RCS] cooldown.

RCP [RCS] Cooldown

The RCP [RCS] cooldown to the RIS/RRA [SIS/RHRS] mode connection temperature is achieved by decreasing the VDA [MSRT] (F1B) setpoints of the SG.

The RBS [EBS] has to be operated in parallel with the RCP [RCS] cooldown to offset the reactivity increase caused by the cooldown. Because of the steady contraction of the primary coolant as its temperature falls, RBS [EBS] injection can proceed without letdown, avoiding any substantial increase of pressuriser level or challenge to the pressuriser safety relief valves.

The required flow rate of 2.78 kg/s with one RBS [EBS] train in operation and a cooldown rate of 25°C/h (see Sub-chapter 6.7) allow the shutdown state to be reached without overfilling the pressuriser.

13.5. SYSTEMS SIZING

The volume of the RBS [EBS] tanks and the RBS [EBS] flow rate are sized by this PCC event.

SECTION 14.3.13 - TABLE 1 (1/2)

Initial Boron Concentrations

These data refer to natural boron, not accounting for B-10 enrichment.

The boron concentrations are given in ppm without uncertainties

Initial BC in power operation

	BLX	EOL
Cycle 1	697	10
Cycle 2	1319	10
Cycle 3	1537	10
UO_2 – INOUT – 18 months	1610	10
UO_2 – INOUT – 22 months	1535	10
MOX – INOUT – 18 months	1649	10

The boron concentrations are calculated at nominal power, with all rods out (ARO), and with equilibrium xenon

Required BC in standard hot shutdown

	BOL	EOL
Cycle 1	392	< 0
Cycle 2	882	< 0
Cycle 3	1109	< 0
UO_2 – INOUT – 18 months	1224	< 0
UO_2 – INOUT – 22 months	1142	< 0
MOX – INOUT – 18 months	1339	< 0

The boron concentrations are calculated at hot shutdown conditions, with all rods in (ARI), in the absence of xenon poisoning.

The subcriticality criterion of -3400 pcm required for the Steam Line Break leads to a definition of boron concentration for all fuel management schemes.

The uncertainties to take into account (not including those due to onsite measurements) are 220 ppm for UO_2 schemes and 250 ppm for MOX schemes.

SECTION 14.3.13 - TABLE 1 (2/2)

Initial Boron Concentrations

These data refer to natural boron, not accounting for B-10 enrichment.

The boron concentrations are given in ppm without uncertainties

Required BC in standard cold shutdown

	BOL	EOL	Subcriticality Criterion
Cycle 1	925	401	- 3200 pcm
Cycle 2	1264	432	- 2900 pcm
Cycle 3	1464	483	- 2400 pcm
UO_2 – INOUT – 18 months	1528	497	- 2100 pcm
UO_2 – INOUT – 22 months	1425	499	- 1700 pcm
MOX – INOUT – 18 months	1599	580	- 1400 pcm

The boron concentrations are calculated at cold shutdown conditions, with all rods in (ARI), in the absence of xenon poisoning.

The subcriticality criterion required for the Rod Ejection at cold shutdown leads to a definition of boron concentrations for all fuel management schemes.

The uncertainties to take into account (not including those due to onsite measurements) are 200 ppm for all schemes.

Required BC in cold shutdown for refuelling and maintenance

	BOL to EOL
UO_2	2195
MOX	2440

The boron concentrations are calculated to just ensure the subcriticality at cold conditions, with all rods out (ARO), in the absence of xenon poisoning. This guarantees the core remains subcritical if the rods are withdrawn by error when the vessel closure head is lifted.

The UO_2 retained value is a bounding value for all UO_2 management schemes, and allows for the IRWST [In-containment Refuelling Water Storage Tank] design. The bounding UO_2 management scheme is the 'UO_2 – INOUT– 18 months' scheme.

The uncertainties to take into account (not including those due to onsite measurements) are 210 ppm for UO2 schemes and 160 ppm for MOX schemes.

SECTION 14.3.13 - TABLE 2 (1/2)

Critical Boron Concentrations

These data refer to natural boron, not accounting for B-10 enrichment.

The boron concentrations are given in ppm without uncertainties

Critical BC for power operation

	BOL	EOL
Cycle 1	164	< 0
Cycle 2	547	< 0
Cycle 3	763	< 0
UO_2 – INOUT – 18 months	839	< 0
UO_2 – INOUT – 22 months	790	< 0
MOX – INOUT – 18 months	825	< 0

The critical boron concentrations are calculated at hot zero power (303.3°C and 155 bar), with all rods in except the highest worth rod (ARI - 1), in the absence of xenon poisoning.

The calculation uncertainty is 100 ppm

Critical BC for standard hot shutdown state

	BOL	EOL
Cycle 1	64	< 0
Cycle 2	413	< 0
Cycle 3	591	< 0
UO_2 – INOUT – 18 months	689	< 0
UO_2 – INOUT – 22 months	593	< 0
MOX – INOUT – 18 months	688	< 0

The critical boron concentrations are calculated at hot zero power (303.3°C and 155 bar), with all rods in (ARI), in the absence of xenon poisoning.

The calculation uncertainty is 100 ppm

SECTION 14.3.13 - TABLE 2 (2/2)

Critical Boron Concentrations

These data refer to natural boron, not accounting for B-10 enrichment.

The boron concentrations are given in ppm without uncertainties

Critical BC for standard cold shutdown state

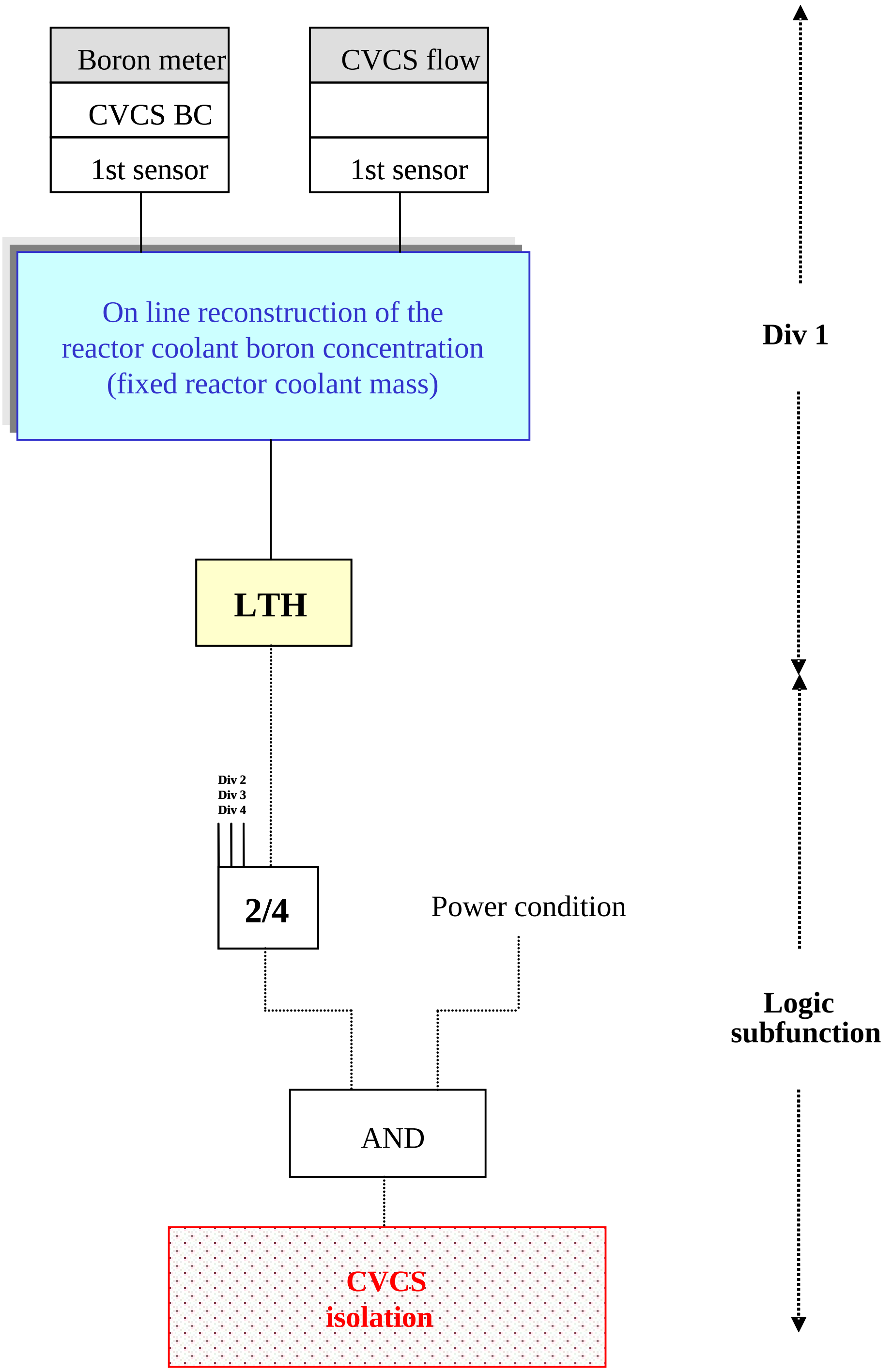
	BOL	EOL
Cycle 1	653	144
Cycle 2	975	190
Cycle 3	1203	268
UO_2 – INOUT – 18 months	1292	304
UO_2 – INOUT – 22 months	1232	342
MOX – INOUT – 18 months	1405	422

The critical boron concentrations are calculated at cold zero power (15°C and 1 bar), with all rods in (ARI), in the absence of xenon poisoning.

The calculation uncertainty is 100 ppm

SECTION 14.3.13 - FIGURE 1

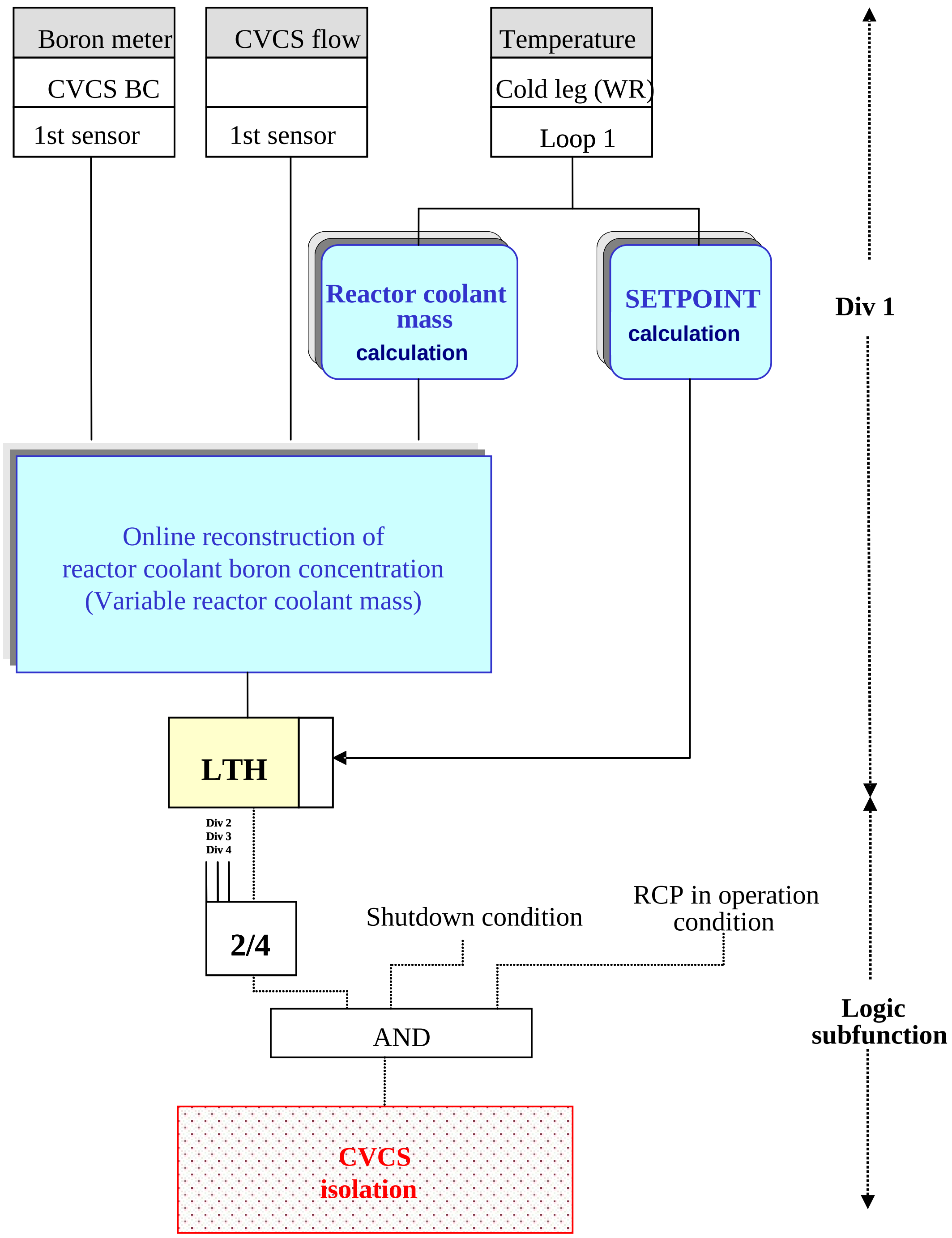
Anti-Dilution in Power Conditions Protection Channel[2]



[2] LTH is for Low Threshold

SECTION 14.3.13 - FIGURE 2

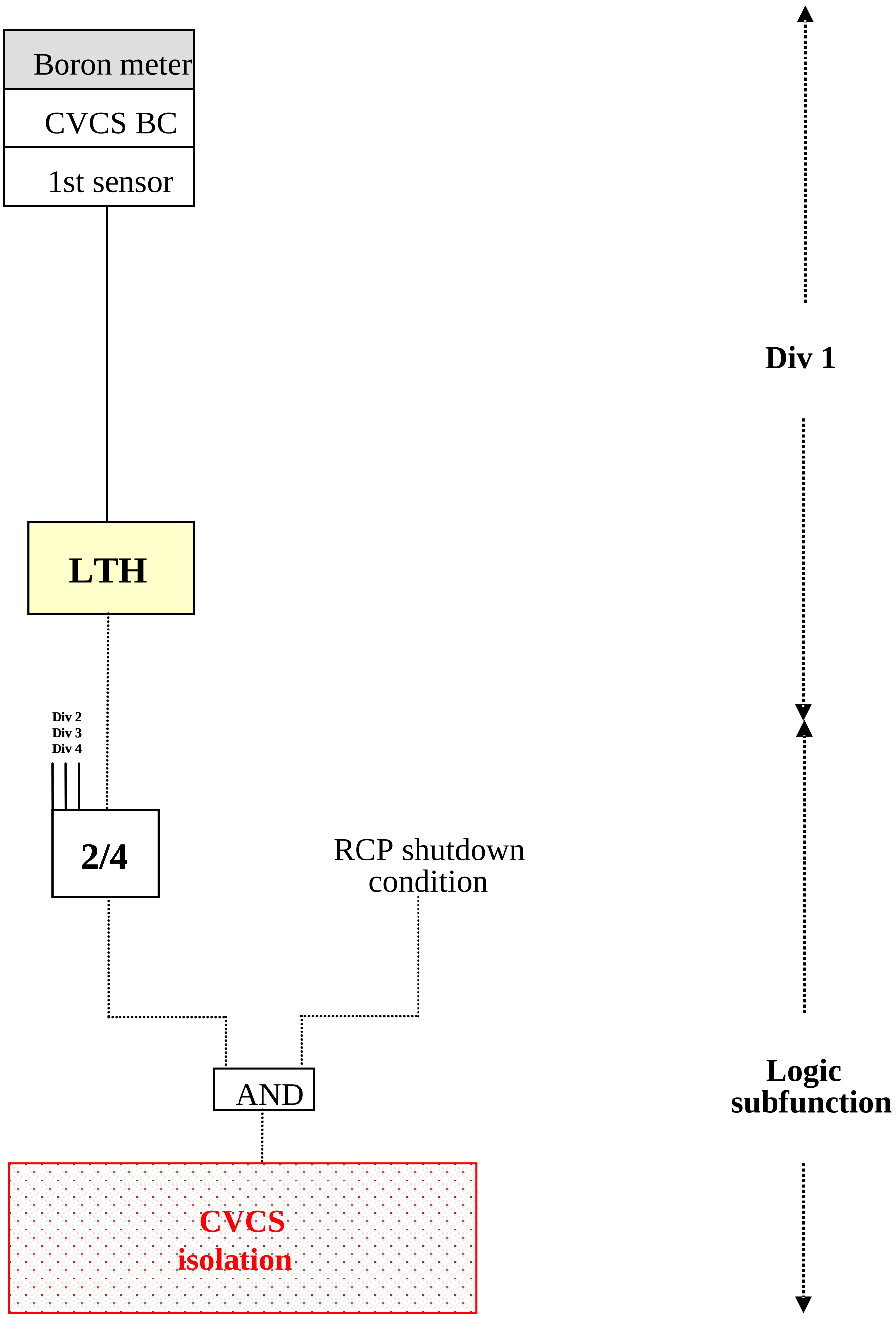
Anti-Dilution in Standard Shutdown States Conditions Protection Channel[3]



3 LTH is for Low Threshold

SECTION 14.3.13 - FIGURE 3

Anti-Dilution in Shutdown Conditions with Reactor Coolant Pumps Not in Operation Protection Channel[4]



[4] LTH is for Low Threshold

14. RCV [CVCS] MALFUNCTION CAUSING INCREASE OR DECREASE IN REACTOR COOLANT INVENTORY

14.1 ACCIDENT DESCRIPTION

The control of reactor coolant system inventory is performed by controlling the pressuriser water level. The actuators to control the pressuriser water level are the RCV [CVCS] letdown control valve, also referred to as the High Pressure (HP) reducing station, and the RCV [CVCS] charging pumps. In normal operation, one out of two pumps is in operation. The position of the letdown control valve is continuously adjusted as a function of the pressuriser water level.

The pressuriser water level control could fail due to inadvertent opening or closing of the RCV [CVCS] letdown control valve. This would lead to either a spurious increase or decrease of RCP [RCS] inventory. If this occurs, independent dedicated limiting functions will be activated automatically to stop the inventory change. The following staggered automatic limitation functions are provided to keep the pressuriser water level in the allowable range. Thus, the availability of the plant is increased by avoiding the need to activate safety functions, such as reactor trip or the opening of pressuriser safety valves.

The RCV [CVCS] malfunction is classified as a PCC-2 event.

Inadvertent RCP [RCS] inventory increase

- If the pressuriser water level exceeds the first setpoint MAX1, which typically corresponds to the limiting condition of operation and is about 80 cm above the setpoint, the RCV [CVCS] letdown receives an additional “opening” signal [Ref-1].

- If the pressuriser water level exceeds the second setpoint MAX2, which is still sufficiently below the protection function setpoint, all possible sources for further water level increase are isolated [Ref-1]. These include pressuriser spray (normal and auxiliary) and the RCV [CVCS] charging line. These automatic countermeasures prevent reactor trip and avoid pressuriser safety valves being challenged as a result of the over-pressurisation caused by the increase in water level.

Inadvertent RCP [RCS] inventory decrease

- If the pressuriser water level drops below the first setpoint MIN1, which typically corresponds to the limiting condition of operation and is about 80 cm below the setpoint, the RCV [CVCS] letdown receives an additional “closing to a minimum flow” signal [Ref-1].

- If the pressuriser water level drops below the second setpoint MIN2, which is set sufficiently high to preclude a switch-off of the pressuriser heaters or even a draining of the pressuriser, two automatic signals are initiated [Ref-1]. These initiate closing the letdown line and starting the second RCV [CVCS] charging pump.

Note that the signals remain active as long as the water level is outside the normal range. When the pressuriser water level returns to the normal range, these signals are deactivated automatically.

If the systems described above fail, e.g. due to failure of mechanical components, the plant cannot continue power operation. The actuation of safety functions, such as reactor trip, results in lowering the pressuriser RCP [RCS] pressure. This is true for both the inadvertent increase and decrease of reactor coolant inventory.

14.2 SYSTEM SIZING

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

15. PRIMARY SIDE PRESSURE TRANSIENTS (SPURIOUS OPERATION OF PRESSURISER SPRAYS OR HEATERS)

15.1. ACCIDENT DESCRIPTION

The control of the reactor coolant system RCP [RCS] pressure is performed by controlling the pressuriser pressure. Proportional heaters and ON-OFF heaters are used as needed to increase RCP [RCS] pressure. Two main sprays that originate from the cold legs of two RCP [RCS] loops, and one auxiliary spray that originates from the RCV [CVCS], are used to reduce the RCP [RCS] pressure. The setpoint for the pressure control is fixed at 155 bar during power operation and is automatically reduced with decreasing reactor coolant temperature in order to fulfil brittle fracture related requirements.

Two events that affect the RCP [RCS] pressure due to the malfunction of the pressure control system are considered here. Spurious operation of the pressuriser heaters results in a reactor coolant system pressure increase. Spurious operation of pressuriser sprays results in a reactor coolant system pressure decrease.

These events are classified as PCC-2 events. Transients involving inadvertent pressuriser spraying or inadvertent pressuriser heating are not limiting. Both of these events will be terminated by specific signals and actuations.

The following staggered automatic limitation functions are typically provided to keep the pressure in the required range and thus increase plant availability by avoiding activation of safety functions, such as reactor trip (RT) or opening of pressuriser safety valves.

Inadvertent Pressure Decrease

If the pressuriser pressure falls below the MIN setpoint, which typically is set below the limiting condition of operation (LCO), at about 5 bar below the setpoint, the pressuriser heating is activated by a specific signal and at the same time both spray systems are isolated [Ref-1].

Inadvertent Pressure Increase

If the pressure increases above the MAX setpoint, which typically is set above the LCO limit at about 5 bar above the setpoint, the pressuriser heaters are cut off and the normal pressuriser spray is activated by a specific signal [Ref-1]. In addition, if one or all of the reactor coolant pumps are lost and consequently the normal spray is partly or completely unavailable, the auxiliary spray from the RCV [CVCS] is available for pressure limitation [Ref-1].

If the above mentioned limitations fail, e.g. due to failure of mechanical components, and not taking in to account any possible manual countermeasures, the plant cannot continue power operation. This is because of the actuation of safety functions such as reactor trip with either inadvertent pressure decrease or increase and pressuriser safety valve opening, with pressure increase or safety injection at very low RCP [RCS] pressures would occur.

15.2. ACCEPTANCE CRITERIA

For the spurious actuation of the pressuriser heaters event, RCP [RCS] over-pressurisation and the potential for overfilling the pressuriser are of most concern. The departure from nucleate boiling is not expected to be challenged by this event.

For the spurious actuation of the pressuriser sprays event, the departure from nucleate boiling is the main consideration.

15.3. SYSTEM SIZING

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

16. UNCONTROLLED RCP [RCS] LEVEL DROP (STATES C, D)

16.1. ACCIDENT DESCRIPTION

The event considered is an uncontrolled level drop event during RIS/RRA [SIS/RHRS] operation at lowered RCP [RCS] level (¾ loop) in states C and D.

Mid-loop (¾ loop operation) is necessary to reduce the inventory of the RCP [RCS] upper plenum and the U-tubes of the steam generators during plant start-up, and to drain the pressuriser and purge the RPV head with nitrogen before opening to the atmosphere during plant shutdown (states C and D). This mode of operation is also required to maintain the water level below the maximum level in the RCP [RCS] during maintenance of the steam generators or the reactor coolant pumps.

Experience shows that most accidents and incidents occurring during operation in these states have been related to problems in level measurements and level monitoring [Ref-1]. Therefore, the monitoring of these operational modes and the measures necessary to avoid or mitigate spurious draining of the RCP [RCS] are specifically addressed.

The Uncontrolled RCP [RCS] Level Drop is classified as a PCC-2 event.

16.2. DESCRIPTION OF THE EVENT SEQUENCE

The draining of the RCP [RCS] during level drop down to ¾ loop can be initiated by either:

- an operator error during manual draining of the RCP [RCS],
- a control system failure leading to an excessive letdown flow rate through the low pressure RCV [CVCS] line.

For these cases, a conservative draining flow rate of approximately 25 l/s is credited [Ref-1].

16.3. DESCRIPTION OF THE INITIAL STATE

State C with lowered RCP [RCS] level is characterised as cold shutdown with three RIS/RRA [SIS/RHRS] trains in operation (RCP [RCS] temperature ≤55°C; P_{RCS} = 1 bar), as described in sub-section 1.2 of Sub-chapter 14.0. The RCP [RCS] is partly open to the atmosphere but it can be rapidly re-closed so that the SGs can be used for residual heat removal. The RCP [RCS] is at ¾ loop level.

The automatic protection system function for the RIS/RRA [SIS/RHRS] is the safety injection system RIS [SIS] actuation on RCP [RCS] loop level < MIN. In this state injection is only performed by the MHSI.

State D, described in sub-section 1.2 of Sub-chapter 14.0, is defined as cold shutdown with three RIS/RRA [SIS/RHRS] trains in operation with the RCP [RCS] temperature ≤ 55°C and P_{RCS} = 1 bar. The RCP [RCS] is open to the atmosphere so the SGs cannot be used for decay heat removal. The RCP [RCS] level can be at ¾ loop or higher.

In State D, the automatic protection system function for the RIS/RRA [SIS/RHRS] is the RIS [SIS] actuation on "RCP [RCS] loop level < MIN1". In this state, the RCP [RCS] injection is only performed by the MHSI. No automatic actuation of the LHSI trains occurs.

The decay heat removal is performed by three RIS/RRA [SIS/RHRS] trains, and the fourth train is on stand-by for manually actuated operation.

The letdown flow is routed via the low pressure (LP) reducing station, which is connected to the two discharge points downstream of the RIS/RRA [SIS/RHRS] heat exchangers (trains 3 and 4). In this plant shutdown state, the RCV [CVCS] pumps will normally be stopped and the make-up flow is re-injected into the RCP [RCS] by the RIS/RRA [SIS/RHR] train 3 or 4 via the RCV [CVCS] pumps bypass line and the volume control tank.

The MHSI pumps (trains 1 to 4) are in stand-by for safety injection; their delivery head is reduced to 40 bar before the start of RIS/RRA [SIS/RHRS] operation by opening their dedicated large mini flow line [Ref-1].

In these plant conditions, a single failure, as well as an unavailable system such as the second MHSI train, has to be considered.

16.4. RCP [RCS] LEVEL MEASUREMENTS AND CONTROL

The RCP [RCS] level measurement and control systems are defined in [Ref-1]. The main characteristics of these systems are summarised below.

There are four water level instruments, one in each RCP [RCS] hot leg. These instruments are dedicated to mid-loop operation, i.e. their measuring range covers the loop diameter from top to bottom.

The RCP [RCS] level during mid-loop operation is maintained by the loop level control. The reference value is set at ¾ (approximately 600 mm above the lower edge) of the loop. This is a preliminary value that must be confirmed by a commissioning test. The actual value will be generated by a selector which uses the four level measurements as an input in order to avoid a control malfunction if one level measurement fails. The difference between the reference and actual values is an input to the controller. This controller generates the control signal for the low-pressure reducing station, which adjusts the letdown flow rate and thus the RCP [RCS] level.

A comparison of the actual charging flow with the control signal will be performed to accelerate the letdown flow response.

The control range, including the uncertainties of the measuring points and the control device itself, will ensure that unacceptable level conditions for RRA [RHRS] suction are avoided.

16.5. CONDITIONS FOR SAFE LHSI/RHRS PUMP OPERATION

There are three phenomena that can threaten residual heat removal operations [Ref-1].

- Air intake into the RIS/RRA [SIS/RHRS] suction line due to vortex formation;
- Cavitation within the RIS/RRA [SIS/RHRS] suction line, valves and fittings;
- Cavitation (insufficient NPSH) at the suction of LHSI pumps.

These phenomena depend on the geometric construction and layout of the suction nozzles and lines, as well as physical parameters such as RCP [RCS] pressure and temperature, RCP [RCS] level, and RIS/RRA [SIS/RHRS] flow rates.

The present design of the EPR arrangement is characterised by the following data [Ref-1].

	RIS/RRA [SIS/RHRS] trains 1 to 4
RCP [RCS] suction nozzle	DN 300
Suction line	DN 250
Admissible suction flow * at RCP [RCS] pressure/temperature 0.8 bar/55°C	150 l/s
Admissible suction flow * at RCP [RCS] pressure/temperature 0.25 bar/55°C	120 l/s

* Note: These flow rates refer to operation at the ¾ loop level.

These parameters are determined by the vortex formation at 0.8 bar at higher RCP [RCS] pressure, and by the cavitation characteristics into the suction lines at 0.25 bar. These parameters are suitable to cope with the EPR cool down requirements [Ref-1], i.e. the cooldown to ≤ 55°C can be performed within 16 hours after shutdown by use of the first two trains, and for cooldown below 100°C by use of four RIS/RRA [SIS/RHRS] trains. Mid-loop operation is anticipated after about 23 hours by use of three RIS/RRA [SIS/RHRS] trains (one RIS/RRA [SIS/RHRS] train is on stand-by).

The admissible flow rates are conservatively assessed for operation at the mid-loop level and to limit vortex formation [Ref-1]. Globe valves have been assumed in the pipe layout and valve construction when calculating the admissible flow. The calculations show that the operational flow rate of 150 l/s that is assumed for residual heat removal does not need to be reduced. This result is valid even when the RCP [RCS] pressure and level are reduced.

The LHSI/RHRS flow must be reduced when the primary circuit is being drained. This draining is done during the start-up phase, prior to the filling up of the SG tubes.

16.6. LIMIT VALUES, ALARMS, AND INTERLOCKS

The mid-loop level control system automatically adjusts the operational level, such that the lowest limit value of the control range still ensures correct operation. In the event of a further level drop, before the onset of vortex formation, an operational alarm will be generated by the loop level measurements. In addition the RCV [CVCS] letdown line isolation valves and the low pressure reducing valve will receive an operational interlock generated signal (F2 classified) to close. The next low limit value of the four loop level measurements generates the "RIS [SIS] signal by RCP [RCS] loop level" signal. This signal is a F1A classified signal that performs the RCPB isolation (so as to isolate the RCV [CVCS] letdown line) and starts the MHSI pumps.

The staggering of these limit values will be optimised and will be verified by commissioning tests addressing the following aspects:

- The operational level control system has to maintain residual heat removal operation.
- The first operational alarm and interlock has to be set sufficiently below the control range to allow for measurement uncertainties and random variations in level.

- The safety injection signal for make-up by the MHSI pumps should be initiated at a low enough level to avoid spurious initiation, but early enough to allow continued operation of the LHSI pumps.

16.7. UNCONTROLLED DRAINING OF THE RCP [RCS], DESCRIPTION AND COUNTERMEASURES

Draining of the RCP [RCS] is performed when the cooldown has finished and the RCP [RCS] is at atmospheric pressure. The RCV [CVCS] letdown flow is routed via the low pressure reducing station to the coolant storage system for this purpose. This procedure is performed by the operator. The maximum flow rate for draining via this path is approximately 25 l/s [Ref-1]. The operator must carefully monitor the level measurements. The operator must manually activate automatic level control once the control range of the loop level control system is reached.

If an operator error is assumed, allowing continuous draining at the maximum inventory loss rate of 25 kg/s (as stated in section 2), the level would drop by about 0.4 mm/s. In these circumstances, the suction condition limits of the RIS/RRA [SIS/RHRS] pump would be reached after about 9 to 10 minutes at the earliest.

However, the operational interlock described above would isolate the letdown line and stop the draining before these conditions are reached. If this interlock and alarm are assumed to fail, draining will continue. Subsequently, the safety injection signal (Loop Level < MIN) would be initiated. This performs the RCV [CVCS] letdown line isolation, which stops the draining before reaching RIS/RRA [SIS/RHRS] pump stop conditions. Moreover, the four MHSI pumps, on stand-by, start and help to recover the RCP [RCS] inventory.

Consequently, the RCP [RCS] water level would be immediately increased and residual heat removal operation can continue without damage occurring to the RIS/RRA [SIS/RHRS] pumps.

16.8. CONCLUSION

It can be concluded that an uncontrolled RCP [RCS] level drop in states C and D would be successfully mitigated as either the RCP [RCS] draining is stopped immediately by an operational interlock, or a RIS [SIS] actuation on “RCP [RCS] loop level < MIN” signal by isolating the RCV [CVCS] letdown line and starting the MHSI. Although closure of the RCV [CVCS] letdown line isolation valves takes 40 seconds (conservative value), the level drop is stopped early enough to avoid RIS/RRA [SIS/RHRS] pump trip conditions. Moreover MHSI start-up leads to RCP [RCS] inventory recovery.

16.9. SYSTEM SIZING

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

17. LOSS OF ONE COOLING TRAIN OF THE RIS/RRA [SIS/RHRS] IN RESIDUAL HEAT REMOVAL MODE (STATES C, D)

17.1. ACCIDENT DESCRIPTION

The cooldown of the plant is described in [Ref-1]. The cooldown is normally performed by RIS/RRA [SIS/RHRS] trains 1 and 4 for RCP [RCS] temperatures between 120°C and 100°C. Below 100°C RCP [RCS] temperature, the RIS/RRA [SIS/RHRS] trains 2 and 3 are used as well. When the cooldown is finished, three RIS/RRA [SIS/RHRS] trains are sufficient to keep the RCP [RCS] temperature at 55°C.

The plant condition with lowered RCP [RCS] level, i.e. operation at ¾ loop level, which takes place at about 23 hours after shutdown, is the most onerous situation for heat removal. In these conditions, the water inventory in the RCP [RCS] is low and the suction conditions for the RIS/RRA [SIS/RHRS] are sensitive to temperature variations. The saturation pressure increases with increasing temperature, and the conditions within the RIS/RRA [SIS/RHRS] suction lines down to the LHSI/RHR pumps have to be monitored. This ensures that flashing to steam which would impair the residual heat removal function does not occur.

The failure of one RIS/RRA [SIS/RHRS] train is assumed. In addition, one train of LHSI/RHR is on stand-by in mode D and remains unavailable in both states.

This event is categorised as PCC-2. It is analysed in states C3 and D.

17.2. POSTULATED INCIDENT SEQUENCE

As mentioned above, in states C3 and D with lowered RCP [RCS] level, three out of four RIS/RRA [SIS/RHRS] trains are required to be in operation to maintain a RCP [RCS] temperature of below 55°C. The fourth RIS/RRA [SIS/RHRS] train is on stand-by. In these plant states, the failure of one RIS/RRA [SIS/RHRS] train in RHR operation has to be assessed. Therefore, for continuous primary side residual heat removal operation, only two RIS/RRA [SIS/RHRS] trains remain available.

The start-up of the RIS/RRA [SIS/RHRS] from stand-by is not claimed in the short term analysis.

It can be shown that the two RIS/RRA [SIS/RHRS] trains are able to maintain a RCP [RCS] temperature in a range which ensures their continuous proper operation without additional counter measures.

17.3. BOUNDARY CONDITIONS

The following conservative boundary conditions are considered:

- RCP [RCS] open with the lowest water inventory (¾ loop operation)
- Decay heat corresponding to 23 hours after reactor trip {CCI Removed} b {CCI Removed}
- One failed RIS/RRA [SIS/RHRS] train;

- Two RIS/RRA [SIS/RHRS] trains in operation cooled by the assigned trains of the RRI [CCWS] and the SEC [ESWS]
- No consideration of the RIS/RRA [SIS/RHRS] train that is on stand-by
- Maximum service water temperature (30°C) [Ref-2]
- RCP [RCS] temperature at about 55°C
- RCP [RCS] pressure slightly decreased for gas purging (0.8 bar)

The main data for each cooling chain train are [Ref-2]:

- LHSI heat exchanger capacity 1.1 MW/°C
- RRI [CCWS] heat exchanger capacity 2.5 MW/°C
- LHSI flow rate in RHR-mode 150 kg/s
- RRI [CCWS] flow rate 500 kg/s
- SEC [ESWS] flow rate 750 kg/s
- SEC [ESWS] suction temperature 30°C

17.4. OPERATIONAL CONDITIONS OF THE REMAINING RIS/RRA [SIS/RHRS] TRAINS AND CONCLUSION

The remaining two RIS/RRA [SIS/RHRS] trains are able to remove the decay heat of approximately 27.5 MW maintaining the RCP [RCS] temperature below 70°C.

Analyses for this transient have been performed at 4900 MW. The residual power is lower for the EPR_{4500} than for the EPR_{4900} (30 MW, BDR-99, Appendix 14B). In EPR_{4900} analysis, [Ref-1], the two trains of RIS/RRA [SIS/RHRS] can remove decay heat whilst maintaining the RCP [RCS] temperature below 70°C.

Without modifying the operating conditions in states C3 and D, the primary temperature will therefore remain well below 70°C for the EPR_{4500}, and there will be only a slow temperature increase after failure of one RIS/RRA [SIS/RHRS] train.

The operational flow rate of 150 kg/s need not be reduced, i.e. the NPSH available at RIS/RRA [SIS/RHRS] suction exceeds the NPSH required, so that continued operation of the unaffected RIS/RRA [SIS/RHRS] trains is ensured.

17.5. SYSTEMS SIZING

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

18. LOSS OF ONE TRAIN OF THE FUEL POOL COOLING SYSTEM (PTR [FPCS]) OR OF A SUPPORTING SYSTEM (STATE A)

Note: The classification upgrade of the PTR [FPCS] system following the GDA has not been taken into account in this sub-chapter.

18.1. OVERVIEW

The fuel pool cooling system (PTR [FPCS]) is described in Sub-chapter 9.1 “Fuel handling and storage”. The residual power in the fuel pool is higher with a single zone design for the fuel storage rack, as a result of an increase in the number of fuel storage cells; the decay heat has therefore been re-evaluated [Ref-1].

The rules for analysis of PTR [FPCS] accidents are described in Sub-chapter 14.0 of the PCSR.

The main characteristics of the system and of the electrical supply of the pumps are summarised below [Ref-2].

The two main PTR [FPCS] F1B trains are designed to remove decay heat from the fuel pool during normal operation (PCC-1) and during PCC-2 to PCC-4 events. Each of the two main PTR [FPCS] trains consists of two PTR [FPCS] pumps in parallel and one heat exchanger. Each heat exchanger can be cooled by two Component Cooling Water System (RRI [CCWS]) trains via a common header. A third PTR [FPCS] train, which is F2 classified, is also installed to remove decay heat from the fuel pool. This train is composed of one pump and one heat exchanger cooled by the intermediate EVU [CHRS] system, which in turn is cooled by a dedicated cooling chain (SRU [UCWS]).

The PTR [FPCS] mechanical flow diagram is shown in Section 14.3.18 - Figure 1.

Pumps 1 and 2 of the first main PTR [FPCS] train are connected to a dedicated switchboard, which is supplied by a main switchboard from electrical division 2. Additionally, during maintenance work on the main switchboard in division 2, the electrical supply for pumps 1 and 2 is provided via cross-connection no. 28 from the neighbouring division 1.

Pumps 3 and 4 of the second main PTR [FPCS] train are connected to a dedicated switchboard, which is supplied by a main switchboard from the electrical division 4. Additionally, during maintenance work on the main switchboard in division 4, the electrical supply for pumps 3 and 4 of the second main PTR [FPCS] train is provided via cross-connection no. 20 from the neighbouring division 3.

The extra (third) train is supplied by a dedicated switchboard from the main switchboard in electrical division 1 and can be connected via the cross-connection no. 21 to division 2 during maintenance work. Also, it is possible to supply this train (including the cooling chain) from the SBO diesel generator for electrical division 1 during station blackout.

An overview of the electrical supply of PTR [FPCS] pumps is shown in Section 14.3.18 – Figure 2.

18.2. PRINCIPLES USED FOR ANALYSIS OF PTR [FPCS] ACCIDENTS

The following methodology is used for all analysed PCC and RRC-A PTR [FPCS] accidents:

18.2.1. Initial conditions

An initial fuel pool water temperature of 50°C is assumed, which bounds all operating states [Ref-1].

18.2.2. Grace period

A grace period is calculated accounting for the thermal inertia during fuel pool transients. The grace period is calculated assuming no pool cooling is provided for the cases studied. The grace periods are calculated based on the time taken to reach a water temperature of 80°C or for the start of boiling.

To estimate this grace period, only heating of the fuel pool water by the decay heat from the spent fuel elements is considered, with a safety margin included. Heat losses to the structures in the pool are conservatively not taken into account.

18.2.3. Verification of the decoupling criteria for PTR [FPCS] studies

For PTR [FPCS] PCC and RRC-A studies, it must be confirmed that the final fuel pool temperature does not exceed the decoupling criteria imposed for the PTR [FPCS] design, namely [Ref-2][1]:

- 80°C for PCC transients without fuel pool draining,
- Avoidance of boiling for PCC transients involving fuel pool draining; 80°C in the long term when a main PTR [FPCS] train has been restored
- 95°C for RRC-A conditions.

The equipment available to provide fuel pool cooling is assessed by considering:

- possible loss of PTR [FPCS] equipment and/or support system equipment (e.g. RRI [CCWS], electrical power supplies) affected by the initiating event,
- possible loss of equipment due to application of the single failure criterion or LOOP or from consideration of preventive maintenance.

Applying the following PCC rule: “If the transient has no impact on performance of an F2 or NC system (no change of status, no change of operating and environmental conditions), and if the system was operating prior to the accident, the system may be assumed to continue at normal operation. No spurious commands from the I&C need to be assumed in these conditions” (see section 2.6 of Sub-chapter 14.0), certain F2 equipment can be claimed.

[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.

On this basis, the extra (third) train can be claimed in the PCC, and since the extra train including all its support systems and components does not change its state during the entire transient (i.e. no starting of pumps, no opening/closing of valves and no switches in the system, etc) the active single failure does not have to be applied to it.

Thus, to ensure the availability of adequate cooling at key times, the third train will be started as a preventive measure during maintenance work on the other PTR [FPCS] systems. It is started, for example, during maintenance on a heat exchanger or pump or on relevant support systems including the RRI [CCWS] trains or electrical power supplies.

The fuel pool water temperature is then calculated at steady state conditions, with decay heat corresponding to MOX fuel with a safety margin. A SEC [ESWS] or a SRU [UCWS] inlet temperature of 30°C is assumed. A RRI [CCWS] temperature of 38°C is also assumed [Ref-1] [Ref-2][2].

The volume of water in the fuel pool used in the studies is 1463 m^3 in cases without draining of the pool {CCI Removed} [a] [Ref-2][2].

18.3. INITIAL CONDITIONS IN NORMAL OPERATION (PCC-1)

In state A, which corresponds to reactor power operation, it is necessary to perform analysis considering 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.

Case 1: At EOC, i.e. before the beginning of shutdown for refuelling, maintenance of a main PTR [FPCS] train can be scheduled as the power in the pool is the minimum for state A. In these conditions the maximum heat load to be removed from the fuel pool is 2.99 MW.

Note: For more information on preventive maintenance see Sub-chapter 14.0.

Case 2: At BOC, i.e. at the start of a new reactor operating cycle, maintenance can be performed on a support system, for example, the RRI [CCWS]. The heat load in the fuel pool is the maximum for state A at this time. In these conditions the maximum heat load to be removed from the fuel pool is 5.85 MW.

Note: BOC is the most onerous time for this type of maintenance and therefore bounds the impact of maintenance during the complete cycle.

To ensure its availability, the third train will be started as a preventive measure during maintenance work on the other PTR [FPCS] systems. This includes maintenance on a heat exchanger or pump or on relevant support systems, e.g. on the RRI [CCWS] trains or electrical power supplies.

An initial fuel pool water temperature of 50°C is assumed which bounds all operating states (as described in sub-section 18.2.1).

[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.

18.4. GRACE PERIOD

An initial water volume in the fuel pool of 1463 m^3 during normal operation is assumed {CCI Removed} [a] with no passive failure (leakage) considered for PCC-2. In these conditions, the grace period without any means of cooling will be [Ref-1]:

For EOC: The water temperature in the fuel pool will reach 80°C 16.5 hours after total loss of the cooling function, and boiling of fuel pool water will start after a longer grace period of 27.4 hours.

For BOC: The water temperature in the fuel pool will reach 80°C 8.4 hours after total loss of the cooling function, and boiling of fuel pool water will start after 14.0 hours.

18.5. BOUNDARY CONDITIONS

The transient is analysed using conservative assumptions, consistent with the approach adopted for all other PCC events. Hence the fuel pool heat load for EOC and BOC is calculated using the bounding decay heat value for BOC, i.e. 5.85 MW with safety margin included.

Single failure and preventive maintenance are combined with the event.

A loss of offsite power (LOOP) occurring following an earthquake, without an additional failure, is assumed in the analyses, consistent with the general study rules discussed in Sub-chapter 14.0.

The precautionary start-up of the third PTR [FPCS] train during maintenance work is assumed in the study.

18.6. DECOUPLING CRITERIA

The PTR [FPCS] system is designed using a PCC-2 decoupling criterion which states that the water temperature in the fuel pool must not exceed 80°C (see section 2.10.1 of Sub-chapter 14.0).

18.7. TRANSIENTS

For events that cause loss of pool cooling, given the significant grace periods before a fuel assembly becomes exposed, it is considered that the controlled state has been reached at the time of the initiating event. The countermeasures identified below can be used to restore the unit to a safe state.

The most onerous configurations are analysed below. The two cases analysed bound all other initiating events, as well as all possible combinations of single failure and preventive maintenance [Ref-1][3] [Ref-2].

[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.

Case 1: EOC: in this situation, maintenance may be performed on either a complete main PTR [FPCS] train or sections of a main train (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 to be taking place on the heat exchanger of a main train, (for example main train 2), and the whole main train is then declared unavailable. (Case 1 is calculated using the bounding decay heat value for BOC, i.e. 5.85 MW, see section 18.5)

If the initiating event in this situation is a loss of the dedicated electrical switchboard which supplies PTR [FPCS] pumps 1 and 2 in the first PTR [FPCS] main train, PTR [FPCS] main train 1 is also lost: as a result the two main PTR [FPCS] trains are both unavailable.

According to the study rules, the extra (third) train remains available and in operation during the entire transient and cools the fuel pool. As a result, for this configuration and the given decay heat in the fuel pool, the stabilised fuel pool temperature will not exceed 47°C (assuming a SRU [UCWS] temperature of 30°C).

In the case of LOOP (without applying the single failure criterion), the third PTR [FPCS] train also shuts down and all pool cooling is lost. The emergency diesel generators are then started up, restoring power to the third train and its support systems, enabling pool cooling to be restored (safe state reached). Assuming that fuel pool cooling is lost for a fixed period of one hour, the average temperature of the pool will not exceed 54°C. In the long term, the average temperature of the water in the fuel pool does not exceed 47°C (assuming a SRU [UCWS] temperature of 30°C).

Case 1 : EOC		
	“Standard” PCC-2 event	**PCC-2 event with LOOP**
Initiator	Loss of the dedicated electrical switchboard	Loss of the dedicated electrical switchboard
Preventive maintenance	Yes (on one PTR [FPCS] train)	Yes (on one PTR [FPCS] train)
Single failure	Yes	No
LOOP	No	Yes
Main PTR [FPCS] train 1	Lost due to initiator	Lost due to initiator
Main PTR [FPCS] train 2	In maintenance	In maintenance
Third PTR [FPCS] train	Available for cooling the pool (Single failure not applied as the third PTR [FPCS] train is in operation when the initiator occurs)	Lost due to LOOP but re-powered for cooling the pool (no single failure)

Case 2: BOC: In this situation maintenance may be performed on one support system, for example the RRI 3 [CCWS 3] line. Initially the first main PTR [FPCS] train is in operation, cooled by RRI 1 [CCWS 1].

If the initiating event in this situation is the failure of the electrical switchboard which supplies the two pumps of the first PTR [FPCS] main train, both pumps of this train are lost. As a result the first PTR [FPCS] main train is lost.

If the single failure is now applied on the RRI 4 [CCWS 4] line, the second PTR [FPCS] main train can no longer provide cooling and, consequently, the two main PTR [FPCS] trains are lost.

Taking into account that the extra (third) PTR [FPCS] train will be operating as a precautionary measure as in case 1, and using the approach described previously, this extra (third) train remains available and in operation during the entire transient and cools the fuel pool. As a result, for the given configuration and decay heat in the fuel pool, the long-term fuel pool water temperature will not exceed 47°C [Ref-1][4].

In case of LOOP (without applying the single failure criterion), all PTR [FPCS] trains lose power. The emergency diesel generators are then started up, restoring power to the operational main PTR [FPCS] train and its support systems, enabling pool cooling to be restored (safe state reached). Thus, for the given configuration and decay heat for the fuel pool, the average long-term temperature of the water in the fuel pool will not exceed 52°C (assuming an RRI [CCWS] temperature of 38°C).

Case 2 : BOC		
Items	**“Standard” PCC-2 event**	**PCC-2 event with LOOP**
Initiator	Loss of the dedicated electrical switchboard	Loss of the dedicated electrical switchboard
Preventive maintenance	Yes (on a support system)	Yes (on a support system)
Single failure	Yes	No
LOOP	No	Yes
Main PTR [FPCS] train 1	Lost due to initiator	Lost due to initiator
Main PTR [FPCS] train 2	Lost due to single failure applied on support system (RRI [CCWS])	Lost due to LOOP and re-powered for cooling the pool
Third PTR [FPCS] train	Started and available for cooling the pool	Lost due to LOOP

[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.

18.8. CONCLUSION

Since, in all cases (BOC and EOC), the water temperature in the fuel pool remains lower than 80°C throughout the transient, the PCC-2 decoupling criterion is fulfilled for the transient 'Loss of one train of the fuel pool cooling system or of a supporting system (state A)'.

18.9. EFFECT OF A REACTOR EVENT ON THE PTR [FPCS]

In addition to the studies concerning the PTR [FPCS], a PCC-2 event affecting the reactor has been assessed with regard to its effect on PTR [FPCS] and pool temperature.

The event has been assessed by calculating the pool temperature at full load [Ref-1][5] [Ref-2], considering:

- Normal operation of the PTR [FPCS] cooling trains,
- Different PTR [FPCS] configurations: beginning and end of cycle, and end of refuelling, with either one or two main trains in service,
- Maximum decay heat (MOX fuel management) with and without a safety margin,
- a PTR [FPCS] pool water volume for normal operation of 1463 m^3,
- an RRI [CCWS] temperature of 40°C. This decoupling temperature is representative of the maximum temperature which may arise in the RRI [CCWS] during a PCC-2 transient affecting the reactor core.

The maximum fuel pool temperature calculated is 54°C, at 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-2 core events do not have a significant effect on the PTR [FPCS] system.

[5] 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.

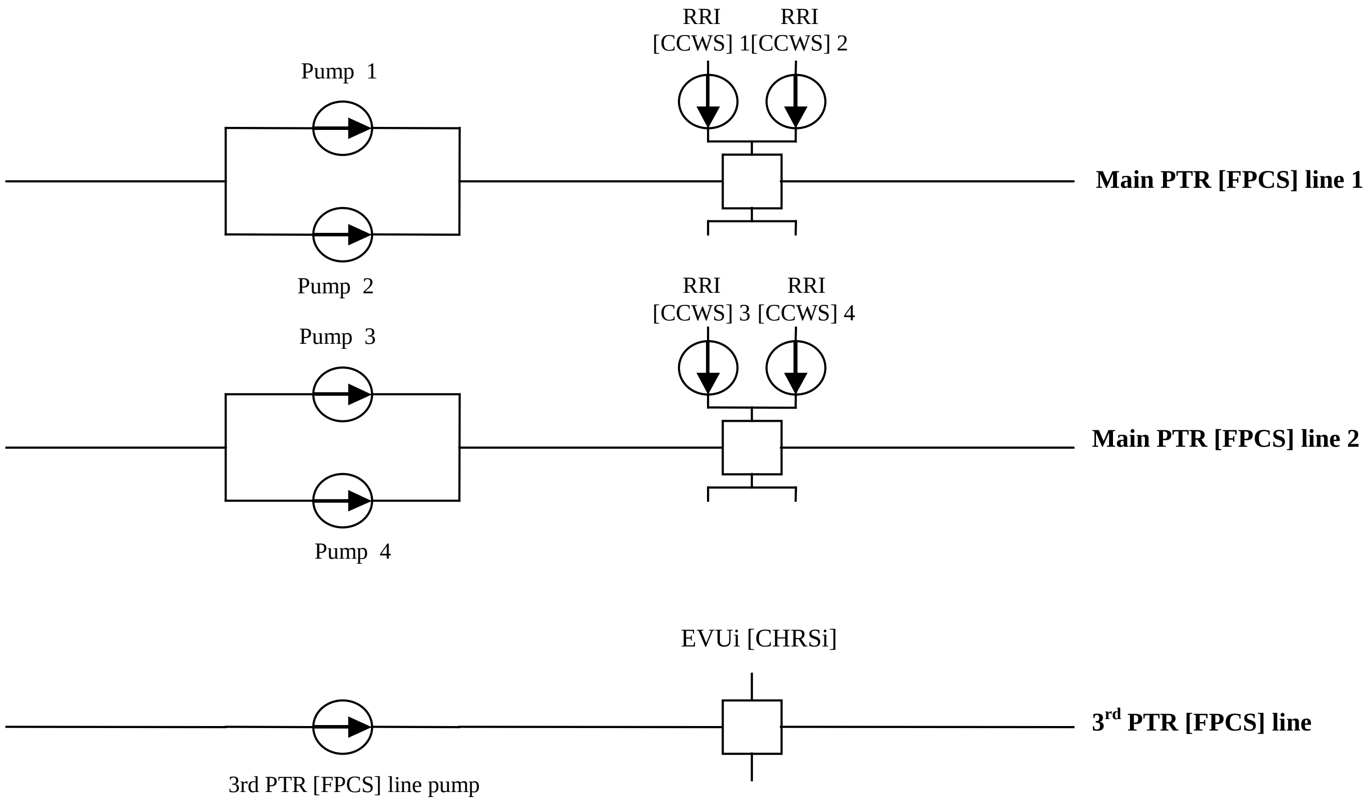
SECTION 14.3.18 - TABLE 1

Main assumptions and results (4500 MWTH)

		End of cycle	Beginning of cycle
Decay heat (MW)	**With margins**	**2.99**	**5.85**
Decay heat used for calculation(MW)	**With margins**	**5.85**	**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**
$T_{fuel\ pool}$ (final) (°C)			
During maintenance : Cooling with the extra (third) PTR [FPCS] train		**47**	**47**
Cooling with a main PTR [FPCS] train		**52**	**52**
Decay heat (MW)	**With margins**	**2.99**	**5.85**
Grace period without any cooling (hours)			
	To reach 80°C	**16.5**	**8.4**
	To reach 100°C	**27.4**	**14.0**

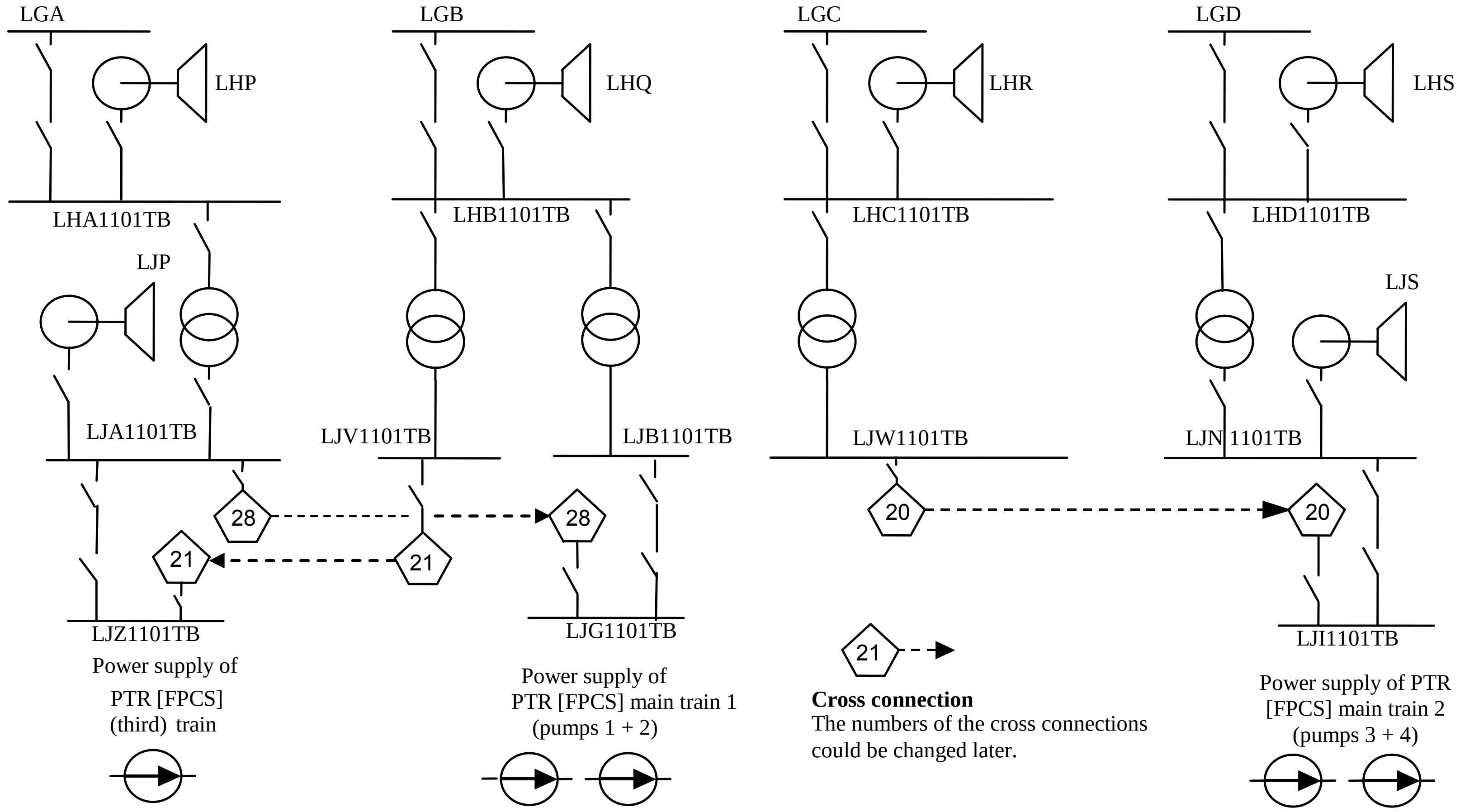
SECTION 14.3.18 - FIGURE 1

Simplified diagram of the PTR [FPCS] cooling lines [Ref-1]



SECTION 14.3.18 - FIGURE 2

Simplified diagram of the PTR [FPCS] electrical power supply [Ref-1]



19. SPURIOUS REACTOR TRIP (STATE A)

19.1. ACCIDENT DESCRIPTION

This transient is defined as an occurrence of a spurious reactor trip, or a spurious Automatic Reactor Trip signal, triggered when the unit's neutronic and thermal-hydraulic parameters are at their nominal operating values (including normal fluctuations and uncertainties).

A spurious reactor trip is less severe than any other PCC-2 event studied in Sub-Chapter 14.3. For all other PCC-2 events at power, an automatic reactor trip occurs when at least one of the unit's parameters deviates from its normal operating range.

The PCC-2 transient that follows a spurious reactor trip is similar to the PCC-2 event studied in section 5 of this sub-chapter for the loss of condenser vacuum, and its consequences are bounded by that transient:

- A loss of condenser vacuum causes an automatic reactor trip when all of the unit's parameters are within their nominal ranges, except for the primary system and steam generator pressures that are increased due to the turbine trip as the result of the loss of condenser vacuum.
- In a loss of condenser vacuum event, the turbine trip is quickly followed by a reactor trip. This scenario is similar to a spurious reactor trip in which a reactor trip automatically initiates the turbine trip. The only difference is a reversal of the reactor and turbine trip chronology, as these events are only separated by a few seconds.

19.2. SYSTEMS SIZING

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

SUB-CHAPTER 14.3 – 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.

3. EXCESSIVE INCREASE IN SECONDARY STEAM FLOW

3.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

3.1.1. General concern

3.1.1.1. Spurious actuation of partial cooldown (PC)

[Ref-1] S. Laurent. Fuel management – Neutronic design report (SCIENCE calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.1.2 Typical sequence of events

3.1.2.2. From the controlled state to the safe shutdown state

RCP [RCS] cooldown

[Ref-1] T. Godefroid. EPR – Basic Design – Extra boration system – System manual – D2.4 Physical phenomena determining the operating conditions. ITSR DC 162 Revision A. AREVA. November 1998. (E)

3.5. DESCRIPTION OF SENSITIVITY CASES (FROM THE INITIATING EVENT TO THE CONTROLLED STATE)

3.5.3. Specific assumptions

3.5.3.1 Neutronic data and decay heat

[Ref-1] S. Laurent. Fuel management – Neutronic design report (SCIENCE calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

[Ref-2] S. Laurent. Neutronic data for transient analysis (Update 4500 MWth). NFPSC DC 286 Revision B. AREVA. February 2006. (E)

3.5.4. Results

Neutronics calculation: EPR$_{4250}$ SMART calculation

[Ref-1] N. Nicaise. PSAR Sections 15.2.2D, 15.2.3D, 15.2.4D. PSRR DC 6 Revision B. AREVA. December 2003. (E)

Global extrapolation of the results from EPR_{4250} to EPR_{4500}

[Ref-2] J. Feingold. PSAR section 15.1 Plant characteristics assumed in the accident analyses. EPRR DC 1693 Revision C. December 2003. (E)

[Ref-3] S. Laurent. Fuel management – Neutronic design report (SCIENCE calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

[Ref-4] C. Panefresco. Fuel management – Neutronic design report (SCIENCE calculations). NFEPC DC 15 Revision C. AREVA. September 2003. (E)

5. LOSS OF CONDENSER VACUUM

5.2. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

[Ref-1] T. Godefroid. EPR – Basic Design – Extra boration system – System manual – D2.4 Physical phenomena determining the operating conditions. ITSR DC 162 Revision A. AREVA. November 1998. (E)

6. SHORT-TERM LOSS OF OFFSITE POWER (<2 HOURS)

6.2. METHODS AND ASSUMPTIONS

6.2.1. Important Phenomena and Qualification of the NLOOP and PANBOX/COBRA Models

__Qualification__

Qualification of NLOOP Models for Primary and Secondary Side Phenomena

[Ref-1] Dr Gerth, Dr Ro. Verification of the Korea specific Version of NLOOP by recalculation of the “Station Blackout Accident” at KNU 1. KWU Work-Report ST14-87-e2168 . AREVA. (E).
This document contains proprietary information and can be accessed only within AREVA offices

[Ref-2] Seitz, Hofmann. Verifikation des Rechenprogramms NLOOP für die Anlage Biblis-B am Verlauf der Störung TUSA ohne FDU vom 08.02.84.
[Verification of the NLOOP code for the Biblis-B Nuclear Power Station using a turbine trip incident without main steam by pass which occurred on Feb. 8. 1984.]
KWU Arbeitsbericht R15-84-2137 from 26.10.84. AREVA.
This document contains proprietary information and can be accessed only within AREVA offices.

[Ref-3] Richter . Vergleichsrechnung zu dem PKL III versuch "A5.2, totaler Ausfall des Speisewassers bei einem intakten DE" mit dem midifizierten Rechenprogramm NLOOP. [Comparison of results from the modified NLOOP code with the test outcome of the PKL III experiment" A5.2 Complete Loss of main feedwater in an intact steam generator"]. Siemens Arbeitsbericht KWU E411/91/2066. AREVA.
This document contains proprietary information and can be accessed only within AREVA offices.

Qualification PANBOX/COBRA Models for DNBR Related Phenomena

[Ref-4] R. Muller. Analysis of Grafenrheinfeld Pump Shaft Break Event with PANBOX 2 including COBRA 3-CP. Work Report A1C-1 305860-0, dated April 26, 1999. AREVA. (E). This document contains proprietary information and can be accessed only within AREVA offices.

[Ref-5] Finnemann . Proceedings of a Specialists Meeting on Calculation of 3-Dimensional Rating Distributions in Operating Reactors, Paris, 26th-28th November 1979 (Nodal Expansion Method for the Analysis of Space-Time Effects in LWRs). AREVA. (E)
This document contains proprietary information and can be accessed only within AREVA offices.

6.2.2. Short-Term Study (Evaluation of Minimum DNBR)

Flow Coast-Down and Heat Transfer to the Fuel-Clad

[Ref-1] Dr. Goll. Mikrostrukturelle Bestimmung der Brennstoffzentraltemperatur im Normalbetrieb.
[Determination of the fuel temperature in normal operation].
Work Report A1C-1 307321, dated January 17, 2000. AREVA.
This document contains proprietary information and can be accessed only within AREVA offices.

8. PARTIAL LOSS OF CORE COOLANT FLOW (LOSS OF ONE REACTOR COOLANT PUMP)

8.2. METHODS AND ASSUMPTIONS

8.2.1 Important Phenomena and Qualification of the Models Used in NLOOP and PANBOX/COBRA

[Ref-1] R. Muller. Analysis of Grafenrheinfeld Pump Shaft Break Event with PANBOX 2 including COBRA 3-CP. Work Report A1C-1 305860-0, dated April 26, 1999. AREVA. (E). This document contains proprietary information and can be accessed only within AREVA offices.

8.2.2 Initial and Boundary Conditions

Flow Coast-Down and Heat transfer between Cladding and Coolant

[Ref-1] Dr. Goll. Mikrostrukturelle Bestimmung der Brennstoffzentraltemperatur im Normalbetrieb. Work Report A1C-1 307321, dated January 17, 2000. AREVA. This document contains proprietary information and can be accessed only within AREVA offices.

10. UNCONTROLLED RCCA BANK WITHDRAWAL FROM HOT ZERO POWER CONDITIONS

10.2. METHODS AND ASSUMPTIONS

10.2.2. Boundary Conditions and Assumptions

[Ref-1] EPR Preliminary Safety Analysis Report (PSAR 4250) - section 15.2.2M. “Uncontrolled RCCA Withdrawal (State A)”. Edition 2003. AREVA. (E)

SECTION 14.3.10 - TABLE 2

[Ref-1] EPR Preliminary Safety Analysis Report (PSAR 4250) - section 15.2.2M. “Uncontrolled RCCA Withdrawal (State A)”. Edition 2003. AREVA. (E)

SECTION 14.3.10 - FIGURE 2

[Ref-1] EPR Preliminary Safety Analysis Report (PSAR 4250) - section 15.2.2M. “Uncontrolled RCCA Withdrawal (State A)”. Edition 2003. AREVA. (E)

11. RCCA MISALIGNMENT UP TO ROD DROP, WITHOUT CONTROL SYSTEM ACTION

11.3. RESULTS AND CONCLUSIONS

11.3.1 Results for the EPR_{4250}

[Ref-1] EPR Preliminary Safety Analysis Report, Section 15.2.2P “RCCA misalignment up to rod drop without limitation” (PSAR 4250). Edition 2003. AREVA. (E)

SECTION 14.3.11 - TABLE 1

[Ref-1] EPR Preliminary Safety Analysis Report, Section 15.2.2P “RCCA misalignment up to rod drop without limitation” (PSAR 4250). Edition 2003. AREVA. (E)

14. RCV [CVCS] MALFUNCTION CAUSING INCREASE OR DECREASE IN REACTOR COOLANT INVENTORY

14.1. ACCIDENT DESCRIPTION

[Ref-1] L. CARFANTAN. Definition of P/S I&C functions. NEPR-F DC 469 Revision A. AREVA. April 2009. (E)

15. PRIMARY SIDE PRESSURE TRANSIENTS (SPURIOUS OPERATION OF PRESSURISER SPRAYS OR HEATERS)

15.1. ACCIDENT DESCRIPTION

[Ref-1] L. CARFANTAN. Definition of P/S I&C functions. NEPR-F DC 469 Revision A. AREVA. April 2009. (E)

16. UNCONTROLLED RCP [RCS] LEVEL DROP (STATES C, D)

16.1. ACCIDENT DESCRIPTION

[Ref-1] Experience Feedback from test facilities and plants on RCS loop level measurements. DNM03451. AREVA. August 1998. (E)

16.2. DESCRIPTION OF THE EVENT SEQUENCE

[Ref-1] R. Gagner. EPR sizing 4500 MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

16.3. DESCRIPTION OF THE INITIAL STATE

[Ref-1] J. Feingold. EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 « Plant characteristics assumed in the accident analyses ». EPRR DC 1693 Revision C. AREVA. December 2003. (E)

16.4. RCP [RCS] LEVEL MEASUREMENTS AND CONTROL

[Ref-1] Experience Feedback from test facilities and plants on RCS loop level measurements. DNM03451. AREVA. August 1998. (E)

16.5. CONDITIONS FOR SAFE LHSI/RHRS PUMP OPERATION

[Ref-1] Safety injection System Manual, Chapter D2.3, Description, equipment characteristics. DNM01808. (E)

16.7. UNCONTROLLED DRAINING OF THE RCP [RCS], DESCRIPTION AND COUNTERMEASURES

[Ref-1] R. Gagner. EPR sizing 4500 MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

17. LOSS OF ONE COOLING TRAIN OF THE RIS/RRA [SIS/ RHRS] IN RESIDUAL HEAT REMOVAL MODE (STATES C, D)

17.1. ACCIDENT DESCRIPTION

[Ref-1] System Design Manual - Reactor Coolant System (RCP [RCS]) - Part 2: System Operation. NESS-F DC 538 Revision A. AREVA. May 2009. (E)

NESS-F DC 538 Revision A is the English translation of NFPMS DC 1132 Revision F.

17.3. BOUNDARY CONDITIONS

[Ref-1] S. Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

[Ref-2] R. Gagner. EPR Operation at 4250MWth. EPRR DC 1701 Revision B. AREVA. February 2004. (E)

17.4. OPERATIONAL CONDITIONS OF THE REMAINING RIS/RRA [SIS/RHRS] TRAINS AND CONCLUSION

[Ref-1] Safety Injection System (SIS) System Manual, Chapter D2.3, Description, equipment characteristics. DNM01808. (E)

18. LOSS OF ONE TRAIN OF THE FUEL POOL COOLING SYSTEM (PTR [FPCS]) OR OF A SUPPORTING SYSTEM (STATE A)

18.1. OVERVIEW

[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)

[Ref-2] 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.

18.2. PRINCIPLES USED FOR ANALYSIS OF PTR [FPCS] ACCIDENTS

18.2.1. Initial conditions

[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.

18.2.3. Verification of the decoupling criteria for PTR [FPCS] studies

[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.

[Ref-2] 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)

18.4. GRACE PERIOD

[Ref-1] EPR FA3 Preliminary Safety Report, Section 15.2.2x “PCC 2: Loss of a PTR cooling train or a PTR support system (State A)”. Edition 2006. EDF. (E)

18.7. TRANSIENTS

[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] EPR FA3 Preliminary Safety Report, Section 15.2.2x “PCC 2: Loss of a PTR cooling train or a PTR support system (State A)”. Edition 2006. EDF. (E)

18.9. EFFECT OF A REACTOR EVENT ON THE PTR [FPCS]

[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] EPR FA3 Preliminary Safety Report, Section 15.2.2x “PCC 2: Loss of a PTR cooling train or a PTR support system (State A)”. Edition 2006. EDF. (E)

SECTION 14.3.18 - FIGURE 1

[Ref-1] EPR FA3 Preliminary Safety Report, Section 15.2.2x “PCC 2: Loss of a PTR cooling train or a PTR support system (State A)”. Edition 2006. EDF. (E)

SECTION 14.3.18 - FIGURE 2

[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.